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# *Design and development status of small and medium reactor systems 1995*

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# FOREWORD

There is an increasing interest among Member States in the potential for deployment of smaller nuclear power plant units as energy sources for power production, heat generation, co-generation of heat and electricity, desalination, etc., and the IAEA has made an updated survey of the design and development status of small and medium power reactors (SMR) systems.

This publication presents material submitted by different vendors and organizations and conclusions drawn from the discussions of these contributions at a number of consultants meetings and an Advisory Group meeting. In this context, it should be noted that the role of IAEA is not to promote any particular design or solution, but to provide a forum for the exchange of information, and to compile reports on the results of such information exchanges.

The objectives of this report are to provide a balanced review of the current discussion on SMR potential and common features to both high level decision makers and technical managers. The report presents a review of the economic market and financial aspects of such systems. It also provides highlights of the incentives for the developments, as well as the main objectives and requirements currently under discussion in many Member States that are interested in nuclear power based on the deployment of small and medium power reactors.

International co-operation is considered an important and integral part of the development effort for deployment of nuclear energy for power production and/or heat applications. Mechanisms for such co-operation and possible activities that could be carried on an international level are also discussed in the report.

Detailed design descriptions and design status information on the major systems currently under development, as provided by the different vendors and organizations in accordance with a specified format, are also provided.

As noted, the role of the IAEA is to provide a forum for information exchange and to disseminate information to its Member Sates. Member States with existing nuclear programmes and other Member States with an interest in the future application of these systems could establish an international consensus on many subjects of mutual interest. This publication addresses the state-of-the-art that has been attained in the design of small and medium power reactors, their safety characteristics and development status. It also supplies information about design objectives, plant design alternatives and regulatory requirements which are topics that are of prime importance, particularly to developing countries.

## *EDITORIAL NOTE*

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# 1. INTRODUCTION

## 1.1. PURPOSE OF THE REPORT

Since the last IAEA Status Report on Small and Medium Power Reactors was issued nearly eight years ago[1], many designs have been introduced and several designs have matured. The main incentives for these designs have been improvements in safety, reliability and economics. In response to important commercial developments, the energy range of small and medium reactors (SMRs) is now taken as being up to around 700 MW.

The purpose of this report is to provide up-to-date balanced technical information to engineers and scientists involved in the development, design or licensing of SMRs. It brings out the design approaches and features of SMRs, in particular their simplicity, their larger flexibility for a variety of applications and the use of passive safety features as fundamental to the designs.

Part of the report is addressed to policy makers who are planning to evaluate SMRs it gives to them an overview of the present status of SMR development and the requirements and situations that would make their deployment most beneficial.

## 1.2. STRUCTURE OF THE REPORT

The bulk of the material in this report is contained in design descriptions of a large number of SMR projects. This material is contained in chapters 5, 6 and 7. Chapters 2 and 3 give an overview of the SMR field. Chapter 2 discusses the market opportunities, the constraints and the extent of the development activity. Chapter 3 summarizes the activity on a national basis and on an application and reactor technology basis and includes a brief summary of some activities of the international organizations.

A format for the reactor descriptions was developed over several consultancies and has been followed by all the contributions. The descriptions have been supplied by the reactor development teams in Members States. The descriptions are structured in the six sections listed in Table 1.1.

TABLE I.1: STRUCTURE OF THE DESIGN DESCRIPTIONS IN THIS TECDOC

1. Design objectives and special features
2. Design description
3. The safety concept of the design
4. Extended design data listing
5. Design and developmental status
6. Statement on the economics of the described system.

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[1] *IAEA-TECDOC-445, Small and Medium Power Reactors (1987).*

These areas had to be structured in such a way that they could address different design approaches and different technology lines (i.e. WCR, GCR, LMCR) efficiently. In order to give a real impression of the design and development status a special section is dedicated to this aspect. In addition, the depth of coverage varies according to the design status. Design descriptions are presented in Chapters 5, 6 and 7 corresponding to detailed, basic and conceptual designs respectively. References are given at the end of each design description.

# 2. OVERVIEW OF THE SMR MARKET

## 2.1. THE SMR POSITION IN 1995

### 2.1.1. The SMR Market

Energy consumption is increasing all over the world. This is true both in developing countries and in more developed economies. United Nations estimates indicate that world population is due to increase more in the next three decades than ever before in a similar period. A reference scenario for future electricity demand drawn up by the World Energy Conference, Madrid [1] predicts close to a doubling of the world's generation capacity from 1990 to the year 2010. This growth of energy demand is dominated by developing countries. There are many places and applications where this increased demand will be best met by power plants in the SMR range, due to a small grid system or for application in a remote area or for a special purpose.

The world primary energy consumption amounts to well over 300,000 peta Joules and over half of that is used as hot water, steam and heat. Only a few nuclear power plants are being used for heat applications. Heat applications include desalination, hot water for district heating, heat energy for oil recovery, petroleum refining, petrochemical industries, and methanol production from hard coal. Clearly nuclear heat production could play a major and important role. Nuclear power at present is used mainly for electrical power generation which only forms 30% of the energy market. There have been numerous studies on the use of SMRs for heat applications rather than electrical generation and some of these studies have shown the SMR option to be viable both technically and economically[2].

If this expected increase in power demand were to be met from fossil fuel sources, there would be an increase in the release of $CO_2$ to the atmosphere. There are very strong concerns about the effect of $CO_2$ and other gases on global warming. Other emissions from fossil fuel combustion lead to atmospheric pollution and acid rain. Nuclear power clearly has the potential to reduce these harmful environmental effects and since the projected growth in energy demand is dominated by the growth in developing countries, there is a large opportunity for reactors in the SMR range.

### 2.1.2. SMR Projects

With such a range of possible applications in many different parts of the world, a large number of different R & D and design projects have been set up. Fig 2.1 lists most of these projects, many of which there are design descriptions in chapter 5, 6 and 7 of this report. Fig 2.2 lists those for which descriptions have been submitted by the design teams to this report and indicates the status of their development. LWRs, HWRs, GCRs and sodium cooled reactors all have active development work in various Member States. Information on other designs can be found in the OECD report on small and medium reactors [3].

Over the past 30 years there have been many market surveys for SMRs. They have shown a potential for sales of a large number of reactors before the turn of the century. These estimates of the market have turned out to be grossly over optimistic but have encouraged developers to continue their efforts. In spite of a moderate response from the market, there is still a very large development effort continuing but few of the advanced SMR designs have yet been in operation to demonstrate their capabilities. Indeed, few of them have been funded through the detailed design stage to make them ready for construction. They do, however, present a variety of solutions to the problems of

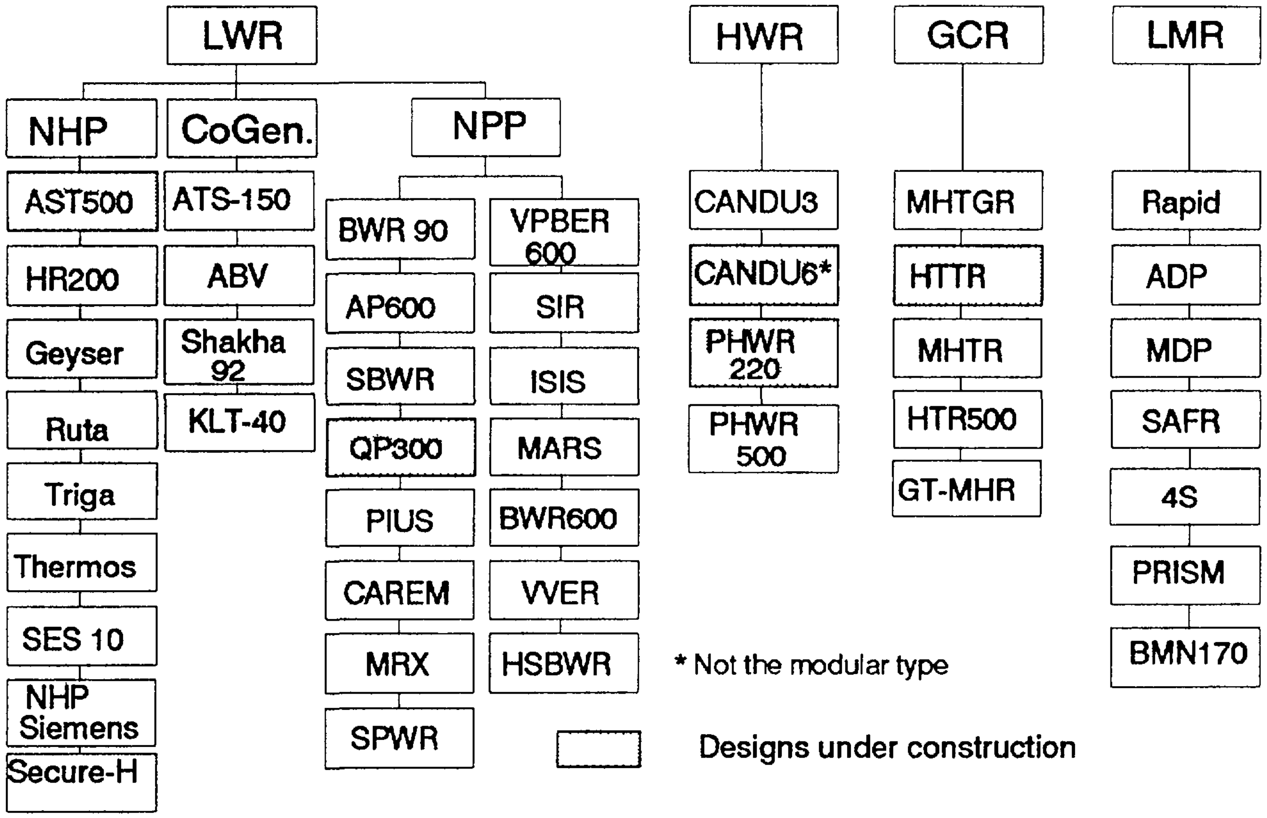



*Fig 2.1 SMR Development Lines in the World*

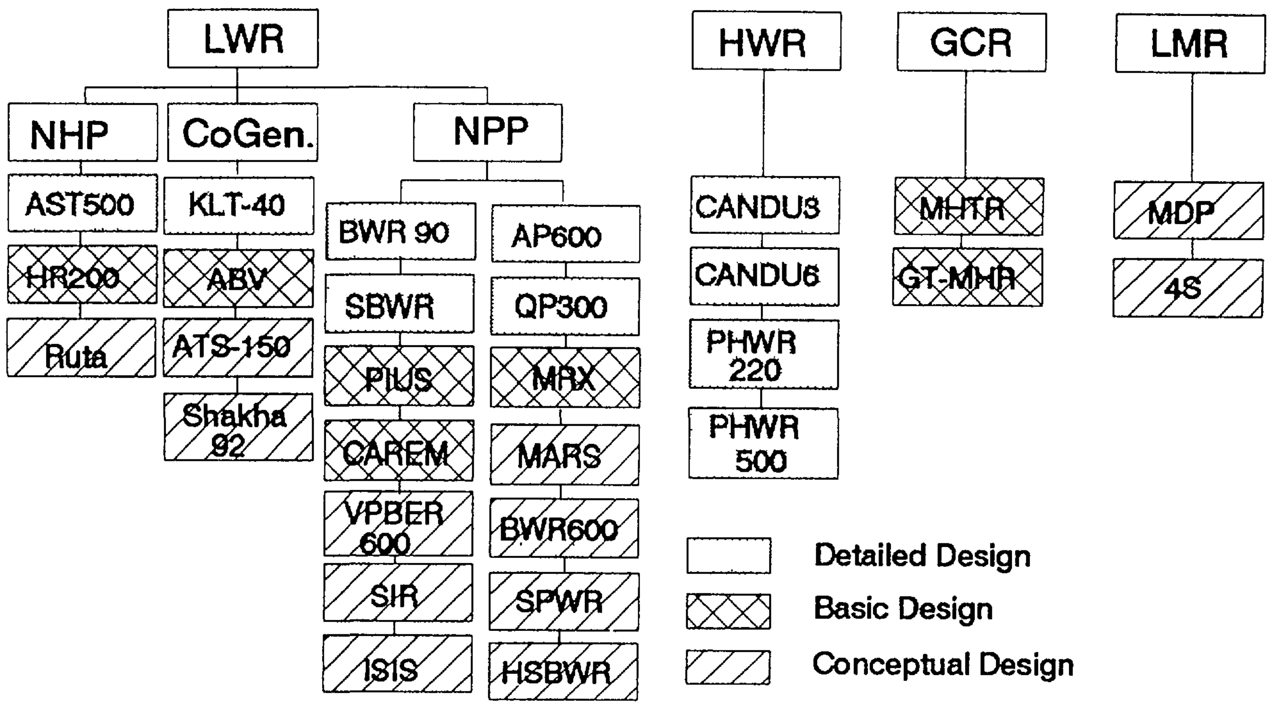



*Fig 2.2 SMR Designs in this TECDOC and their Design Status*

reactor design for future designers to draw on and to give an impression to purchasers of the capabilities of current designs, which could be developed to meet their needs.

There is thus a gap between the designs available but not built, and their exploitation in what appears to be a potentially large market.

### 2.1.3. Bridging the gap

One possible way of bridging the gap would be for vendors to collaborate on a design to spread the design and development costs and for users to collaborate to define an SMR requirements document for particular applications. There have been some notable vendor collaborations in industrialised countries demonstrating that this is a possible way forward. Requirements documents have been produced for power generation and requirements have been harmonised on a regional basis (Asia, Eastern Europe and North America). Developing Member States having similar technological and financial circumstances could establish their version of requirements for an identified market. Such requirements could be taken up in some of the developing projects to increase their prospects for constructions.

## 2.2. INCENTIVES FOR DEVELOPMENTS

Small and medium size reactor development has many incentives; some are economic others are safety related. The motivation for these developments has included the need to influence public acceptability of nuclear power. The simplicity of reactor designs should improve the transparency of their reactor safety. Another incentive to SMR development has been its suitability for the implementation of new design approaches. Innovative and evolutionary designs with novel features have been implemented in the SMR range. A passive safety approach has so far been the technology of small and medium reactors. Some Member States have been or are interested in SMR developments since TMI and Chernobyl as an answer to utility, as well as public requirements, in particular to safety and public acceptance issues. The economics of nuclear reactors are summarised in Appendix I. SMRs have particular characteristics which can enable them to be economically viable in spite of losing the advantage of the economics of scale. These economic incentives are included in the list below.

The incentives for the development of SMRs can be summarised as follows:

- Simpler design,

  An SMR can be modularised more easily and constructed in a shorter time than larger plants, thus reducing constructions costs (including interest during construction) and generating earlier revenues.

- Increased safety margins leading to a longer grace period,

  Passive safety features simplify the design and attain the required safety objective in a different way compared to large plants with more active safety systems. This could reduce cost and simplify the presentation of the safety of the reactor to both regulatory authorities and the public.

- Lower severe core melt frequency and minimum accident consequences.

- Better match to grid requirements,

  SMRs can provide a better match to small grids or to a slow growth of energy demand.

- Better use of nuclear industry infrastructure and manpower skills in countries with smaller nuclear programmes.

  One 600MWe unit every 2 years is preferable to one 1200MWe unit every 4 years.

- SMRs could open up energy markets,

  There is interest for process heat, desalination, district heating and oil extraction as well as power generation.

- Easier multi-unit siting and bulk ordering,

  Economies of series and collocation.

- Lower financial risk due to:

  lower financing requirements per unit,

  shorter and better predictable construction schedule.

## 2.3. OBJECTIVES AND REQUIREMENTS FOR SMRs

Development or deployment of SMRs will take place in a programme which includes some aspects of the following general objectives:

1. The size of reactor is appropriate to a geographical location, distribution network or application.

2. It should be economic within the constraints of the other objectives.

3. It must be demonstrably safe and licensable.

These general objectives are applicable to reactors of any size but there are particular aspects of reactors in the SMR range which help in meeting them and which are worth repeating.

1. Size. SMRs are appropriate for remote regions with limited load. They are appropriate for utilities with small grid systems. They are appropriate for some dedicated applications such as desalination, district heating or process heat possibly in a co-generation mode.

2. Economics. SMR designs all claim simplification of design to reduce costs and offset to some extent the economies of scale. Modularisation allows a greater element of factory construction and assembly and is generally less expensive than work on site. It leads to shorter construction times and savings in interest during construction. The reduced capital requirements compared with large plants may well be attractive to purchasers.

3. Safety. Most SMRs make extensive use of inherent safety features and passive safety systems. Such systems are appropriate to SMRs and are harder, if not impossible, to engineer on large reactors. They tend to be simpler than active systems resulting in a simpler safety case and easing the problems of public acceptability.

While objectives provide for general and long-term applicable targets for nuclear reactors of present and future designs, requirements provide more specific, clear and complete statements by utilities in a given country. The requirements are usually grounded on well proven technology and long experience of commercial operation. The design requirements usually take into consideration problems of the past and incorporate new features assuring simple, robust, and more forgiving designs. It also provides for a common ground for regulators and vendors on licensing issues. Well defined requirements agreed upon by regulators, vendors and utilities provide for investor confidence. The design requirements usually cover the whole plant (i.e. NSSS, BoP, safety systems etc.) and provide clear specifications with regard to performance, maintainability and plant economics. Clearly such requirements are specific to a given type of reactor. Taking into consideration infrastructure and experience, requirements in most Developing Countries and some Industrialised Countries are expected to be easily fulfilled by a small or medium reactor.

## REFERENCES

[1] Draft Summary Global Report, World Energy Conference, Madrid (1992)

[2] International Atomic Energy Agency Nuclear Applications for Steam and Hot Water Supply, IAEA-TECDOC-615, Vienna (1991).

[3] Organisation for Economic Cooperation and Development, Small and Medium Reactors: 2 volumes, OECD (1991).

# 3. PROGRAMMES FOR SMR DEVELOPMENT

## 3.1. CURRENT ACTIVITIES IN MEMBER STATES

Nuclear energy plays an important role in supplying a significant portion of the world electricity demand. Reactor generated heat has been utilized in several parts of the world for district heating, process heat application, and seawater desalination. It should be noted here that over 50% of the world energy demand is utilized for either hot water or steam production. Such processes could be carried out more efficiently and cleanly utilizing nuclear energy. In spite of the slow down or stoppage of nuclear programmes in many Member States in the last decade, utilization of nuclear power is picking up momentum at various bases in South East Asia, Eastern Europe and the Russian Federation.

Some South East Asian countries believe strongly that nuclear power will be a principle source of energy for years to come. Small and medium reactors form a major part of this activity. The People's Republic of China has a well developed nuclear capability having designed, constructed and operated reactors used in submarines. These reactors can be regarded as SMRs and the skills needed to implement them are the same as those needed for terrestrial power plants. China has some 10,000 nuclear engineers in three major centres in different parts of the country as well as other centres which make a major contribution. There is a particular interest in small power generation and district heating reactors to help ease the current enormous logistic problems in distributing 1.1 billion tonnes of coal around the country each year. A 300 MWe PWR (QP300) has been in operation for three years and two 600 MWe reactors are under detailed design and site preparation. All three reactors are of the evolutionary reactor type. Longer term plans call for development of a 600 MWe passive system (AC600). A 5 MWt integrated water cooled reactor has been built and operated for five winter seasons (since 1989) for district heating. Another purpose of the 5 MW reactor is the development work for other applications such as desalination. A 200 MWt demonstration heating reactor construction has been started aiming at start of operation by the year 1998. A 10 MWt high temperature gas cooled reactor for process application is also under construction. The test Module HTR will be operated by the Institute of Energy and Technology at Tsinghua University (INET) and is expected to go critical by 1998. The system is used to accumulate experience in plant design, construction and operation. Several applications, such as electricity generation, steam and district heat generation are planned for the first phase. A process heat application, to produce methane, is planned for the second phase.

India has some early reactors of the CANDU type supplied by Canada but India has adopted a prime policy target of self reliance in nuclear power development, based on heavy water moderated reactors. Five units of the 220 MWe PHWR type are under construction and all are expected to be in operation by the year 1997. An additional four units of the same type and an extra four units of a scaled up 500 MWe type are planned. This would fulfill 57% of phase one of the programme targeted at 10,000 MWe generated by nuclear power. The main objective is to make maximum economical use of the uranium natural resources in the first phase. The second phase is planned to utilize fast breeder reactors fueled by plutonium generated in phase one. A 500 MWe prototype is in the detailed design stage. India also has large reserves of thorium which exceed its reserves of uranium. The heavy water reactor with its very good neutron economics is well suited to the thorium/U233 cycle and a programme of R&D work for phase three, aiming at utilization of the U233/Th cycle in an advanced heavy water reactor, has been initiated.

Japan has a high population density and a shortage of suitable sites for nuclear reactors due to the large fraction of the landmass covered by mountainous terrain. This has led to a preference for large reactors on the available sites to maximise the power output from them. There is a very strong and diverse programme of reactor development supported by the big industrial companies, by the national laboratories and by the universities. Three large industrial companies have developed their own LWR designs in the SMR range and Japan Atomic Energy Research Institute (JAERI) has several more innovative designs. Twelve units are under various stages of construction or commissioning, but only two of the twelve units are in the SMR range. The MONJU fast breeder reactor (280 MWe), a prototype demonstration plant, started commercial operation in April 1994. Full scale operation is planned for December 1995. The SIGA 540 MWe BWR started commercial operation in July 1993. The guidelines of the programme in LWR technology call for improvement in the use of uranium resources, flexibility in the fuel cycle to allow for long range uncertainty, and improvement in safety including utilization of passive safety systems. At least seven different designs are currently being worked on in the SMR range; namely SPWR, MRX, MS 300/600, HSBWR, MDP, 4S and RAPID. SPWR and the marine reactor MRX are integrated PWRs. The MS series are simplified PWRs. HSBWR is a simplified BWR. MDP, 4S and RAPID are small sodium-cooled fast reactors. Preliminary investigations have shown a high level of safety, operability and maintenance. The economics of these systems have been promising and they are expected to form part of Japan's next generation of reactors.

Japan has also a development programme where gas cooled reactors in the small and medium size range are under development. A High Temperature Engineering Test Reactor (HTTR) has been under construction since 1991 at Oarai. The 30 MWt reactor will be the first of its kind to be connected to a high temperature process heat utilization system with an outlet temperature of 850°C. The system will be used as a test and irradiation facility and will also be utilized to establish the basic technology for advanced HTGR for nuclear process heat applications. The system is expected to go critical by 1997. However, the main trend in power generation is still taking the line of larger (1000-1300 MWe) evolutionary light water reactors. The guidelines of the programme put user-friendliness, improvement in operability, and flexibility of core design as prime design objectives.

Korea has nine nuclear power plants in operation and has an ambitious programme for the further development of nuclear power. The country is not well blessed with indigenous resources of fossil fuel and has to rely on imports. Furthermore 80% of the countryside consists of mountainous terrain which encourages the installation of large stations to make optimum use of the available sites. Most of the existing plants are of the PWR type, but, since April 1984, there has been a policy to install a PHWR (~700 MWe) to give some diversification in supply and operation. A second PHWR (700 MWe) is under construction and an extra two are in the stage of seeking a construction permit. Large size PWRs (1000 MWe) are expected to form the main component of nuclear power installation in Korea well into the next century. The optimal combination of PWRs and PHWRs is mainly to maximize the usage of the uranium resources through the use of spent PWR fuel in the PHWR without chemical reprocessing. This choice has been the first phase of a strategy of reactor development in Korea. The study "The outlook and development strategy of nuclear energy for the 21st century in the Republic of Korea" draws up four different phases. The second phase recommended a review of several different types including integral PWRs, the CANDU-3 and the modular HTGR to supplement the programme. The energy sources in this case would be also considered for other purposes such as district heating. High conversion reactors are recommended for the 3rd phase up until breeder reactors (the forth phase) become more economically viable with construction foreseen in 2030. Medium size reactors are expected to play an important role in technology development specially if the Modular HTGR programme is pushed forward.

In phase 1 six medium PHWR are expected to be constructed by the year 2006. The mid- size plant of the PHWR would certainly form part of the Korean power source, but the (PWR) Korean standard nuclear power plant KSNP with 1000 MWe rating is expected to form the main stream of the power generation industry in Korea.

Indonesia has a very rapid growth of population spread over 13,000 large and small islands. There is a clear future potential for reactors in the SMR range. However, the main island has over half the current population and could take a large station; a feasibility study covering this and all aspects of Indonesia's possible nuclear programme has been undertaken. The outcome is in favour of the nuclear power option. 7000 MWe of nuclear capacity is being considered up to the year 2015. Optimal plant size is being looked at and a large number of 600 MWe units are being considered. Indonesia has deposits of tar sands for which extraction based on nuclear heating using HTGRs is being investigated. The first nuclear power programme on public acceptance is being executed.

Thailand has just started a feasibility study on the construction of a nuclear power plant. In the Philippines the restart of construction of the 651 MWe PWR by Westinghouse is under consideration.

A particularly interesting stage in the development of the international trade in nuclear power is a contract between Pakistan and China for the supply of a 300 MWe PWR of Chinese design. In many other parts of the world the commitment of money needed to bring small reactor designs to the stage where such a contract would be feasible has been lacking.

In the Russian Federation, according to a report-understood to have the endorsement of the Ministry of Atomic Energy of the Russian Government- the future strategy is drawn up in three stages. Stage one covers 1990-2000. This period would mainly be concerned with the renovation of the existing reactors, and the development of a new generation of reactors with enhanced safety as a prime target. The second stage, covering the following ten years, will see continuation of further development and evolutionary improvement of existing reactor designs; in addition, prototype plants incorporating new technology will be designed and constructed. A large scale expansion of nuclear power utilization is called for. Post 2010, depending on how environmental and fuel supply problems in the electricity generation industry evolve, the plan calls for a large expansion of nuclear power on the basis of innovative reactors designed and built in the second stage. There are at present a large number of SMR types under development in Russia which are all presented as viable projects at international meetings. It seems likely that there will be a consolidation into a smaller number to take forward into the prototype stage over the next few years.

The current Russian programme is largely based on 1000 MWe units but the 500-600 MW range is well represented in the development programme. Two units of 600 MWe each are planned in the Far East region of the country for the period 2000-2010. Two others in Karel'ska are planned for the same period.

Russia is a country with a clear scope for the deployment of smaller plants due to its huge land mass with well separated communities living in areas with harsh winters. The nuclear energy option seems to have much better economics compared to conventional sources for application in remote areas, especially for domestic heating. Several reactors of small size (10-30 MW) are planned for construction around the year 2000.

Eastern Europe has VVER units at the 440 MWe size but for the future larger units are mostly looked at. The main trend in the region, however, is to improve the safety and I&C systems on the reactors currently in operation to bring them to acceptable levels of environmental performance, reliability and safety.

In Western Europe, almost all utilities have opted for nuclear power plant of the large size (1000-1500 MWe) if they have opted for nuclear at all. France is the only Western European country that has maintained a large scale programme of systematic development, installation and operation of nuclear power. Economic, political or infrastructure problems have caused suspension or intermittent progress in programmes in other countries. In this climate considerable efforts are under way also to cover severe accidents as design basis accidents for large and medium sized reactors. On the basis of several different national development programmes on SMRs, many innovations using a wide variety of coolants, fuel, containments and safety features have been worked out. SMR-specific development effort in Western Europe has decreased because of reductions in governmental funding.

The North American nuclear industry has suffered from a lack of orders since the Three Mile Island Accident and even before that due to problems of insurance coverage against major accidents amongst other issues. In the USA, the government, working closely with the Electric Power Research Institute (EPRI), took steps to capitalise on developments within the industry designed to break this situation. Following a design selection process, the Government is partly financing the reactor certification process and the costs of first of a kind engineering. In addition to a large reactor design (ABWR), the AP600 in the SMR range is being supported and aggressively marketed worldwide. In Canada a preceived need for a simpler, cheaper reactor which could be more easily demonstrated to the public as safe has led to the development of a smaller version of the CANDU line. Design and safety requirements for the next generation of reactors have been identified both in Canada and in the USA by the utilities and governmental agencies. In North America, Medium Size Reactors are expected to supply a significant nuclear share of electricity. Passive systems in the mid-size range are being promoted by researchers and vendors. AP600, CANDU 3 and SBWR are the main medium size reactors designed to fulfill electricity needs for utilities in the USA and Canada. Heat producing reactors such as SES-10 in Canada, and the TRIGA Power System in the USA are under safety review and design processing respectively.

In Argentina the work on Atucha 2 (745 MWe PHWR) has continued. Argentina has carried out a development effort for the design of a small pressurized water cooled reactor "CAREM". The system has a total power of 100 MWth and it is of the modular integrated type. The basic design of the system is complete and it is currently undergoing detailed design. Juragua 1 & 2 (440 MWe VVER) construction plans have come to a halt due to financing difficulties in Cuba.

Middle Eastern countries have identified a strong need both for electricity and for power for desalination and several of them are looking at the nuclear option. The reserves of fossil fuel are massive in some countries but in others there is total reliance on imports. The prospects for further imports may be limited by the policy of the exporting country and there are concerns about atmospheric pollution. The water problem is confounded by low rainfall, a rising population with increasing expectations in its standard of living and by a lowering of the water table in the traditional sources under the desert sands. A study for the North African countries of the economic feasibility for nuclear desalination has been completed. In Egypt, a feasibility study has been completed for a medium sized NPP. Nuclear Policy is expected to be based on ordering existing designs in the SMR range from the world market.

From information provided by Member States (see Table 3.1), it can be seen that several nuclear power plants in the SMR range are under construction around the world. These data show that SMRs could play an important role in many industrialised and developing countries.

TABLE 3.1: SMRs UNDER CONSTRUCTION

| Country | Number of units | Name | Type | Power | Expected commissioning date |
|---|---|---|---|---|---|
| Argentina | 1 | AtuchaII | PWR | 692 MWe | 1998 |
| India | 5 | PHWR 220 | HWR | 220 MWe | 1995-1997 |
| Republic of Korea | 3 | | PHWR | 650 MWe | 1997-1998-1999 |
| Pakistan | 1 | Chashnupp | PWR | 300 MWe | 1999 |
| Romania | 5 | | HWR | 660 MWe | 1995-2001 |
| Slovak Republic | 4 | Mochovce | PWR | 388 MWe | 1996 |
| Russian Federation | 2 | AST 500 | PWR | 500 MWth | 1998 |

## 3.2. SUMMARY OF TECHNICAL DEVELOPMENT

### 3.2.1. Passive principles

As mentioned above, the use of safety systems operating on passive principles is a feature of many SMR designs. The original incentive was to produce designs which could cope with any accident initiating event coupled with the failure of all engineered safety systems. Reliance would be on natural processes, such as gravity and natural convection, only. There should be no need for operator intervention for a long, perhaps indefinite period. Two reactors have achieved this in their design, the Swedish PIUS reactor and the Modular HTGR. Some information on these designs is given later. In these two designs, the safety systems are passive both in initiation and in operation. Other designs need some form of stored energy (e.g. batteries, springs or hydraulic reservoirs) to initiate the passive systems but are then passive in operation, requiring no safety grade power sources such as diesel generators. There are thus different degrees of passivity and the recent IAEA document on reactor terminology has gone to some lengths to include all the different types of system for which their designers claim passivity [1].

The important feature of all these systems, however, is not their degree of passivity but their performance and reliability in carrying out the function for which they were designed. All reactors have to achieve the same standards of safety as a minimum but the passive systems may be able to achieve this standard more easily provided their performance and reliability can be demonstrated. The driving forces of natural convection are generally lower than those of pumped circulation systems and the flow in natural convection does not always follow the path which a first analysis might suggest. Programmes of experimental verification are needed (see for example the programmes connected with the AP600 and SBWR designs). There are further issues on whether a single natural convection system, relying only on the force of gravity for its operation, is adequate or is some element of redundancy still needed ?

### 3.2.2. Dedicated nuclear heating plants (NHP)

The power range of nuclear heating reactors is generally lower than SMR power reactors. They are rated between 2 to 500 MWth. Apart from the high temperature reactors which are discussed in section 3.2.5., their supply temperature is aimed mainly at district heating or sea water desalination and does not exceed 130°C. This corresponds to a primary circuit temperature of around 200°C, and a power density ranging from 2 to 60 kW/l.

The smaller size and lower pressure resulting from these requirement leads to simplification of the overall design and allows for the maximum utilization of natural processes.
Simplifications have been achieved through a less massive RPV, through integration of the primary circuit in the RPV, and in the safety systems and containment. Further simplifications have been made in the use of natural circulation for normal heat removal (made possible by the large safety margins in the NHP design) and by the use of passive safety systems.

Over a dozen reactor designs are known worldwide, most of which have originated in developing Member States. The economics of these reactors, however, can only be justified in remote regions isolated from a national grid. Only a few of the concepts have been constructed (e.g. AST-500 in Russia, HR5 in China and SLOWPOKE in Canada). As a result operational experience has been limited. It is not expected that all currently proposed designs will be implemented.

A renewed interest in gas-cooled reactors for heat applications is evolving specially in Japan, China and Indonesia.

### 3.2.3. Water-cooled and moderated nuclear power plants (NPPs)

It is this area of SMR development to which most attention has been given. Common to most SMR developments is the pursuit of passive safety systems based on the premise that such systems are easier to implement in plants smaller than the current 1000 - 1500 MWe units and that they will lead to savings in overall plant cost. The prime objective is to prolong the grace period from the current 30 minutes, which is commonly required by safety authorities, to a period of several days before active measures initiated by operators are required for long-term cooling. The grace period is practically inversely proportional to the nominal thermal power and directly proportional to the amount of water in the passive cool down systems. The decay heat is removed from the core by natural circulation. Heat removal from the core cooling water is accomplished by the emergency heat removal loops which also operate by natural circulation. The ultimate heat sink is either the atmosphere or large water tanks within or outside the containment.

Designers have sought innovation in the areas of:

- Shut-down,
- Residual heat removal,
- Make up water supply,
- Protection against LOCA.

In addition the goal of simplification has been pursued with vigour.

In the PWR field there have been two main approaches; by development of smaller loop type reactors or through the integral reactor route. There are more designs of the integral type in the SMR range (SPWR, MRX, VPBER-600, NHR-200, SIR, SBWR, etc.), but the loop approach (AP-600, AC-600) is more advanced in terms of market readiness. Boiling water reactors are by their very nature of the integral type and some of their traditional features, such as pressure suppression containment systems have been adopted by some of the PWRs.

In the pursuit of more readily demonstrable safety, design objectives have been to increase the design margins and to enhance operating flexibility in comparison with larger reactors. The SMR designs envisage larger specific pressuriser volumes and water inventories above the core (in terms of $m^3/MW$). Contributions to these ends are also expected from the lower power densities of the SMR cores that are 10 to 60 per cent lower than those of their larger counterparts.

Amongst the various designs similar system concepts keep appearing. The following discussion will therefore look at designs in terms of the design areas listed above.

*Shut-down*

A common approach is to design the core to ensure that it always has a substantially negative moderator temperature coefficient (NMTC). The principle means for achieving this in PWRs is to eliminate, or at least to reduce, the level of boron in the moderator. This leads to use of large quantities of burnable poison and to installation of more control rods to ensure shut-down at the cold shut-down state of the reactor. There is also scope to reduce the NMTC by careful attention to the lattice design parameters. Extra margin to dry-out is often designed for by operating at a lower rating than in the large reactors and this gives further scope for a tighter lattice. All the PWR designs have taken steps to keep the NMTC at a low level. Some, for example JPSR and SIR, have gone to the limit of eliminating boron altogether.

A large negative MTC can lead to difficulty with cool-down accidents such as steam line break or inadvertent operation of emergency primary make-up systems. The problem can be handled by ensuring a large coolant inventory, which slows any temperature drop, and by limiting the maximum rate at which cool make-up water can be admitted.

*Residual Heat Removal (RHR)*

Residual heat can be removed directly from the primary coolant, through the secondary system (in a PWR) or from the containment. There is thus scope for diversity without being extravagant in the complexity of the systems. There will normally be the systems that are used in normal operation, including shut-down and refuelling, and those provided in the safety grade systems, which may be passive in their operation. The systems for normal operation require power for their operation and can be regarded as systems for protection of the investment in the reactor power station. Almost all the SMRs go for passive RHR safety systems.

Heat may be removed from the primary coolant system by means of a heat exchanger positioned in the primary vessel and which operates by natural convection to an external heat exchanger in the atmosphere or in a large tank of water within the primary containment. In some cases the system requires valve operation to bring it into service while in others it operates continuously. A valved system may operate in a water solid mode or in a boiling and condensing mode and will need some form of reservoir to prime it. Where the system operates continuously, so that there is no problem of start up in an accident situation, there is a steady loss of around 6% of heat output which has a significant effect on plant efficiency and hence on economics. In some cases this has been partially overcome by extracting some of the heat into a feed heater. The Russian VPBER-600 is an integral design with this arrangement of heat exchangers and feed heaters.

Alternatively, primary water may be diverted to an auxiliary heat exchanger. An example of this arrangement is in AP600 where there is a natural circulation loop to a heat exchanger in the in-containment refuelling water storage tank which is brought into operation automatically. The tank has capacity for 72 hours of residual heat removal.

In the case of integral PWRs an obvious route is to make use of the steam generators (SGs) which already exist in the primary vessel. Valve movements are needed to reconfigure the steam generators, or some of them, to a natural circulation heat exchanger. The ultimate reliability of this system, and indeed of all systems requiring valve movements, is dominated by the reliability of the valves and therefore some degree of redundancy is needed. If there are several SGs in the reactor, this presents no great problem apart from finding space for all the valve gear and arranging access for its maintenance. The SIR design has twelve steam generators. When needed, four of them can be reconfigured by single valve movements into four independent loops with boiling in the SGs and condensation in condensers positioned in the refuelling water storage tanks situated within the containment. The condensate returns by natural circulation and there is sufficient water in the tanks to absorb decay heat for 72 hours.

Additionally, a bleed system coupled with a make-up system provides a diverse route for residual heat removal from either the primary or secondary circuit (if there is one). The AST500 design has such a system and the SIR design has a feed and bleed system on both primary and secondary.

Perhaps most ingenuity has been exercised in the provision of heat removal systems from the containment. Heat can be removed either through the walls of the containment itself or by means of a heat exchanger arrangement. An example would have condensing surfaces within the containment, to condense steam released from the reactor pressure vessel, and cooled by water circulating by natural convection to an air cooler. The latter system allows the use of a full double containment as is required in some countries.

Direct heat transfer through the containment structure usually implies a single steel containment. The limit on heat transfer through it and hence on the reactor power that can be accommodated within the containment, depends on the area of the containment surface, the temperature inside it and the measures provided to enhance heat transfer on either side of the vessel. An outer protective structure is often provided with a convective air flow between it and the containment vessel. Means to enhance heat transfer may be a water spray on the outside of the containment vessel (AP600) or water tanks between the containment and the outer protective structure (VVER500/600).

*Make-Up*

There are many variants in the design of gravity fed accumulator or make-up tanks. Some designs have relied on steam injectors (SIR, HSBWR) which take steam from the pressuriser or steam space at the top of the RPV and use it to inject water from a storage tank into the RPV. This is an old technology used in steam locomotives but required here at higher pressures and over a greater pressure range. The injectors are also required to start up reliably when needed without operator intervention. They satisfy the requirements of a passive system since they have no moving parts and do not call for external power. There have been a number of experimental programmes to prove their effectiveness. Accumulator tanks supplying water by gravity are more usual (AP600, AC600); they require some form of pressurisation but do not necessarily need valve movement to bring them into use if the prepressurisation is well chosen and the start up procedures are correctly specified.

*LOCA*

The first line of defense in depth is prevention. There will always be some piping runs as part of the primary circuit but if they can be kept small and few in number then the possibility and consequences of a LOCA are very much reduced. BWRs do not normally need to take more than a small fraction of the recirculation flow outside the primary vessel and this can be reduced even more by adopting natural circulation inside the vessel. The only pipe connections are then for the chemical and volume control and make up systems. Large break LOCA is eliminated as a possible accident initiator. The SBWR in its various versions takes advantage of this system. In PWRs the same result is achieved in integral reactors since the entire primary circuit is enclosed within the primary vessel apart from chemical and volume control piping and provision for makeup water in the event of an accident. This is a great attraction of the integral reactor type which has economic and other benefits on containment since it entirely eliminates the large break LOCA problem.

The second line of protection is in ensuring that the core does not uncover. A first approach is to avoid penetrations of the vessel below the level of the top of the core. Preferably there will be several meters of water above the core giving a large inventory for boil off and avoiding the need for very rapid large injection of water in the event of a LOCA. The integral reactor designs can have over 8 meters of water above the top of the core. When it comes to providing extra water to keep the core covered there are two basic approaches; either by providing pumped or passive safety grade water supplies or by providing for flooding the pressure vessel externally. Water supplies have been discussed under make-up above. The other approach is to provide an outer vessel which is either permanently flooded (eg the PIUS approach) or can readily be flooded (eg the guard vessel approach in several Russian designs). The Guard Vessel performs other useful functions such as contributing to the localisation of fission products and providing a degree of pressure suppression in some classes of accidents. For loop type PWRs, high pressure accumulators are needed as in the larger reactors.

Another source of small leakage in older designs is through the pump seals. Many of the SMR designs eliminate this problem by specifying canned rotor pumps. There is sufficient operational experience of these at the sizes needed to have confidence in their reliability. The penetrations for installation of pumps may be the largest penetrations of the pressure vessel. These penetrations must be designed to the same standards of integrity and inspectability as the main vessel head. A decision also has to be made whether to install them at the bottom of the vessel, to maximise the inlet pressure at the pump to avoid cavitation (VPBER-600), or near the top to avoid having penetrations below the top of the core (SIR, SPWR).

*The PIUS Concept*

The Swedish PIUS concept deserves separate mention since it is the most innovative of all the advanced SMR designs. The objective was to produce a design which is completely passive in initiation and in operation in its response to accidents including the most severe ones. The PWR reactor is placed inside a concrete pressure vessel which contains boronated water. Primary water is pumped through the core. At the top and bottom of the reactor vessel there are hydraulic seals between the primary and the boronated water. As long as the primary is functioning correctly, boronated water is prevented from entering the primary by a hydraulic balance between the circulating hot water inside and the static cold water outside. Any disturbance to normal flow due to any significant event causes the hydraulic balance to be broken and boronated water enters the primary. This immediately shuts the reactor down and initiates an alternative flow path by natural circulation between the core and the large boronated reservoir thus providing an assurance that the core will not uncover and providing a heat sink for the residual heat. The PIUS system eliminates safety-grade make-up systems altogether through a large water inventory in the RPV that can sustain full blow-down without uncovering the core. There is a further passive path for heat from the large tank to be removed to an air cooled cooler on top of the building by natural convection. Design and experimental verification of the PIUS system is well advanced.

Other designers have appreciated the technical attractions of the PIUS system but have been concerned about possible high price or felt that it went too far from established practice in one step. Two other designs using some of the PIUS principles have been produced at the concept stage, the Italian ISIS and the Japanese ISER design. In ISIS one of the hydraulic seals is replaced by a novel form of check valve.

*Simplification*

Generally, the term plant simplification means simplification of the arrangement of systems and equipment, of operations, inspections, maintenance and quality assurance requirements, resulting in significant reductions in equipment and bulk material quantities. Significant simplification of the systems throughout the plant as well as an increased application of modularised and prefabricated construction are key design features of the advanced SMR technologies. There is no new design which does not lay emphasis on simplification and its benefits. Factory prefabrication of modularised components, including sections of reinforcing structures, and ease of decommissioning (small components, cast iron vessels, boron tanks) are further key claims of the SMR technologies. All principal components are to be built in a factory where full quality control and production line techniques can be used. They are completed as modules which are then installed on site.

The use of passive safety systems leads directly to simplification in design since it eliminates the need for multiple redundant safety systems with their redundant safety grade power supplies. A system which relies only on gravity for its operation has no problem about the availability of its power supplies and has a reliability determined only by the integrity of its piping and the confidence with which it will perform under all required conditions, which admittedly may be difficult to prove to the satisfaction of the safety authority. But there are other steps which designers have taken in the interests of simplicity.

Traditional control rod drives require a lot of space either above or below the core. There are possibilities to use liquid absorber materials, which do not require the space for rod drives and for in vessel storage when withdrawn. There have also been designs for in vessel mechanical drives (PSR, MRX, HR 200). These eliminate the need to consider control rod ejection which is one of the main, but unlikely, reactivity accident initiators. A more radical solution is in the JAERI SPWR design where liquid filled tubes are used instead of control rods.

The elimination of large primary circuit pipes in integral PWRs and hence of the need to contain the rapid increase in containment pressure in a LOCA allows an easing of the containment specification. Pressure suppression systems for PWRs become feasible and several versions have been proposed. The SPWR design proposes a wet containment through which steam passes and condenses following a LOCA. In the SIR design pressure suppression using water tanks was adopted but to reduce the cost of containment, it was modularised into a number of factory constructed tanks connected to a close fitting containment round the pressure vessel by a ducting system.

Most designs have sought to reduce the number of components, such as valves or the number of cable runs, by as much as 80% in the most favourable cases.

There are very significant developments in instrumentation and control systems allowing simplification and an increase in reliability at the same time. Many proposals use digital electronics leading to a complete redesign of the architecture of the control system. The use of computers is also having a marked effect and raises problems about establishing the reliability of computer software. It seems unlikely that there will be a precise procedure for quantifying software reliability but techniques of diversity, redundancy and very strict quality control in software production have been shown to be adequate.

One of the simplest reactor systems is the SBWR using natural circulation in the primary vessel and with no need for steam generators, one of the most troublesome major components of large PWRs.

### 3.2.4. Heavy water reactors

Heavy water reactors have demonstrated their safety, reliability and economical viability in several countries. Their neutron economy gives them a wide flexibility in the choice of the fuel cycle paving the way for a better uranium utilization. Natural uranium, slightly enriched uranium, recovered uranium from reprocessing MOX fuel, thorium or spent LWR fuel form options for the fuel cycle of HWRs. Japan has operated the 165 MWe Fugen reactor with high capacity and has the largest MOX fuel experience. Pu utilization in HWRs could be seen as a link to future fast breeder reactors. Most of the HWR designs are of the channel type allowing on power refuelling making the excess reactivity of the core small at all times. These features, along with the presence of the moderator at low temperature and pressure, give these type of reactors unique safety characteristics. HWR have only been designed and constructed in the SMR range so far with power ranges from approximately 200-700 MWe. The Canadian CANDU design and the PHWR type reactor from India form the main technology development activities at the commercial level. The CANDU-9 design is based on the CANDU-3 design and is the only large HWR with a power rating of 900 MWe.

### 3.2.5. Gas-cooled reactors

The gas-cooled reactors discussed here are all thermal reactors with a graphite moderator. There has been work on gas-cooled fast reactor designs but these have been generally outside the SMR range.

Commercial operation of gas cooled reactors has been mainly in the UK and France. Magnox Reactors (in the SMR range) have been operated in the UK since 1956. They are based on uranium metal fuel rod technology with magnesium alloy cladding and $CO_2$ as the coolant. This design puts a limitation on maximum outlet temperature and consequently on the efficiency of the plant. The later AGR reactors (also in the SMR range) obtained much higher efficiency through a high gas temperature and stainless steel clad UO2 fuel rods.

High Temperature Gas cooled Reactors on the other hand are based on ceramic coated particle fuel allowing for high outlet temperature. The basic fuel design utilizes a uranium oxide or carbide particle coated by pyrolytic carbon and silicon carbide able to withstand 800 bar of internal pressure. The stability of this fuel at high temperatures has permitted the design of reactors with a truly passive response to loss of all safety systems, including all gas cooling, provided that the overall diameter of the reactor and its vessel is small enough. The fuel heats up and this heat passes by convection and conduction to the vessel which is in turn cooled by natural convection to the atmosphere. The centre temperature of fuel in the middle of the core can rise to 1600°C without any problem of fission product retention. If the steam generator is positioned at a lower level than the core, there is no danger of water ingress onto the hot graphite. This concept is suitable for relatively low powered reactors. HTGRs constructed to date have been more conventional in layout and in safety features.

The main design variants of this type of reactor have depended on whether fuel is of the pebble bed type or of the prismatic type. The pebble bed type consists of a large number of spherical fuel elements. The fuel element matrix is graphite and the fuel kernels are imbedded within the inner layer of the matrix. A fuel free graphite reflector shell is located inside the RPV.

The prismatic type has the coated particles in a graphite matrix forming a fuel rod which is inserted into vertical holes in the moderator blocks. Solid unfueled blocks make up the reflector zone surrounding the active core area.

Several development programmes led to demonstration of high temperature gas cooled reactor features, in particular the coated fuel concept. This has been based on the Dragon project in the UK, Peach-bottom No1 and Fort St. Vrain in the US and the THTR in the Federal Republic of Germany. Currently, the HTTR, of the prismatic type, in Japan will be the first to be used for a process heat application with an outlet temperature of 850°C or above.

The Russian Federation also has designs for gas cooled reactors in the range of 50 to 400 MWe. The VGM reactor (200 MWe) is intended for process heat applications with outlet temperature up to 900°C.

A gas cooled reactor based on the Brayton cycle is in the development stage in the US. The GT-MHR has a power rating of 550 MWe. The energy conversion system has a projected efficiency of approximately 48% producing electricity by directly driving a gas turbine generator.

Coordinated research programmes on heat transfer and decay heat removal, reactor physics and validation of predictive methods for fuel behaviour are being conducted under the auspices of the IAEA. Actinide waste and plutonium burning gives another possible application for HTGRs and worldwide attention is being given to this possibility.

Almost all gas-cooled reactors are in the SMR range. This along with their safety features provide this reactor design line with immense potential for its deployment in developing and industrialised Member States. Minimized staffing for operation, particularly for low power reactors, could be possible due to the safety margins.

The main design features can be summarized as follows:

*Fuel*

The ceramic fuel and the multiple coating of the fuel kernels results in a micro pressure vessel capable of maintaining the integrity of the fuel and guaranteeing fission product retention in conditions much more severe than postulated accident conditions.

*Coolant*

The coolant is a single phase having no corrosive action on the system nor does it have a reactivity effect. Moreover, chemical and energetic reactions between the fuel and the coolant are not possible.

*Core*

The core material and geometry ensures that the core integrity cannot be challenged even in the case of complete loss of coolant and no scram.

*Safety*

Emergency heat removal in a passive manner has been made possible and enhanced by the slim reactor pressure vessel geometry.

### 3.2.6. Liquid metal reactors (LMRs)

A fast neutron spectrum allows production of more fissile material than that consumed for heat generation. In a fast reactor liquid metal such as sodium is normally used to remove the heat, and it has a minimum effect on the moderation of fission neutrons. Sodium as a coolant has an excellent heat capacity, low operating pressure and natural convection capability.

Sodium coolant has very good thermal conductivity. In the event of failure of the main sodium pumps, heat can be transferred to the vessel boundary by conduction and natural convection without large increases in temperature. Provided the reactor is small enough, decay heat can then be transferred through the vessel wall to a natural convection air flow. Alternatively a small additional heat exchanger in the sodium pool can be used to take heat to an external heat sink by natural convection. Thus the sodium and the small size permit a passive decay heat removal system.

Small fast reactors can be designed to have totally passive safety systems that do not require power and may not require valve movements to initiate them. The characteristics of these designs are metal fuel, sodium coolant, small size and a passive decay heat removal system.

Early reactors used uranium or U/Pu alloy fuel since its high thermal conductivity would avoid excessive central temperatures in a high power density core. However, this fuel suffered severe irradiation growth and a change was made to oxide, carbide or nitride fuels. The growth problem was later solved in metal fuel by the development of a ternary alloy of U/Pu/Zr which shows adequate stability under irradiation. It has a further attraction in that it can be reprocessed in relatively simple plant using electrolytic separation. This allows a closed breeding cycle with reactor, reprocessing and fabrication plants within a single site boundary which is attractive from a non-proliferation point of view. There are cost savings if LMRs are constructed in large plants consisting of several modules in order to share the fuel and component handling facility as well as the sodium engineering auxiliaries.

Fast neutron capture in fertile actinides is one way of disposing of them, or at least reducing their quantity. Several countries are funding fast reactor R&D to develop and evaluate LMR actinide burning capability.

Seventeen LMR prototype reactors and power plants have been built and most of them have accumulated many years of operating experience. Breeding capability and the closed fuel cycle have also been demonstrated in large scale plants. LMR based technology is available today for the design and construction of improved new reactors in the SMR range.

There are four modular small or medium-sized liquid metal reactor concepts under consideration:

- The "Advanced Liquid Metal Reactor (ALMR)" by General Electric et al. (former PRISM Project) with 155 MWe/module and 465 MWe per power block;
- The 325 MWe -- "Modular Double Pool Reactor (MDP)" being designed by the Japanese Central Research Institute of Electric Power Industry (CRIEPI);
- The Super, Safe, Small and Simple LMR (4S) being designed by CRIEPI (50 MWe);
- BMN-170 is under design in Russia.

There are other designs such as SAFR and the RAPID design from the USA and Japan respectively.

The advanced Liquid Metal Reactor (ALMR) is a modular plant which is typical of current developments. A 1275 MWe station would have 9 modules each giving 140 MWe. The reactor is an integral pool type where the main reactor vessel contains the reactor, the primary pumps, the intermediate heat exchanger, the in-vessel fuel transfer machine and the necessary ducting to channel the sodium flow. The reactor vessel is below ground level and is surrounded by a guard vessel with a small gap between it and the reactor vessel for sodium leak detection. The outside of the guard vessel is cooled by natural air circulation with down flow and up flow separated by a baffle. The fuel is Pu/U/Zr alloy and the core diameter is smaller than the height to avoid positive sodium void coefficient. The core restraint system is designed to alter the natural bowing feed-back so that the reactivity feedback contribution is generally small and negative.

The RAPID system has a central 300 mm diameter channel which is normally filled by a small sodium flow. In the event of sodium boiling in the core, the channel fills with sodium vapour providing a neutron streaming path to offset the positive sodium void effect. Positive sodium void reactivity effects present one of the major safety issues in LMRs since they could cause an overheating event to escalate. There has been much attention to ways of reducing the effect, such as the RAPID proposal.

The Double Pool concept has been driven by an objective to reduce fast reactor construction costs to the same level as those of an LWR. A major contribution has been to reduce the overall size of the intermediate heat transport system by installing the steam generators in the sodium filled annular space between the primary vessel and the guard vessel. These two examples serve to show some of the ways in which the basic modular concept can be modified to meet different objectives.

The long-term potential of increasing the available fuel supply and reducing the long term radiation level of nuclear waste are fundamental reasons for development of this type of reactor. The time-scale for its development provides an opportunity to expand international cooperation on the R&D required for this technology. Joint programmes could significantly reduce future expenditure by individual countries and effectively use technology that has been developed in all countries during the past 40 years. Apart from the restart of Superphenix (1200 MWe) and the starting of Monju in 1994, most of the work in this field has concentrated on research work in the areas of material-coolant interactions and material movement and relocation.

## 3.3. DEVELOPMENT STATUS BY REGIONS

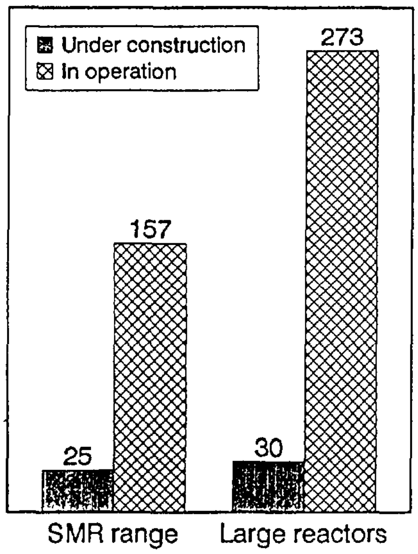
Looking at the overall status of development it can be categorized in three different ways. The developed countries, namely the OECD countries with a long established technology base, Eastern Europe and the former Soviet Union with a significant potential for nuclear power and third world countries with very small nuclear capacity. Each has particular needs to retain or establish a future for nuclear energy.

The OECD countries can be further divided into two categories. France, Japan and the Republic of Korea have large scale successful programmes. For most of the OECD countries, programmes have either been stalled or it has been decided not to pursue nuclear power for reasons of public opinion, costs, or regulatory obstructions. The dominant factor has been the image of nuclear power in the eyes of the public. A great amount of effort is needed to restore confidence in these countries. In areas where nuclear power still has a positive image, the public will have to be assured that the nuclear industry is reliable, economical and, more importantly, is safe.

In the second category, existing nuclear power plants require immediate upgrading which requires a large amount of effort and money. As a first step the whole industry needs to be brought to an acceptable level of efficiency, reliability and safety. It will also call for a major restructuring of the managerial system. The task of tackling these aspects all at once is enormous. Deployment of new reactors does not seem likely for some years.

Developing countries with a modest infrastructure and technological base, will need extensive assistance to set up a well defined programme. Sharing experience between these countries and countries in the other two categories will not only address their energy needs but also assure the establishment of high standards of safety culture from the beginning. This would ensure that the countries interested in developing nuclear power could set up their programmes and execute them successfully.

Nuclear power investment on a worldwide basis has preferred large units due to the economy of scale, especially in the industrialized countries. This can be clearly seen from the number of nuclear power plants in operation today (Fig. 3.1.). The number of units currently under construction in the SMR range is in the same range as the big power plants. Moreover most of these reactors are being deployed in developing countries.




| No. of reactors | Under construction | In operation |
|---|---|---|
| SMR range | 25 | 157 |
| Large reactors | 30 | 273 |

Figure 3.1 Nuclear Reactors in operation and currently under construction

## 3.4. RELEVANT ACTIVITIES OF INTERNATIONAL ORGANIZATIONS

The activities of the International Atomic Energy Agency (IAEA) in the development and design of nuclear reactors, and in particular, the organization of programmes are independent of the reactor size. The programme of the IAEA in advanced reactor power technology promotes technical information exchange between Member States that have interests in exploratory, research or development programmes. The Agency also publishes reports available to all Member States, and provides assistance to Member States with an interest in nuclear technology. For Member States with current programmes, the IAEA activities are coordinated by the International Working Groups (IWGs) on the particular reactor technology line of interest. The international working groups on fast reactors, gas-cooled reactors and advanced technologies for water-cooled reactors meet periodically to review the national programmes and advise the IAEA on its technical programmes and various activities in the field. The international working group meetings are held in an open forum where progress, problems and operating experience can be frankly discussed. This provides for an opportunity to share the lessons learned and bring national experiences to a truly global level.

The activities planned by the IAEA vary from small consultancies on a specific topic to large symposia with a broader base of technical aspects. Technical committee meetings on a specific topic provide the means to exchange information and coordinate work on areas of mutual interest.

Several forms of IAEA support are also available to the Member States. Research groups in different countries are given the opportunity to exchange information and communicate through coordinated research programmes on specific areas of interest upon Government request. Technical assistance to provide expert advice, give training, or fellowships could be arranged for engineers and researchers of developing Member States. Special equipment could be provided by the Agency for developing Member States with research programmes to assist them in the successful execution of their research programmes.

The IAEA has continued to sponsor and follow up on new developments, to collect information on the constraints on SMR introduction, and to provide assistance on the utilization of nuclear power and development of technical human resources. Heat power applications such as district heating, oil recovery and seawater desalination are examples of Agency sponsored programmes for nuclear energy application. Surveys on the status of development of small and medium size reactors, provide examples of the Agency follow up on new developments. On the other hand, the development of nuclear power systems of a larger size (1200 - 1500 MWe) has shifted the medium size from the 100 - 500 MWe range to include 700 MWe reactors [2]. One important result is that most of the new developments, both evolutionary and innovative, fall in to the SMR range. This means that a review of the SMRs becomes an up-to-date overview of the main reactor technology trends.

The international activity has played an important role in the development of reactors, not only with technical information exchange and coordination of research, but also with collaboration in the construction and operation of small experimental reactors to demonstrate their technical feasibility.

The area of SMRs have been the subject of several major studies in recent years, principally carried out by international organizations such as the IAEA, European Union (EU) and Nuclear Energy Agency of the Organisation for Economic Co-operation and Development (NEA/OECD). These activities can be summarized as follows:

- In 1985, the IAEA initiated a study of Small and Medium Power Reactors [3], which was continued over a number of years on the basis of a series of technical meetings, first on SMRs, then on Advanced Water Reactors. These studies provided a comprehensive analysis of the benefits and options for small power reactors, but also identified many of the obstacles to their use, including the relative economics of larger reactors and fossil stations, the problems of finance in developing countries, and public acceptability. Recent studies have concentrated on the potential applications of SMRs in desalination and the technical review of SMRs of which this report is part.

- In 1988, the NEA began a detailed assessment of the role of SMRs in OECD countries over the next few decades. This was carried out by an Expert Group from different OECD member countries which examined the status of the various SMR designs that were available and the prospects for their application for electricity and heat. The study looked particularly at the economic aspects, the advantages of passive safety, and specific applications of small reactors.

- In 1990, the EU placed a short contract for a study of the potential for SMRs in Europe up to 2020 which was carried out by a small team from the UK and Germany. The study concentrated mainly on the situation in Western Europe but also made a preliminary assessment of the likely opportunities for SMRs in the new Eastern Europe. It was concluded that there were some opportunities for SMRs in Europe, although they would find it difficult at first to compete against larger reactors and combined cycle gas stations for electricity, and gas boilers for heat production. It was noted, however, that conditions from country to country varied so widely that a true assessment could only be based on more detailed country analysis.

- Also in 1990 the EU placed a number of contracts, through the Small Reactor Interest group, for assessment of the possible potential applications in Germany, France, Spain and the UK. These were mainly based on HTRs but analyses of the heat markets in these countries could also have some relevance for other SMRs, although these would not be capable of supplying the higher temperature process heat which is available from HTRs and which would be necessary for some applications.

In 1994, the IAEA published a case study which it had performed on the feasibility of SMRs in Egypt [4] in which the economics, environmental and health impacts, organizational and manpower requirements, national participation (industry) aspects, and financial requirements of an SMR project were analyzed. It was concluded that the lifetime levelized electricity generation costs of a twin unit SMR plant would be in the same range as those of a coal plant of the same size.

In 1995 the IAEA published a technical document on safety principle for the design of future nuclear power plants [5]. The report represents the conclusion of a long effort of many experts from different countries and different organizations (authorities, designers, utilities, etc) to formulate a proposal of safety objectives and principles for the design of the nuclear power plants. The proposed safety objective stresses the importance of the explicit consideration of severe accidents and the minimization of the off-site consequences, even in the case of severe accidents.

The report is rather general and fully applies to SMRs, the majority of which are already designed to achieve a very high level of prevention and mitigation of severe accidents.

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# 4. FORMAT OF THE DESIGN DESCRIPTIONS

## 4.1. FORMAT DESCRIPTION

This chapter discusses the logic and content of the detailed design descriptions of the reactors included in chapers 5-7 of this report.

Nuclear power technology development is a wide and diverse area. Three main development lines have existed for a long time and all have developed designs that are either readily available or will be available in the future. The technology of each line is substantially different in the physics and the hydraulics of the reactor.

These three main lines, as defined by their primary coolant are water cooled, gas cooled, and liquid metal cooled. Water cooled reactors can be further categorized as heavy water or light water moderated reactors. The design approach for a given system can be substantially different from another system within the same technology line. The small and medium reactor area has to deal with all technological lines and all varying approaches. The relevant reactor information for the purpose of this TECDOC has been divided into six parts:

1. Design objectives and special features,
2. Design description,
3. The safety concept of the design,
4. Extended design data listing,
5. Design and developmental status,
6. Statement on the economics of the described system.

Chapters 5-7 provide the technical descriptions of some of the small and medium size reactors which have been developed or are currently under development in various Member States. These descriptions include more detailed information in a consistent and systematic way, than has been provided in earlier Agency publications. The following sections are descriptions of the format used for the six different areas mentioned above.

Item one gives the main design objectives and highlights the main design features as seen by the vendor or designer. The design description section is broken down into the different systems and features as follows:

- Nuclear steam supply system

  This section deals with information about the reactor vessel, the core, control rods and control rod drive mechanism, vessel head and head mounted components. It discusses the systems and equipment employed to handle and store both fresh and spent fuel. It also covers information on the primary, secondary and tertiary circuit as applicable, including associated systems e.g. steam generators, pressurizer, reactor coolant pumps etc.

- Balance of plant systems

  This section provides information on the secondary coolant, turbine, condenser, steam isolation, auxiliary feed water systems and addresses the radioactive waste management system.

- Instrumentation, control and electrical systems.

  This section covers the strategies for reactor control and describes the electronic systems used to monitor, control and provide for emergency protection. A general description of the control room layout is highlighted.

- Safety consideration and emergency protection.

  This section gives a general explanation of the safety systems' operation and the type of systems deployed. It generally describes the sequence of operation of the safety systems. Under emergency conditions, accident propagation and counter balance measures are discussed.

- Building and structures.

  This section covers the building arrangements and discusses the provision for accident localization. It also discusses the containment structure, the ultimate barrier for defence-in-depth. The section also covers the accessibility of plant equipment and gives an overview on radiation exposures. The seismic protection of the nuclear island is briefly discussed.

Section three provides a structured summary of the bases for the safety systems, the design basis accidents and how the systems address these accidents on the prevention, protection and mitigation levels. Mitigation of severe accidents is presented in a general sense. In section four, a comprehensive listing of design data is provided for the reader to form a complete picture of the design. Graphical representations are limited as far as possible to two figures; one figure of the reactor system cross section and another one presenting the general plant layout or schematic diagram. In cases where this is not possible figures are limited to two pages.

The project status section is presented according to the design status definitions described in section 4.2. A summary of the research and development work carried out, or not yet covered is given. Activities carried out in the area of licensing are presented in this section also. The organisations involved in the work (the entities) are listed.

The economics of the project are very difficult to provide since they depend on whether the project is to be carried out in the country of origin or abroad, on site specific conditions and on the infrastructure available. A section on the project economics is left open for the designers or vendor to provide either qualitative or quantitative information.

## 4.2. DESIGN CLASSIFICATION

### 4.2.1. Terminology

Several dozen new reactor projects of advanced design mostly in the small to medium power range are referred to in a variety of publications. The status of these projects are defined by the different authors by reference to terms like conceptual, preliminary, basic, engineering, and detailed designs. Although these terms are used, they are used with widely different interpretations. The use of these or similar terms leads to confusion, and does not permit a reasonably clear and unambiguous understanding of the real development status achieved. The problem appears clearly when reactors at a different development stages are described in a similar manner in the same publication, giving the impression of the same development status.

In order to present a balanced overview of what is actually happening in the area of small and medium reactors, and to provide the information in an objective manner to the interested technical community as well as decision makers, it is necessary to make a realistic presentation on the status of development of the different reactor designs.

It is recognized that the kind of information needed to define the status of development of a reactor design is not easily obtained. Information on the different aspects of individual development status would be very useful in order to better understand and qualify the design approach and the time when a reactor may become available for industrial application. The main aspects to be considered are:

a. Design approach
b. Development effort
c. Design status
d. Licensing status

### 4.2.2. Design approach[1]

Advances in technology and the lessons learned from experience have always been introduced into the new designs. Reactor designs under development implement these advances in technology differently and to a varying extent. The design approach can be classified from a technology implementation point of view as follows:

a) Evolutionary design

Evolutionary designs based on proven technology demonstrated in practice, incorporating some improvements, but no substantial changes, modifications, or novel features. These designs are fundamentally similar to the latest models operating or under construction, and are also perceived in this way. In principle, they are available for construction without the need for plant demonstration.

Evolutionary designs based on proven technology, incorporating not only minor improvements, but also some novel features, which may need to be further developed and/or demonstrated in practice. Regarding the need for demonstration, opinions sometimes differ between designers, vendors, regulators and utilities. These designs are intended to offer substantial real improvements with respect to current reactors. They could be available for construction on a short term, subject to approval by regulators and acceptance by prospective investors.

b) Innovative design

Innovative designs, which use current technology and take advantage of accumulated experience, but in addition incorporate or are based on new features. There is a recognized need for demonstration, shared by all parties concerned. These designs constitute medium to long-term options, with potential major improvements regarding safety, reliability and economics compared to current reactors and evolutionary designs.

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[1] Currently an effort is being made by IAEA to establish an international consensus on defining design stages of advanced reactors.

### 4.2.3. Development effort

The development effort strongly depends on the design approach adopted. For an evolutionary design approach with no novel features the development effort may be very small. On the other hand, for an evolutionary or innovative design approach with novel design features the effort may be very large indeed. To assess the status of development of the different design approaches at present, it is necessary to compile information on the overall development effort required, the current situation, and what is still needed. This will give an important input to the development status of a given design and in most cases provide an indication of the increase or slow-down of the design effort. Conclusions on the development status of a design will need an in-depth assessment of the overall situation, since there may be overriding factors such as financial constraints, public acceptance, political opposition or unjustifiable improvements with regard to the investment involved which alter the long term viability of the project.

### 4.2.4. Design and licensing status

The implementation of a nuclear power plant project proceeds with the design, construction, and operation stages. The licensing process goes in parallel with these activities and varies from one Member State to another. The amount of design and licensing work to be conducted before start of actual construction has no universally accepted rules or practices. Earlier reactors started with little design work completed. Nowadays, there seems to be a general understanding that 60 to 80% of the total design effort must have been completed before start of construction. No design decisions of importance are expected to be left open especially if they are subject to the results of R&D work. Clearly, the more the design is complete the less the risk for design changes, schedule delay, licensing problem and hence cost overruns. The licensing and design status are the best indicators of real development status, but the indiscriminate use of various terms to describe design status and differing licensing procedures leads to confusion, and does not permit a reasonably clear understanding of the real status

For the purpose of the present IAEA report a simple classification for the design and licensing status has been proposed to provide a coherent approach for clear understanding and unified status classification of the different designs, Three levels of designs are defined namely *conceptual, basic* and *detailed* designs. Figure 4.1 gives a general description of these levels. Four milestones or

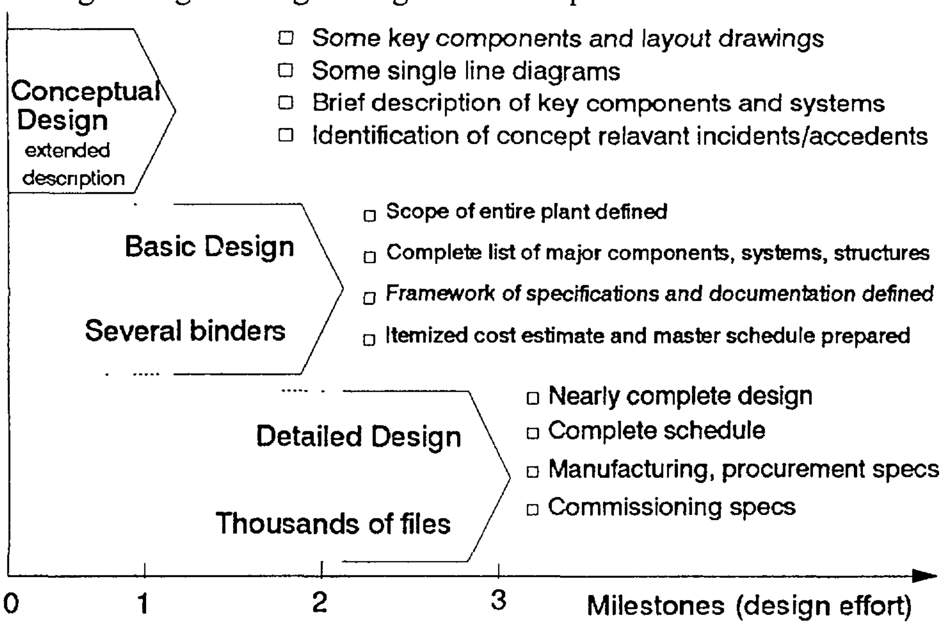



*Fig 4.1 Definitions of design status*

phases have been generally defined for the licensing procedure. Preliminary licensability assessment of the design is considered the first milestone. The second is a formal submission of the licensing application. If the review by regulators is underway the design is considered to be in the prefinal third stage of the licensing process. The final or fourth stage is the issue of the final license. A more detailed design and licensing status definition is presented in Appendix II and has been used in some of the design descriptions in chapters 5-7.

## 4.3. SAFETY CHARACTERISTICS

### 4.3.1. Defence in depth and its realization

The presentation of the safety related characteristics of the concept, is organized in two steps to show, in a simplified manner, the overall concept of the safety approach:

a. Implementation of defense in depth
b. Correspondence between the safety functions and the safety related features

Worldwide, designers, safety authorities and advisory groups (e.g. INSAG) recommend that the design of NPPs should be based on the Defense in Depth principle.

Plants must be highly resistant to accidents (prevention level) and the implemented features (systems and/or inherent characteristics) must be effective at preventing degradation of the reactor core (protection level) and, if necessary, able to mitigate the consequences of severe accidents (mitigation level).

The information presented in the design descriptions structures the information in such a way that the implementation of Defense in Depth can be clearly identified.

### 4.3.2. Prevention

Accident prevention is realized in three ways:

a. suppressing the initiating events, e.g., the use of a canned motor pump suppresses pump seal small LOCA

b. reducing their frequency, e.g. enlargement of pressurizer reduces the number of relief valve actuations which reduces LOCA frequency

c. reducing the potential for significant consequences from an accident, e.g. use of small diameter pipe makes the consequences less pronounced.

### 4.3.3. Protection

This Defense in Depth level aims at protecting the reactor against core damage by improving design features and/or the counter measures to cope with abnormal situations. The concept characteristics favourable for this are classified according to the initiating event families in a similar way to the prevention level.

Functional redundancies (two or more systems able to realize the same function) are generally implemented in order to give the required level of reliability and to minimize the common mode failure risks. Adequate protection can be created with active and/or passive, safety and/or non-safety graded systems.

### 4.3.4. Mitigation

The Defense in Depth approach requires demonstration of the plant safety taking into account core degradation. The reasons for this requirement can be interpreted as follows:

a. to cover the possible lack of completeness of the selected deterministic sequences in the safety analysis,

b. to demonstrate the potential of the concept for mitigating severe accidents,

c. to demonstrate the avoidance, by design, of any cliff edge effect[1]

Moreover it must be demonstrated that no accident sequence, whether it is of low or high probability, contributes to risk in a way that is excessive in comparison with other sequences.

The design characteristics that are related to mitigation of accident consequences are identified in the descriptions in relation to the safety functions that they must assure during severe accidents.

In order to indicate these relationships and the implemented functional redundancies it is suggested that the main plant features (systems and inherent characteristics) should be classified versus the safety functions for Design Basis Accidents and for further certain hypothetical conditions (Beyond Design Basis/BDB). Their passivity and the distinction between safety grade and non-safety grade are identified.

[1] The cliff edge is a discontinuity in the relationship between the frequency and the consequences that define the risk (risk-frequency x consequences)

# 5. DESIGN DESCRIPTIONS FOR REACTORS IN THE DETAILED DESIGN STAGE

## 5.1. REACTOR DESIGN DESCRIPTION AND DEVELOPMENT STATUS OF BWR-90

### 5.1.1. Basic objectives and features

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. This way the "remaining risk" for the public is reduced to an extremely low value.

The BWR 75 design included two standard sizes, with nominal thermal power of 2,000 $MW_{th}$ and 3,000 $MW_{th}$, 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 2,350 and 3,300 $MW_{th}$, respectively. These standard sizes have later been supplemented by a larger unit - with a nominal thermal power of 3,800 $MW_{th}$ - taking advantage of the margins that are gained by utilization of the new generation of ABB Atom BWR fuel.

The net electrical output of the smallest BWR 90 version amounts to 830 MWe at a coastal site with very cold circulating water ($\leq 5\,°C$); at sites with warmer circulating water the output will be lower - a typical "low" value of 700-720 MWe puts it in the upper range of SMRs. For that reason, a description of the smallest BWR 90 version has been included in this SMR report.

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.

### 5.1.2. Design description

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 probably 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 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.

#### *5.1.2.1. Nuclear steam supply system*

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 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 for the plant operation since it reduces the amount of in-service inspection to be carried out during the refuelling outage. The reactor vessel length is 20.6 m and the width is 5.6 m.

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 (for more than three million operating hours) since 1978. Within a couple of years such internal pumps will be in use also by other BWR vendors, in the ABWR plants.

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 manoeuvring 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 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. Daily load following in a 100-40-100 % cycle with (1 -) 2 hour ramps can be accommodated.

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.

The reactor core is a typical ABB Atom BWR core, made up of 500 fuel assemblies of the SVEA-100 type. In the BWR 75, the core design was based on traditional 8x8 fuel assemblies with a rod diameter of about 11 mm; the SVEA fuel assemblies introduced 4x4 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 4x4 subassemblies are replaced by 5x5 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 separate electro-mechanical and hydraulic functions, the former used for normal, continuous fine motion of the control rod and the latter for rapid 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 diversified means of control rod actuation and insertion (together with a generous reactor pressure relief capacity) in combination with a capability of rapid reduction in the recirculation flow rate (recirc. pump run-back) has led to regulatory acceptance of the system as being a sufficient ATWS (anticipated transient without scram) measure; the CRD design is "ATWS proof".

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.

The steam generated in the core region is separated from the reactor coolant in the steam separators on top of the moderator tank cover, and its content of water droplets and moisture is lowered on the passage through the steam driers. The "dried" steam collects in the top portion of the RPV, from where it is conveyed 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 distributers are of a special ABB Atom design that ensures a "thermal sleeve" protection against the "cold" feedwater for the RPV wall, and 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; in these situations the feedwater flow is routed via smaller nozzles that can easier withstand thermal transients.

A four train auxiliary feedwater system, or high pressure coolant injection system, with piston pumps is also provided, drawing water from the condensation pool in the containment wetwell and injecting it into the vessel. The capacity of each pump is sufficient to ensure that the water loss that may arise from a rupture of the largest nozzle at the bottom of the RPV can be counteracted by two trains. There is also a four train low pressure coolant injection system with centrifugal pumps that draw water from the condensation pool; two pumps have sufficient capacity to keep the core flooded following any design basis event, including large LOCAs.

The RPV is provided with a pressure relief system which consists of 12 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 over pressurization 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).

A shutdown 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 cleanup system, with a radial type precoat filter, heat exchangers (one of regenerative type), and pumps, draws water from the shutdown cooling system nozzles and returns it as purge flows through the control rod drives and the recirc. 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 cleanup system (RWCU) 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". However, no very significant changes of auxiliary systems have been introduced in the BWR 90.

The primary system, the reactor coolant pressure boundary, and important ancillary systems are enclosed in the primary containment, a cylindrical prestressed concrete structure that incorporates an embedded steel liner to ensure adequate leaktightness; a steel dome is provided as a "removable" closure of the shaft above the reactor pressure vessel.

The primary containment (Cf. figure5.1.1.) is of pressure-suppression 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 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 a 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.

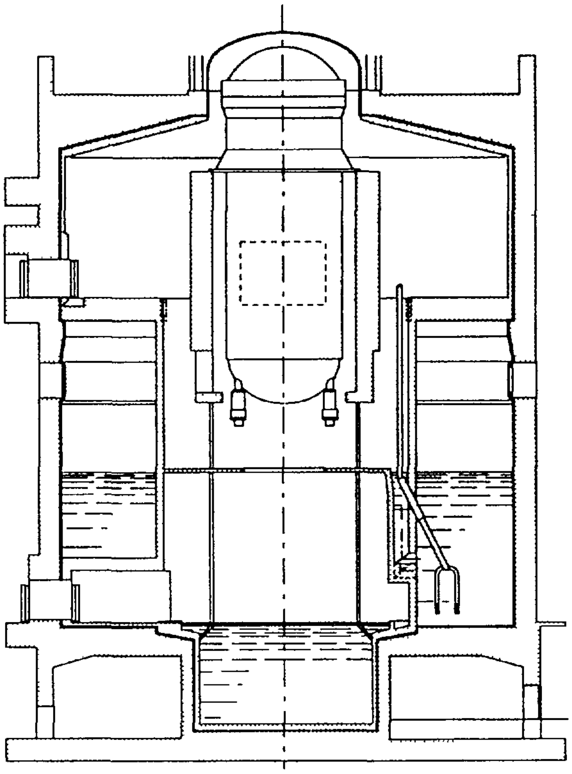


*Fig 5.1.1. BWR 90 - Containment Arrangment*

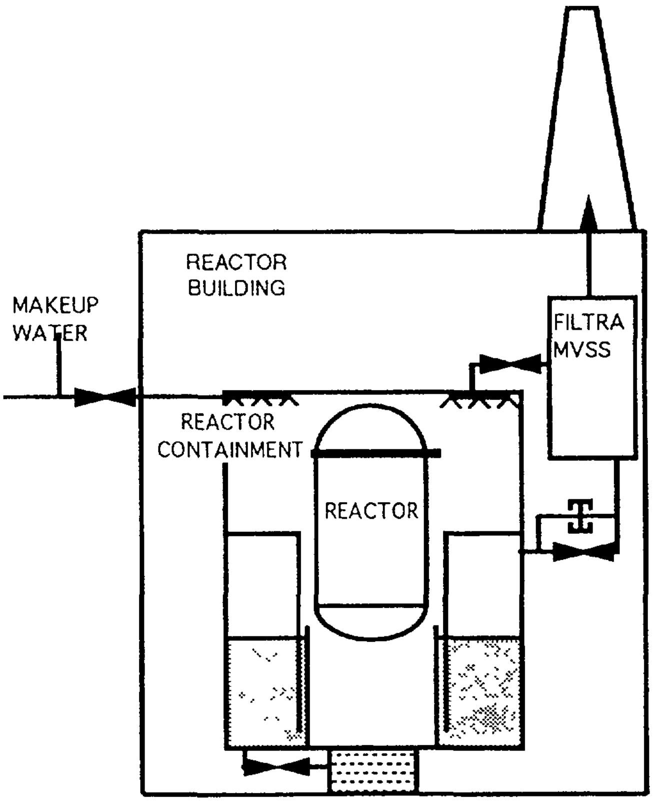
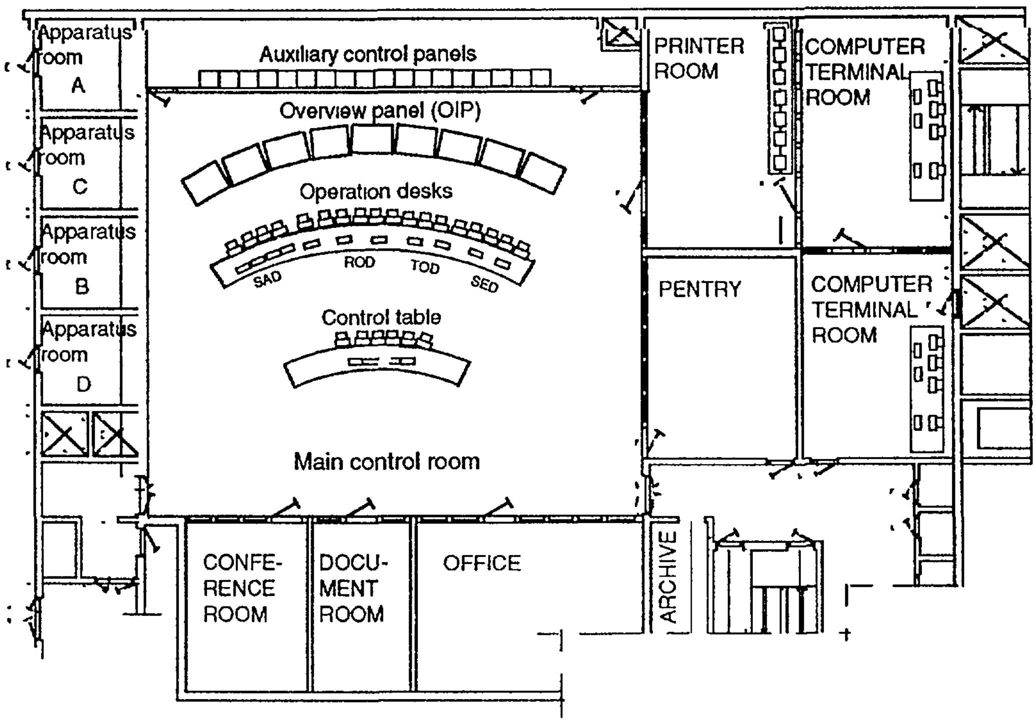



*Fig 5.1.2. BWR 90 - Severe Accident Mitigation*




**Operation desks**
- VDU displays and keyboards
  - Safety systems desk (SAD);
  - Reactor operation desk (ROD),
  - Turbine operation desk (TOD),
  - Service desk (SED)

**Overall indication/overview panel (OIP)**
- VDU displays of process information
- VDU displays with large characters
  - Process overviews;
  - Safety overviews;
  - Fire alarm overview

*Fig 5 1.3. BWR 90 - Control Room Arrangement*

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 reactor building encloses the primary containment completely and is designed to serve as a secondary containment, kept at underpressure by ventilation systems which can be rerouted to filter the exhaust air 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 The design strength of the reactor building 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, the aircraft should preferably be prevented from penetrating the first line of defence - the walls or the roof of the reactor building

The containment design of BWR 90 incorporates, as depicted in Figure 5 1 2 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 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.

*5.1.2.2. Balance of plant systems*

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 720-850 MWe depending on the site conditions, in particular with respect to circulating water temperature.

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 (or four) 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. The condensate is pumped forward to the dearator (or the feedwater tank) through low pressure heaters and a condensate cleanup system with ion exchange filters by means of three 50% condensate pumps. The drainage from the heaters is pumped forward through the cleanup 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. 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.

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., 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.

*5.1.2.3. Instrumentation, control and electrical systems*

Modern process control and communication technology is applied to the BWR 90 - its control and instrumentation systems are mainly based on micro-computers. Process communication with the control room is realized by means of distributed functional processors. These in turn interact, via serial communication links, with a number of object-oriented (object = process component) process interface units. Thus, the protection and control system configuration is characterized by decentralization and the use of object-oriented intelligence. The arrangement satisfies the requirements of redundancy and physical separation. It includes intelligent self-monitoring of protective circuits.

The use of serial communication links guarantees interference-free performance and reduces cabling. Standardization of the object-oriented circuits 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.

A very important aspect is that the software is also standardized to simple program functions. This makes it easy even for 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, Cf. Figure5.1.3., 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). The overview presentation shows the main process in the form of a flow diagram and indicates the status (normal, disturbed or failed) of various plant functions in correspondence with the operating instructions for the plant. It is visible to all operators in the control room.

The 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 computer-based reactor scram function via the reactor protection system (RPS) has been supplemented by a scram backup system that is implemented in hard-wired equipment.

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".

The electrical power systems for safety-related objects are strictly divided into four separated sub-divisions - a principle that is implemented in the operating BWR 75 plants and maintained in the BWR 90.

For the ordinary power distribution, some simplifications have been introduced. Voltage stability considerations limit the ratio between direct-on-line starting motor loads and available short circuit power on each busbar; this has been a design constraint in previous plants. 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 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 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.

*5.1.2.4. Safety considerations and emergency protection*

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 favor 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 may be a good idea not to deprive the operator of his ability to respond intelligently to emergencies.

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: However, passivity is also embodied in some design features. For example, no operator action is needed within 30 minutes of a disturbance that could threaten safety barriers. Furthermore, systems introduced to mitigate the effects of a severe accident (see below) were designed for passivity.

*5.1.2.5. Buildings and structures*

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. With respect to building layout and arrangement ABB Atom has traditionally favoured a coordinated 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 characterized by a division into an essentially nuclear and safety-related portion, consisting 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 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.

The reactor building encloses the reactor primary containment and forms a secondary containment. The building also 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.

The BWR 90 pressure-suppression containment consists of 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 and is statically free from the surrounding reactor building, except for the common foundation slab.

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 Finland and Sweden. The essential features of the BWR 90 containment to achieve enhanced environmental safety, including protection during 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 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.

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" of 0,25 g.

### 5.1.3. Safety concept

The engineered safety systems in BWR 90 are characterized by their consistent division into four redundant and physically separated subsystems. This concept that was introduced already in TVO I and II and further developed in Forsmark 3 and Oskarshamn 3, has been reconfirmed as constituting an optimal arrangement with respect to safety, layout and maintainability. 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.

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. 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 electromechanical insertion (using the motor drives of each rod), and by using the stand-by liquid control system (borating the primary loop). The stand-by 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 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 point estimate core melt frequency is significantly below $10^{-5}$, in accordance with international guidelines for new reactors (eg. as expressed in INSAG 3). [The point estimate core damage frequency was calculated to be about 2 $10^{-6}$, using Forsmark 3 site data,; application of a TVO model for the electrical systems yielded 7,4 $10^{-6}$, with an increased contribution from loss of offsite power ]

A summary of safety features etc. is presented in the Tables 5.1.1 to 5 1.4.

TABLE 5.1.1 : MAIN SAFETY-RELATED SYSTEMS IN THE BWR 90 PLANT

| Name | Safety graded | Main characteristics |
|---|---|---|
| All structures in RCPB | x | Reactor vessel, piping up to isolation valves |
| Containment structure | x | Concrete structure with liner, penetrations and access openings, isolation valves |
| Containment support systems | | Condensation pool cooling and cleanup system, containment spray system, containment atmosphere control system, overpressure protection by filtered vent system |
| Primary loop, integrity | x | Steam & FW lines to isolation valves, pressure relief valves and blowdown pipes, etc |
| Primary loop, in-vessel internals | x | Flow guiding structure, steam separators, steam driers, spray nozzles, feedwater spargers, core support plate and grid |
| Reactivity control system | x | Reactor scram function; RPS, scram valves, hydraulic insertion of control rods (non-safety scram backup via el mech drives), boron injection system |
| Emergency core cooling systems | x | Independent four train high and low pressure injection systems |
| Residual heat removal system | x | Two high pressure and one low pressure loop for shutdown and RHR |
| Main control room ventilation system | x | Coolers, filters, fans |

TABLE 5.1 2.: MAIN ACCIDENT INITIATORS FOR THE BWR 90 PLANT

| % contribution | Initiating event |
|---|---|
| 1,4 | LOCA (primary) Loss of Primary Coolant Accident |
| N/A | LOCA (secondary) Pipe Rupture (water or steam) |
| N/A | LOCA (interfacing eg SGTR (Steam Generator Tube Rupture) |
| 35 | Loss of feedwater |
| 21 | Loss of main heat sink |
| 15 | Reactor shutdown (ATWS Anticipated Transients Without Scram) |
| 16 | Reactor pressure vessel rupture |
| 10 | Loss of electric sources (partial)--> loss of offsite power |
| 1,6 | Miscellaneous Primary Transients (incl (Secondary) Transients such as turbine trip), |
| | Station blackout |

TABLE 5.1.3.: DESIGN FEATURES FOR PREVENTION AND PROTECTION LEVELS THAT REDUCE (R) OR SUPPRESS (S) THE INITIATORS' FREQUENCY, OR LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES (L)

A

| INITIATING EVENT | PREVENTION LEVEL FEATURES |
|---|---|
| LOCA (primary): | - Elimination of longitudinal welds, and reduction of number of other welds, reduce initiating frequency for severe RCPB leaks and ruptures;<br>- elimination of all large nozzles below the core top, eliminates large LOCAs at low location;<br>- direct steam cycle & normal coolant voiding; limits LOCA effects on core and fuel;<br>- wet motor RCPs; eliminates small LOCAs due to shaft seal failure |
| LOCA (secondary): | - not applicable |
| LOCA (interfacing): | - not applicable |
| Primary (& sec.) Transients: | - increased design margins; reduce initiator frequency; |
| Reactor pressure vessel rupture | - Elimination of longitudinal welds, and reduction of number of other welds, reduce initiating frequency for severe RCPB leaks and ruptures; |
| Loss of electric sources: | - feeder from independent "startup" grid, with gas turbine backup, & four train redundancy in electrical busbars, standby power generators, batteries and DC/AC converters; reduce sensitivity to loss of preferred power supply |
| Loss of main heat sink | - four train redundancy and separation, also with respect to water intakes, & possibility of drawing water from water channel outlet, if inlet becomes choked; reduces probability of total heat sink loss |
| Loss of (SG) feedwater: | - operational redundancy in normal feedwater pumps, & four train aux. FW supply system (HP coolant inj. system), pressure relief system permitting depressurization and operation of low pressure emergency coolant injection system; reduce initiating frequency |
| ATWS | - diversified means of shutting down; hydraulic scram insertion, el.- mech. scram backup insertion, combined with RCP runback, & automatic boron injection; "eliminates" ATWS |
| Station blackout | - feeder from independent "startup" grid, with gas turbine backup, & four train redundancy in electrical busbars, standby power generators, batteries and DC/AC converters; reduce probability of total loss of electric sources |

TABLE 5.1.3.: DESIGN FEATURES FOR PREVENTION AND PROTECTION LEVELS THAT REDUCE (R) OR SUPPRESS (S) THE INITIATORS' FREQUENCY, OR LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES (L)

B

| INITIATING EVENT | | PROTECTION LEVEL FEATURES |
|---|---|---|
| LOCA (primary): | - | automatic initiation of containment isolation & coolant injection to the RPV; if necessary, automatic depressurization to get low pressure coolant inj. system in operation, combined with elimination of below core large nozzles, facilitates reliable core flooding |
| LOCA (secondary): | - | not applicable |
| LOCA (interfacing): | - | not applicable |
| Primary (& sec.) Transients: | - | not important, due to increased design margins and "inherent" self-protection by negative reactivity coefficients |
| Loss of electric sources: | - | feeder from independent "startup" grid, with gas turbine backup, & four train redundancy in electrical busbars, standby power generators, batteries and DC/AC converters; reduce sensitivity to loss of preferred power supply |
| Loss of main heat sink | - | four train redundancy and separation, also with respect to water intakes, & possibility of drawing water from water channel outlet, if inlet becomes choked; reduces probability of total heat sink loss |
| Loss of (SG) feedwater: | - | operational redundancy in normal feedwater pumps, & four train aux. FW supply system (HP coolant inj. system), pressure relief system permitting depressurization and operation of low pressure emergency coolant injection system; reduce initiating frequency |
| ATWS | - | diversified means of shutting down; hydraulic scram insertion, el.-mech. scram backup insertion, combined with RCP runback, & automatic boron injection; "eliminates" ATWS |
| Station blackout | - | feeder from independent "startup" grid, with gas turbine backup, & four train redundancy in electrical busbars, standby power generators, batteries and DC/AC converters; reduce probability of total loss of electric sources |

## TABLE 5.1.4.: DESIGN FEATURES FOR MITIGATION LEVEL OF BWR 90 PLANT

| Safety functions | Systems (Cf Tab 5 1 1 ) | Passive/active | Design features/Remarks |
|---|---|---|---|
| Design Basis<br>Fission product containment | Containment of PS type<br>containment spray<br>condensation pool cooling | passive<br>active<br>active | The surrounding reactor building serves as a secondary containment, kept at a below-atmospheric pressure, ventilation exhaust air passes through filters |
| Coolant inventory | Water losses are made up by coolant injection systems | active | Water is drawn from condensation pool, pool water can be replenished from outside, |
| Decay heat removal | Via the main heat sink or via the RHR system, exceptionally, via coolant evaporation | active | Heat removal via turbine condenser to heat sink<br>Heat removal to heat sink or to containment |
| Reactivity control | Insertion of control rods, and RCP runback, + injection of boron | active<br>active | Backup, ever-present, boron injection is automated, safety-grade backup |
| Primary circuit pressure control | Safety relief valves (auto )<br>Manually controlled valves | active<br>active | Spring-force, self-activated on pressure, but active,<br>Manual order//manual actuation |
| Severe accident<br>Containment temperature and pressure control | One train of containment spray is normally operating, temperature and pressure initiates full operation | active | |
| Heat removal | Condensation pool cooling and external water supply | active | External cooling system, and water and energy supply |
| Tightness control | Filtered vent system ensures confinement also at high pressures | passive | At overpressure the gas compression chamber in wetwell will vent via rupture disc to the filter system, protecting the rest of the containment against rupture A valve arrangement permits reclosure of the vent path |
| Inflam gas control | The containment has inerted atmosphere during operation | passive | Containment is always nitrogen-filled, besides, a recombiner system prevents accumulation of oxy/hydrogen |
| Fission product containment | See tightness control above | | Tightness is primarily ensured by filtered vent system, leakages are collected by secondary containment venting Containment spray of drywell help wash out" fission products from atmosphere |
| Corium management | "Core catcher arrangement | passive | coolable pit for collecting, confining and cooling debris and molten material |
| Others | | | Not relevant |

### 5.1.4. Design data questionnaire (for Water-Cooled Reactors)

## I. GENERAL INFORMATION

1. Design name: BWR 90
2. Designer/Supplier address: ABB Atom AB Sweden
3. Reactor type: BWR / Number of modules: 1
4. Gross thermal power (MWth)/reactor: 2350 MWth
5. Net electrical output (MWe)/reactor: 720-820 MWe
6. Heat supply capacity (MWth): 100-500 MWth

## II. BASIC DESIGN DESCRIPTION

### A. CORE AND REACTIVITY CONTROL

7. Fuel material: $UO_2$
8. Fuel inventory (tonnes of heavy metal): 86
9. Average core power density: 53 kW/l
10. Average fuel power density: 27,2 kW/kgU
11. Maximum linear power: 38,0 kW/m
12. Average discharge burnup: 50 000 MWd/t
13. Initial enrichment or enrichment range: 1,9-2,25 Wt%
14. Reload enrichment at equilibrium: 3,1 Wt%
15. Refuelling frequency: 12 months
16. Type of refuelling (on/off power): off
17. Fraction of core withdrawn: ≈ 22%
18. Moderator material and inventory: $H_2O$
19. Active core height: 3,68 m
20. Equivalent core diameter: 3,90 m
21. Number of fuel assemblies: 500
22. Number of fuel rods/assembly: 100
23. Rod array in assembly: 4x5x5
24. Cladding material: Zr 2
25. Clad thickness: 0,6 mm
26. Number of control rods or assemblies: 121
27. Type: cruciform
28. Additional shutdown systems: Boron injection
29. Control rod neutron absorber material: $B_4C$/Hf
30. Soluble neutron absorber: Boron acid
31. Burnable poison material and form: $Gd_2O_3$ in $UO_2$

### B. REACTOR COOLANT SYSTEM

#### B1. Coolant

32. Coolant medium and inventory: $H_2O$/360 $m^3$
33. Design coolant mass flow through core: 9350 kg/s
34. Cooling mode (forced/natural): forced
35. Operating coolant pressure: 7,0 MPa
36. Core inlet temperature: 278°C
37. Core outlet temperature: 286°C

#### B2. Reactor pressure vessel

38. Overall length of assembled vessel: 20,6 m
39. Inside vessel diameter: 5,6 m
40. Average vessel thickness: 130+5 mm
41. Vessel material: ASTM steel
42. Lining material: stainless steel
43. Design pressure: 8,5 MPa
44. Gross weight: 600 $10^3$kgs

#### B3. Steam generator

45. Number of steam generators: N/A
46. Type: N/A
47. Configuration (horizontal/vertical): N/A
48. Tube material: N/A
49. Shell material: N/A
50. Heat transfer surface: SGN/A
51. Thermal capacity: SGN/A
52. Feedwater pressure: 7,0 MPa
53. Feedwater temperature: 215°C
54. Steam pressure: 6,9 MPa

55. Steam temperature: 285°C

**B4. Pressurizer**

56. Pressurizer total volume: N/A
57. Steam volume: N/A

**B5. Main coolant pumps**

58. Number of recirculation pumps: 6
59. Type: internal, glandless, wet, asynchr. motor
60. Pump mass flow rate: 1560 kg/s
61. Pump design rated head: 0,2 MPa
62. Pump nominal power: 750 kW
63. Mechanical inertia: low

## C. CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS) [REACTOR WATER CLEANUP SYSTEM (RWCU) IN BWRs]

64. Number of extraction lines: 2
65. Number of pumps: 2
66. Number of injection points: 2+
67. Feed and bleed connections: N/A

## D. CONTAINMENT

68. Type: Pressure-suppression
69. Overall form: cylindrical
70. Structural material: concrete
71. Liner materialcarbon steel: (stainless steel in pools)
72. Single/double wall: single (reactor building ensures secondary cont. function)
73. Dimensions (diameter/height): D=26,1/H=40 m
74. Design pressure: 0,6 MPa
75. Design temperature: 172°C (drywell)
76. Design leakage rate: 1 vol-% per day

# III. SAFETY RELATED SYSTEMS

## A. DESIGN CONDITIONS

### A1. Fission products retention

77. Containment spray system (Y/N): Yes
    a. Duration (h): one loop continuously; start of all after acc.
    b. Flow rate: each loop 77 kg/s
    c. Mode of operation: active
    d. Safety-graded (Y/N): Yes
78. F.P. sparging (Y/N): Yes
79. Containment tightness control (Y/N): Yes
    periodical insp. (embedded steel liner, 5 mm thick)
80. Leakage recovery (Y/N): ??
81. Guard vessel (Y/N): No

**A2. Reactivity control**

82. Absorber injection system (Y/N): Yes
    a. Absorber material: boron
    b. Mode of operation: active
    c. Redundancy: Yes
    d. Safety-graded: active inj. from storage vessel, via pumps and valves
83. Control rods (Y/N): Yes

a. Maximum control rod worth (pcm): not imp. for BWR [reactivity worth of all rods17%]
b. Mode of operation (active): Active (hydraulic pressure for scram; el.mech. act. as backup)
c. Redundancy: Yes
d. Safety-graded: Yes

**A3. Decay heat removal**

*A3-1 Primary side*

84. Water injection
   a. Actuation mode (manual/automatic): M&A
   b. Injection pressure level: 7,0 MPa
   c. Flow rate: 4x22,5 kg/s
   d. Mode of operation (active/passive): Active
   e. Redundancy: Yes
   f. Safety-graded (Y/N): Yes

84b. Water injection
   a. Actuation mode (manual/automatic): M&A
   b. Injection pressure level: 1,0 MPa
   c. Flow rate: 4x170 kg/s
   d. Mode of operation (active/passive): Active
   e. Redundancy: Yes
   f. Safety-graded (Y/N): Yes

85a. Water recirculation and heat removal (hot standby) [Steam relief to cond. pool, HP inj. for makeup.] a.Intermediate heat sink Condensation pool & cooling system
   b. Mode of operation (active/passive): Active
   c. Redundancy: Yes
   d. Self-sufficiency (h): < 1h
   e. Safety-graded (Y/N): Yes

85b. Water recirculation and heat removal (shutdown)
   a. Intermediate heat sink: Closed cooling system
   b. Mode of operation (active/passive): Active
   c. Redundancy: Yes
   d. Self-sufficiency (h): > 1 h
   e. Safety-graded (Y/N): Yes

*A3-2 Secondary side*

86. Feedwater [Aux. FW supply in 84 above]
   a. Actuation mode (manual/automatic): Automatic
   b. Flow rate (kg/s): as needed
   c. Mode of operation (active/passive): active
   d. Redundancy: 3x50%
   e. Self-sufficiency (h): limited
   f. Safety-graded (Y/N): No

87. Water recirculation and heat removal
   a. Ultimate heat sink (cold source): Sea or lake
   b. Mode of operation (active/passive): active
   c. Redundancy: 3x33%
   d. Self-sufficiency (h)-
   e. Safety-graded (Y/N): No

*A3-3 Primary pressure control*

88. Implemented system (Name): Reactor pressure relief system
   a. Actuation mode: 12 safety valves automatically (on pressure directly or via measuring channels), and/or manual ignition; 2 control valves for pressure control o fisolated system: A (&/or M)
   b. Side location: primary
   c. Max. depressurization rate/valve:100 kg/s

control valves: no

**B. SEVERE ACCIDENT CONDITIONS**
*[A severe accident is a beyond DBE accident.]*

**B1. Fission products retention**

89. Containment spray system (Y/N): Yes
90. F.P. sparging (Y/N): see item 78
91. Containment tightness control: Indirectly, by vent system (& filters), see item 79
92. Leakage recovery (Y/N): reclosure of vent system valves
93. Risk of recriticality (Y/N): ??

**B2. Recriticality control**

94. Encountered design feature
    a. Mode of operation (active/passive): N/A
    b. Safety-graded (Y/N): N/A

**B3. Debris confining and cooling**

95. Core debris configuration: Pit, with (core catcher) cooled walls
96. Debris cooling system
    a. Mode of operation: passive water supply off condensation pool active water supply from external sources
    b. Self-sufficiency: 10 h
    c. Safety-graded : Special safety class

**B4. Long term containment heat removal**

97. Implemented system: Pumps, HXs, valves
    a. Mode of operation (active/passive): active
    b. Self-sufficiency (h): --
    c. Safety-graded (Y/N): No
98. Intermediate heat sink
    a. Self-sufficiency (h): --
    b. Safety-graded (Y/N): No
99. External coolant recirculation
    a. Implemented components: Pumps, HXs, valves
    b. Mode of operation (active/passive): Active
    c. Self-sufficiency (h): --
    d. Safety-graded (Y/N): no
100. Ultimate heat sink
    a. Self-sufficiency (h): --
    b. Safety-graded (Y/N): No

**B5. Combustible gas control**

101. Covered range of gas mixture concentration not relevant for inerted BWR containments
102. Modes for the combustible gas control
    a. Containment inertation: Yes
    b. Gas burning: N/A
    c. Gas recombining: Yes
    d. Others: N/A

**B6. Containment pressure control**

103. Filtered vented containment (Y/N): Yes
    a. Implemented system: rupture discs & valves, filters
    b. Mode of operation: passive
    c. Safety-graded (Y/N): Special safety class
104. Pressure-suppression system (Y/N)
    a. Implemented system: Yes
    b. Mode of operation: Passive

b. Mode of operation: Passive
c. Safety-graded (Y/N): Yes

## C. SAFETY-RELATED I&C SYSTEM

Automatic load following (Y/N): Yes
* range: 30% power
* maximum rate: 20%/min

Load rejection without reactor trip: Yes
Full Cathode Ray Tubes (CRT) display: Yes
Automated startup procedures: Yes, largely
Automated off-normal shutdown procedures: Yes
Use of field buses and mart sensors: Yes
Expert systems or artificial intelligence advisors: Yes
Protection system backup: Yes

## D. EMERGENCY POWER SUPPLY SYSTEM
***[rather backup power supply]***

105. Type (diesel, gas, grid conn.): conn. to 2nd grid + 4 dg units
106. Number of trains: 4

## E. AC/DC SUPPLY SYSTEM

107. Type (rectifier/converter/battery): AC from rectifier/battery/converter trains; DC by local rectifiers
108. Estimated time reserve: 1 hour

# IV. CONVENTIONAL THERMAL CYCLE

## A. TURBINE SYSTEM

109. Type
110. Overall length (m): depends on make
111. Width (m): depends on make
112. Number of turbines/reactor: 1 (or 2)
113. Number of turbine sections per unit: 1 HP/ 3LP
114. Speed (rpm) (- will depend on site): 3 000 rpm

## B. STEAM CHARACTERISTICS

115. HP inlet pressure: 6,78 MPa
116. HP inlet temperature: 283 °C
117. HP inlet flow rate: 1260 kg/s
118. LP inlet pressure: ≈1,5 MPa
119. LP inlet temperature: ≈265 °C
120. LP inlet flow rate (per section): ≈255 kg/s

## C. GENERATOR

121. Type: 3-phase, synchr.
122. Apparent power: 1010 MVA
123. Active power: 858 MW
124. Frequency (depends on site): 50 Hz
125. Output voltage: 23 kV
126. Total generator mass (tonnes): depends on make
127. Overall length: depends on make
128. Stator housing outside diameter: depends on make

D. CONDENSER

129. Number of tubes: depends on make
130. Heat transfer area: 24300 $m^2$
131. Flow rate: $\approx$36 $m^3/s$
132. Pressure: $\approx$3,8 kPa
133. Temperature of circulating water: 5°C

E. CONDENSATE PUMPS

134. Number: 3
135. Flow rate: $\approx$380 kg/s
136. Developed head: $\approx$1,43 MPa
137. Temperature: $\approx$30°C
138. Pump speed: depends on make

### 5.1.5. Project status

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 is 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 are two of the top performing nuclear power plants in the world with an average capacity factor of 91,3% over the last 8 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. 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, in accordance with the general stipulations presented by the organizers, is not easily done; the status of the BWR 90 project lies somewhere between D2 and D3 - pretty close to D3 even though all D3 activities have not been completed. To be more specific, the activities "design/engineering 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 been completed.

With respect to licensing activities, reference is made to the comment above with respect to 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.

### 5.1.6. Project economics

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 indicates a total construction time of 54 months for the plant - from pouring of the first concrete to start commercial operation.

Detailed turnkey cost estimates have "confirmed" that the general rule regarding "economics of scale" will apply; the overnight cost per net KWe output is some 10% higher than for a 1000-1100 MWe unit, and about 15% higher than for the 1350 MWe unit. The fuel cycle costs of the three versions are roughly equal, but the operating and maintenance costs will again be highest for the smallest unit, in particular with respect to the cost of personnel. As a consequence, the energy cost, the cost per net kWh, will differ between the three versions in approximately the ratios indicated above.

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## 5.2. AP600 REACTOR SYSTEM DESCRIPTION AND DEVELOPMENT STATUS

### 5.2.1. Basic objectives and features

The basic AP600 design objectives are as follows:

A. Safety

- Use of simple, dedicated, independent passive safety systems that require no operator action for 72 hours post accident, and maintain core and containment cooling indefinitely without AC power.
- Predicted core damage frequency < $10^{-5}$ per year and a significant release frequency <$10^{-6}$ per year. [Numbers listed are the objectives: Current estimates are core & damage frequency of 3 x $10^{-7}$ and significant release frequency of 9 x $10^{-8}$]

B. Reliability

- To simplify design, operation and maintenance
- Robust design with at least 15 percent operating margin on core parameters and designed for rated performance with up to 10 percent of steam generator tubes plugged at a maximum hot leg temperature 316°C.
- Use of proven power generating system components which require no plant prototype.
- Plant design objective of 60 years without reactor vessel replacement,
- Occupational radiation exposure < 70 man-rem per year.
- Overall plant availability goal greater than 90 percent considering forced and planned outages. Objectives for unplanned reactor trips less than one per year. Objective for a refueling outage free of major problems is completion in 17 days or less (breaker to breaker).

C. Economics

- Net electrical output of at least 600 MWe.
- U.S. overnight capital cost less than $ 1500/kW in 1993 $ and a power generation cost that is competitive with coal on an average US site.
- A five year total project schedule (owner's commitment to commercial operation) and a three year construction schedule for replicate units. First unit would have a six year overall schedule.

Many design upgrades (lesson-learned and state-of-the-art features have been incorporated into the present AP600 plant design. The major innovative plant features are as follows:

- Low power density reactor design which provides increased operating margins and improved fuel economy.
- Simplified primary loop configuration employing canned motor pumps mounted on the steam generator lower heads.
- Simple, passive safety systems, not required for normal operation which use "natural" driving forces only, require one-time alignment of active valves and no support systems after actuation.
- State-of-the-art digital instrumentation and control systems plus an advanced man-machine interface control room with sit-down, console-type work stations, soft controls

and integrated, prioritized alarms and procedures

- Enhanced overall plant arrangement designed to minimize cost and construction schedule and to meet safety, operational and maintenance criteria
- Modular construction and fabrication concepts to shorten construction schedule and reduce construction cost

### 5.2.2. Design description

*5.2.2.1. Nuclear steam supply system*

A Primary circuit

The primary circuit of the AP600 reactor has retained most of the general design features of current generation designs, the design adapted some evolutionary features that enhance the safety and maintainability of the design Two cold leg pipes of each coolant loop, and elimination of primary piping between the primary pump and the steam generator all significantly contribute to such enhancement These features also simplify the support structure for the primary systems, reduce inservice inspections and provide a better access for the maintenance staff

- Reactor Pressure vessel

The reactor pressure vessel is the pressure boundary of the reactor core It houses a variety of internals that support and locate the core and provide for flow distribution It is fabricated from forged sections to avoid longitudinal welding It also avoids horizontal welds around the belt line Specific design characteristics for the AP600 pressure vessel are given in the general design data (in section 5 2 4)

Penetrations in the reactor pressure vessel (RPV) allow cooling and make up water to enter and exit Four (22 in ID) inlets spaced 90° apart, two (31 in ID) outlet nozzles 180° apart, and two (6 81 in ID) safety injection nozzles are incorporated in the upper cylindrical region of the reactor vessel The RPV is some what larger in size than a conventional two loop plant, (3 99m ID, 11 05m high), provides a larger primary water inventory, and increases the reactor vessel gap This significantly reduces the total neutron fluence over the life time of the RPV The rest of the RPV specifications are similar to the conventional RPV of a two loop plant of the same type Reactor pressure internals such as the structures for control of hydraulic flow, fuel support devices, fuel assemblies, and control rods, and control drive mechanisms are also similar A cross-section schematic of the AP600 Reactor System is shown in Figure 5 2 1

- Reactor core

The reactor core of the AP 600 has been studied over a wide range of parameters and their effect on the main design objectives (simplification, reduction of cost, and reliability) has been determined The resulting design has 20% more fuel assemblies than a conventional two loop plant, namely 145 fuel assemblies of the 17 X 17 (optimized fuel assemblies VANTAGE V fuel) with an active fuel length of 3 65m The design is based on well developed low enrichment fuel core technology (VANTAGE 5H design) This results in a low power density design (78 8 kW/l) and provides for increased margins and better fuel utilization The design also features a radial neutron reflector consisting of steel and a somewhat larger water gap at the core periphery These features provide for better fuel cycle cost and provide for a lower total neutron fluence to the RPV ($< 2 \times 10^{19}$ n/cm$^2$ for a 60y life time compared to $5 \times 10^{19}$ n/cm$^2$ for a conventional plant with 40 year lifetime)

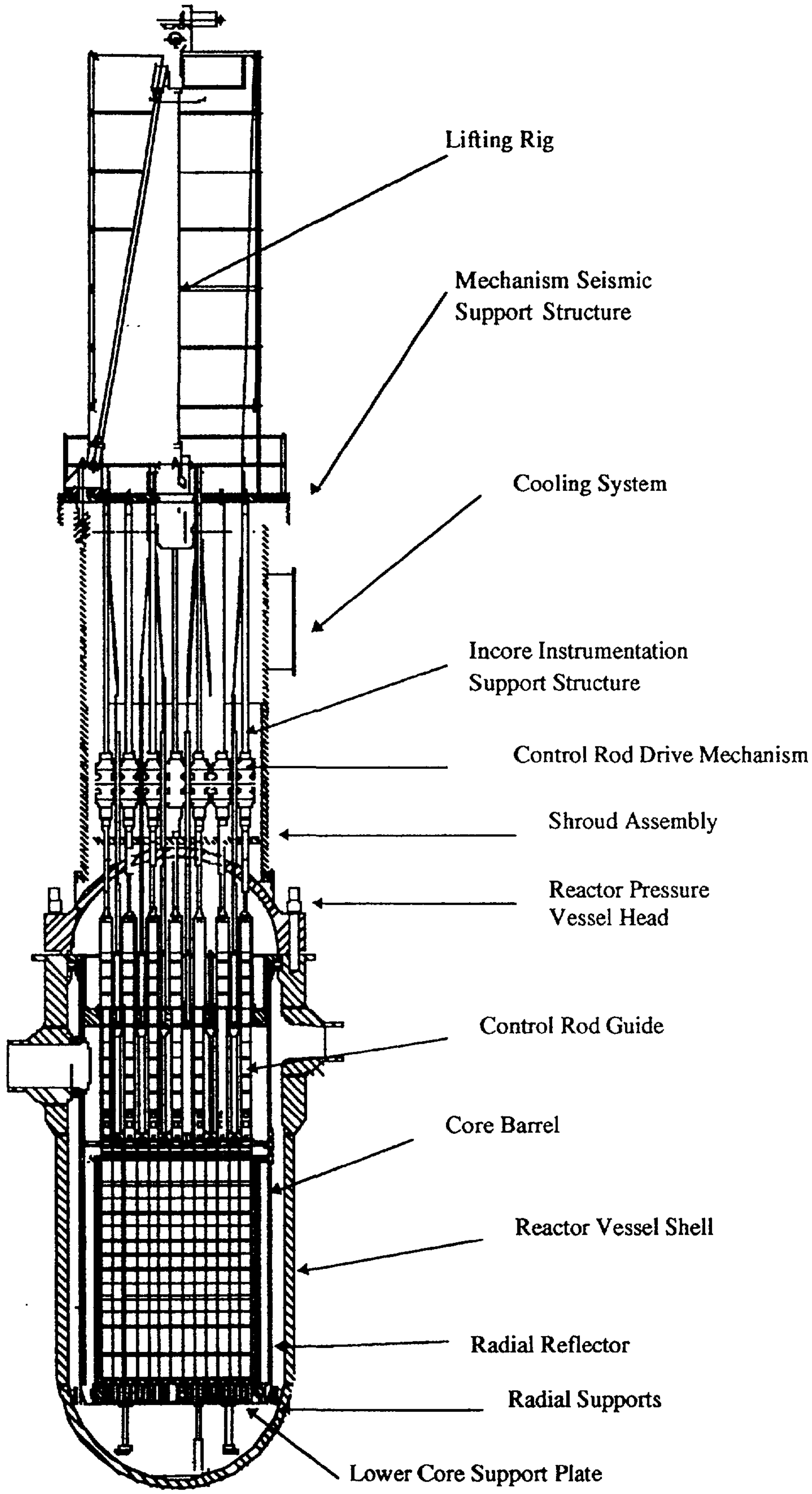


*FIG. 5.2.1. AP600 reactor system cross section.*

Soluble boron and burnable poisons are utilized for shutdown and fuel burnup reactivity control. Low worth gray control rods (16 clusters) are used for power regulation and load follow. The core consists of 3 radial regions differing in enrichment, and has a negative temperature coefficient of reactivity. The enrichment of the fuel ranges from 2 to 4%. The core has a fuel cycle of 18 to 24 months with region discharge burnups as high as 55000 MWd/t. (Fuel assembly burnups can be 65,000 MWd/t).

Neutronic and thermohydraulic data will be obtained from fixed in-core instrumentation, fixed incore doctors and thermocouples, providing on-line core performance monitoring.

- The integrated head package

The reactor pressure head and head mounted components structure are replaced by an Integrated Head Package (IHP). The integrated head package consists of the reactor vessel head, control rod drive mechanism (CRDM), seismic supports, cables and cable bridge, and messenger tray. The IHP also consists of the shroud assembly and cooling system. The in-core instrumentation system is also integrated into the head package and could be moved with it in the case of refueling. The conoseals have been replaced by individual swage fittings which are usually used in plants with bottom mounted instrumentations. Permanently installed reactor cavity seals and a multiple stud tensioner substantially reduce the time for reactor vessel head closure, and ORE (Occupation Radiation Exposure).

- Steam generators

The steam generators are Westinghouse Delta 75 "W" Model with "F" tube bundles made of Inconel - 690 TT steel U shaped, vertically placed and consisting of 6307 tubes 0.688 in OD with a wall thickness of 0.04 in. The total heat transfer area is approximately 75,180 $ft^2$ tubes (~6984 $m^2$). Each SG is rated at 970 $MW_{th}$ and capable of generating 1.91 X $10^6$ kg/hr of saturated steam at 777 psia (5.36 MPa).

The coolant flow rate of the primary circuit totals to approximately 381$m^3$/min with a hot leg temperature of 316°C and a cold leg temperature of ~ 277°C.

The primary side head has been modified to allow for the direct attachment of two reactor coolant pumps. Access and space for tube inspection, plugging and sleeving is provided via a multi-port nozzle dam. 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. The design has led to only ~ 12% increase in SG total weight. The secondary side of the SG is maintained the same. Maintenance of the steam generator could be carried out by using robotic arms or by maintenance personal at the time of refueling.

- Reactor coolant pumps

The Westinghouse canned motor pump model 8006 employed at Shippingport and in a large number of nuclear and fossil applications has been adapted for the AP600. The pump is a single speed (1800 rpm) designed to supply 3300 shp. The pump motor is three phase, designed for voltage of 4000 V at the terminals. The pump by design has very little coast down capability. In the case of complete loss of flow accident, the flow coast down required to satisfy the departure from nuclear boiling criteria is a function of the time needed to initiate rod trip and rod insertion. This is estimated to be in the order of ten seconds. In order to meet the coast down requirement the canned motor pump design has been modified to provide for high inertia rotor incorporating a relatively large thrust runner assembly. The runner is fitted with a heavy depleted uranium disc (canned) within the runner. The rotating inertia of the pump in this case is 210 kg $m^2$ (5000 lb $ft^2$). This provides the required inertia to avoid exceeding minimum DNB requirements for postulated design basis accidents.

The advantages of utilization of the canned motor pump in such mode and configuration are significant. The new high-inertia rotor has gone through a testing program in a full size prototype. The canned motor pump itself has demonstrated its excellent performance and reliability. Last but not least, the design approach for its employment in the system simplifies the chemical and volume control system because it has no seals, does not require a seal water system or continuous changing pump operation.

*5.2.2.2. Balance of plant*

- Turbine unit and associated systems

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 tandom -compound, 1800 rpm machine. The turbine generator is intended for base load operation and also has load follow capability.

Mechanical design of the turbine blade 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 two moisture separator reheaters (MSR) are integral components in the main turbine system. Each MSR has a single stage of reheat. The system is designed to allow operation with one MSR in service.
- The condenser and circulating water systems were optimized. It is a twin-shell, multipressure unit with one double-flow, low-pressure turbine exhausting into the top of each shell.
- The hot wells of the two shells are interconnected and function basically as a single unit. Condensate polishing is provided in a sidestream to remove inpurities in the condensate and improve secondary side chemistry.
- Two feedwater pumps are used. Each has the capability to operate the plant at 70% power level
- Steam generator blowdown has been simplified (no filters, no separate demineralizer). The blowdown fluid is sent to the condensate polishers for clean-up.

- Radioactive waste management system

The radioactive wastes are processed within the system with equipment and systems that have been designed for easy maintenance. High exposure due to operation has been substantially reduced compared to current plants of similar type.

From a construction point of view the radioactive waste treatment system has been removed from seismic I category building and its components have been simplified and optimized as much as possible for modularization.

Several engineered features have been implemented that resulted in either simplification of the system or increased safety margin. Local removable shields are used to house portions of pumps, filters, or valves providing for better access. In other cases submersible pumps are used eliminating pump shielding all together. Another factor contributing to volume reduction is the elimination of valve and instrument rooms with this equipment being placed in shield boxes within piping areas.

The waste processing system has been tailored according to waste categories to produce effluent suitable for reuse, discharge or final disposal.

The processing system i.e filters, and ion exchanges, are sized for high process rate and capacity. All processes including gaseous waste systems are operated at atmospheric pressure and ambient temperature minimizing leakage possibility. In-liner process is used for solidification of liquid wastes providing maximum packing efficiency.

Other features of the system are the mobile evaporation and solidification equipment for abnormal conditions. The elimination of detergent waste through the use of dry cleaning laundry is another feature of the waste treatment system.

*5.2.2.3. Instrumentation, control and emergency protection systems*

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 computer system of the station control and data acquisition is a distributed micro processer based systems. A digital multiplexed control system takes the place of hard wired analogue control. This accounts for a significant reduction in cable usage. Built-in diagnostics and board level maintenance makes restoration of operability of any fault in the system a matter of replacement of printed circuit cards. Automatic control systems and procedures are deployed to simplify these procedures and power level manoeuvers. In case of unsafe conditions the reactor protection system (PMS) takes over and automatically scrams the reactor and actuates the relevant safety systems. Diverse methods are used to assure the shutdown of the reactor in hypothetical situations. The systems also provide for post-accident monitoring.

The PMS consists of the following:

- Integrated protection cabinets. These contain the reactor trip subsystem, the engineered safety features actuation subsystem and the communication subsystem.
- Engineered safety features actuation cabinets, which perform system-level logic calculations such as initiation of safety injection.
- Protection logic cabinets, which provide the capability for on-off control of individual plant loads for the Class 1E applications. These cabinets receive inputs from the engineered safety features actuation cabinets and the main control room via the main control room multiplexers.
- Qualified display processing cabinets, which receive inputs from qualified input signal cabinets and the integrated protection cabinets. This data is displayed in the main control room and remote shutdown area on qualified video display units.

The plant control system provides for non safety plant control functions. The Diverse Actuation System provides for an alternative means for initiating the reactor trip and the emergency safety features by integrating a complete diverse actuation system within the plant control system.

Another I&C system is the special monitoring system. This system provides for long term monitoring and diagnostic. Functions such as sensor response, rod drop test, transient monitoring, corrosion/erosion monitoring and individual component diagnostics are some examples of the system functions. The I&C architecture contains also the in-core instrumentation system for 3-D power mapping.

The data display and monitoring system provides display, alarm generation, interactive procedures and several backup functions. The operation and control centers system includes the complete operational scope of the main control room, the remote shut down area, and the technical support center. The system, actually defines the layout of the control room, work stations, the remote shut down area, and designs the lay out, and content of displays, alarms, controls, and procedures.

The main station computer communicates with the I&C systems and compares the conditions (devices inputs, out puts, self generated functions and constants) and the control action taken to verify that it is producing the expected responses. Any faults found could be transferred to the alarm system that brings it to the attention of the operator.

- Control room

The AP600 control room is centrally located within the plant. Its main features are:

1. A wall-panel information station which provides plant status displays.
2. Two independently powered work-stations capable to start up the plant, obtain full power operation, manoeuvre, and shut down the plant along with a full display panel.
3. A supervisor work station identical to the two work stations except that its controls are locked so that no control action can be taken in normal conditions. Failure of either of the two work stations will enable the supervisor work station to become the active one.

Data links are used to communicate between the control stations and the control and protection system. The alarm system which categorizes, prioritizes, 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 in their work stations (sit down see-over operation). The operator can select any function and the relevant displays will appear on the screen. The operator can interrogate that function for any information. The operator could also select any component and alter its current status.

*5.2.2.4. Safety considerations*

a. Inherent safety features

Inherent hazards, eliminated by design, make the AP600 inherently safe with respect to these hazards. The AP600 has some inherently safe characteristics -- the negative reactivity coefficient at all times which halts the chain reaction in the event of an expected power increase, the large water inventory, and the provision to limit any release from the containment by placing

Residual Heat Removal (RHR) and safety injection systems entirely within the containment building -- are all inherent safety characteristics with respect to reactivity control, LOCA, and off-site releases respectively.

b. Passive safety

The name AP600, standing for advanced passive, gives the design name and the main design emphasis. The design provides for all the required safety functions following plant transient or accidents through passive safety features. For the AP600 design, passive residual heat removal, inventory control for the reactor coolant system, short and long-term LOCA injection, and containment heat removal are all passive and integrated within the containment system. All passive safety systems have been verified using PSA and DBA analysis. Utilization of previous experience of similar designs on the conventional PWR and BWR technology and test programs carried out by Westinghouse, and in collaboration with other entities, is aimed at meeting NRC safety criteria and already have been verified. These systems have no pumps and have 34 remotely operated valves to provide core decay heat removal and safety injection. For a typical active system, 6 pumps and over 100 remote valves are needed to perform the same functions. A schematic drawing of the AP600 Passive Safety injection systems is shown in Figure 5.2.2.

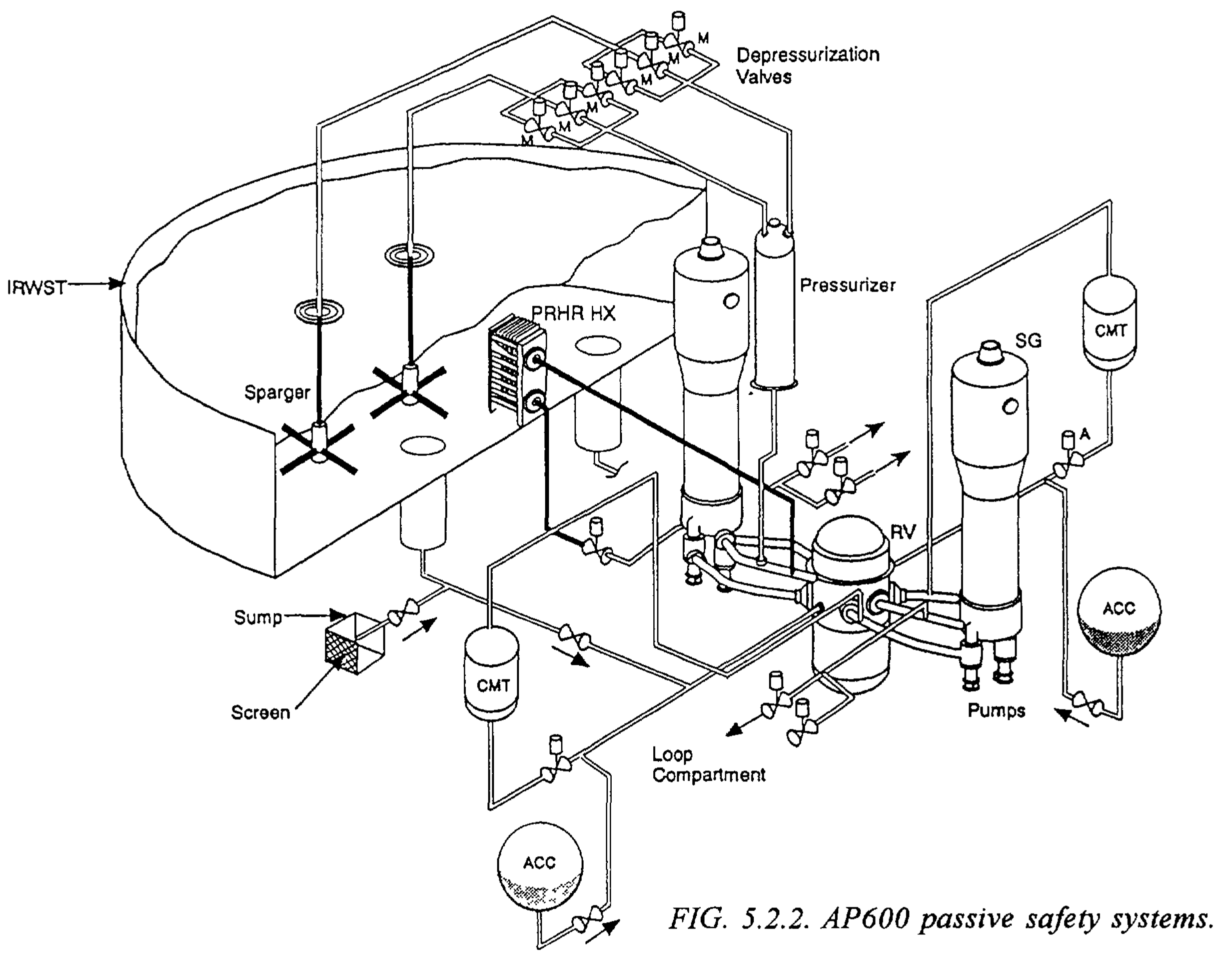


*FIG. 5.2.2. AP600 passive safety systems.*

- Passive residual heat removal (PRHR) system:

If feed water systems or steam generator heat removal is not available, two PRHR heat exchangers provide decay heat removal. The system is located above the Reactor Coolant System (RCS) and forms a closed loop at full system pressure using natural convection circulation. The heat sink for the system is the In-containment refueling water storage tank. The system is designed for non-LOCA events and is automatically activated on a low steam generator water level signal by opening one of two 25cm normally closed, fail-open air-operated valves. The systems, in conjunction with the passive containment cooling systems (PCCS), can provide cooling indefinitely. The IRWST can provide decay heat removal for approximately 2 hours before the water starts to boil. At boil-off, the PCCS would act as the alternate heat sink and the condensate would be collected and piped back into the IRWST.

- Inventory control for the reactor coolant system

If the reactor makeup system is unavailable, passive reactor makeup is provided by two core make up tanks (CMT). The tanks are designed for full RCS pressure and are located above the RCS cold loop piping (2.4 m). In case of a small LOCA, water can flow by gravity into the reactor vessel through two direct vessel insertion nozzles. The CMT's activate on a low water level signal from the pressurizer and from the safety injection signal. In either case, the reactor is scrammed, the RCP are tripped, and the CMT's isolation air-operated valves are automatically opened. Each CMT tank has a capacity of (~ 56.6 $m^3$) and is maintained full of borated water. The pressure balance is provided for through an overhead connection to the top of the pressurizer. The system is mainly designed for small leaks or small LOCA. If a large LOCA occurs, the accumulators are needed to provide a high make up flow rate. The 48.1 $m^3$ of borated water under nitrogen pressure of 48.3 bars ensures the filling of the reactor vessel lower plenum and down comer following a RCS blow down. Additional water required for longer term cooling is provided by the in-core refueling water storage tank.

- Long term cooling

After the initial injection of the limited water capacity of the CMT's and accumulators, the longer term source of water for cooling is from the (IRWST). The IRWST is located in the containment above the RCS and has a storage capacity of about 500,000 gal (2000 $m^3$). The injection in this case is gravity driven and could only be established after the RCS has been depressurized below the elevation head. The depressurization could be attained by either the automatic depressurization system (ADS) or by large LOCA. The IRWST would provide water flow into the reactor for at least 6 to 10 hours. By the time the water level in the IRWST goes low, the containment water level would have increased sufficient for water to flow into the RCS through the gravity recirculation lines.

- The automatic depressurization system (ADS)

The ADS is made of four stages of valves. Through this arrangement slow controlled depressurization is attained. The first three stages are directly connected to the pressurizer and discharges to the IRWST through two spargers. The fourth stage is directly connected to each hot leg and discharges directly (through redundant isolation valves) to the containment. Actuation of the depressurization is initiated as CMT levels decrease.

- The passive containment cooling system (PCCS)

The atmosphere is the ultimate heat sink for the AP600 reactor system. Computer calculations have shown that under design basis accident scenarios the containment steel structure would provide sufficient cooling to prevent exceeding the containment design pressure. In performing this function, the heat is transferred through the containment vessel to the environment. The containment concrete structure with the baffle provides for natural circulation of outside air. The air enters into the gap between the concrete containment and the baffle at the top, flows down the outside of the baffle, and rises up along the steel containment vessel. The heat transfer is enhanced by a water film formed by the gravity drain of water onto the containment shell from the PCCS water storage tank (1514 $m^3$) located at the top of the concrete structure. The tank has sufficient water to provide three days of cooling. Analysis carried out by Westinghouse has shown that after three days with only the dry air circulation, the containment steel structure stays well below the design pressure (45 psig). The PCCS system is activated in the event of high pressure reading within the containment. Such an event would take place if the normal heat removal (the containment fan coolers) is unavailable for an extended period of time, or as a result of design basis accidents with large energy releases.

d. Safety analysis

The AP600 design has been confirmed by the results of transient and accident analysis to meet its safety objectives. Core melt down frequency and severe accident evaluation show that the passive systems are effective in mitigating the consequences of design basis accidents.

Analysis has shown that the peak clad temperature following a large break (LOCA) is about 800°C which is substantially lower than a PWR plant of current design and well below NRC limit of 1204°C. For small LOCA ($\leq$ 8 inches) analysis show that core uncovery does not occur.

The AP600 has a core melt frequency an order of magnitude less than the best of the current US plants. Containment failure has been shown to be about 100 times lower. This is due to the low probability of containment failure during any core melt scenario and the initiation of containment isolation early, at the start of an accident.

c. Active safety

Consistent with current practice, active systems are used as first defense level against more probable events. These systems "termed as non-safety systems" establish and maintain safe shutdown conditions. This requires at least one non-safety AC power source to be available. These systems include the Chemical and Volume Control System (CVCS), the Startup Feed Water System (SFWS), the Normal Heat Removal System (RNS), Spent Fuel pit cooling System (SFS), and the Diverse Actuation System (DAS). The later provides the defense in depth function of system actuation to back up the Safety Monitoring System (PMS) and to protect against common mode failure.

The Diverse Indication System (DIS) is part of the (DAS) which provides for safety related system automatic actuation. The DIS provides for alarms to the main control room so that manual actuation can be taken.

*5.2.2.5. Buildings and structures*

a. Buildings

The AP600 plant consists of six principal building structures (the nuclear island, the turbine building, the annex building, the diesel generator building, the radioactive waste building and the access and control building). The nuclear island including the containment, the shield, and auxiliary buildings is built on one basemat. The buildings volumes have been efficiently utilized keeping in mind the accessibility and maintenance jobs to be carried out. The nuclear island is relatively small compared to a conventional 2-loop plant of the same size. All safety related equipment are located in the containment building, and the auxiliary building. All reactor coolant support systems including the chemical and volume control systems are within the nuclear island. This substantially reduces the shielding needed outside the containment.

The plant layout has a single personnel entry with separate access corridors for clean and potentially contaminated passages. A well defined area for control of personnel radiation exposure is consistent with the ALARA principle.

A modular construction approach is used for the equipment, cable trays, HVAC ducts, and a uniform spacing between walls and floors to accommodate such equipment is provided. This provides for an efficient construction schedule and efficient space utilization. Facilities with long experience for module fabrication such as ship yards are anticipated to construct the equipment. The schedule of the flow of materials, shop production, and assembly of modules will be closely integrated and supervised.

This modular construction approach and simpler design will allow for substantial reduction in construction time; a 36 month construction period from the lay down of the first structural concrete to fueling of the reactor is expected.

b. Structures

Conventional concrete structures, and prefabricated steel structures assembled on site and filled with concrete after placement are used in various structures. Precast concrete modules would also be used. Wide use of removable formwork is employed to limit steel exposure to potential corrosion. Prefabrication of reinforced rebar modules is extensively used. In some places mechanical rebar splices are used to reduce the weight of prefabricated modules. All of these techniques have been employed in previous nuclear power plant construction. Steel structures such as the air baffle, and the containment vessel are constructed of steel panels. Panels are made of stiffened steel or corrugated plate depending on availability and cost; panels for the baffle are designed with thermal expansion of the containment vessel in mind. Future inspection and maintenance are taken into consideration.

c. Seismic behaviour

Buildings and structures of a nuclear power plant are usually classified into three groups with regard to seismic design. Group I includes building and structures which house the nuclear reactor system, and the nuclear steam supply system. This includes systems, mechanisms, electrical devices, and control panels intended to ensure safety and preclude release of radioactivity at all times. Group II refers to buildings and structures intended to house equipment whose failure may result in power generation interruption but does not affect the equipment housed in group I building, e.g. failure of such equipment does not result in release of radioactivity materials. Group III refers to all other structures other than group I and II.

Certain aspects of the AP600 civil engineering practice are inherently favourable from the seismic stand point. The reactor building center of gravity is maintained as low as possible.

Simple symmetry of equipment is practiced. A single reinforced concrete slab is laid under the reactor building. Other engineered practices used are:

- The stiffness of piping is chosen such that the natural frequency will be as far as possible from the characteristic motion of the building
- Support for major equipment have provisions for freedom of motion, other restraining devices such as shock absorbers are used to limit deflections during an earth quake
- Shock absorbers are also used to increase stiffness of main pipes while still allowing freedom of motion to accommodate thermal expansion

Most major equipment has gone through full or reduced scale tests to verify their seismic resistant design

The reactor vessel is the only fixed element in the system The rest of the piping system is allowed to move to accommodate thermal expansion

### 5.2.1. Safety concepts

TABLE 5 2 1 MAIN SAFETY RELATED SYSTEMS IN THE AP600 CONCEPT

| Name | Safety graded | Main characteristics |
|---|---|---|
| Primary Circuit | X | reactor vessel<br>4 recirculation pumps<br>2 steam generators |
| Diverse Reactivity Control System | X | boron injection, via passive safety injection system |
| Control Rods | X | 45 Rod cluster control assemblies |
| Passive Residual Heat Removal system | X | 2 HX within the In-containment Refueling Water Storage Tank (IRWST) Permanently connected to the primary circuit |
| Passive Core Cooling System | X | Safety Injection, Depressurization and Passive Residual Heat Removal |
| Passive Containment Cooling System | X | Safety related ultimate heat sink |
| Passive Main Control Room Maintainability System | | Provides fresh air, cooling and pressurization |

TABLE 5 2 2 MAIN ACCIDENT INITIATORS FOR THE AP 600

- LOCA (Primary) Loss of Primary Coolant Accident
- LOCA (Secondary) Secondary Pipe Rupture (water or steam)
- LOCA (Interfacing) e g SGTR Steam Generator Tube Rupture
- ATWS Anticipated Transients Without Scram,
- Primary Transients,
- Secondary Transients (Turbine trip),
- Loss of Electric Sources (all AC sources),
- Total loss of the cold sources,
- Total loss of the Steam Generator Feedwater,
- Station blackout

TABLE 5 2 3 DESIGN FEATURES FOR PREVENTION AND PROTECTION THAT REDUCE, SUPPRESS, THE INITIATORS FREQUENCY OR LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES

**PREVENTION LEVEL**

**LOCA** (Primary)
- Reduced vessel fluence reduce initiator frequency
- Primary circuit integration limits small LOCA consequences
- Canned pumps suppress initiator (seal caused small LOCA)
- Adaptation of the leak before break limits accident consequences,

**LOCA** (Secondary)

**LOCA** (Interfacing)
- Simplified systems and improved materials reduces initiator frequency

**Primary transients:**
- Increased design margins reduces initiator frequency

**Secondary transients:**

**Loss of electric sources:**

**Total loss of the cold source** (Water)
- Passive ultimate Decay Heat Removal (air) suppress initiator

**Total loss of the S G feedwater:**

**Station Blackout:**

**PROTECTION LEVEL**

**LOCA (Primary):**
- Primary circuit integration > leakage limitation
- Increased water inventory around the core,
- Safety injection automatic and passive

**LOCA (Secondary):**

**LOCA (Interfacing)**
- Easy S G isolation

**ATWS:**
- Strong negative temperature coefficient

**Primary Transients.**
- Greater primary inertia

**Secondary Transients:**
- Greater primary inertia

**Loss of Electric Sources:**
- Implementation of passive systems (battery power sources)

**Total Loss of heat sink:** not significant*

**Total Loss of S.G. feedwater:** not significant*

**Station Blackout:** not significant*

NB The possibility for easy natural convection in the primary and secondary circuits is a favourable common factor for all the abnormal situations

* The passive DHR and the large P C inertia, contribute to make these three situation not significant for the short term

TABLE 5.2.4: DESIGN FEATURES FOR MITIGATION LEVEL

| Safety Functions | Systems (Cf. Tab.5.2.1) | Passive/Active | Design Features/Remarks |
|---|---|---|---|
| **Design Basis**<br>Fission product containment | Clad<br>Primary Circuit<br>Containment | Passive<br>Passive<br>Passive | Fist barrier of defense<br>Second hold-up area<br>Natural removal mechanisms |
| Coolant inventory | Passive Safety Injection System (PSIS) | Passive | Fourteen times coolant system inventory |
| Decay heat removal | Passive Residual Heat Removal System (PRHR)<br>PSIS/Containment<br>Passive Containment Cooling System (PCCS) | Passive<br>Passive<br>Passive | Non-LOCA heat removal<br>LOCA heat removal<br>Ultimate heat sink (atmosphere) |
| Reactivity control | Control Rod<br>Boron Injection (PSIS) | Active<br>Passive | -<br>Passive core cooling system |
| Primary circuit pressure control | Pressurizer Safety Relief Valves<br>Automatic Depressurization System (ADS) | Passive<br>Passive | Provides overpressure protection<br>Reduces pressure for passive safety injection |
| **Severe Accident**<br>Containment temperature and pressure control | Containment/PCCS** | Passive | Ultimate DHR*<br>totally passive |
| Heat removal | PSIS/Containment/PCCS** | Passive | |
| Tightness control | Containment | Passive | Simplified containment geometry |
| Inflam. gas control | Ignitors | Passive | |
| Fission product containment | Flooded cont. | Passive | FP sparging within the water that quench the PC. |
| Corium management | | | Corium management: Containment floor geometry |
| Others | | | Meets URD and NRC requirements |

* The success of this action is conditioned by the good behaviour of the cooling through the containment (steel wall + PCCS)

** Self sufficiency: 72 hours (PCCS refeeding)

### 5.2.4. Design data questionnaire

## I. GENERAL INFORMATION

1. Design name - AP600
2. Designer/Supplier address - Westinghouse
3. Reactor type - PWR  Number of modules/per plant - 1
4. Gross thermal power (MW-th) per reactor - 1940 MWt
5. Net electrical output (MW-e) per reactor - 600 $MW_e$
6. Heat supply capacity (MW-th)

## II. BASIC DESIGN DESCRIPTION

### A. CORE AND REACTIVITY CONTROL

7. Fuel material - $UO_2$
8. Fuel inventory (tones of heavy metal) - 66.9 MTU
9. Average core power density (kW/liter) - 78.82
10. Average fuel power density (kW/kgU) - 28.89
11. Average linear power (KW/ft) - 4.10
12. Average discharge burnup (MWd/t) - 40,000 Mwd/Mt (Nominal)
13. Initial enrichment or enrichment range (Wt%) - 2.0-3.0%
14. Reload enrichment at the equilibrium (Wt%) - 3.55%
15. Refueling frequency (months) - 18 or 24 Months
16. Type of refueling (on/off power) - off power
17. Fraction of core withdrawn (%) - 33%
18. Moderator material and inventory
19. Active core height (m) - 3.658 m
20. Core diameter (m) - 3.361 m
21. Number of fuel assemblies - 145
22. Number of fuel rods per assembly - 264
23. Rod array in assembly
24. Clad material - Zircaloy
25. Clad thickness (mm) - 0.57
26. Number of control rods or assemblies - 24
27. Type - Rod cluster control
28. Additional shutdown systems
29. Control rod neutron absorber material - Ag-In-Cd
30. Soluble neutron absorber - Boric acid
31. Burnable poison material and form - WABA Wet annular burnable abs.
 - IFBA Integral fuel burnable abs.

### B. REACTOR COOLANT SYSTEM

**B1. Coolant**

32. Coolant medium and inventory - Borated water
33. Design coolant mass flow through core (kg/s) - $9.19 \times 10^3$Kg/s
34. Cooling mode (forced/natural) - Forced
35. Operating coolant pressure (bar) - 155 bar
36. Core inlet temperature (°C) - 276.1
37. Core outlet temperature (°C) - 312.4

**B2. Reactor pressure vessel**

38. Overall length of assembled vessel (m) - 11.59
39. Inside vessel/diameter (m/mm) - 3.988m
40. Average vessel thickness (mm) - 0.203m
41. Vessel material - carbon steel
42. Lining material - stainless steel
43. Design pressure (bar) - 2500 psia = 172.4 bar
44. Gross weight (tone/kg) - finished weight vessel + head 364.3 mtonne, 803,000lbs with studs

**B3. Steam generator**

45. Number of steam generators - 2
46. Type - Delta 75
47. Configuration (horizontal/vertical) - vertical

48. Tube material - Inconel 690 - TT
49. Shell material - Carbon steel
50. Heat transfer surface per steam generator ($m^2$) - 6,984.45 $m^2$
51. Thermal capacity per steam generator (MW) - 970 MWt
52. Feed water pressure (bar) - 1030 psig = 72.1 bar
53. Feed water temperature (°C) - 435°F = 223.9°C
54. Steam pressure (bar) - 777 psia = 53.6 bar
55. Steam temperature (°C) - 514.9°F - 268°C

**B4. Pressurizer**

56. Pressurizer total volume ($m^3$) - 36.82 $m^3$
57. Steam volume (full power/zero power, $m^3$) - 14.16 $m^3$

**B5. Main coolant pumps**

58. Number of cooling or recirculation pumps - 4
59. Type - Canned motor
60. Pump mass flow rate (kg/s) - 4.97 x $10^3$ Kg/s
61. Pump design rated head - 240 ft (7.173 bars)
62. Pump nominal power (kW) - 2240 KW, 7,173 bars
63. Mechanical inertia (kg $m^2$) - 5000 lb-$ft^2$, 210.0 kg $m^2$

## C. CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS)

64. Number of extraction lines - 1
65. Number of pumps - 2
66. Number of injection points - 1
67. Feed and bleed connections

## D. CONTAINMENT

68. Type - Free standing steel
69. Overall form (spherical/cyl.) - cylindrical
70. Structural material - steel
71. Liner material - steel
72. Simple/double wall - single
73. Dimensions (diameter, height) (m) - 39.6 m, 57.6 m
74. Design pressure (bar) - 3.16 Kg/$cm^2$
75. Design temperature (°C) - 137.8°C
76. Design leakage rate (% per day) - 0.12% per day

# III. <u>SAFETY RELATED SYSTEMS</u>

## A. DESIGN CONDITIONS

**A1. Fission product retention**

77. Containment spray system (Y/N) - No
    a. Duration (h)
    b. Flow rate ($m^3$/h)
    c. Mode of operation (active/passive)
    d. Safety graded (Y/N)
78. F.P. sparging (Y/N)
79. Containment tightness control (Y/N) - Yes
80. Leakage recovery (Y/N) - (not credited)
81. Guard vessel (Y/N) - No

**A2. Reactivity control**

82. Absorber injection system (Y/N) - Yes
    a. Absorber material - boron
    b. Mode of operation (active/passive) - passive
    c. Redundancy - Yes
    d. Safety graded - Yes
83. Control rods (Y/N) - Yes
    a. Maximum control rod worth (pcm) - 4% scram worth
    b. Mode of operation (active/passive) - passive
    c. Redundancy - 1 set scram rods N
    d. Safety graded - Yes

**A3. Decay heat removal**

*A3-1 Primary side*

84 Water injection
- a Actuation mode (manual/automatic) - automatic
- b Injection pressure level (bar) - 2500 PSIA
- c Flow rate (kg/s) - proprietary
- d Mode of operation (active/passive) - passive
- e Redundancy - Yes
- f Safety graded (Y/N) - Yes

85 Water recirculation and heat removal
- a Intermediate heat sink (or heat exchanger) - Yes
  - b Mode of operation (active/passive) - passive
- c Redundancy - Yes
- d Self sufficiency (h) - NATL CIRC
- e Safety graded - Yes

*A3-2 Secondary side*

86 Feed water
- a Actuation mode (manual/automatic) - automatic
- b Flow rate (kg/s) - Proprietary
- c Mode of operation (active/passive) - active
- d Redundancy - Yes
- e Self sufficiency (h) - Pumps
- f Safety graded - No

87 Water recirculation and heat removal
- a Ultimate heat sink (cold source) - Atmosphere
- b Mode of operation (active/passive) - passive
- c Redundancy - Yes
- d Self sufficiency (h) - 72 hours
- e Safety graded - Yes

*A3-3 Primary pressure control*

88 Implemented system (Name) -
Automatic Depressurization System
- a Actuation mode (manual/automatic) - Automatic
- b Side location (primary/secondary circuit)- Primary
- c Maximum depressurization rate (bar/s) - 2200 psi/40 min = 0 0632 bar/sec
- d Safety graded - Yes

## B. SEVERE ACCIDENT CONDITIONS*

**B.1 Fission products retention**

89 Containment spray system (Y/N) - No

90 F P Sparging (Y/N) - No

91 Containment tightness control (Y/N) - Yes

92 Leakage recovery (Y/N) - Yes (hold up)

93 Risk of recriticality (Y/N) - No

**B.2 Recriticality control**

94 Encountered design feature
- a Mode of operation (A/P) - Passive
- b Safety graded - Yes

**B.3 Debris confining and cooling**

95 Core debris configuration (core catcher) - RV or containment floor

96 Debris cooling system (name) - PCCS + containment water
- a Mode of operation (A/P) - passive
- b Self sufficiency - indefinite
- c Safety graded (Y/N) - Yes

**B.4 Long term containment heat removal**

97 Implemented system - PCCS
- a Mode of operation (A/P) - P
- b Self sufficiency (h) - indefinite
- c Safety graded (Y/N) - Yes

---

* All systems must be qualified to operate under the accident

98. Intermediate heat sink - No
  a. Self sufficiency (h)
  b. Safety graded (Y/N)

99. External coolant recirculation - No
  a. Implemented components
  b. Mode of operation (A/P)
  c. Self sufficiency (h)
  d. Safety graded (Y/N)

100. Ultimate heat sink
  a. Self sufficiency (h) - indefinite
  b. Safety graded (Y/N) - Yes

**B.5 Combustible gas control**

101. Covered range of gas mixture concentration - 0.13%

102. Modes for the combustible gas control
  a. Containment inertation - No
  b. Gas burning - Yes
  c. Gas recombining - Yes
  d. Others - No

**B.6 Containment pressure control**

103. Filtered vented containment (Y/N) - No
  a. Implemented system
  b. Mode of operation (A/P)
  c. Safety graded

104. Pressure suppression system (Y/N) - Yes
  a. Implemented system - PCCS
  b. Mode of operation - Passive
  c. Safety graded (Y/N) - Yes

## C. SAFETY RELATED I&C SYSTEM

Automatic load following (Y/N) - Yes
* range (% power) - 100-50-100%
* maximum rate (%/min) - 50%/2HR

Load rejection without reactor trip (Y/N) - Yes
Full Cathode Ray Tubes (CRT) display (Y/N) - Yes
Automated start-up procedures (Y/N)
Automated normal shutdown procedures(Y/N)
Automated off normal shutdown procedures
- Interactive Procedures, but automatic plant control

Use of field buses and smart sensors(Y/N) - No
Expert systems or artificial intelligence advisors (Y/N) - No
Protection system backup (Y/N) - Diverse actuation system NonIE

## D. EMERGENCY POWER SUPPLY SYSTEM

105. Type (diesel, gas, grid connection) - diesel/batteries
106. Number of trains - 4 divisions 11 batteries

## E. AC/DC SUPPLY SYSTEM

107. Type (rectifier, converter, battery) - Class 1 uninterruptable power supply
108. Estimated time reserve (hr) - class 1E, 24h adn 72h
Non class 1E 2hr

# IV. CONVENTIONAL THERMAL CYCLE

## A. TURBINE SYSTEM

109. Type - Tandem-compound, 4 flow, 47 in. last-stage blade
110. Overall length (m) - 47.5
111. Width (m) - 8.6
112. Number of turbines/reactor - 1/1
113. Number of turbine sections per unit (e.g. HP/LP/LP) - 1HP, 2LP
114. Speed (rpm) - 1800 rpm

## B. STEAM CHARACTERISTICS

115. H.P. inlet pressure - 777 psia = 53.6 bar
116. H.P. inlet temperature - 268.3 °C

117. H.P. inlet flowrate - 8.44 X $10^6$lb/hr = 3.83 x $10^6$ kg/hr
118. L.P. inlet pressure - 164 psia = 11.3 bar
119. L.P. inlet temperature - 255.6°C
120. L.P. inlet flowrate (per section) - 2.87 x $10^6$lb/hr = 1.302 x $10^6$ kg/hr

**C. GENERATOR**

121. Type (3-phase synchronous, DC) - 3-phase synchronous
122. Apparent power (MVA) - 780
123. Active power (MW)
124. Frequency (hz) - 60
125. Output voltage (kV) - 22
126. Total generator mass (t)
127. Overall length - 12m
128. Stator housing outside diameter

**D. CONDENSER**

129. Number of tubes
130. Heat transfer area
131. Flowrate ($m^3$/s) - 386
132. Pressure (m bar) - 85
133. Temperature (°C) - 42

**E. CONDENSATE PUMPS**

134. Number - 3
135. Flowrate - 5800 gpm, 365.91 liter/sec
136. Developed head - 875 ft, 26.15 bar
137. Temperature - 42°C
138. Pump speed

### 5.2.5. Project status

#### *5.2.5.1. Entities involved*

Under the sponsorship of the US Department of Energy and the Electric Power Research Institute, Westinghouse has designed the AP600 Nuclear Power Plant. The design team apart of Westinghouse includes Bechtel and Burns & Roe Co as architect engineers.

Avondale Industries - module design.
CBI Services structural steel - Containment vessel design.
M-K Ferguson Co. - Constructability, schedule and cost estimation.
Southern Electric Co. - Turbine Island buildings and systems.

ENEA Energy Research Center of CASACCIA to carry out tests for the blow-down tests for the ADS system. SIET, SPES Facility, Italy and Oregon State University are conducting the two large passive system design verification tests.

Westinghouse Energy Systems conducts the passive containment cooling test analysis program.

### 5.2.5.2. *Design status*

The AP600 development program is set to obtain final design approval from the NRC by January 1996 and certification by 1997. The standard safety analysis report have been submitted to NRC for review on 26 of June 1992. Systems and equipment unique to the AP600 design have been supported by R&D results especially for the passive features employed in the basic design (see 6.5.3).

Westinghouse is currently working on the detailed design for, FOAKE (First Of A Kind Engineering) project. Detailed design for a standard AP600 is expected to be ready for construction within the next 3 years. This would make the AP600 available for construction as a standard plant on commercial basis with no further development needed before the end of this century.

### 5.2.5.3. *Research and development work*

A. Executed R&D work

In the conceptual design phase a research and development program has been executed to confirm the passive safety features of the AP600 system. This program included tests for the PCCS, 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 passive containment cooling system, the passive safety injection system, and components design verification program. The program also looked into design availability for commercialization. The main test programs and their results were as follows:

1. PCCS Program

- Heated plated test

The test results have confirmed the results obtained from simulation and calculation of the containment for wetted and dry conditions. The experiments further confirmed the ease with which a water film could be attained. The film has also been confirmed to be stable with regard to high air velocities.

- Integrated PCCS test

A steel pressure vessel 3ft in diameter and 24ft high was used to simulate the entire PCCS system. The test has demonstrated the capability of the design to effectively remove heat from the containment even with a dry surface.

- Large scale integrated PCCS test

A ⅛ - scale containment was constructed. The experimental model took into consideration the internal containment structures and volumes also. This was mainly done to establish internal circulation patterns for air and steam mixture. Numerous instrumentations were used to obtain temperature, pressure velocities, evaporation and condensation rates.

- Air flow path resistance

A 14° section of the containment was built with 1:6 scaling factor to quantitatively assess the air flow resistance. This resulted in modification of the air inlet and exhaust to be a rounded entrance and a rounded air baffle supports.

- Water distribution test

A ⅛th section full sized model was used to test the water film formation for the expected water flow range. The results are being used to carry out the detailed design for the water delivery and distribution system.

- Wind tunnel test

Test carried out at the wind tunnel facility of the University of Western Ontario indicate that wind speed and direction would have no significant effect on the natural convection draft within the containment annulus.

2. PSIS program

- Passive residual heat removal heat exchanger

Thermal performance of the heat exchanger, and the thermal mixing behaviour of the IRWST have been verified. The results were used to establish the model for heat transfer characteristics that were used for the transient analysis in the SSAR submitted to NRC.

- Automatic depressurization system

The Automatic Depressurization System Test is a full sized simulation of one of the two AP600 depressurization system flowpaths from the pressurizer which will duplicate or conservatively bound the operating conditions of the AP600 ADS valves, sparger, and quench tank. The ADS test is being performed at ENEA's VAPORE Test Facility in Casaccia, Italy, and will be conducted in two parts, Phase A and Phase B.

- Check valve test

Tests to verify the capability of PSIS system check valves to open at low pressure differential conditions have been carried out. Satisfactory results have been obtained.

3. Component design verification program

- High inertia RCP rotor test

A full scale canned motor pump was assembled using a high inertia shaft. The shaft used a depleted uranium disk encased within the runner. The pump was tested for the full operational speed. The test results showed no significant vibration and was satisfactory.

- In-core instrumentation system test

A full scale control rod drive was installed in a test stand with a simulated fixed detector to verify any interference from the CRDM. The test results showed that the in-core instrumentation system would suffer no interference from the electromagnetic field of the CRDM.

- Reactor internal airflow test

Flow pattern within the core was investigated in a 1/9 scale model made of clear plastic. Tests were carried out at different flow rates. The test was mainly carried out since bottom oriented instrumentation currently used in current plants were removed from the AP600 design.
D work

B. Ongoing and planned R & D work

- Check valve test

Actual plant operation conditions would be used for in-site tests.

- Long-term cooling tests

The long term cooling test being conducted at Oregon State University is designed to experimentally investigate the integral system and long term cooling behaviour of the Westinghouse AP600 nuclear steam supply system. The OSU test program is a ¼ scale. lower pressure ($\sim$ 78 bar) program which includes all the AP600 safety systems and faithfully represents the AP600 geometry.

- Reactor coolant pump test

An air flow test has been performed on the SG channel head and on the RCP. Further tests on the RCP to establish the water flow hydraulic performance are planned for 1994.

- Core makeup tank test

A $\sim$1/100 volume scale CMT is under construction. The facility is designed to simulate CMT operating condition over a wide range and to verify the gravity drain mechanisms. The facility would also test the operation of instrumentation for CMT level indicator.

- Full pressure integral system test

A full-height, full-pressure systems test at the SPES facility in Italy is being conducted. This test is at operating temperature and provides full simulation of AP600 passive core cooling safety features.

C. R & D work needed

All known R&D work required has been included in the tests completed, or is part of the ongoing and planned test program. However, until the licensing process has been completed unknown issues may arise requiring additional tests.

### *5.2.5.4. Licensing process*

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 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. The plan intention is to have a design certification by the end of 1995. This licensing process consist of six parallel activities. The first part of which 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 had been submitted to NRC on 26 June 1992. The second part consists of the inspections, tests, analysis and acceptance criteria (ITAAC). Compliance with the ITAAC would demonstrate that public safety interests and investment are sound. The third part is submission of the Probabilistic Risk Assessment (PRA). The ITAAC and PRA for the AP600 have been completed and submitted to NRC. The NRC reviewal of the SSAR results in a report (NRC safety evaluation report). NRC final approval of this report results in a final design approval. The design certification is a rule making processing that leads to the AP600 design certification. Such certification would be valid for 15 years. A certification program summary has been submitted by the design team. The new procedure allows for more interaction between the design team and their NRC counterpart. The next and final milestone (in the US) is the application 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.

## 5.2.6. Economics

The capital, fuel and O&M cost assumptions for an $N^{th}$ of a kind AP600 reactor, built in the US in single and twin unit configurations and competing fossil technologies are shown in Table 5.2.5. Capacity factors of 80% and levelization over 30 years are assumed for each technology. Levelized generation costs are shown for each technology in Table. 5.2.5. The AP600 shows levelized costs of 3.94 c/Kwh for the twin reactor and 4.45 c/Kwh for the single unit.

The AP600 Overnight Construction Cost is estimated to be $899.6 million. Major item of the overnight cost are given in Table 5.2.6.

## TABLE 5.2.5: US AP600 ECONOMICS, 1993 $

| | Single Unit | Twin Unit | 500 MW Pulv | 2 X 300 MW Combined |
|---|---|---|---|---|
| | AP 600 | AP 600 | Coal | Cycle |
| Vendor overnight cost, $/kW | 1500 | 1364 | 1189 | 500 |
| Owner's cost, $/kW | 204 | 162 | 118 | 50 |
| Total overnight cost, $/kW | 1704 | 1526 | 1307 | 550 |
| Fixed O&M cost, $/kW yr | 65 7 | 52 1 | 31 0 | 8 8 |
| Var O&M cost, mills/kwh | 0 5 | 0 4 | 4 3 | 0 6 |
| Fuel cost, C/million BTU | 48 8 | 48 8 | 146 | 238 |
| Real fuel cost escalation, %/yr | 0 | 0 | 1 1 | 3 0 |
| Heat Rate, BTU/kWh | 10,400 | 10,400 | 9,700 | 7,000 |
| Interest rate, %/yr (Inflation free, before tax) | 6 2 | 6 2 | 6 2 | 6 2 |
| 30-year Levelized generation cost, C/kWh | 4 45 | 3 94 | 4 96 | 4 96 |
| Capital cost | 2 85 | 2 55 | 2 13 | 0 88 |
| Fuel cost | 0 51 | 0 51 | 1 96 | 3 89 |
| O&M cost | 0 99 | 0 78 | 0 87 | 0 19 |
| Decommissioning cost | 0 10 | 0 10 | ---- | ---- |

## TABLE 5.2.6.: MAJOR ITEMS OF AP600 OVERNIGHT CONSTRUCTION COST

| Item | $ Millions, 1993$ |
|---|---|
| Structures | 122.9 |
| Reactor Equipment | 216.8 |
| Turbine Equipment | 148.3 |
| Electrical Equipment | 68.6 |
| Miscellaneous Equipment | 19.1 |
| Main Condenser and Heat Rejection | 31.6 |
| Direct Costs | 607.3 |
| Indirect Costs | 192.6 |
| Total Base Construction Cost | 799.9 |
| Contingency | 99.7 |
| Total Overnight Cost | 899.6 |
| $/Kw | $1500/Kw |

## 5.3. SBWR REACTOR SYSTEM DESCRIPTION AND DEVELOPMENT STATUS

### 5.3.1. Basic objectives and features

The design of the Simplified Boiling Water Reactor (SBWR) represents a complete design for a nominal 600 MWe power plant. The rated thermal output of the reactor core is 2000 MWt, and the rated NSSS thermal output is 1996.1 MWth. The SBWR incorporates 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 SBWR Design objectives include: 60 year plant life from full power operating license date, 87% or greater plant availability, 24 month refuelling interval with 17 days or less critical path activities, personnel radiation exposure limit of 100 man-rems/year, providing safety-related functions primarily through passive means, core damage frequency of less than $10^{-5}$/reactor year, limiting significant release frequency to $10^{-6}$/reactor year, and requiring no operator action for 72 hours following a design basis accident to maintain safe stable condition except for operator actions necessary to maintain main control room habitability.

*Principal design criteria*

The principal design criteria governing a SBWR Standard Plant are presented in two ways. First, the criteria are classified as applicable to either a power generation function or a safety-related function. Second, they are grouped according to system. Some significant design criteria are listed below.

General power generation design criteria

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

Heat removal systems are provided with sufficient capacity and operational 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 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 so 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 assure 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 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 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 the 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.

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 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 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 inaccessible, it is possible to bring the reactor from power range operation to cold shutdown conditions by utilizing alternative controls and equipment that are available outside the control room.

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

### 5.3.2. Design description

#### *5.3.2.1. Nuclear steam supply*

Reactor pressure vessel

The SBWR reactor pressure vessel (RPV) assembly consists of the pressure vessel and its appurtenances, supports and insulation, and the reactor internals enclosed by the vessel (excluding the core, in-core nuclear instrumentation, neutron sources, control rods, and control rod drives).

The RPV, together with its internals, provides guidance and support for the fine-motion control rod drives (FMCRDs). Certain of the reactor internals distribute sodium pentaborate solution delivered by the Standby Liquid Control (SLC) System when necessary to achieve core subcriticality via means other than insertion of control rods.

The RPV is a vertical, cylindrical pressure vessel of welded construction, with a removable top head, and head flanges, seals and bolting. The vessel also includes penetrations, nozzles, shroud support, and venturi shaped flow restrictors in the steam outlet nozzles.

The reactor vessel is 6 meters (236 in.) in diameter minimum, with a wall thickness of about 158 mm (6.2 in.) with cladding, and a height of 24.5 m (80.4 ft.) from the inside of the bottom head (elevation zero) to the inside of the top head. The bottom of the active fuel location is 3750 mm (147.6 in.) from elevation zero and the active core is 2743 mm (108 in.) high. The overall RPV height of approximately 25 m (82 feet) permits natural circulation driving forces to produce abundant core coolant flow. An increased internal flow-path length relative to prior GE\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 reserve of water above the core, which translates directly into a much longer period of time (compared to prior BWRs) before core uncovery can 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 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 the normal heat sink which eventually leads to actuation of the safety-relief valves.

Reactor core design

The SBWR 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 manoeuverability. Since the SBWR is a natural circulation reactor, the reactivity control is maintained by movement of control blades and burnable poison being introduced into 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 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. Each HCU is designed to scram up to two FMCRDs. 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 177 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/sec 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 89 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.

Nuclear boiler system

The primary functions of the Nuclear Boiler System (NBS) are: (1) to deliver steam from the RPV to the turbine main steam system (TMSS), (2) to deliver feedwater from the condensate and feedwater system (C& FS) to the RPV, (3) to provide overpressure protection of the RCPB, (4) to provide automatic depressurization of the RPV in the event of a 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 TMSS; the feedwater lines (FWLs) to direct feedwater from the C&FS to the RPV; the RPV instrumentation to monitor the conditions within the RPV over the full range of reactor power operation.

The main steam line flow limiter, a flow restricting venturi built into the RPV MSL nozzle of each of the two 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 two MSLs. On each MSL there is one MSIV in the containment and one MSIV outside the containment. The MSIVs are Y-pattern globe valves. The Y-pattern configuration permits the inlet and outlet flow passages to be streamlined to minimize pressure drop during normal steam flow.

The nuclear pressure relief system consists of safety/relief valves (SRVs) located on the MSLs between the RPV and the inboard MSIV. There are four SRVs per MSL. The SRVs 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 six 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) injecting flow to replenish core coolant to maintain core temperature below design limits in the event of a LOCA. It also maintains the reactor depressurized for continued operation of GDCS after an accident without need for 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 enhancing the passive resupply of coolant by the GDCS.

The use of a combination of SRVs and DPVs to accomplish the ADS function provides an improvement in ADS reliability against hypothetical common-mode failures of otherwise non-diverse ADS components. It also minimizes components and maintenance as compared to using only SRVs or only DPVs for this function.

Reactor servicing equipment

The reactor building fuel handling floor is serviced with a fuel handling platform, refuelling machine, and an auxiliary platform.

The refuelling machine is a gantry-type crane which spans the reactor vessel cavity and fuel and storage pools to handle fuel and perform other ancillary tasks. It is equipped with a traversing trolley on which is mounted a telescoping tabular mast and integral fuel grapple. An auxiliary hoist is also provided. The machine 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. The fuel handling platform is only used for fuel servicing and transporting tasks. It is equipped with a trolley and telescoping grapple and is manually operated. 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.

New fuel storage racks are aluminum and are constructed for floor mounting. For dry vault storage the racks are loaded from the top, while those in pools are side loaded. The storage vault capacity is 19% of core load while the storage pool is 39% core load.

Spent fuel storage racks are of stainless steel laminate construction with neutron absorbing material. This ensures that a full array (285% of full core) or loaded spent fuel will remain subcritical by 5% of $\Delta k$, under all conditions.

The thermal-hydraulic design of the rack provides sufficient natural convection cooling flow to remove 19,929 W/bundle (68,000 Btu/hr/bundle) of decay heat.

### *5.3.2.2. Balance of plant systems*

Main turbine

The main turbine is a tandem compound, two flow, 52 inch (1320 mm) last stage bucket with one high pressure (HP) turbine and one low pressure (LP) turbine. The steam passes through an in-line high velocity moisture separator (HVS) prior to entering the LP turbine. Steam exhausted from the LP turbine is condensed and degassed in the condenser. The turbine uses steam at an atmospheric pressure of 6.79 MPa (985 psia) from the reactor and rotates at 1800 RPM. Steam is bled off from each turbine and is used to heat the feedwater. The steam and power conversion system is designed to operate at 105% of maximum guaranteed turbine throttle flow for transients and short-term loading conditions.

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 and prevents the escape of radioactive steam from the turbine shaft/casing penetrations and valve stems and prevents air in-leakage through subatmospheric turbine glands. 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 40% of the turbine generator rated load without reactor trip or operation of a SRV. 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 (TBS).

The main condenser, which does not serve or support any safety function and has no safety design basis, is a single-shell type deaerating unit with its shell located directly beneath the low pressure turbine. The 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.

Noncondensable gases are removed from the power cycle by the Main Condenser Evacuation System (MCES). The MCES removes power cycle noncondensable 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 start-up. 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.

Condensate and feedwater system

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 condensate pumps take the deaerated condensate from the condenser hotwell and deliver it through the condensate demineralizer and through a string of four low pressure feedwater heaters to the reactor feed pump suction. The reactor feed pumps discharge through two high pressure feedwater heaters to the reactor. The condensate portion of the C&FS has three motor-driven, constant speed centrifugal pumps, each rated at 33% to 60% of total system-rated flow.

Radioactive waste management system

The radioactive waste management system consists of liquid, solid, detergent, and laundry waste management, and mobile systems. Liquid waste processing is done on a batch basis. Equipment drains and other low-conductivity water 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. Laundry drain wastes and other 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. Connection 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 sludges 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 controlled-access enclosed area for temporary storage.

Connection are provided for mobile processing systems that could be brought in to augment the installed waste processing capability.

### *5.3.2.3. Control and instrumentation, and electrical systems*

The SBWR Control and instrument systems provide manual and automatic means to control plant operations and initiate protective actions should plant upset conditions occur. The SBWR 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-tolerent 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, reduces the construction cost, and facilitates automation of plant operations.

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 soft-ware-controlled logic processors.

The Process Computer System provides performance monitoring and control, and power generator control subsystem functions to provide efficient plant operation and automation.

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 shut down 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.

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 for reactor scram on rising excessive neutron flux or too short a period for flux generation.

Main control room panels

The main control room (MCR) panel is comprised 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 main control room 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, refuelling, safe shutdown, and maintaining the plant in a safe shutdown condition. Human factors engineering principles have been incorporated into all aspects of the SBWR 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.

Station electrical system

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.

Six individual voltage regulating transformers supply 120 Vac non-safety-related control and instrument power.

Direct current power supply

The Class 1E dc power supply provides power to the Class 1E vital ac buses through inverters, and to 125 Vdc loads required for safe shutdown. Each of the four divisions of class 1E dc power is separate and independent. Each division has a 125 Vdc battery and a battery charger fed from its divisional 480 Vac Motor Control Center (MCC). This system is designed so that no single failure in any division of the 125 Vdc system will prevent safe shutdown of the plant.

During a total loss of off-site power, the Class 1E system is powered automatically from two non-Class 1E standby diesel generators. If these are not available, each division of Class 1E isolates itself from the non-Class 1E system, and power to safety-related loads is provided uninterrupted by the Class 1E batteries. The batteries are sized to power safety-related loads for a 72 hour period.

Non-Class 1E dc power is supplied through four non-Class 1E 480 Vac MCCs in the same manner as the class 1E dc power. 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 125 Vdc bus through a battery charger. A 125 Vdc station battery provides backup to the supply from the battery charger.

The second MCC in each group provides power to a 250 Vdc bus through a battery charger. A 250 Vdc station battery provides backup to the supply from the battery charger.

Standby AC power supply

The non-safety-related Standby ac Power Supply consists of two diesel generators. Each diesel generator (DG) provides 6.9 kVac 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 safety- related loads and to non-safety-related investment protection loads. Other non-safety-related loads are not powered from the standby power source.

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.

Instrument and control power supply

The Instrument and Control Power Supply provides 120 Vac single phase power to instrument and control loads that do not require an uninterruptible power source.

*5.3.2.4. Safety considerations*

The basic SBWR safety design philosophy is to provide inherent margins (e.g., larger volumes and water inventory) to eliminate system challenges such as no SRV opening following reactor isolation. 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. 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 (active) 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 except for operator actions are necessary to maintain main control room (MCR) habitability. A description of some important passive safety-related systems follows:

The 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 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 ICS is initiated automatically on either a high reactor pressure, or MSIV closure, or a level 2 signal. To start an IC into operation, a condensate return valve is opened whereupon the standing condensate drains into the reactor and the steam-water interface in the IC tube handle moves downward below the lower headers to a point in the main condensate return line. The ICS can also be initiated manually by the operator from the MCR.

The IC/PCC 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/PCC pool inventory. The ICS passively removes sensible and core decay heat from the reactor (i.e., heat transfer from the IC tubes to the surrounding IC/PCC pool water is accomplished by natural convection, and no forced circulation equipment is required) when the normal heat removal system is unavailable.

Emergency core cooling - Gravity-driven cooling system (GDCS)

Emergency core cooling is provided by the Gravity-Driven cooling system (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 maintain the fuel cladding temperatures below temperature limits defined in 10CFR 50.46.

The GDCS injects water into the downcomer annulus region of the reactor after a LOCA and reactor vessel depressurization. It provides short-term gravity-driven water makeup from three separate water pools located within the upper drywell at an elevation above the active core region. The system also provides long-term post-LOCA makeup from the suppression pool to meet long-term core decay heat boil-off requirements. 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). During severe accidents the system floods the lower drywell region with water if the core melts through the RPV.

The GDCS is completely automatic in actuation and operation. A backup to automatic actuation is the ability to actuate by operator action.

The GDCS is composed of three identical divisions completely independent of each other both electrically and mechanically. A confirmed RPV level 1 signal will actuate the ADS to reduce RPV pressure. Simultaneously, 150-second short-term system timers, and 30-minute long-term system timers in the GDCS logic are started, which after time-out, actuate squib valves providing an open flow path from the respective water sources to the vessel.

Passive containment cooling system (PCCS)

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.

Each PCCS condenser is located in a subcompartment of the IC/PCCS pool, an all pool subcompartments communicate at their lower ends to enable full use of the collective water inventory, independent of the operational status of any given PCCS loop.

Each loop which is open to the containment, contains a drain line to the GDCS pool, and a vent discharge line the end of which is submerged in the pressure suppression pool.

The PCCS loops are driven by the pressure difference created between the containment drywell and the suppression pool during a LOCA and require no sensing, control, logic or power actuated devices for operation. Together with the pressure suppression containment system, the three PCCS condensers limit containment pressure to less than its design pressure for at least 72 hours after a LOCA without make-up to the IC/PCC pool.

The PCCS condensers are in a closed loop extension of the containment pressure boundary. Therefore, there are no containment isolation valves and they are always in "ready standby".

*5.3.2.5. Buildings and structures*

The principal plant structures are: Reactor building - houses all structures, components, equipment and systems providing safety-related functions. This includes the reactor containment, safety envelope, the refuelling area with spent fuel storage, the control room, and auxiliary equipment area. the reactor building structure is integrated with that of a stepped cylindrical reinforced concrete containment vessel (RCCV); the RCCV is located on a common basemat and surrounded by three concentric boxes: the inner box (safety envelope), the intermediate steel frame, and the outer box. The inner and outer boxes are made of reinforced concrete shear walls and the intermediate steel frame is made of structural steel framework with non-structural walls as required for radiation shielding, separation, etc. The building is partially embedded. All SBWR safety-related equipment is housed in the reactor building safety envelope, main steam tunnel, and pools located beneath the operating floor, with the non-safety-related systems and areas (including the MCR) surrounding this envelope. The reactor building is Seismic Category I structure.

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. The turbine building is a reinforced concrete structure up to the turbine operating deck, and steel frame and metal siding thereafter. It is built at grade level. 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.

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 level 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 the radwaste building below grade is designed to the requirements of Regulatory Guide 1.143, and the balance of the structure is Non-Seismic Category NS.

Electrical building - houses the two non-safety-related standby diesel generators and their associated auxiliary equipment, and the solid-state adjustable speed drive units powering the feedwater pump motors and others powering the Reactor Water Cleanup/Shutdown Cooling System pumps.

### 5.3.3. Safety concepts

The safety concepts of the SBWR are outlined by the following tables.

TABLE 5.3.1: MAIN SAFETY RELATED SYSTEMS IN THE SBWR CONCEPT

| Name | Safety graded | Main characteristics |
|---|---|---|
| Reactor pressure vessel system | x | Reactor pressure vessel |
| Nuclear boiler system | x | 8 safety relief valves, 6 depressurization valves |
| Isolation condenser system | x | Passive high pressure decay heat removal (3 independent heat exchanger loops) |
| Control rod drive system | x | 177 fine motion control rod drives |
| Standby liquid control system | x | Neutron absorbing fluid injection |
| Gravity driven cooling system | x | Passive emergency core cooling (3 independent divisions) |
| Flammability control system | x | Uses passive autocatalytic recombiners to recombine oxygen and hydrogen in the nitrogen-inerted containment |
| Passive containment cooling system | x | Passive containment heat removal (3 independent heat exchangers) |

TABLE 5.3.2: MAIN ACCIDENT INITIATORS FOR SBWR

| | | |
|---|---|---|
| LOCA | - | GDCS injection line break |
| | - | Main steam line break |
| | - | Bottom head drainline break |
| | - | DPV stub tube break |
| Transients | - | Coolant temperature decrease |
| | - | Reactor pressure increase |
| | - | Loss of preferred power |
| | - | Loss of feedwater flow |
| | - | Rod withdrawal error/subcritical/low power |
| | - | Rod withdrawal error at power |
| Station blackout | - | Plant is designed such that this event probability is less than $10^{-3}$/yr. |
| ATWS | - | Events: Limiting (e.g., MSIV valve closure); Moderate Impact (e.g., loss of condenser vacuum); Minimum Impact (inadvertent opening of all turbine bypass valves) |

TABLE 5.3.3A: DESIGN FEATURES FOR PREVENTION AND PROTECTION LEVELS THAT REDUCE (R), SUPPRESS (S) THE INITIATORS FREQUENCY OR LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES (L)

| |
|---|
| **Prevention level:** |
| **LOCA:** Design of lines to high quality engineering codes and standards, and to seismic and environmental requirements; Stringent water quality and material specification to limit corrosion; Elimination of longitudinal welds; reduction in number of welds; Inspection and maintenance. |
| **Transients:** - coolant temperature decrease: Caused by (1) steam extraction line to heater is closed, or (2) feedwater is bypassed around heater. SBWR is designed such that no single operator error of equipment failure shall cause a loss of more than 55.6°C feedwater heating.<br>- Reactor pressure increase: Several events may cause this e.g., inadvertent closure of one turbine control valve, pressure regulator downscale failure, generator load rejection, turbine trip MSIV closure, loss of condenser vacuum, loss of non-emergency AC power to station auxiliaries, loss of feedwater etc. All these have been analysed. Features are included in the instrumentation and control systems or redundancies to maintain reactor pressure through a combination of component automatic responses or operator actions, depending on the identified cause.<br>- Loss of preferred power: When this happens, Standby AC Power Supply (two diesel generators) and DC Power Supply (battery) will supply power.<br>- Rod withdrawal error/subcritical/low power: the reactor control and instrumentation system (RCIS) and automatic power regulator (APR) systems prevent this event from occurring.<br>- Rod withdrawal at power may be caused by an operator procedural error, causing a malfunction of a control rod(s) withdrawal logic or a automated rod withdrawal logic malfunction. The automated thermal limit monitor system (ATLM) performs the rod block monitoring system. |
| **Station blackout:** Startup shall be possible when AC power is restored (no equipment damage, or inspection required) within two hours; or capability exists for safe reactor shutdown and maintaining shutdown cooling under a postulated loss of all AC power for more than 72 hours. |
| **ATWS:** The following are provided for prevention/mitigation: Alternate Rod Insertion (ARI) system that utilizes sensors and logic, independent of Reactor Protection System (RPS); Electrical insertion of FMCRDs with sensors and logic independent of RPS; Automatic feedwater runback under conditions indicative of an ATWS; Automatic initiation of SLCS under conditions indicative of an ATWS. |

TABLE 5.3.3B: DESIGN FEATURES FOR PREVENTION AND PROTECTION LEVELS THAT REDUCE (R), SUPPRESS (S) THE INITIATORS FREQUENCY OR LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES (L)

| **Protection level:** |
|---|
| **LOCA:** Automatic initiation of containment isolation and passive systems actuation (ADS followed by GDCS; and PCCS) |
| Transients: - Coolant temperature decrease: Caused by (1) steam extraction line to heater is closed, or (2) feedwater is bypassed around heater. SBWR is designed such that no single operator error or equipment failure shall cause a loss of more than 55.6°C feedwater heating.<br>- Reactor pressure increase: Several events may cause this e.g., inadvertent closure of one turbine control valve, pressure regulator downscale failure, generator load rejection, turbine trip MSIV closure, loss of condenser vacuum, loss of non-emergency AC power to station auxiliaries, loss of feedwater etc. All these have been analysed. Features are included in the instrumentation and control system or redundancies to maintain reactor pressure through a combination of component automatic responses or operator actions, depending on the identified cause.<br>- Loss of preferred power: when this happens, Standby AC Power Supply (two diesel generators) and DC Power Supply (battery) will supply power.<br>- Rod withdrawal error/subcritical/low power: the Reactor control and instrumentation System (RCIS) and Automatic Power Regulator (APR) systems prevent this event from occurring.<br>- Rod withdrawal at power may be caused by an operator procedural error, causing a malfunction of a control rod(s) withdrawal logic or a automated rod withdrawal logic malfunction. The automated thermal limit monitor system (ATLM) performs the rod block monitoring system. |
| **Station blackout:** Startup shall be possible when AC power is restored (no equipment damage, or inspection required) within two hours; or capability exists for safe reactor shutdown and maintaining shutdown cooling under a postulated loss of all AC power for more than 72 hours. |
| **ATWS:** The following are provided for prevention/mitigation: Alternate Rod Insertion (ARI) system that utilizes sensors and logic, independent of Reactor Protection System (RPS); Electrical insertion of FMCRDs with sensors and logic independent of RPS; Automatic feedwater runback under conditions indicative of an ATWS; Automatic initiation of SLCS under conditions indicative of an ATWS. |

TABLE 5.3.4: DESIGN FEATURES FOR MITIGATION LEVEL OF SBWR

| Safety function | Systems | Passive/ Active | Design features/remarks |
|---|---|---|---|
| **Design basis:**<br>Fission product containment | Containment vessel | passive | Pressure suppression type with minimum penetrations |
| Coolant inventory (high pressure) | - Control rod drive system<br>- Feedwater system<br>- Isolation condenser pool | active<br>active<br>passive | 4 ICS loops |
| (low pressure) | - GDCS loops<br>- GDCS pool<br>- FAPCS (LPCI mode)<br>- Automatic depressurization system (ADS) | passive | - 2 GDCS independent loops (short and long term) The short term loop has 3 divisions drawing from 3 separate GDCS pools Long term loops draws from the suppression pool |
| Decay heat removal | - PCCS<br>- IC pool | passive | |
| Reactivity control | - FMCRD<br>- SLCS (ATWS) | active<br>passive | Control rod insertion<br>Boron injection |
| Reactor coolant pressure control | ADS | passive | SRVs and DPVs |
| **Severe accident:**<br>Containment temperature/pressure | PCCS<br>Containment vent (pressure control) | passive | |
| Heat removal | PCCS and ICS | passive | |
| Tightness control | Containment | passive | Minimum penetrations and leak-tight seal design of penetrations |
| Inflammable gas control | Flammability control system | passive | Passive auto-catalytic recombiners |
| Fission product containment | Containment | Passive | Leak tight design |
| Corium management | "Core catcher" arrangement | | Lower drywell design includes features to contain and cool debris and molten material |
| Others | | | |

### 5.3.4. Design data questionnaire

## I. GENERAL INFORMATION

1. Design name: Simplified Boiling Water Reactor
2. Designer/Supplier address: GE-Nuclear Energy
3. Reactor type: BWR Number of modules/per plant: 1
4. Gross thermal power (MW-th) per reactor: 2000
5. Net electrical output (MW-e) per reactor: 600
6. Heat supply capacity (MW-th)

## II. BASIC DESIGN DESCRIPTION

### A. CORE AND REACTIVITY CONTROL

7. Fuel material: $UO_2$
8. Fuel inventory (tonnes of heavy metal): 109 uranium metal
9. Average core power density (kW/liter): 41.5
10. Average fuel power density (kW/kgU): 22
11. Average linear power (W/m): 16.6 kW/m
12. Average discharge burnup (MWd/t)
13. Initial enrichment or enrichment range (Wt%): 3.95
14. Reload enrichment at the equilibrium (Wt%)
15. Refueling frequency (months): 24
16. Type of refueling (on/off power): off power
17. Fraction of core withdrawn (%)
18. Moderator material and inventory: Water, 463.4 $m^3$
19. Active core height (m): 2.743
20. Core diameter (m): 4.73
21. Number of fuel assemblies: 732
22. Number of fuel rods per assembly: 60
23. Rod array in assembly: 8x8
24. Clad material: Zircaloy-4
25. Clad thickness (mm): 0.8128
26. Number of control rods or assemblies: 177
27. Type: $B_4C$ compacted in st. steel tubes
28. Additional shutdown systems: --
29. Control rod neutron absorber material: $B_4C$
30. Soluble neutron absorber: Sodium Pentaborate
31. Burnable poison material and form: Gadolinia Urania fuel rods

### B. REACTOR COOLANT SYSTEM

**B1. Coolant**

32. Coolant medium and inventory: Water
33. Design coolant mass flow through core (kg/s): 7555
34. Cooling mode (forced/natural): Natural
35. Operating coolant pressure (bar): 71.7
36. Core inlet temperature (C): 278.5
37. Core outlet temperature (C): 287.7

**B2. Reactor pressure vessel/tube**

38. Overall length of assembled vessel/tube (m): 26.616
39. Inside vessel/diameter (m): 6000 mm
40. Average vessel thickness (mm): 154
41. Vessel material: Low alloy steel
42. Lining material: Ni-Cr-Fe Alloy clad
43. Design pressure (bar): 8.62
44. Gross weight (kg): 781,000

**B3. Steam generation**

45. Number of steam generators
46. Type
47. Configuration (horizontal/vertical) Not applicable
48. Tube material
49. Shell material
50. Heat transfer surface per steam generator ($m^2$)
51. Thermal capacity per steam generator (MW)
52. Feed water pressure (bar): 71.7
53. Feed water temperature (C): 215.6
54. Steam pressure (bar): 71.7
55. Steam temperature (C): 287.7

**B4. Pressurizer** N/A

56 Pressurizer total volume ($m^3$)

57 Steam volume (full power/zero power, $m^3$)

**B5. Main coolant pumps** N/A

58 Number of cooling or recirculation pumps

59 Type

60 Pump mass flow rate (kg/s)

61 Pump design rated head

62 Pump nominal power (kW)

63 Mechanical inertia (kg $m^2$)

## C. CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS) N/A

64 Number of extraction lines

65 Number of pumps

66 Number of injection points

67 Feed and bleed connections

## D. CONTAINMENT

68 Type Pressure suppression

69 Overall form (spherical/cyl ) Cylindrical

70 Structural material Reinf concrete

71 Liner material Steel

72 Simple wall Single

73 Dimensions (diameter, height) (m)
Upper drywell (D=31 5, 12 1, 1 2, H=5 95, 11 25, 7 25) m
Lower drywell (D=78 2, 10 9, 9 2, H=1 07, 1 55, 1 97, 9 38, 3 2) m
Drywell head (D=8 9, H=3 4) m
Drywell cylinder (D=8 9, H=2 77) m

74 Design pressure (bar) 3 79

75 Design temperature (C) 1 71

76 Design leakage rate (% per day) 0 5%

# III. <u>SAFETY RELATED SYSTEMS</u>

## A. DESIGN CONDITIONS

**A1. Fission product retention**

77 Containment spray system (Y/N) Yes
- a Duration (h) Variable
- b Flow rate ($m^3$/h) Variable
- c Mode of operation (active/passive) Active
- d Safety graded (Y/N) No

78 F P sparging (Y/N)

79 Containment tightness control (Y/N) Yes

80 Leakage recovery (Y/N) No

81 Guard vessel (Y/N) No

**A2. Reactivity control**

82 Absorber injection system (Y/N) Yes
- a Absorber material Sodium Pentaborate
- b Mode of operation (active/passive) Passive
- c Redundancy No
- d Safety graded Yes

83 Control rods (Y/N)
- a Maximum control rod worth (pcm)
- b Mode of operation (active/passive) Active
- c Redundancy Yes
- d Safety graded Yes

**A3. Decay heat removal**

*A3-1 Primary side*

84 Water injection
- a Actuation mode (manual/automatic)
- b Injection pressure level (bar)
- c Flow rate (kg/s)
- d Mode of operation (active/passive)
- e Redundancy
- f Safety graded (Y/N)

85 Water recirculation and heat removal

a Intermediate heat sink (or heat exchanger)
b Mode of operation (active/passive)
c Redundancy
d Self sufficiency (h)
e Safety graded

*A3-2 Secondary side*

86 Feed water
a Actuation mode (manual/automatic)
b Flow rate (kg/s) 1076
c Mode of operation (active/passive) Active
d Redundancy 3x33 to 60%
e Self sufficiency (h)
f Safety graded No

87 Water recirculation and heat removal
a Ultimate heat sink (cold source) Sea or lake
b Mode of operation (active/passive) Active
c Redundancy 3x50%
d Self sufficiency (h)
e Safety graded No

*A3-3 Primary pressure control*

88 Implemented system (Name) Automatic Depressurization System
a Actuation mode (manual/automatic)· Automatic
b Side location (primary/secondary circuit)
c Maximum depressurization rate (bar/s)
SRVs 1028 kg/s, DPVs-1436 kg/s, 8 771 MPag
d Safety graded Yes

## B. SEVERE ACCIDENT CONDITIONS*

**B.1 Fission products retention**

89 Containment spray system (Y/N) Yes
90 F P Sparging (Y/N)
91 Containment tightness control (Y/N) Yes
92 Leakage recovery (Y/N) No
93 Risk of recriticality (Y/N) No

**B.2 Recriticality control**

94 Encountered design feature
a Mode of operation (A/P)
b Safety graded

**B.3 Debris confining and cooling**

95 Core debris configuration Lower drywell
96 Debris cooling system (name) GDCS Deluge line
a Mode of operation (A/P) Passive
b Self sufficiency --
c Safety graded (Y/N) Yes

**B.4 Long term containment heat removal**

97 Implemented system PCCS
a Mode of operation (A/P) Passive
b Self sufficiency (h)
c Safety graded (Y/N), Yes
98 Intermediate heat sink
a Self sufficiency (h)
b Safety graded (Y/N)
99 External coolant recirculation
a Implemented components
b Mode of operation (A/P)
c Self sufficiency (h)
d Safety graded (Y/N)
100 Ultimate heat sink
a Self sufficiency (h)
b Safety graded (Y/N)

**B.5 Combustible gas control**

101 Covered range of gas mixture concentration
102 Modes for the combustible gas control
a Containment inertation Yes
b Gas burning
c Gas recombining Yes
d Others

* All systems must be qualified to operate under the accident conditions

**B.6 Containment pressure control**

103. Filtered vented containment (Y/N): Yes
   a. Implemented system: Vent line, Rupture disk, Filter
   b. Mode of operation (A/P): Passive
   c. Safety graded: Yes
104. Pressure suppression system (Y/N)
   a. Implemented system
   b. Mode of operation
   c. Safety graded (Y/N)

## C. SAFETY RELATED I&C SYSTEM

Automatic load following (Y/N): Yes
* range (% power): 100-50%
* maximum rate (%/min): ±5%

Load rejection without reactor trip (Y/N): Yes
Full Cathode Ray Tubes (CRT) display (Y/N): Yes
Automated start-up procedures (Y/N): Yes
Automated normal shutdown procedures (Y/N): Yes
Automated off normal shutdown procedures (Y/N): Yes
Use of field buses and smart sensors (Y/N): Yes
Expert systems or artificial intelligence advisors (Y/N): Yes
Protection system backup (Y/N): Yes

## D. EMERGENCY POWER SUPPLY SYSTEM

105. Type (diesel, gas, grid connection): Diesel
106. Number of trains: 2

## E. AC/DC SUPPLY SYSTEM

107. Type (rectifier, converter, battery): Battery/Inverter/Rectified
108. Estimated time reserve (hr): 72 hrs Battery for safety related loads

# IV. CONVENTIONAL THERMAL CYCLE

## A. TURBINE SYSTEM

109. Type: Tandem compound, 2 flow
110. Overall length (m)
111. Width (m)
112. Number of turbines/reactor: 1
113. Number of turbine sections per unit: 1HP/1LP
114. Speed (rpm): 1800

## B. STEAM CHARACTERISTICS

115. H.P. inlet pressure bar: 64.7
116. H.P. inlet temperature (C)
117. H.P. inlet flowrate (kg/s): 1081.2 kg/s
118. L.P. inlet pressure: 11.2
119. L.P. inlet temperature
120. L.P. inlet flowrate: 879.6

## C. GENERATOR

121. Type (3-phase synchronous, DC): 3Ph, 60Hz, 4 poles, Synchronous
122. Apparent power (MVA)
123. Active power (MW): 669.75 MW
124. Frequency (hz): 60
125. Output voltage (kV): 25
126. Total generator mass (t)
127. Overall length
128. Stator housing outside diameter

## D. CONDENSER

129. Number of tubes
130. Heat transfer area: 45,940 $m^2$
131. Flowrate ($m^3/s$): 29.36
132. Pressure (m/bar): 11.82 kPa
133. Temperature (°C): 33.3

## E. CONDENSATE PUMPS

134. Number: 3x (33-60%)
135. Flowrate
136. Developed head
137. Temperature
138. Pump speed

### 5.3.5. Project status

Entities involved

The SBWR program is sponsored by the US Department of Energy and the Electric Power research Institute. GE has the overall responsibility of SBWR engineering, licensing and programme management. Also, GE has formed an international team comprising some of the best available talent and experience in BWR technology.

The team members from the US are: Bechtel, Southern Company Services Inc., Burns and Roe, Foster Wheeler Energy Application, Inc. Massachusetts Institute of Technology and University of California, Berkeley provide technical reviews, consultation and perform testing.

International technical associates providing engineering and/or testing support include: EdF, HEW, EdF and VDEW (Germany); Ansaldo, ENEA, ENEL and Fial (Italy); hitachi, Toshiba and JAPC (Japan); CFE (Mexico), ECN, GKN, KEMA and NUCON (Netherlands); CIEMAT, ENSA, ENUSA, Technatom, UNESA and UTE (Spain); PSI (Switzerland); and Batan (Indonesia).

Design status

Current SBWR schedule targets licensing topical report approval by January 1996, final design approval by May 1998 and certification in November 1999.

Research and development work

R&D efforts are focused on analyses and tests which GE feels necessary for certification. The work is directed toward demonstration of the passive safety systems (both individually and in a simulated SBWR integrated systems environment), data acquisition, and validating the TRACG analysis directed toward modelling of SBWR transient phenomena and containment performance.

The Gravity Driven Cooling System (GDCS) Integral Systems Test (GIST) demonstrates SBWR integral response during the post-blowdown and GDCS operation phase 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 full-scale PANTHERS facility will demonstrate the thermal, hydraulic and mechanical performance of the Passive Containment Cooling System (PCCS). Extensive instrumentation has been added to facilitate qualification of the TRACG code and also monitor individual tube performance.

GIRAFFE, a 1:400 scale facility at Toshiba, representing major SBWR compartments in the nitrogen-inerted containment, proves PCCS performance under a range of SBWR conditions following a LOCA.

PANDA is a 1:25 scale, full height facility which simulates reactor vessel, wetwell, drywell, GDCS pool, IC and PCCS pool and heat exchangers. In addition, single tube condensation test programs were conducted at UC, Berkeley.

Executed R&D work

GIST, GIRAFFE and UC, Berkeley single tube test programmes have been completed. PANDA test facilities fabrication and construction is complete. PANTHERS/IC facility fabrication is in progress. Bulk of the TRACG qualification studies are complete, showing good agreement with test data. Confirmatory tests are planned in PANDA and PANTHERS facilities.

Licensing process

The overall goal of the SBWR programme is the Standard Design Certification by USNRC which requires meeting the requirements specified in 10CFR52, Early Site Permits; Standard

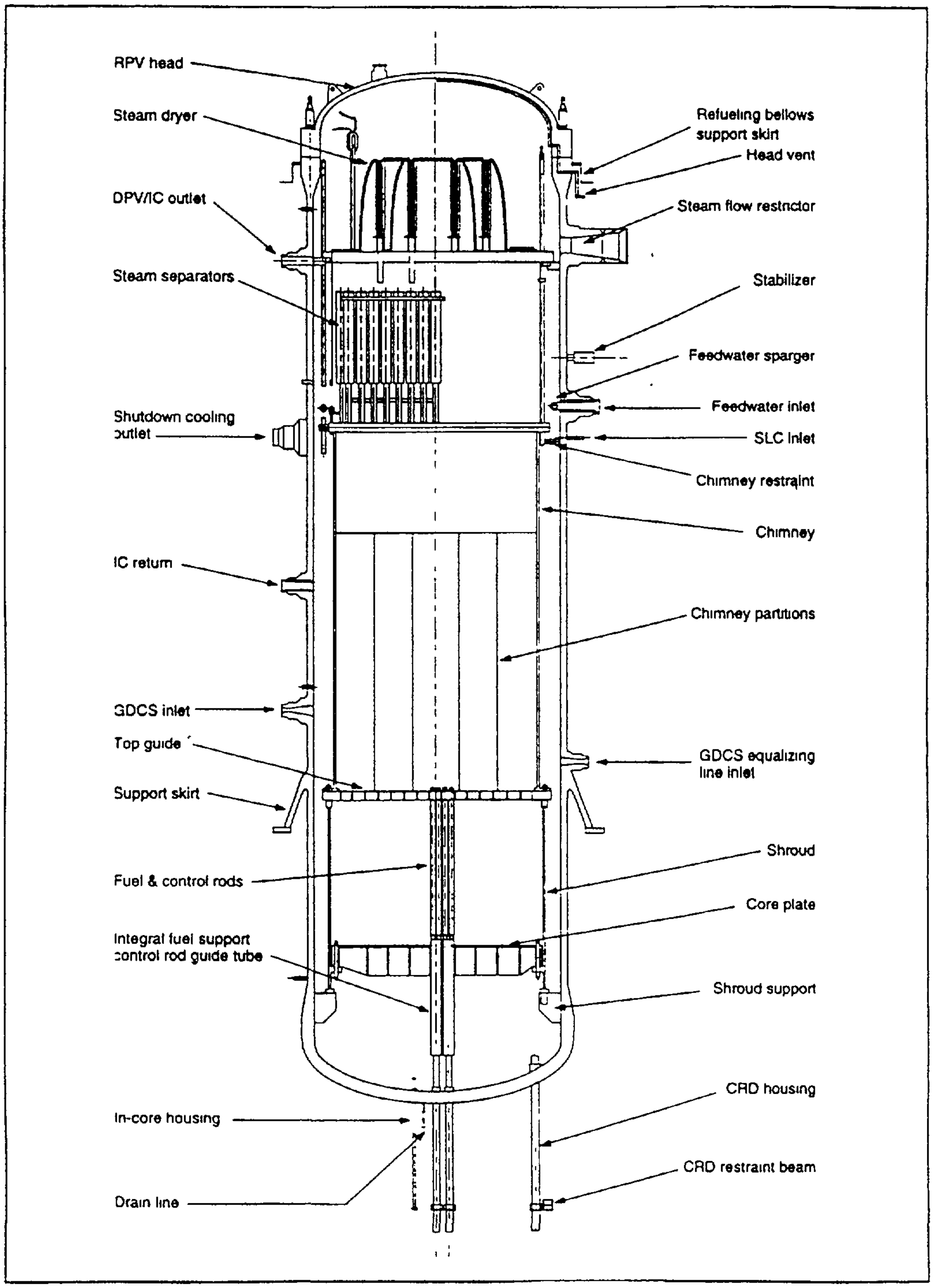


*Fig. 5.3.1. SBWR Reactor Cross Section*

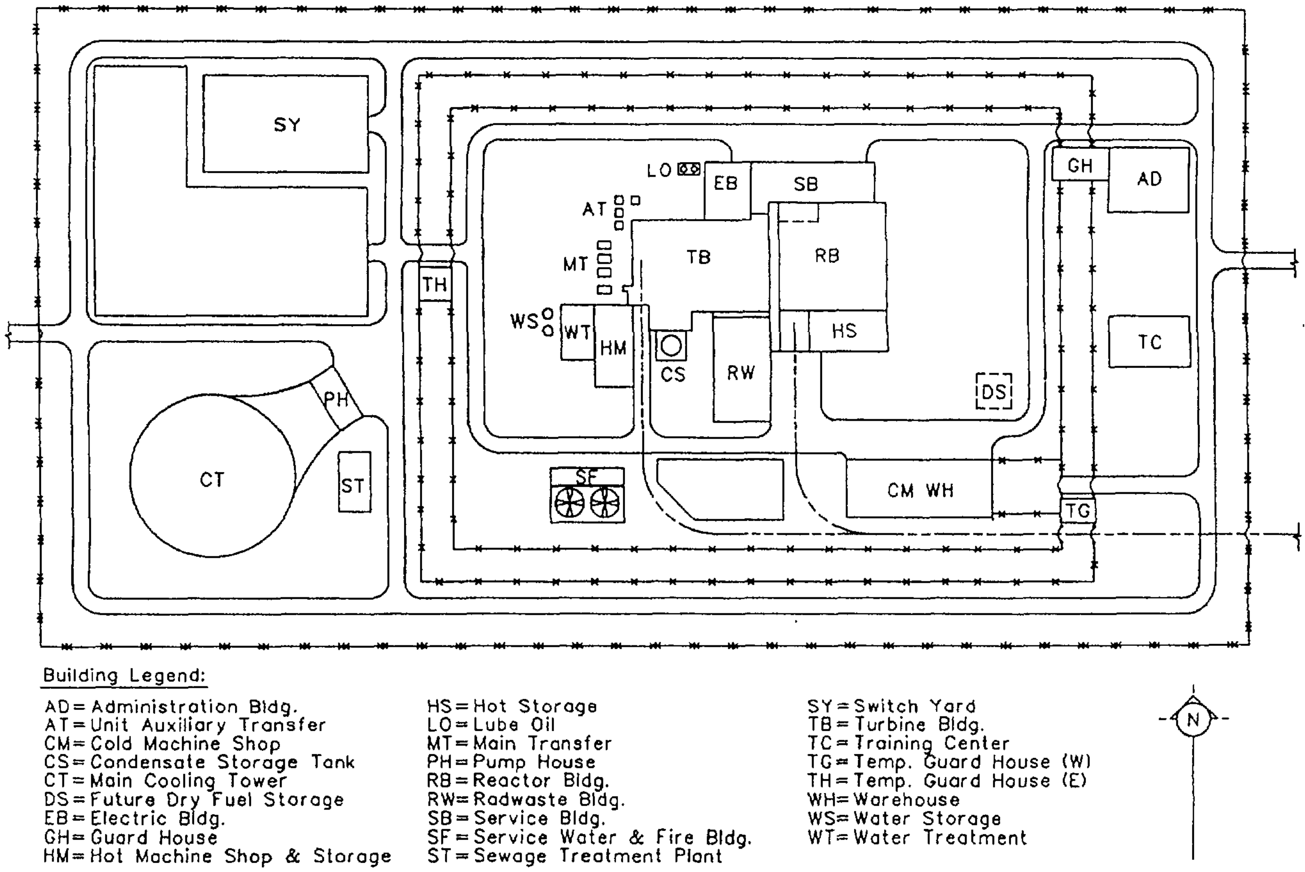


*Fig 5.3.2. SBWR Plant Layout*

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; (4) Proposed inspection, tests, analyses and acceptance criteria (ITAAC). Following the completion of review of the SSAR, 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. SBWR certification will then be issued following a rule making process.

### 5.3.6. Project economics

The estimated 1993 realistic Nth-of-a-kind base construction cost (inclusive of contingency) of the 600 MWe SBWR plant is $1840/kW.

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## 5.4. QP 300 REACTOR SYSTEM DESCRIPTION AND DEVELOPMENT STATUS

### 5.4.1. Basic objectives and features

The basic design objectives and features for QP300 are as follows:

a) To follow the IAEA safety codes and standards for NPP design in the QP300 NPP Project.
b) To follow the defence-in-depth safety concept.
c) To apply the ALARA (as low as reasonably achievable) principle in radiation protection.
d) To use proven power generating system components and technology which require no plant prototype.
e) To have a plant design objective of 40-year lifetime without reactor vessel replacement.
f) To attain a net electrical output of at least 300MWe.
g) To design for a low power density with a large enough safety margin to provide increased operating margins and safety.
h) To design the plant to minimize cost and construction time and to meet safety, operational and maintenance criteria.

### 5.4.2. Design description

#### *5.4.2.1. Nuclear Steam Supply System*

The nuclear steam supply system (NSSS) of the QP300 consists of a pressurized water reactor, reactor coolant system (RCS) and associated auxiliary systems. The NSSS has retained the general design features of current PWR plant design.

A. Reactor Coolant System

The RCS is arranged as two closed reactor coolant loops connected in parallel to the reactor vessel, each containing a reactor coolant pump and a steam generator. An electrically heated pressurizer is connected to the hot leg of one reactor coolant loop.

High pressurized water, driven by the reactor coolant pumps, circulates through the reactor core to remove the heat generated by the nuclear chain reaction. The heated reactor coolant enters the steam generator to transfer the heat energy to the feedwater of the secondary circuit through the U tubes to generate steam for the turbine generator. The cycle on the primary side is completed when the water from the steam generator is pumped back to the reactor vessel. The entire RCS is composed of leaktight components to ensure that all fluids are contained in the system pressure boundary.

*Reactor Pressure Vessel*

The reactor pressure vessel (RPV) is part of the pressure boundary of the RCS. It contains a variety of internals that support and locate the core and provide flow distribution.

The RPV consists of an upper head and a vessel. It is designed and manufactured in accordance with section III of the ASME Boiler and Pressure Vessel Code. The vessel and the upper head are made of low alloy steel and fabricated from forged sections. All the inner surface of the RPV in contact with reactor coolant is clad with stainless steel.

Two inlet nozzles (700 mm ID) spaced 180° apart, and two outlet nozzles (700 mm ID) 180° apart, are incorporated in the upper cylindrical region of the vessel. The incore flux measuring instrumentation penetrations are located in the bottom head of the vessel. The inside diameter of the vessel is 3.37m.

The upper head and head mounted components structure form an integrated head package. The head package consists of the upper head, control rod drive mechanism, seismic supports etc.

The total height of the RPV is 10.7m. Specific design characteristics for the QP300 reactor pressure vessel are given in the General Design Data (in Section 6.4).

*Reactor Core*

The reactor core of the QP300 consists of 121 fuel assemblies of the 15 x 15 type with an active fuel length of 2.90 m. The average volumetric power density of the core is 70.9 kw/l.

Core reactivity is controlled by means of chemical poison dissolved in the coolant, burnable poison rods and control rod assemblies. Soluble boron and burnable poison rods are utilized for shutdown and fuel burnup reactivity control. Control rod assemblies (37 clusters) are used for power regulation and hot shutdown. The core consists of 3 regions with enrichments of 2.4%, 2.67 % and 3.0%. It has a negative temperature coefficient of reactivity. The core has a fuel cycle of 12 to 16 months with a discharge burnup of 30,000 MWd/tU.

Neutronic and thermohydraulic data will be obtained from fixed in-core instrumentation, including fixed in-core detectors and thermocouples, which provide core performance monitoring data.

*Steam Generators*

The steam generator is of the U-tube, vertical type, and produces saturated steam. The (approximately 3,000) heat transfer tubes are made of Incoloy-800 sized at -22mm in OD with a wall thickness of 1.2mm. The total heat transfer area is approximately 3,073 $m^2$ for each SG. Each SG is rated at 501 MWt and capable of generating 9.775 x $10^5$kg/hr of saturated steam at 5.54 MPa.

The coolant flow rate of the primary circuit totals approximately 32,200 $m^3$/hr with a hot leg temperature of ~ 315.5°C and a cold leg temperature of ~ 288.5°C.

The steam generator is designed and manufactured in accordance with section III of the ASME and related Chinese standards. There are one inlet and one outlet on the lower channel head. A partition plate divides the channel head into inlet and outlet chambers. An access opening, for inspection and maintenance is provided in each chamber of the channel head. A feedwater inlet nozzle is located at the top of the upper shell.

An integral steam flow structure is installed in the steam nozzle to limit the quantity of steam flow in the event of main steam line break.

*Reactor Coolant Pumps*

The reactor coolant pump is a vertical, single speed (1500 rpm), bottom suction, horizontal discharge casing unit with controlled leakage mechanical shaft seals and driven by a vertical drip-proof three phase induction motor designed for a voltage of 6000V. The pump motor is provided with an anti-reverse rotation device and a flywheel. The rotating inertia of the pump is 1750 kg $m^2$ to meet the flow coast down requirement in the case of a complete loss of flow accident to avoid exceeding the minimum DNB.

The motor is coupled to the pump by means of a flexible spacer type gear coupling. The coupling is provided with a spacer sleeve of sufficient length to ensure easy maintenance of the shaft seals and the graphite bearing.

*Pressurizer*

The pressurizer is a vertical, cylindrical vessel with semi-spherical upper and lower heads. The matrix material of the pressurizer is SA508-III low alloy carbon steel. The internal surface is clad with stainless steel. A spray line connection nozzle, a relief valve connection nozzle and two safety valve connection nozzles are provided in the upper head. An access port on the upper head is provided for inspecting the spray nozzle. The replaceable electric heating elements are located on the lower head. The surge line connection nozzle is located on the lower head.

B. Auxiliary Systems

There are various auxiliary systems in the QP300, which support the safe operation of the RCS and the plant.

The auxiliary systems can be divided into five categories.

a. The systems used to ensure normal power operation of the reactor and the RCS are:
Chemical and volume control system
Boron recycle system
Reactor coolant pump seal water injection system
Steam generator blowdown system
Sampling system
b. The systems used to remove the residual heat after reactor shutdown and to cool the various components and the spent fuel are:
Component cooling water system
Spent fuel pool cooling and cleanup system
Residual heat removal system
Essential service water system
c. The systems used to protect the reactor and the containment during accidents so that the environment will not be contaminated seriously by radioactive substances are:
Safety injection system
Containment spray system
Containment hydrogen recombiner system
Containment isolation system
Auxiliary feedwater system
d. The system used for reactor refuelling and for storage of spent and new fuel is:
Fuel handling, and storage system
e. The systems used to maintain environment conditions within limits prescribed within the buildings are:
HVAC systems in the containment (containment air cooling, containment air cleanup, CRDM cooling, reactor cavity cooling, containment purge supply and exhaust,

containment hydrogen mixing, and ventilation).
HVAC systems in the nuclear auxiliary building.
HVAC systems in the electrical building (main control room habitability ventilation system)
Fuel storage building ventilation system
Diesel generator building ventilation system

*5.4.2.2. Balance of Plant Systems*

A. Conventional Systems

*Turbine and Auxiliaries*

The turbine is a three cylinder, tandem compound, quadruple exhaust, 3000-rpm, condensing turbine. There are two moisture separator-reheaters. The turbine-generator has a maximum guaranteed rating of 325 MWe gross at 1906.64t/h steam flow with inlet steam conditions of 5.34 MPa(a), 0.5% moisture, exhausting at 0.00574MPa(a), and zero percent makeup. There are six stages of feedwater heating. The turbine generator is intended for base load operation but also has load follow capability.

The turbine is equipped with a digital electro-hydraulic control system which uses a solid-state electronic controller and a high pressure fluid system to control valve movement.

The condenser is of the surface, twin pass type. It is of a twin shell construction. There are two deaerators that utilize extraction steam from the low pressure turbines, five low pressure feedwater heaters that utilize extraction steam from the low pressure turbines, three high pressure heaters that utilize extraction and exhaust steam from the high pressure turbine, three one-half-sized condensate pumps and condensate booster pumps, and three one-half-sized feedwater pumps. Heater drains from the three high pressure feedwater heaters are cascaded to the deaerator, drains from the five low pressure heaters are cascaded to the condenser.

*Main Conventional Systems*

The main conventional system is composed of main steam system, turbine bypass system, main feedwater system, and other auxiliary systems.

The turbine bypass system enable the NSSS to following large turbine load reductions, bypassing the turbine and discharging steam directly to the condensers. It is also used for temperature control of the RCS during hot shutdown and cooldown until start up of the reactor residual heat removal system.

B. Radioactive Waste Management System

The radioactive waste management system of the QP300 consists of gaseous waste treatment system, liquid waste treatment system, solid waste treatment and storage system and radioactive waste solidification system.

The radioactive waste treatment systems provides all equipment necessary to collect, process, monitor, and dispose of radioactive gaseous, liquid, and solid wastes that are produced during reactor operation.

The gaseous waste treatment system provides storage of radioactive effluent until it decays down to a value allowable for discharge to the environment through the HEPA filter and iodine filter into the plant stack.

The liquid waste treatment system collects, processes and monitors all potentially radioactive liquid waste produced during normal plant operation and maintenance. The liquid effluent is discharged, reused or finally disposed in the plant according to its radioactive level.

The solid waste treatment and storage system provides treatment, encapsulation and storage for the solid radioactive waste produced during, plant normal operation and maintenance.

The radioactive waste solidification system is designed to provide a means to solidify various radioactive liquid concentrates in cement, so as to become a solidified product with appropriate physicochemical properties.

*5.4.2.3. Instrumentation Control and Electrical System*

*Plant Control Systems*

The plant control scheme is based on the "reactor follows turbo generator" principle. The plant control system is used to maintain certain parameters at setpoint value by the automatic operation of actuators so as to ensure the safe and stable operation of the plant. It comprises the following major subsystems:

reactor power control system, pressurizer pressure control system, pressurizer water level control system, steam generator feedwater control system, steam dump control system and turbo controller.

*Protection System*

In case of unsafe conditions the reactor protection system takes over and automatically scrams the reactor and actuates the relevant safety systems. The reactor protection system includes the reactor trip system and the engineered safety features actuation system.

The reactor trip system automatically prevents operation of the reactor in an unsafe region by tripping the reactor whenever the allowable limits are reached.

The engineered safety features actuation system is a limit protection system. When a serious accident occurs in the plant, it actuates the corresponding engineered safety systems to provide emergency cooling for the reactor core so as to ensure containment integrity and to prevent radioactive contamination. The actuation system consists of five parts: safety injection actuation, steam line isolation actuation, feedwater line isolation actuation, containment spray actuation, containment isolation actuation system.

*Instrumentation System*

The instrumentation system provides all necessary information for operation of the plant, various signals for displaying, recording, controlling, protecting, annunciating, and the alarm function for the equipment and operational systems. It includes the in-core instrumentation system, nuclear instrumentation system, process instrumentation system, control rod position indication system and seismic instrumentation system.

*Plant Radiation Monitoring System*

The system is intended to check continuously that the operational radioactive limits coupled with operational activities are sufficient to prevent excess exposure of public and plant personnel. It includes the process radiation monitoring system, the effluent radiation monitoring system, and the in-plant area radiation monitoring system.

*Electrical Power System*

The electrical power system will include a power transmission and generation system and an electric power distribution system. The system design will be dependent on the local electricity supply arrangements on site.

*5.4.2.4. Safety considerations and emergency protection*

a. Inherent safety features

QP300 has some inherent safe characteristics The negative reactivity coefficient above hot zero power at all times halts the chain reaction in the event of an unexpected power increase. The low power density (13.59 kW/m) provides increased operating margins and improves the plant safety.

b. Passive safety

The accumulators filled with boronated water inject water (2x 40 $m^3$) into the RCS by nitrogen pressure of 50 bars in the event of a large LOCA.

c. Active safety

The design of QP300 uses the active safety system to mitigate accidents.

*Safety Injection System*

The safety injection system, which has a large capacity, consists of 2 charging pumps, 4 high head injection pumps, 2 low head injection pumps and 2 accumulators (passive action). If a LOCA occurs, the boronated water will be made to pass into RCS by the safety injection system.

*Core Cooling*

The boronated water for safety injection is provided by the refuelling water storage tank (RWST) (1500 $m^3$). After low level is reached in the RWST, the safely injection system is switched to recirculation, and the long term core cooling is provided.

*Containment System*

The large containment is designed so as to enlarge the capacity of the hot well. The containment spray system may remove heat from the containment atmosphere to reduce containment pressure and temperature following a LOCA or MSLB, and remove the airborne iodine.

*Auxiliary Feedwater System*

Redundancy and diversity have been considered in the design of the auxiliary feedwater system which includes 2 motor pumps and 2 diesel pumps. After loss of outside power, the feedwater can be provided by the diesel pumps.

*Emergency Power Supply*

After loss of outside power, various systems and components important to safety will be supplied with emergency power. Two batteries and three emergency diesel generators are installed in the plant. The third diesel generator could counter station black out.

d. Safety analysis

The analysis of all design basis accidents has been performed. The analysis shows that the consequences of all design basis accidents are well below NRC limits. The peak clad temperature following a large-break (LOCA) is about 1080°C (< NRC limit 1204°C). For small LOCA, the peak clad temperature is about 530°C, due to the large capacity of the high head injection system.

*5.4.2.5. Buildings and structures*

A. Buildings

The QP300 plant consists of nine principal building structures (the reactor building, the nuclear auxiliary building, the fuel storage building, the electrical building, the turbine generator building, the diesel generator building, the liquid radwaste solidification building, the solid radwaste storage building, and the administration building). The nuclear island including the reactor building, the nuclear auxiliary building, the fuel storage building and the electrical building is built on a common foundation mat (~ 87m x 90m). The reactor building is located at the center of the island and the nuclear auxiliary building as well as other buildings are arranged around the reactor building. The turbine generator building is arranged near to the electrical building on the nuclear island.

B. Structures

The containment is a prestressed reinforced concrete structure in the shape of a cylinder with a torispherical dome and a flat foundation mat. The cylindrical portion of the containment is prestressed by a post tensioning system consisting of horizontal tendons and vertical tendons. The dome is prestressed with crossed tendons that are anchored at the dome stiffening ring girder. The foundation mat is conventional reinforced concrete.

The interior surface of the containment is steel-lined for leaktightness. A protective layer of concrete covers the portion of the liner above the foundation mat.

The containment building is designed so that leakage to the environment will not exceed 0.3% per day of the mass of gas contained in the containment, even in the unlikely event of a LOCA.

The containment structure concrete provides biological shielding for normal and accident conditions. The internal structures and compartment arrangement provide missile protection for the equipment and biological shielding for maintenance personnel.

The internal structures of the containment and other buildings of the nuclear island are conventional reinforced concrete structures.

C. Seismic behaviour

For the purpose of design and analysis, the building structures are designed by seismic requirements according, to their relation to plant safety Seismic classification is as follows

SSE, The structures are designed and constructed in accordance with the seismic requirements for both the safety shutdown earthquake (SSE) and the operating basis earthquake (OBE).

OBE, The structures are designed and constructed in accordance with the seismic requirements for the operating basis earthquake.

NA, Not-applicable, meaning the structures are not required to be designed either for SSE or for OBE

The nuclear island structure is designed and constructed in accordance with seismic class SSE

### 5.4.3. Safety concept

TABLE 5 4 1 MAIN SAFETY RELATED SYSTEMS IN THE QP300 CONCEPT

| Name | Safety | Main characteristics |
|---|---|---|
| Primary circuit | X | Reactor vessel, 2 main pumps, 2 steam generators |
| Diverse reactivity control system | X | Boron injection, via safety injection system |
| Control rods | X | 37 rod cluster control assemblies |
| Residual heat removal system | X | 2HX, 2 pumps |
| Component cooling water system | X | 4HX, 4 pumps |
| Emergency core cooling system | X | 2 charging pumps, 4 high head injection pumps, 2 low head injection pumps |
| Containment spray system | X | 2 spray pumps |
| Auxiliary feedwater system | X | 2 motor pumps, 2 diesel pumps |
| Main control room habitability ventilation system | X | Provides fresh air, cooling and pressurization |
| Emergency power supply system | X | 2 batteries, 3 diesel generators |

TABLE 5.4.2 MAIN CONCEPT INITIATORS FOR THE QP 300

- LOCA (Primary): Loss of Primary Coolant Accident
- LOCA (Secondary): Secondary Pipe Rupture (water or steam)
- LOCA (Interfacing). e.g.: SGTR Steam Generator Tube Rupture
- ATWS· Anticipated Transients Without Scram
- Primary Transients
- Secondary Transients (Turbine Trip)
- Loss of Electric Sources (all AC sources)
- Total loss of the Cold Sources
- Total Loss of the Steam Generator Feedwater
- Station Blackout

TABLE 5 4.3 DESIGN FEATURES FOR PREVENTION AND PROTECTION LEVELS. THAT REDUCE (R), SUPPRESS (S). THE INITIATORS FREQUENCY OF LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES (L)

| PREVENTION LEVEL |
|---|
| LOCA (Primary)<br>- Reduced vessel fluence, reduce initiator frequency<br>- Primary circuit integration. limits small LOCA consequences<br>- Adaptation of the leak before break. limits accident consequences<br><br>LOCA (Secondary)<br><br>(LOCA (Interfacing)<br>- controlled water quality and improved materials reduces initiator frequency<br>Primary transients<br>- Increased designed margins reduces initiator frequency<br>Secondary transients<br>Loss of electric sources<br>Total loss of the cold course (Water)<br>Total loss of the SG feedwater<br>Station blackout |
| **PROTECTION LEVEL** |
| (LOCA (Primary)<br>- Primary circuit integration > leakage limitation<br>- Safety injection<br><br>LOCA (Secondary)<br><br>LOCA (Interfacing)<br>- N16 etc monitoring<br>ATWS<br>- Strong negative temperature coefficient<br>- ATWS mitigation system<br>Primary transients<br>- Multiple reactor trips<br>Secondary transients<br>- Multiple reactor trips<br>Loss of electric sources<br>- Emergency power supply system (batteries, diesel generators)<br>Total loss of heat sink<br>Total Loss of S.G feedwater<br>Station blackout The third diesel generator for station blackout |

TABLE 5.4.4: DESIGN FEATURES FOR MITIGATION LEVEL

| Safety Functions | Systems (Cf.Tab 5.4.1) | Passive/Active | Design Features/Remarks |
|---|---|---|---|
| Design Basis<br>Fission Product containment | Clad<br>Primary Circuit<br>Containment/Containment Spray System | Passive<br>Passive<br>Passive/Active | |
| Coolant inventory | Safety injection system (SIS) | Active | |
| Decay Heat Removal | Residual Heat Removal System (RIIRS)<br>SIS/Containment<br>Containment Spray System (CSS) | Active<br>Active/Passive<br>Active | |
| Reactivity control | Control rod<br>Boron injection (SIS) | Active<br>Active | |
| Primary circuit pressure control | Pressurizer depressurization system | Active | |
| Severe Accident<br>Containment temperature and pressure control | Containment/venting system | Passive | |
| Heat Removal | SIS/Containment/CSS | Active/Passive | |
| Tightness control | Containment | Passive | |
| Inflam. gas control | Recombiner/Igniters | Passive | |
| Fission product containment | Filter | Passive | |

### 5.4.4. Design data questionnaire

## I. GENERAL INFORMATION

1. Design name: QP300
2. Designer/Supplier address: Shanghai Nuclear Engineering Research & Design Institute, No. 29 Hong Cao Road, Shanghai, China.
3. Reactor type: PWR
4. Gross thermal power (MW-th) per reactor: 998.6
5. Net electrical output (MW-e) per reactor: 300
6. Heat supply capacity (MW-th):----

## II. BASIC DESIGN DESCRIPTION

### A. CORE AND REACTIVITY CONTROL

7. Fuel material: $UO_2$
8. Fuel inventory (tones of heavy metal): 35.9
9. Average core power density (kW/liter): 70.9
10. Average fuel power density (kW/kgU): 27.8
11. Maximum linear power (W/m): $3.917 \times 10^4$
12. Average discharge burnup (MWd/t): 30000
13. Initial enrichment or enrichment range (Wt%): 2.4-3.0
14. Reload enrichment at the equilibrium (Wt%): 3.4
15. Refueling frequency (months): ~ 12
16. Type of refueling (on/off power): off power
17. Fraction of core withdrawn (%): 33
18. Moderator material and inventory borated water: ~ 150 $m^3$
19. Active core height (m): 2.9
20. Core diameter (m): 2.486
21. Number of fuel assemblies: 121
22. Number of fuel rods per assembly: 204
23. Rod array in assembly: 15 x 15
24. Clad material: 4 alloy
25. Clad thickness (mm): 0.7
26. Number of control rods or assemblies: 37
27. Type: clusters
28. Additional shutdown systems: boron injection
29. Control rod neutron absorber material: Ag-In-Cd
30. Soluble neutron absorber: boron
31. Burnable poison material and form: Borosilicate glass

### B. REACTOR COOLANT SYSTEM

#### B1. Coolant

32. Coolant medium and inventory: borated water, ~ 150 $m^3$
33. Design coolant mass flow through core (t/h): 24000
34. Cooling mode (forced/natural): forced
35. Operating coolant pressure (bar): 15.20
36. Core inlet temperature (°C): 288.5
37. Core outlet temperature (°C): 315.5

#### B2. Reactor pressure vessel/tube

38. Overall length of assembled vessel/tube (m): 10.705
39. Inside vessel/diameter (m/mm): 3.374m
40. Average vessel/tube thickness (mm): 175
41. Vessel/tube material: SA 508 III
42. Lining material: stainless steel
43. Design pressure (bar): 17.16
44. Gross weight (tone/kg): 216 t

#### B3. Steam generator

45. Number of steam generators: 2
46. Type: inverted U-tube

47. Configuration (horizontal/vertical): vertical
48. Tube material: Incoloy 800
49. Shell material: S271
50. Heat transfer surface per steam generator ($m^2$): 3072.9
51. Thermal capacity per steam generator (MW): 517.5
52. Feed water pressure (bar): ~ 6.8
53. Feed water temperature (°C): 216
54. Steam pressure (bar): 5.54
55. Steam temperature (°C): 267.5

**B4. Pressurizer**

56. Pressurizer total volume ($m^3$): 35
57. Steam volume (full power/zero power, $m^3$): 17.5 (full)

**B5. Main coolant pumps**

58. Number of cooling or recirculation pumps: 2
59. Type: Vertical, single-speed, shaft seal units
60. Pump mass flow rate (kg/s): 12000
61. Pump design rated head: 60 m, $H_2O$
62. Pump nominal power (kW): 4500
63. Mechanical inertia (kg $m^2$): 1750

## C. CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS)

64. Number of extraction lines: 2
65. Number of pumps: 2
66. Number of injection points: 2
67. Feed and bleed connections

## D. CONTAINMENT

68. Type: Cylinder with a shallow dome roof and flat foundation
69. Overall form (spherical/cyl.): Cylindric
70. Structural material: Prestressed reinforced concrete
71. Liner material: steel liner
72. Simple/double wall: single wall
73. Dimensions (diameter, height) (m): 36 x 57
74. Design pressure (bar): 0.26
75. Design temperature (°C): 127
76. Design leakage rate (% per day): 0.3

## III. <u>SAFETY RELATED SYSTEMS</u>

## A. DESIGN CONDITIONS

**A1. Fission product retention**

77. Containment spray system (Y/N): Yes
    a. Duration (h)
    b. Flow rate ($m^3$/h): 300
    c. Mode of operation (active/passive): Active
    d. Safety graded (Y/N): Yes
78. F.P. sparging (Y/N)
79. Containment tightness control (Y/N)
80. Leakage recovery (Y/N)
81. Guard vessel (Y/N)

**A2. Reactivity control**

82. Absorber injection system (Y/N): Yes
    a. Absorber material: Boron
    b. Mode of operation (active/passive): Active and passive
    c. Redundancy: Active - 2, passive - 1
    d. Safety graded : Yes
83. Control rods (Y/N): Yes

a Maximum control rod worth (pcm) 2360
b Mode of operation (active/passive) Active and passive
c Redundancy Sufficient without one most worth rod
d Safety graded Yes

**A3. Decay heat removal**

*A3-1 Primary Side*

84 Water injection
a Actuation mode (manual/automatic) automatic
b Injection pressure level MPa (g) 2 94
c Flow rate (kg/s) 450
d Mode of operation (active/passive) both (A/P)
e Redundancy Active 2, passive 1
f Safety graded (Y/N) Yes

85 Water recirculation and heat removal
a Intermediate heat sink (or heat exchanger) 2 SG + 2HX
b Mode of operation (active/passive) Active
c Redundancy 2
d Self sufficiency (h)
e Safety graded Yes

*A3-2 Secondary side*

86 Feed water
a Actuation mode (manual/automatic) automatic
b Flow rate (kg/s) 1077
c Mode of operation (active/passive) active
d Redundancy 2
e Self sufficiency (h)
f Safety graded Yes

87 Water recirculation and heat removal
a Ultimate heat sink (cold source) sea water and air
b Mode of operation (active/passive) both (A/P)
c Redundancy 2
d Self sufficiency (h)
e Safety graded Yes

*A3-3 Primary pressure control*

88 Implemented system (Name) Pressurizer pressure control system
a Actuation mode (manual/automatic) manual/automatic
b Side location (primary/secondary circuit) primary
c Maximum depressurization rate (bar/s) 50
d Safety graded Yes

**B. SEVERE ACCIDENT CONDITIONS***

**B.1 Fission products retention**

89 Containment spray system (Y/N) Yes
90 F P Sparging (Y/N) Yes
91 Containment tightness control (Y/N) Yes
92 Leakage recovery (Y/N)
93 Risk of recriticality (Y/N) No

**B.2 Recriticality control**

94 Encountered design feature
a Mode of operation (A/P)
b Safety graded

**B.3 Debris confining and cooling**

95 Core debris configuration (core catcher)
96 Debris cooling system (name)
a Mode of operation (A/P)
b Self sufficiency

---

* All systems must be qualified to operate under the accident conditions

c. Safety graded (Y/N)

**B.4 Long term containment heat removal**

97. Implemented system
    a. Mode of operation (A/P)
    b. Self sufficiency (h)
    c. Safety graded (Y/N)
98. Intermediate heat sink
    a. Self sufficiency (h)
    b. Safety graded (Y/N)
99. External coolant recirculation
    a. Implemented components
    b. Mode of operation (A/P)
    c. Self sufficiency (h)
    d. Safety graded (Y/N)
100. Ultimate heat sink
    a. Self sufficiency (h)
    b. Safety graded (Y/N)

**B.5 Combustible gas control**

101. Covered range of gas mixture concentration
102. Modes for the combustible gas control
    a. Containment inertation
    b. Gas burning
    c. Gas recombining
    d. Others

**B.6 Containment pressure control**

103. Filtered vented containment (Y/N)
    a. Implemented system
    b. Mode of operation (A/P)
    c. Safety graded
104. Pressure suppression system (Y/N)
    a. Implemented system
    b. Mode of operation
    c. Safety graded (Y/N)

**C. SAFETY RELATED I&C SYSTEM**

Automatic load following (Y/N): Yes
* range (% power): 15 - 100
* maximum rate (%/min): 10

Load rejection without reactor trip (Y/N): Yes
Full Cathode Ray Tubes (CRT) display (Y/N): Yes
Automated start-up procedures (Y/N)
Automated normal shutdown procedures (Y/N)
Automated off normal shutdown procedures (Y/N)
Use of field buses and smart sensors (Y/N)
Expert systems or artificial intelligence advisors (Y/N)
Protection system backup (Y/N)

**D. EMERGENCY POWER SUPPLY SYSTEM**

105. Type (diesel, gas, grid connection): diesels
106. Number of trains: 2

**E. AC/DC SUPPLY SYSTEM**

107. Type (rectifier, converter, battery): rectifier, converter, battery
108. Estimated time reserve (hr): 4

## IV. <u>CONVENTIONAL THERMAL CYCLE</u>

A. **TURBINE SYSTEM**

109. Type: A double flow high pressure turbine and two double-flow low pressure turbine with extraction for seven stages of feedwater heating
110. Overall length (m): 26.14
111. Width (m): 7.82
112. Number of turbines/reactor: 1
113. Number of turbine sections per unit (e.g. HP/LP/LP): HP/LP/LP-HP/LP/LP
114. Speed (rpm): 3000

B. **STEAM CHARACTERISTICS**

115. H.P. inlet pressure: 5.35 MPa
116. H.P. inlet temperature: 268°C
117. H.P. inlet flowrate: 1906 t/h
118. L.P. inlet pressure: 0.738 MPa
119. L.P. inlet temperature: 253.5°C
120. L.P. inlet flowrate (per section): 762 x 2 t/h

C. **GENERATOR**

121. Type (3-phase synchronous, DC): 3 phase synchronous, DC
122. Apparent power (MVA): 388
123. Active power (MW): 310
124. Frequency (hz): 50
125. Output voltage (kV): 18
126. Total generator mass (t)
127. Overall length
128. Stator housing outside diameter

D. **CONDENSER**

129. Number of tubes: 2 x 17266
130. Heat transfer area: 2 x 14028 $m^2$
131. Flowrate ($m^3/s$): 2 x 31270 $m^3/h$
132. Pressure (m/bar)
133. Temperature (°C): 18-31

E. **CONDENSATE PUMPS**

134. Number: 3
135. Flowrate: 748
136. Developed head: (m●$H_2O$) 80
137. Temperature: 50°C
138. Pump speed: 1500 rpm

### 5.4.5. Project status

*5.4.5.1. Entities involved*

Under the overall control of China National Nuclear Corporation (CNNC), Shanghai Nuclear Engineering Research and Design Institute (SNERDI) has designed the QP300 Nuclear Power Plant. The first QP300 nuclear power plant is Qinshan 300 MWe Phase One NPP. SNERDI has played the role of overall designer and has made the NI design for the Qinshan NPP Project. The East China Electrical Power Design Institute (ECEPDI) has carried out CI design for the Qinshan NPP Project. The Nuclear Power Institute of China (NPIC) has undertaken some thermohydraulic tests.

*5.4.5.2. Design status*

The QP300 Reactor Power System has been under development for many years. The No.1 Unit of QP300 has been built in China, i.e. Qinshan 300 MW Nuclear Power Plant. The Final Safety Analysis Report for the Qinshan NPP was submitted in August 1989 and nuclear fuel was loaded in July 1991 after review and approval of the FSAR by the National Nuclear Safety Administration (NNSA) of China. After some commissioning stages, the rated electric power, 300 MWe, was reached in the Qinshan plant. Recently, the CHASHMA NPP (CHASNUPP) of Pakistan, another unit of the QP 300 type NPP supplied by CNNC, has been under construction. The PSAR of CHASNUPP has been reviewed and approved and the Construction Permit (CP) was issued in July 1992. SNERDI is currently working on the detailed design for CHASNUPP. Proven system components and technology will be used in the project. QP300 is available for building on a commercial basis.

*5.4.5.3.Research and development work*

Various R&D projects and tests have been made for the QP300 NPP. The main R&D work and tests are in the following:

Reactor physics research experiments,
Reactor core thermohydraulic tests,
Stress analysis and strength tests for major components and for critical position of those components,
Control rod drive system tests in cold and hot condition,
Material properties research,
Prestressed reinforced concrete containment research and tests,
Auxiliary equipment and instrumentation tests.

*5.4.5.4. Licensing process*

The licence procedure in China is generally divided into stages as follows:

(1) The operating organization of a nuclear power plant should submit a document relevant to the safety matters of the site specified in the "Nuclear Power Plant Feasibility Study Report" before the site is selected.

(2) After the site is selected, the applicant should submit an "Application for the Construction of Nuclear Power Plant" together with the Preliminary Safety Analysis Report" to the NNSA for the "Nuclear Power Plant Construction Permit"

(3) The applicant should submit an "Application for Initial Fuel Loading of Nuclear Power Plant" together with the "Final Safety Analysis Report" to the NNSA for an "Instrument of Ratification for the Initial Fuel Loading".

(4) The applicant should submit an "Application for an Operation Licence for a Nuclear Power Plant" together with the "Revision to the FSAR", to the NNSA. After a twelve month period following initial attainment of full power the operator will receive a "Nuclear Power Plant Operation Licence".

The QP300 has all necessary safety functions and features and is licensable

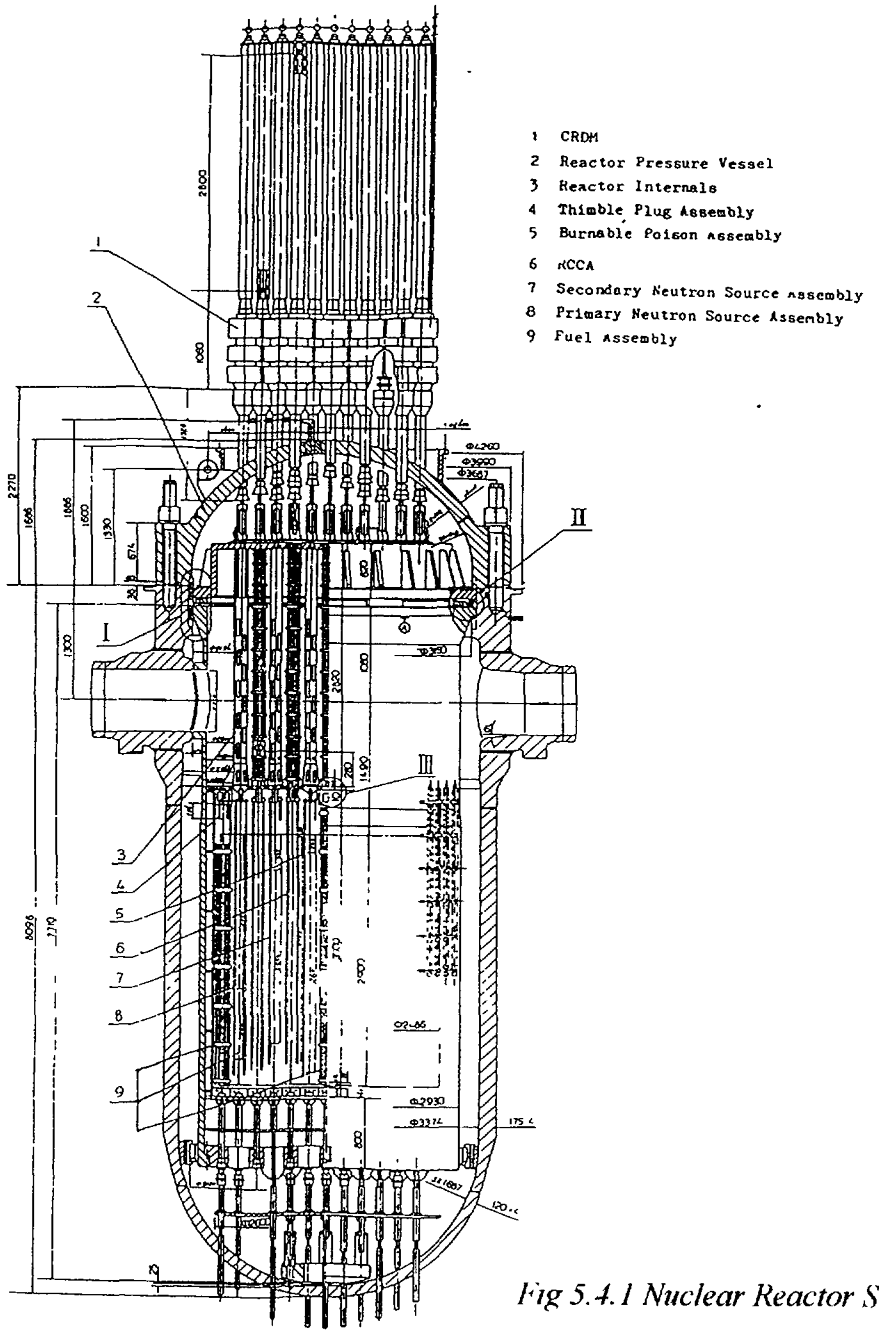
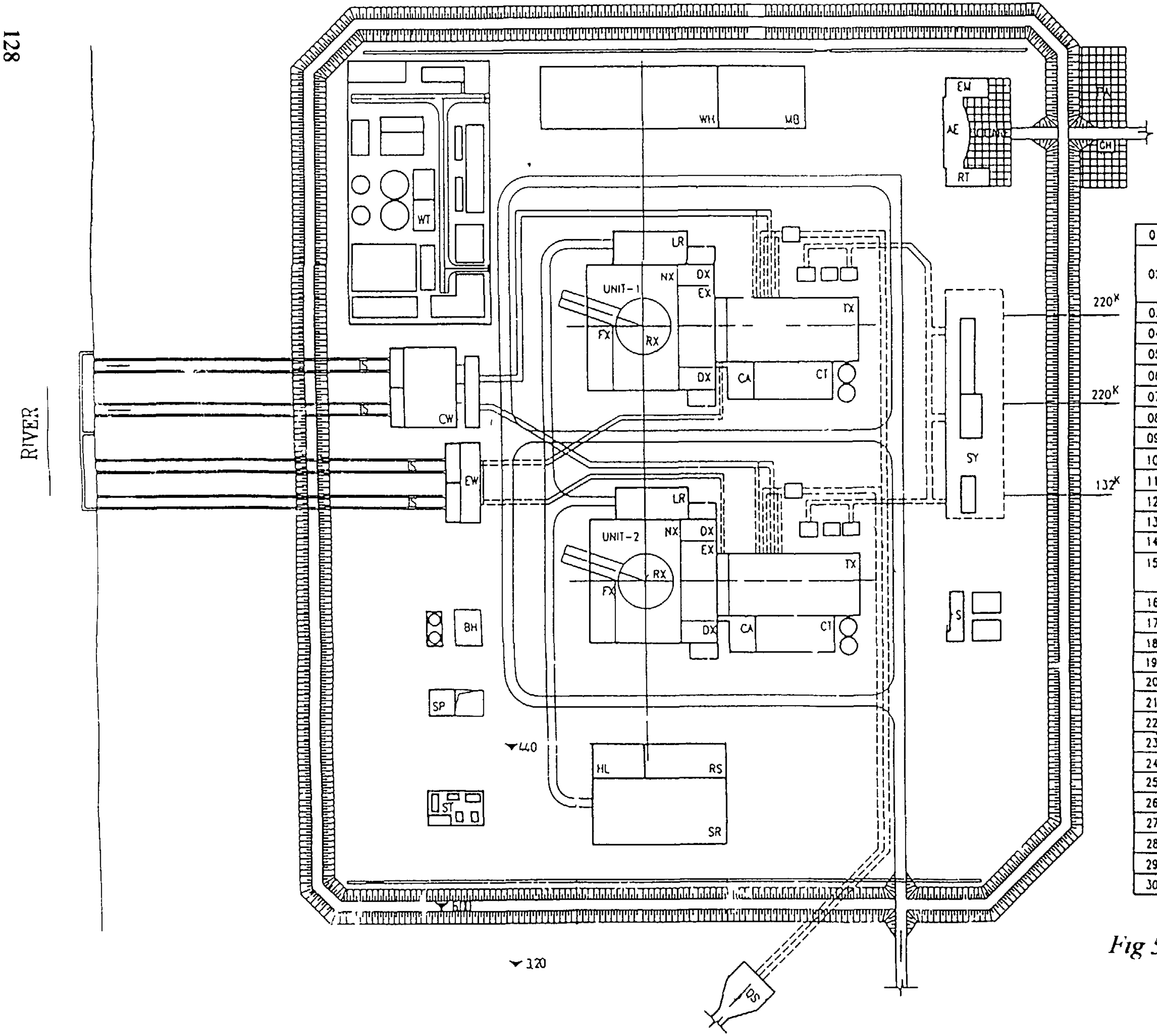



*Fig 5.4.1 Nuclear Reactor Structure*




LIST OF BUILDINGS AND FACILITIES

| | | |
|---|---|---|
| 01 | RX | Reactor Building |
| 02 | NX | Nuclear auxiliary building (including main steam piping gallery refuelling water tank emergency water tank, chilling water station compressed air station intake & exhaust ventilation fans house) |
| 03 | FX | Fuel Storage Building |
| 04 | TX | Turbine Generator Building |
| 05 | EX | Electrical Building |
| 06 | DX | Diesel Generator Building |
| 07 | CA | Personnel control access |
| 08 | SY | Switch yard |
| 09 | CT | Condensate treatment building |
| 10 | LR | Liquid radwaste solidication building |
| 11 | SR | Solid radwaste storage building |
| 12 | RS | Low-level radwaste storage house |
| 13 | HL | Hot laundry |
| 14 | BH | Boiler House |
| 15 | MB | Maintenance building ( maintenance for mechanism, electrical equipment & instrument & oil renew ) |
| 16 | WH | Warehouse |
| 17 | SP | Stormwater pump station |
| 18 | EW | Essential service water pump station |
| 19 | IS | Intake structure |
| 20 | CW | Circulating cooling water pump station |
| 21 | WT | Water treatment plant |
| 22 | ST | Sewage treatment plant |
| 23 | DS | Discharge |
| 24 | FS | Fire pump station |
| 25 | AE | Administration building & emergency center |
| 26 | EM | Environs radiation monitoring building |
| 27 | RT | Restaurant |
| 28 | MS | Meteorological station ( optional ) |
| 29 | PA | Parking area |
| 30 | GH | Guard house |

*Fig 5.4.2. Genral Layout of QP-300 NPP (schematic diagram)*

## 5.5. AST-500 REACTOR SYSTEMS' DESCRIPTION AND STATUS

### 5.5.1. Main Objectives and Features

The nuclear district heating plants (NDHPs) are intended for production of heat in the form of hot water for heating purposes only. The AST reactor plants can be used also for production of process steam of low temperature particularly for seawater desalination. The AST-500 is a 500 $MW_{th}$ power plant, whose prime objective is district heating.

Main features of the reactor plant:

- low parameters of the heat transfer circuits and low heat rating of the core;
- integral layout of primary circuit equipment;
- natural circulation in the primary circuit;
- guard vessel;
- proveness of reactor technology.

In the NDHPs practically all the heat generated is sent to the users. Minimal water consumption and heat losses are to ponds or into the atmosphere.

The efficiency of the heat generated is 95%.

### 5.5.2. Design Description

*5.5.2.1. Reactor Vessel*

The reactor vessel (RV) is fabricated from heat-resistant steel with an anti-corrosion lining. The outer diameter of the cylindrical vessel is 4.90 m, height is 15.310 m, and the design pressure is 2.3 MPa. The vessel weighs 185 t and its service life is approximately 60 years. The fluence of neutrons with $E > 1$ MeV to the RV is $10^{17}n/cm^2$.

Al1 nozzles of the primary circuit auxiliary systems are of small diameter, equipped with flow restrictors and located in the upper part of the vessel.

The Reactor Vessel is attached to a guard vessel. The welds of the reactor vessel can be inspected using ultrasound techniques.

Core

The core consists of 121 fuel assemblies (FAs) of hexagonal shape; The fuel assembly measures 238 mm across flats with a 243 mm spacing pitch. The core height is 3.0 m.

The fuel assembly contains 143 fuel rods (FRs) clad in zirconium alloy 13.6mm in diameter. The fuel pellets are sintered uranium oxide with an enrichment of 2.0%. The core is initially loaded with 49 tonnes of uranium and is partially refuelled every two years. The shrouded fuel assembly with a"chimney" provides for efficient coolant flow distribution in natural circulation.

The absence of soluble boron control leads to simplification of the plant process system and to the enhancement of the reactor self-regulation properties. The use of burnable poison reduces the reactivity margin which has to be compensated for by mechanically driven absorber rods. The mechanical system to control the reactivity, consisting of cluster-type absorber rods, fulfills the functions of reactivity compensation, power control and emergency protection.

Control Rod Drive Mechanism

The reactor uses 36 control rod drive mechanisms (CRDMs) placed on the reactor cover. Each CRDM provides simultaneously the movement of rod-absorbers in three Fuel Assemblies. The CRDM is an electromechanical, rack-type drive. The control rod travelling speed in electric motion mode is 2 mm/s. The mechanism design is analogous to that used for VVER-440 reactors and its reliability has been confirmed by tests of pilot drive mechanisms including life-time tests.

Guard Vessel

The guard vessel is a protective passive device which performs the following safety functions:
- localizes primary coolant in the case of primary circuit depressurization accidents;
- keeps the core under water for reactor depressurization accidents.

Primary circuit

The primary circuit includes the equipment of the main circulation circuit, pressurization system and also auxiliary systems connected to the reactor. The free volume above the reactor water level is used as steam-gas pressurizer for which gas is supplied to the pressurizer before reactor startup. The primary circuit operates in a non-boiling mode.

Secondary Circuit

The secondary circuit is a hermetically closed three-loop system intended for elimination of active primary coolant ingress into the grid water during power operation and into the residual heat removal circuit.

As a part of each secondary circuit loop there are:
- six primary/secondary heat exchangers connected in parallel, placed inside the reactor;
- one circulating pump;
- steam pressurizer;
- grid heat exchangers;
- pipelines with valves;
- emergency cooling system equipment;
- supporting systems.

There are a total of 18 primary/secondary heat exchangers, and six secondary / grid heat exchangers. Total heat transfer surface for the 18 heat exchangers equals 5040 $m^2$.

*5.5.2.2. Heating Grid Circuit*

The heating grid circuit is designed for operation at a pressure of 2.0 MPa and provides for heat transport from the reactor plant to consumers. It can be used also in a residual heat removal mode. The circuit is composed of three loops related to the three secondary loops joined by hot and cold water headers. The grid circuit of the plant contains six heat exchangers.

#### *5.5.2.3. Instrumentation, Control and Electrical Systems*

Automated control of technological processes in NDHP allows for an effective solution of the following tasks:

- reliable heat generation in accordance with the consumers' requirements;
- reduction of generated heat cost;
- assurance of safety for plant personnel, local population, and for the environment;
- effective control of the plant heat output.

These tasks are carried out by use of an automated control system with the following characteristics:

- use of an operator support system to prevent operating personnel errors both at normal operation conditions and emergency situations;
- automated testing of the safety systems thereby assuring their operating reliability and availability, protection of safety systems controllers against common mode failures and dangerous responses associated with design basis effects and personnel errors;
- automation of receiving, processing, recording and presentation of information, simplification of equipment state control, maintenance simplification with total reduction in labor-consuming operations and staff numbers.

#### *5.5.2.4. Safety Considerations and Emergency Protection*

AST-500 is an evolution of the most developed and widely used pressurized water reactors; the embodiment of world tendencies in development of enhanced safety new generation reactors for district heating purposes. The AST-500 safety concept is a combination of inherent safety developed in depth with regard to functional and physical protection.

In the AST-500 all primary circuit equipment is placed inside a common pressure vessel. The reactor integral arrangement allows elimination of large piping, simplifies the primary circuit and facilitates natural coolant circulation in the reactor to reduce RV irradiation and to extend its lifetime.

Safety systems operate on passive principles based on general physical laws (gravity, convection) without large consumption of energy or water and no intervention by operators. These systems consist of independent channels each fulfilling its functions under the assumption of a single failure.

The safety systems and their elements embody redundancy and diversity, through the use of different operating principles in different systems, to provide functional and physical in-depth protection of the reactor. Complete passivity of the safety systems is attained through the wide use of self-actuated devices for the initiation of the systems at deviation of the most important process parameters beyond the set limits.

The low power density of the core, increased design margins, self-regulation of the flow rate through the fuel assemblies and the slow dynamics of tranments ensure that there will be no fuel failures in normal or emergency conditions.

The users of heat from the NDHP are reliably protected from the ingress of radioactivity due to the use the intermediate three loop secondary circuits and by isolation valves.

The intrinsic self-protection of the reactor, the use of self-actuating systems for shut down and cooling of the reactor in combination with passive principles of action ensure immunity to operator errors and equipment failures.

*Means for Reactor Shutdown and Keeping it in a Safe Condition*

- Reactor shut down by controlled insertion of control rods into the core by the CRDMs.
- Reactor shut down by simultaneous insertion of all absorber rods into the core under gravity following de-energization of the CRDMs. This is done automatically and diversely from normal movements of CRDMs in response to changes in physical conditions such as power or water level.
- Reactor shut down by injection of liquid absorber (boron solution) through the active and/or passive channels of the back-up system for transferring the reactor into the "cold" state.
- Self-shutdown of the reactor in emergency conditions due to negative void coefficient of reactivity.
- Self-limitation of power due to negative coefficients of reactivity (reactor "hot" condition). at the balance point between core power and emergency heat removal capacity.

*Means of Decay Heat Removal*

- Through the grid circuit (natural coolant circulation in primary and secondary circuits and forced circulation of grid water).
- Emergency decay heat removal channel permanently connected to the secondary circuit by valves having different principles of action (natural coolant circulation in primary and secondary circuits and forced circulation of service water or evaporation of service water in the tanks).
- Evaporation of secondary coolant through a pilot operated relief valve (PORV) on the secondary circuit steam pressurizer.
- Emergency decay heat removal through the condenser on the reactor upper head.
- Heat exchangers of the primary coolant purification system and the CRDM coolers.
- Large capacity for accumulation of heat in primary, secondary and tertiary circuits.

*Means for Confinement and Reduction of Release of Radioactivity*

- Fuel with low working temperature.
- Hermetic FR cladding (leak-tightness monitoring) under conditions of low fuel heat rating.
- Leak-tight integral reactor and primary circuit.
- Limitation of the cross-section of all pipelines penetrating through the reactor vessel.
- Double valves in primary circuit auxiliary systems' pipelines in combination with self-actuating isolation valves.
- Double isolation valves on the pipelines of systems relating to the accident localizing circuit.
- Guard vessel.
- Leak-tight rooms housing primary circuit related systems. Bubbling of emergency discharges from the primary circuit and from the related rooms.

- Leak-tight protective enclosure (containment) with filters, bubblers, systems for pressure self-limitation, as well as with isolation valves in the pipelines penetrating the containment.

*5.5.2.5. Buildings and Structures*

The NDHP is designed as a twin unit plant. The leak-tight part of the reactor building is shaped as a cylindrical ferro-concrete containment with hemi-spherical dome. The ferro-concrete is not prestressed; the thickness of the containment walls is 1.6 m. The containment is intended for protection against external and internal impacts and is designed for:

- internal gauge pressure of 0.1 MPa at 120°C;
- impact of an air shock wave with a front pressure of 0.05 MPa during one second;
- crash of air plane weighing 20 t flying at 700 km/h velocity.

The equipment and structures of the safety systems are designed for an earthquake of 8 points on the MSK-64 scale.

Refuelling

Reloading the in-vessel structures and fuel is made by means of an universal machine developed by OKBM and fabricated for the pilot NDHP. The reloading machine is installed on railways on the floor of the reactor hall and services the zone where the reactor pit, spent fuel storage pool and equipment storage pits are arranged. When manipulating with fuel the reloading machine is used for reshuffling the FAs in the core, for transferring them between the core and spent fuel storage pools and inside the pools.

All manipulations with FAs beyond the reactor boundary are performed using a special water-filled transfer flask.

### 5.5.3. Safety concepts

TABLE 5.5.1. MAIN SAFETY RELATED SYSTEMS IN THE AST-500

| Name | Safety graded | Main characteristics |
|---|---|---|
| Primary Circuit (PC) | X | Reactor vessel, 18 integrated primary / secondary heat exchangers, natural circulation, leak-tight primary equipment |
| Reactor control and protection system (RCPS) | X | 36 rod claster assemblies moved by individual mechanisms |
| Diverse reactor protection system (DRPS) | X | Passive and active injection of boronated water |
| Passive residual heat removal system (PRHRS) | X | 3 channels: 1 channel on the reactor, 2 channels to the secondary circuit loops |
| Isolation valves (IV) | X | Double self-activated valves on each primary pipeline |
| Guard Vessel (GV) | X | Core uncovering prevention |
| Containment (C) | X | Protection against external and internal impacts |

TABLE 5.5.2. MAIN ACCIDENT INITIATORS FOR THE AST-500

| |
|---|
| - LOCA (primary): Loss of Primary Coolant Accident<br>- LOCA (Secondary: Secondary Pipe Rupture (water or steam)<br>- LOCA (Interfacing): e.g. SGTR Steam Generator Tube Rupture<br>- ATWS: Anticipated Transients Without Scram<br>- Primary Transients,<br>- Secondary Transients (turbine trip),<br>- Loss of electric sources (all AC sources),<br>- Total loss of the cold sources,<br>- Total loss of the steam generator feedwater,<br>- Station blackout |

TABLE 5.5.3. DESIGN FEATURES FOR PREVENTION LEVELS THAT REDUCE (R), SUPPRESS (S), THE INITIATORS FREQUENCY OR LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES (L)

| PREVENTION LEVEL |
|---|
| LOCA (Primary).<br>- Integral primary circuit, reduced fluence, small diameter of nozzles in the reactor vessel, guard vessel, low parameters reduce initiator frequency and limit consequences R |
| LOCA (Secondary) Reduced pressure, leak-tight equipment R |
| LOCA (Interfacing)<br>- Intermediate circuit, low pressure difference between prim and sec circuits limits consequences R |
| Primary transient - increased design margins R |
| Secondary transients - increased design margins R |
| Loss of electric sources - 3 channels, 2 independent off-site power sources |
| Total loss of the cold source (Water) - Passive ultimate heat sink (air) |
| Total loss of the SG feedwater Not relevant |
| Station blackout Natural coolant circulation, passive safety systems, reactor inherent safety features R |
| **PROTECTION LEVEL** |
| LOCA (Primary) - double isolation valves, guard vessel, primary circuit integration I |
| LOCA (Secondary) - 3 independent loops, double isolation valves L |
| LOCA (Interfacing) - intermediate circuit valves L |
| ATWS - Strong negative reactivity coefficients, large heat-accumulating capacity L |
| Primary transients - greater thermal inertia L |
| Secondary transient - greater thermal inertia L |
| Loss of electric sources - passive systems for reactor shut down, residual heat removal, and radioactive discharges localization |
| Total loss of heat sink Not relevant |
| Total loss of SG feedwater Not relevant |
| Station Blackout non-critical |

TABLE 5.5.4. DESIGN FEATURES FOR MITIGATION LEVEL OF AST-500

| Safety Functions | Systems (Cf. Tab. 5.5.1) | Passive/active | Design features/Remarks |
|---|---|---|---|
| **Design Basis**<br>Fission product containment | Primary circuit<br>Guard vessel<br>Containment | Passive<br>Passive<br>Passive | |
| Coolant inventory | Guard vessel<br>Isolation valves | Passive<br>Passive | GV integrates primary systems except clean-up. Quick-acting (5s closure time) valves. |
| Decay heat removal | PRHRS | Passive | 72 h capacity |
| Reactivity control | RCPS<br>DRPS | Passive/active<br>Passive/active | Shut down capability |
| Primary circuit pressure control | PRHRS, RCPC, DRPS | Passive | Without discharge of primary coolant |
| **Severe accident**<br>Containment temperature and pressure control | Containment | Passive | |
| Heat removal | PRHRS | Passive | |
| Tightness control | GV, C | Passive | |
| Inflam. gas control | ---- | | |
| Fission product containment | PC, GV, C | Passive | |
| Corium management | GV | Passive | |
| Others | | | |

**5.5.4. Design data questionnaire (Water Cooled Reactors for AST-500)**

## I. GENERAL INFORMATION

1. Design name: AST-500
2. Designer/Supplier address: OKBM
3. Reactor type: PWR    Number of modules/per plant: 1
4. Gross thermal power (MW-th) per reactor: 500
5. Net electrical output (MW-e) per reactor: Not relevant
6. Heat supply capacity (MW-th): 500

## II. BASIC DESIGN DESCRIPTION

### A. CORE AND REACTIVITY CONTROL

7. Fuel material: $UO_2$
8. Fuel inventory (tones of heavy metal): 49
9. Average core power density (kW/liter): 27
10. Average fuel power density (kW/kgU):
11. Maximum linear power (W/m): 29 000
12. Average discharge burnup (MWd/t): 19 100
13. Initial enrichment or enrichment range (Wt%): 1.0-2.0
14. Reload enrichment at the equilibrium (Wt%): 2.0
15. Refueling frequency (months): 24
16. Type of refueling (on/off power): Off power
17. Fraction of core withdrawn (%): 20-25
18. Moderator material and inventory: Water
19. Active core height (m): 3.0
20. Core diameter (m): 2.965
21. Number of fuel assemblies: 121
22. Number of fuel rods per assembly: 143
23. Rod array in assembly: Hexagonal
24. Clad material: Zirconium alloy
25. Clad thickness (mm): 0.9
26. Number of control rods or assemblies: 36 assemblies
27. Type: Cluster
28. Additional shutdown systems: Boron solution injection
29. Control rod neutron absorber material: $B_4C$
30. Soluble neutron absorber: Boron solution
31. Burnable poison material and form: Rods on boron basis

### B. REACTOR COOLANT SYSTEM

**B1. Coolant**

32. Coolant medium and inventory: Water, 187m$^3$
33. Design coolant mass flow through core (kg/s): 1548
34. Cooling mode (forced/natural): Natural
35. Operating coolant pressure (bar): 1.96 MPa
36. Core inlet temperature (°C): 131
37. Core outlet temperature (°C): 208

**B2. Reactor pressure vessel**

38. Overall length of assembled vessel/tube (m): 16.4
39. Inside vessel/diameter (m/mm): 4.820 m
40. Average vessel/tube thickness (mm)
41. Vessel/tube material: Steel, 15 X 2HMΦA
42. Lining material: Steel, 08 X 18 H10T (18Cr10NiTi)
43. Design pressure (bar): 23
44. Gross weight (tone/kg): 235t

**B3. Steam generator**

45. Number of HX: 18 primary/secondary heat exchangers
46. Type: Straight-tube
47. Configuration (horizontal/vertical): Vertical
48. Tube material: steel, 08 X 18 H10T
49. Shell material
50. Heat transfer surface per HX ($m^2$): 280
    Total = 280 x 18 = 5040 $M^2$
51. Thermal capacity (MW): 500
52. Second water pressure (bar): 12
53. Second water temperature (°C): 87/160

**B4. Pressurizer**

54. Pressurizer total volume ($m^3$)
55. Steam volume (full power/zero power, $m^3$): 28 on full power

**B5. Main coolant pumps (Not relevant)**

56. Number of cooling or recirculation pumps
57. Type
58. Pump mass flow rate (kg/s)
59. Pump design rated head
60. Pump nominal power (kW)
61. Mechanical inertia (kg $m^2$)

**C. CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS) (Not relevant)**

62. Number of extraction lines: 2
63. Number of pumps: 2
64. Number of injection points
65. Feed and bleed connections

**D. CONTAINMENT**

66. Type
67. Overall form (spherical/cyl.): Cylindrical
68. Structural material: Reinforced concrete
69. Liner material: Steel
70. Simple/double wall: Simple
71. Dimensions (diameter, height) (m): 40, 68
72. Design pressure (bar): 1
73. Design temperature (°C): 120
74. Design leakage rate (% per day): 0.2

## III. SAFETY RELATED SYSTEMS

**A. DESIGN CONDITIONS**

**A1. Fission product retention**

75. Containment spray system (Y/N): Yes
    a. Duration (h)
    b. Flow rate ($m^3$/h)
    c. Mode of operation (active/passive): Active
    d. Safety graded (Y/N): Yes
76. F.P. sparging (Y/N)
77. Containment tightness control (Y/N): Yes
78. Leakage recovery (Y/N): Yes
79. Guard vessel (Y/N): Yes

**A2. Reactivity control**

80 Absorber injection system (Y/N) Yes
- a Absorber material $B_4C$
- b Mode of operation (active/passive) Active/Passive
- c Redundancy Yes
- d Safety graded Yes

81 Control rods (Y/N)
- a Maximum control rod worth (pcm)
- b Mode of operation (active/passive) Active/passive
- c Redundancy Yes
- d Safety graded Yes

**A3. Decay heat removal**

*A3-1 Primary side*

82 Water injection
- a Actuation mode (manual/automatic) Manual
- b Injection pressure level (bar) 23
- c Flow rate (kg/s) 3
- d Mode of operation (active/passive) Active
- e Redundancy 3 x 100%
- f Safety graded (Y/N) Yes

83 Water recirculation and heat removal
- a Intermediate heat sink (or heat exchanger) 3 IHX
- b Mode of operation (active/passive) Passive
- c Redundancy 3 x 100%
- d Self sufficiency (h) 24 through each train
- e Safety graded Yes

*A3-2 Secondary side*

84 Feed water intermediate coolant
- a Actuation mode (manual/automatic) Automatic
- b Flow rate (kg/s)
- c Mode of operation (active/passive) Passive
- d Redundancy 3 x 100%
- e Self sufficiency (h) 24 through each train
- f Safety graded Yes

85 Water recirculation and heat removal
- a Ultimate heat sink (cold source) Water tanks and air
- b Mode of operation (active/passive) Passive
- c Redundancy 3 x 100%
- d Self sufficiency (h) 24 through each train
- e Safety graded Yes

*A3-3 Primary pressure control*

86 Implemented system (Name) PCPS, PRHRS
- a Actuation mode (manual/automatic) Automatic
- b Side location (primary/secondary circuit) Primary and sec
- c Maximum depressurization rate (bar/s)
- d Safety graded Yes

## B. SEVERE ACCIDENT CONDITIONS*

**B.1 Fission products retention**

87 Containment spray system (Y/N) Yes

88 F P Sparging (Y/N)

89 Containment tightness control (Y/N) Yes

90 Leakage recovery (Y/N)

91 Risk of recriticality (Y/N) No

* All systems must be qualified to operate under the accident conditions

**B.2 Recriticality control**

92. Encountered design feature: Boron injection system
   a. Mode of operation (A/P): Active/Passive
   b. Safety graded: Yes

**B.3 Debris confining and cooling**

93. Core debris configuration (core catcher): GV
94. Debris cooling system (name)
   a. Mode of operation (A/P): Passive
   b. Self sufficiency
   c. Safety graded (Y/N): Yes

**B.4 Long term Guard Vessel heat removal**

95. Implemented system: Reactor cavity flooding
   a. Mode of operation (A/P): Active
   b. Self sufficiency (h)
   c. Safety graded (Y/N): No
96. Intermediate heat sink
   a. Self sufficiency (h)
   b. Safety graded (Y/N)
97. External coolant recirculation
   a. Implemented components
   b. Mode of operation (A/P)
   c. Self sufficiency (h)
   d. Safety graded (Y/N)
98. Ultimate heat sink: Ambient air
   a. Self sufficiency (h)
   b. Safety graded (Y/N)

**B.5 Combustible gas control**

99. Covered range of gas mixture concentration
100. Modes for the combustible gas control
   a. Containment inertation
   b. Gas burning
   c. Gas recombining
   d. Others

**B.6 Containment pressure control**

101. Filtered vented containment (Y/N): Yes
   a. Implemented system
   b. Mode of operation (A/P): active
   c. Safety graded
102. Pressure suppression system (Y/N): yes
   a. Implemented system: Spray system
   b. Mode of operation
   c. Safety graded (Y/N): Yes

**C. SAFETY RELATED I&C SYSTEM**

Automatic load following (Y/N): Yes
* range (% power): 10-100
* maximum rate (%/min): 0,06

Load rejection without reactor trip (Y/N): Yes
Full Cathode Ray Tubes (CRT) display (Y/N): Yes
Automated start-up procedures (Y/N): No
Automated normal shutdown procedures (Y/N): Yes
Automated off normal shutdown procedures (Y/N): Yes
Use of field buses and smart sensors (Y/N)
Expert systems or artificial intelligence advisors (Y/N): Yes
Protection system backup (Y/N): Yes

## D. EMERGENCY POWER SUPPLY SYSTEM

103. Type (diesel, gas, grid connection): Diesel, Battery
104. Number of trains: 2

## E. AC/DC SUPPLY SYSTEM

105. Type (rectifier, converter, battery): rectifier, battery
106. Estimated time reserve (hr): 168

# IV. CONVENTIONAL THERMAL CYCLE - Not relevant

## A. TURBINE SYSTEM

107. Type: Steam turbine with uncontrolled steam bleeds
108. Overall length (m)
109. Width (m)
110. Number of turbines/reactor
111. Number of turbine sections per unit (e.g. HP/LP/LP)
112. Speed (rpm)

## B. STEAM CHARACTERISTICS - Not relevant

113. H.P. inlet pressure
114. H.P. inlet temperature
115. H.P. inlet flowrate
116. L.P. inlet pressure
117. L.P. inlet temperature
118. L.P. inlet flowrate (per section)

## C. GENERATOR - Not relevant

119. Type (3-phase synchronous, DC)
120. Apparent power (MVA)
121. Active power (MW)
122. Frequency (hz)
123. Output voltage (kV)
124. Total generator mass (t)
125. Overall length
126. Stator housing outside diameter

## D. CONDENSER - Not relevant

127. Number of tubes
128. Heat transfer area
129. Flowrate ($m^3/s$)
130. Pressure (m/bar)
131. Temperature (°C)

## E. CONDENSATE PUMPS - Not relevant

132. Number
133. Flowrate
134. Developed head
135. Temperature
136. Pump speed

### 5.5.5. Project status

The General Designer of the reactor plant of the AST-500 NDHP is OKB Mechanical Engineering, Nizhny Novgorod. The OKBM is a large design and scientific-research centre with production and technological basis for the development of various nuclear power plants.

The General Architect-Engineer of the AST-500 NDHP is "Atomenergoprojekt" Institute, Nizhny Novgorod.

The General Manufacturer of Equipment is the Production Association "Atommash", Volgodonsk.

The project for the reactor unit and the rest of the plant has reached the detailed design stage.

Experimental investigations are being performed related to the special equipment of the plant and specific processes.

Deterministic thermo-physical analysis of emergency conditions as well as probabilistic estimates of safety have been performed.

AST-500 reactor plants have been designed, fabricated and partly erected on Gorky and Varonezh NDHPs.

The production capability and technological potential of Russia permits the delivery of the whole complex of NDHP equipment to the Customer.

As concerns the level of nuclear and radiological safety, the NDHP units meet the international codes and standards and follow the IAEA recommendations.

When developing the AST plant a detailed assessment of design solutions and their practical realization was conducted by an international review team under IAEA auspices (Pre-OSART mission on Gorky NDHP, 1989).

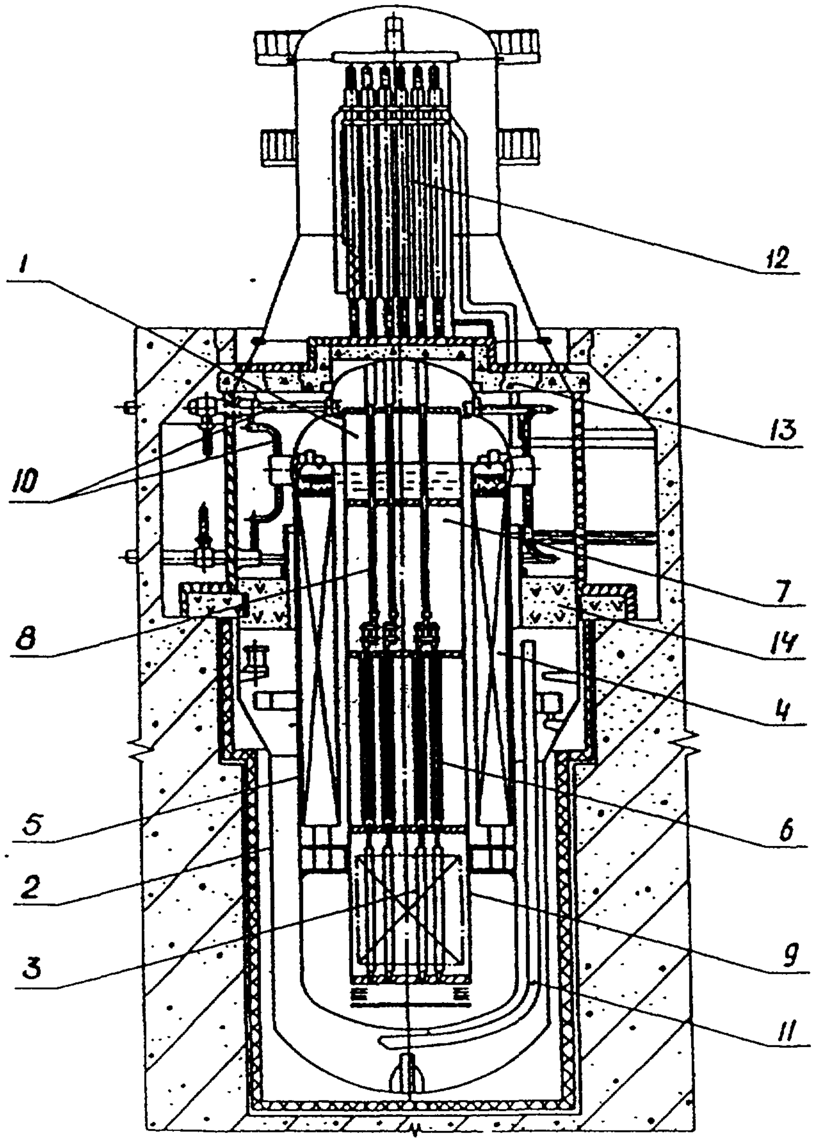
Licensing status. The design is being considered by the state regulatory body.

### 5.5.6. Project economics

The economic attraction of the AST-500 reactor plant results from:

- saving of fossil fuel (up to 800 thousand equivalent tonnes of fuel for two power units);
- effective utilization of the energy produced (up to 95%);
- long service-life of main equipment (60 years), high life-time and reliability of equipment and systems;
- economic nuclear fuel cycle compared to the fossil fuel cycle.

The cost of heat production is approximately 9.0 roubles/Gcal (in 1984 prices).




1. Steam / gas pressurizer
2. Guard vessel
3. Core
4. Primary / secondary heat exchanger
5. Reactor vessel
6. Control rod tie tubes
7. Riser section
8. Guide tubes - connecting devices assembly
9. Core barrel
10. Secondary pipelines
11. RV check rotation device
12. CPS drives
13. Biological shielding units
14. Support ring

*Fig 5.5.1. AST-500 Reactor Unit*

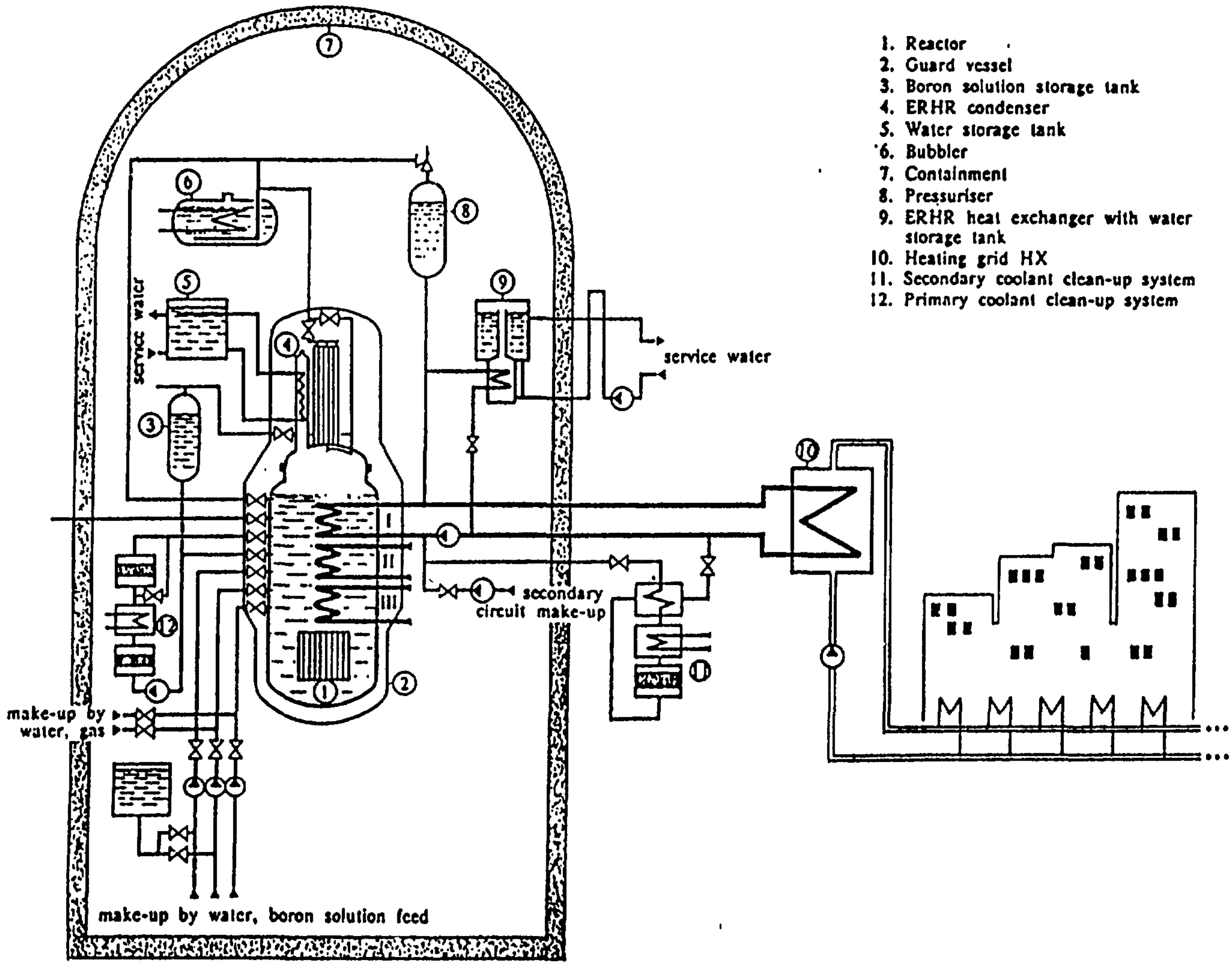



*Fig 5.5.2. AST-500 Principal Flow Diagram*

## 5.6. KLT-40 NUCLEAR STEAM SUPPLY SYSTEM DESCRIPTION OF REACTOR SYSTEMS AND DEVELOPMENT STATUS

### 5.6.1. Basic objectives and features

The KLT-40 is a twin-reactor system intended to produce fresh water and electric power in different proportions. It may also be used for heat production in a co-generation cycle. The KLT-40 design is based entirely on the serially produced marine NSSS being used in the Russian nuclear-powered ice-breakers.

The KLT-40 has the following original features:

1. Primary piping length is minimized;
2. Natural circulation is used in the primary and secondary circuits for all emergency modes.
3. The containment is designed for high over pressure and includes a passive pressure suppression system.
4. Safety is enhanced through fine-tuning of the engineered features proven by operation of the NSSS-prototype and by the use of systems which do not require external power sources.

### 5.6.2. Design description

*5.6.2.1. Nuclear Steam Supply System (NSSS) (see Fig.5.6.1.)*

Reactor vessel

The reactor is a PWR type with forced circulation through the primary circuit. Reactor vessel design pressure and temperature are 16.2 MPa and 350°C respectively. The reactor vessel is made of thermal resistant, high strength feritic steel with corrosion-resistant cladding. The reactor vessel is 3.89 meters high and 2.22 m in diameter.

All reactor nozzles are located in the upper part of the vessel, with restrictors in small diameter nozzles to mitigate an accident caused by guillotine rupture of any primary circuit pipeline.

The reactor vessel lifetime is determined by the allowable neutron fluence ($3.10^{20}$ n/cm$^2$) and could reach 40 years.

Core

The reactor core is a heterogeneous structure that includes 241 fuel assemblies (FAs) and control members (CM) of the reactor control and protection system (CPS). There are 5 control or compensation groups (CG) and 4 groups of emergency protection (EP) rods. Each CG and EP group is controlled by an individual drive. The Core diameter is 1212 mm, height is 0.950 m, and the specific power density is 155 kW/l. The FA structure and manufacturing technology are proven, and their reliability has been verified by long-term operation of similar cores.

The neutron-physical characteristics and the efficiency of the reactivity control system are such that at any moment in the reactor life cycle, cold subcriticality, with no dissolved boron is assured, even in the case of the most effective rod being stuck in its upper position. The liquid absorber injection system is used only in beyond design accidents.

The inherent safety properties of the core cause negative reactivity addition in the event of increase in coolant and fuel temperature or thermal power.

The absence of boron control simplifies the plant layout and improves the reactor self-regulation properties. The use of burnable poison decreases reactivity margin which has to be compensated by the mechanical control system.

Steam generators

The steam generator (SG) is a once through coil heat exchanger intended both to generate steam and to remove residual heat after reactor shutdown. The SG can produce up to 65 t/h of superheated steam. The pressure at the SG outlet is 3.5 MPa, and the steam temperature is 290°C. The SG tubes are made of corrosion-resistant titanium alloy. There are four separate heat exchangers.

Reactor coolant pump

There are four reactor coolant pumps in the NSSS.

Each pump is a centrifugal single-stage glandless one with a canned two-speed synchronous motor intended to circulate primary coolant. The pump capacity is 870 $m^3/h$, the head is 0.38 MPa (at high speed) and the consumed power is not more than 155 kW. Leaks are completely excluded due to the pump design. Continuous gas removal from the upper part of the pump is provided during operation.

*5.6.2.2. Balance of plant systems*

Steam turbine plant (STP)

The steam turbine plant (STP) is intended to provide electric energy to consumers and to transfer thermal energy from the NSSS to a distillation desalination plant (DDP). Each reactor plant operates together with its own STP. The secondary circuit of each STP includes 4 SGs, 2 turbo-generators (TGs) with condensate pumps, main and standby feed pumps, two process condensers with condensate plants, tanks, etc.

In the steam condensers up to 35 MW of thermal energy is transferred to an intermediate circuit. The pressure in the condenser is 0.5 MPa.

The intermediate circuit is intended to receive and transfer thermal energy from the condensers to the DDP SG. Some DDP parameters are: pressure in the circuit is 1.0 MPa, hot leg and cold leg temperatures are 140°C and 125°C, respectively. Over pressure is maintained in the intermediate circuit which excludes radioactivity ingress into it.

*5.6.2.3. Instrumentation, control and electrical systems - Process control system*

The process control system (PCS) is intended to provide for automatic and remote control and protection of the KLT-40 power unit equipment, to monitor key parameters and to present information to the operator during all operation modes.

The PCS concept is based on experience from the development and operation of similar systems, giving high priority to nuclear and radiological safety. It ensures high reliability and system survivability even with equipment failures and operator errors and always provides adequate information to the operators. The system ensures presentation of information in a form convenient for the operators, optimizing the "man-machine" interface.

The concept of the PCS structure for water desalination plants is based on advanced technology and on feedback from experience; it includes all the latest developments in the area of plant operation simplification and minimizing human involvement through enhanced automation and use of modern displays.

*5.6.2.4. Safety consideration and emergency protection*

Safety features ensured by the physical processes

The inherent safety features of the KLT-40 NSSS are:

1. Self-protection, self-regulation and self-limitation of power due to negative reactivity coefficients over the whole range of reactor parameters variation.
2. Natural circulation through primary and secondary circuits ensuring heat removal from the core following the reactor shutdown.
3. The weight of CM and the energy accumulated in the compressed spring of the CM drives assure downward movement from any position at de-energization for reactor shutdown. It also excludes ejection of absorber rods from the core when racks or drive casing loose their tightness.
4. The large heat accumulation capability of the reactor gives the operator a large time margin to analyze emergencies and to organize accident management.
5. Use of passive safety systems.
6. Use of diverse safety systems and redundancy in safety system elements.
7. Wide use of self-actuated devices for initiation of safety system operation, including reactor shut down, when the most important safety related parameters exceed their design limits.
8. Use of the containment structure as a special system for keeping the core under water, and providing for passive heat removal from the reactor following primary circuit loss of integrity.

Passive systems

Passive safety response of the KLT-40 RP is ensured by features and systems that operate without external power sources. These are:

1. CPS CM gravity action and compressed spring energy for emergency reactor shutdown.
2. Passive systems for heat removal by natural circulation in all circuits from heat source to heat sink.

3. Hydroaccumulators in the emergency core cooling system.
4. The containment guard structure and other special systems ensures:
    a) emergency pressure decrease through use of a bubbler tank
    b) the core is kept under water
    c) passive heat removal from the reactor following a primary circuit break
5. Self-actuated devices to initiate operation of protective and confinement systems.

Passive residual heat removal system

The passive system for residual heat removal is intended to remove residual heat from the core following complete loss of power supply when the reactor operates at any power level. The system consists of two loops and operates without power sources to complete cool down without operator intervention.

Each loop includes a water tank of 50 $m^3$ volume, a 700 kW heat exchanger, a 600 kW air-cooled heat exchanger, an expansion tank, pipelines and valves.

Emergency core cooling system and long-term supply of cooling water

The system is intended to remove heat from the core following primary circuit loss-of-integrity accidents.

The operation of the system is based on active and passive principles. The system consists of two independent subsystems, one of them is passive (using hydroaccumulators) operating during the first hour following the accident initiation. The water and gas volumes in the hydroaccumulators are 16$m^3$ and 1$m^3$, respectively at an initial pressure of 8.0MPa.

The second subsystem includes three motor-driven pumps of 10 $m^3$/h capacity and of 3.0 MPa pressure head. Its water inventory is 50 $m^3$. Each system has two channels. A special system for recirculation of water from the condensate collectors back into the reactor is provided using glandless motor pumps. A special structure excludes loss of coolant and ensures long-term passive heat removal from the core and reactor vessel during beyond design base accidents.

The inventory of water in the hydroaccumulators and of coolant condensate is sufficient (provided the circulation is closed) to maintain the water level above the cover of the metal-water shielding tank, ensuring residual heat removal from the reactor to its complete cool-down.

A two-channel feedwater system with three pumps of 1.2 t/h capacity each, and a head up to 25 MPa is provided to compensate for small leaks and, if necessary, to inject liquid absorber into the reactor.

Automated depressurization system

The system is intended to prevent inadmissible increase of pressure in the primary circuit if all subsystems for residual heat removal fail following the reactor shutdown. Primary circuit emergency pressure decrease is provided by two systems. The first system consists of two sets of safety devices (SD) and of a 1.6 $m^3$ cooled dump tank. When the primary circuit pressure reaches 19.6 MPa the SD actuates automatically and discharges a portion of coolant into the dump tank. The system can be used repeatedly following dump tank drainage. Automatic membrane safety devices (ASD) are provided to protect the primary circuit against damage in case of further pressure increase following actuation of the first system. The ASD actuates at pressure exceeding the limit pressure (34 MPa). In this case excess coolant is discharged from the upper part of the dump tank into the bubbler tank. Following pressure decrease the primary circuit tightness is restored by closing the safety valve of the first system.

Containment as a passive system for heat removal

The containment is a passive confinement and protective device of the safety system and performs the following functions:

1. confining radioactive coolant following accidents with primary circuit breaks as well as localization of steam following secondary circuit loss of tightness;
2. keeping the core under coolant following primary circuit loss of integrity.

The containment is provided with a system that prevents it from being damaged in the event of a primary circuit loss-of-integrity accident. The system consists of a 70 $m^3$ bubbler tank with 12 $m^3$ gas volume and bypass channels with a safety valve communicating with different containment rooms.

Safety valve actuation occurs at a pressure differential from 0.13 to 0.17 MPa (gauge).

*5.6.2.5. Buildings and structures*

The reactor with all its equipment has been developed as an ice-breaker propulsion reactor.

### 5.6.3 SAFETY CONCEPTS

TABLE 5.6.1. MAIN SAFETY RELATED SYSTEMS IN THE KLT-40

| Name | Safety graded | Main characteristics |
|---|---|---|
| Primary Circuit (PC) | X | Reactor vessel, 4 steam generators, 4 centrifugal circulating pumps |
| Reactivity control and protection system (CPS) | X | 5 reactivity compensation rod groups, 4 emergency protection groups |
| Liquid absorber injection system (ECCS) | X | Liquid absorber injection, 3 pumps for boron solution injection |
| Emergency core cooling system (PCS) | X | 2 hydroaccumulators, 3 injection pumps |
| Primary circuit pressure control system (PCS) | X | Overpressure protection, PORV |
| Passive Core cooling system (CCS) | X | 2 water storage tanks, 2 air heat exchangers |
| Containment emergency pressure decrease system (CEPS) | X | Containment, bubbler tank, safety valves |
| Containment flooding system (CFS) | X | Flooding valves, Distribution device |
| Secondary circuit cooling system (SCFS) | X | Main condenser |
| Tertiary circuit cooling system (TCCS) | X | Cooler-filter |

TABLE 5.6.2. MAIN ACCIDENT INITIATORS FOR THE KLT-40

- LOCA (primary): Loss of primary coolant accident
- LOCA (Secondary): Secondary pipe rupture (water or steam)
- LOCA (Interfacing): e.g.: SGTR Steam generator tube rupture
- ATWS: Anticipated transients without scram
- Primary transients
- Secondary transients (turbine trip)
- Loss of electric sources (all AC sources)
- Total loss of the cold sources
- Total loss of the steam generator feedwater
- Station blackout.

TABLE 5.6.3. DESIGN FEATURES FOR PREVENTION AND PROTECTION LEVELS THAT REDUCE (R), SUPPRESS (S), THE INITIATORS FREQUENCY OR LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES (L)

| | |
|---|---|
| **PREVENTION LEVEL** | |
| LOCA (Primary) - | Reduced reactor vessel fluence |
| | - Double isolation valves in pipelines of primary circuit |
| | - Isolation valves of containment ventilation system |
| LOCA (Secondary) | - Containment |
| | - Isolation valves in steam pipelines |
| LOCA (Interfacing) - | Double isolation valves in steam, feed water, cooling water pipelines of the tertiary circuit for primary circuit pump, cooler-filter, cooling system pump |
| Primary transient | Safeguard system of liquid absorber injection |
| - | Simple reliable structures of primary circuit pump, cooling system pump, primary circuit valves, cooler-filter |
| Secondary transients - | Simple reliable structure of steam generator |
| Loss of electric sources | - Passive system of residual heat removal (air) |
| Total loss of the source (Water) | - Passive system of residual heat removal (air) cooler-filter |
| Total loss of the SG feedwater | - System of heat removal by means of tertiary water through cooler of purification system |
| Station blackout | |
| **PROTECTION LEVEL** | |
| LOCA (Primary) - | Emergency core cooling system, - Containment |
| | - Restrictors in small nozzles |
| | - Double isolation valves of primary circuit |
| | - Bubbler tank, - Passive system of residual heat removal |
| LOCA (Secondary) | - Double isolation valves of steam generator |
| LOCA (Interfacing) | - Double isolation valves of steam generator, primary circuit pump, cooling system pump |
| ATWS | |
| Primary transients | |
| Secondary transient | |
| Loss of electric sources - | Storage battery, - operation of power sources on residual steam |
| Total loss of heat sink | Non-critical |
| Total loss of SG feedwater | Non-critical |
| Station blackout | Non-critical |

TABLE 5.6.4. DESIGN FEATURES FOR MITIGATION LEVEL OF KLT-40

| Safety functions | Systems (Cf. Tab. 5.6.1.) | Passive/active | Design features/remarks |
|---|---|---|---|
| **Design Basis**<br>Fission product containment | Containment, CEPS using bubbler tank | Passive/active | |
| Coolant inventory | Bubbling system<br>Injection system | Passive<br>Active | |
| Decay heat removal | Secondary circuit cooling system (using main condenser)<br>Tertiary circuit cooling system (using cooler)<br>ECCS (using heat exchanger) | Active<br>Active<br>Passive | |
| Reactivity control | CPS<br>LAIS | Active/passive<br>Active | |
| Primary circuit pressure control | PCS | Passive | |
| **Severe accident**<br>Containment temperature and pressure control | CEPS | Active | |
| Heat removal | ECCS<br>TCCS | Active/passive<br>Passive | |
| Tightness control | | | |
| Inflam. gas control | | | |
| Fission product containment | Bubbling system<br>Injection system | Passive<br>Active | |
| Corium management | | | |
| Others | | | |

### 5.6.4. Design data questionnaire (Water Cooled Reactors for KLT-40)

**I. GENERAL INFORMATION**

1. Design name: KLT-40
2. Designer/Supplier address: OKBM
3. Reactor type: PWR    Number of modules/per plant: 2
4. Gross thermal power (MW-th) per reactor: Up to 160
5. Net electrical output (MW-e) per reactor: Up to 35
6. Heat supply capacity (MW-th): Up to 220 G cal/hr at power decrease down to 0

**II. BASIC DESIGN DESCRIPTION**

**A. CORE AND REACTIVITY CONTROL**

7. Fuel material: Uranium-aluminium alloy
8. Fuel inventory (tones of heavy metal): 199.59 Kg U-235
9. Average core power density (kW/liter): 155
10. Average fuel power density (kW/kgU): 480
11. Maximum linear power (W/m)
12. Average discharge burnup (MWd/t)
13. Initial enrichment or enrichment range (Wt%)
14. Reload enrichment at the equilibrium (Wt%)
15. Refueling frequency (months): 24-36
16. Type of Refueling (on/off power): off power
17. Fraction of core withdrawn (%): 100
18. Moderator material and inventory: Water
19. Active core height (m): 950 mm
20. Core diameter (m): 12 12 mm
21. Number of fuel assemblies: 241
22. Number of fuel rods per assembly: 54 (51)
23. Rod array in assembly
24. Clad material: Zirconium alloy
25. Clad thickness (mm)
26. Number of control rods or assemblies: 5
27. Type
28. Additional shutdown systems: Liquid Absorber System
29. Control rod neutron absorber material: Boron and rare earth metals
30. Soluble neutron absorber
31. Burnable poison material and form: Rods on gadolinium basis

**B. REACTOR COOLANT SYSTEM**

**B1. Coolant**

32. Coolant medium and inventory: Water
33. Design coolant mass flow through core (kg/s): 722
34. Cooling mode (forced/natural): Forced
35. Operating coolant pressure (bar): 130
36. Core inlet temperature (°C): 278
37. Core outlet temperature (°C): 318

**B2. Reactor pressure vessel**

38. Overall length of assembled vessel/tube (m): 3891 mm
39. Inside vessel/diameter (m/mm): 2220 mm
40. Average vessel/tube thickness (mm): 116.5
41. Vessel/tube material: Thermal resistant steel
42. Lining material: Vessel steel with corrosion resistant facing
43. Design pressure (bar): 165 KGf/cm$^2$
44. Gross weight (tone/kg): 72000 Kg

**B3. Steam generator**

45. Number of steam generators: 4
46. Type: Once-through, coil
47. Configuration (horizontal/vertical): Vertical
48. Tube material: Corrosion-resistant metal
49. Shell material: Low-alloyed thermal resistant steel
50. Heat transfer surface per steam generator ($m^2$)
51. Thermal capacity per steam generator (MW)
52. Feed water pressure (bar)
53. Feed water temperature (°C): 170
54. Steam pressure (bar): 3.5 MPa
55. Steam temperature (°C): 290

**B4. Pressurizer**

56. Pressurizer total volume ($m^3$): 8.28
57. Steam volume (full power/zero power, $m^3$): 3.15

**B5. Main coolant pumps**

58. Number of cooling or recirculation pumps: 4
59. Type: Centrigual, Single-stage, Glandless with two-speed asynchronous motor
60. Pump mass flow rate (kg/s): 870 $M^3$/h and 290 $M^3$/h
61. Pump design rated head
62. Pump nominal power (kW): 155 at high speed and 13 at low speed
63. Mechanical inertia (kg $m^2$)

**C. CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS)**

64. Number of extraction lines
65. Number of pumps: 2
66. Number of injection points: 1
67. Feed and bleed connections

**D. CONTAINMENT**

68. Type: Independent, steel
69. Overall form (spherical/cyl.): Cylindrical
70. Structural material: Steel
71. Liner material
72. Simple/double wall: Simple
73. Dimensions (diameter, height) (m): 11, 18
74. Design pressure (bar): 0,4 MPa (not more)
75. Design temperature (°C): 145°C (not more)
76. Design leakage rate (% per day): 1% (not more)

## III. SAFETY RELATED SYSTEMS

**A. DESIGN CONDITIONS**

**A1. Fission product retention**

77. Containment spray system (Y/N): Yes
    a. Duration (h): ~ 20
    b. Flow rate ($m^3$/h): 4.5
    c. Mode of operation (active/passive): Passive/Active
    d. Safety graded (Y/N): Yes
78. F.P. sparging (Y/N): Yes
79. Containment tightness control (Y/N)
80. Leakage recovery (Y/N)
81. Guard vessel (Y/N)

**A2. Reactivity control**

82. Absorber injection system (Y/N)
    a. Absorber material: Boron and rare earth metals
    b. Mode of operation (active/passive): Passive
    c. Redundancy: Yes
    d. Safety graded: Yes

83 Control rods (Y/N) Yes
a Maximum control rod worth (pcm) 15%
b Mode of operation (active/passive) Active/passive
c Redundancy Yes
d Safety graded Yes

**A3. Decay heat removal**

*A3-1 Primary side*

84 Water injection
a Actuation mode (manual/automatic) automatic
b Injection pressure level (bar) 8 0 MPa
c Flow rate (kg/s)
d Mode of operation (active/passive) Passive
e Redundancy Yes
f Safety graded (Y/N) Yes

85 Water recirculation and heat removal
a Intermediate heat sink (or heat exchanger)
b Mode of operation (active/passive) Active
c Redundancy Yes
d Self sufficiency (h)
e Safety graded Yes

*A3-2 Secondary side*

86 Feed water
a Actuation mode (manual/automatic) Automatic
b Flow rate (kg/s)
c Mode of operation (active/passive) Active/Passive
d Redundancy Yes
e Self sufficiency (h) to complete cooling
f Safety graded Yes

87 Water recirculation and heat removal
a Ultimate heat sink (cold source)
b Mode of operation (active/passive) Passive
c Redundancy Yes
d Self sufficiency (h) to complete cooling
e Safety graded Yes

*A3-3 Primary pressure control*

88 Implemented system (Name)
a Actuation mode (manual/automatic) Automatic
b Side location (primary/secondary circuit) Primary Circuit
c Maximum depressurization rate (bar/s)
d Safety graded Yes

## B. SEVERE ACCIDENT CONDITIONS*

**B.1 Fission products retention**

89 Containment spray system (Y/N) Yes
90 F P Sparging (Y/N) Yes
91 Containment tightness control (Y/N)
92 Leakage recovery (Y/N) Yes
93 Risk of recriticality (Y/N) No

**B.2 Recriticality control**

94 Encountered design feature
a Mode of operation (A/P) Active
b Safety graded Yes

**B.3 Debris confining and cooling**

95 Core debris configuration (core catcher)
96 Debris cooling system (name) Metal water shielding tank
a Mode of operation (A/P) Passive
b Self sufficiency Unlimited
c Safety graded (Y/N) Yes

---

* All systems must be qualified to operate under the accident conditions

**B.4 Long term Guard Vessel heat removal**

97. Implemented system
    a. Mode of operation (A/P): Passive
    b. Self sufficiency (h): To complete cooling
    c. Safety graded (Y/N): Yes
98. Intermediate heat sink
    a. Self sufficiency (h): to complete cooling
    b. Safety graded (Y/N)
99. External coolant recirculation
    a. Implemented components
    b. Mode of operation (A/P): Passive
    c. Self sufficiency (h): to complete cooling
    d. Safety graded (Y/N)
100. Ultimate heat sink
    a. Self sufficiency (h)
    b. Safety graded (Y/N): Yes

**B.5 Combustible gas control**

101. Covered range of gas mixture concentration : very low

102. Modes for the combustible gas control
    a. Containment inertation
    b. Gas burning — No
    c. Gas recombining
    d. Others

**B.6 Containment pressure control**

103. Filtered vented containment (Y/N): yes
    a. Implemented system
    b. Mode of operation (A/P): Active
    c. Safety graded: Yes

104. Pressure suppression system (Y/N): Yes
    a. Implemented system
    b. Mode of operation: Passive
    c. Safety graded (Y/N): Yes

**C. SAFETY RELATED I&C SYSTEM**

Automatic load following (Y/N): Yes
* range (% power)
* maximum rate (%/min)

Load rejection without reactor trip (Y/N)
Full Cathode Ray Tubes (CRT) display (Y/N): Yes
Automated start-up procedures (Y/N)
Automated normal shutdown procedures (Y/N)
Automated off normal shutdown procedures (Y/N)
Use of field buses and smart sensors (Y/N)
Expert systems or artificial intelligence advisors (Y/N)
Protection system backup (Y/N)

**D. EMERGENCY POWER SUPPLY SYSTEM**

105. Type (diesel, gas, grid connection)
106. Number of trains

**E. AC/DC SUPPLY SYSTEM**

107. Type (rectifier, converter, battery)
108. Estimated time reserve (hr)

**IV. CONVENTIONAL THERMAL CYCLE**

**A. TURBINE SYSTEM**

109. Type: Steam turbine with uncontrolled steam bleeds

110 Overall length (m)
111 Width (m)
112 Number of turbines/reactor
113 Number of turbine sections per unit (e g HP/LP/LP)
114 Speed (rpm)

## B. STEAM CHARACTERISTICS

115 H P inlet pressure 3 2 MPa
116 H P inlet temperature 285 °C
117 H P inlet flowrate
118 L P inlet pressure
119 L P inlet temperature
120 L P inlet flowrate (per section)

## C. GENERATOR

121 Type (3-phase synchronous, DC)
122 Apparent power (MVA)
123 Active power (MW)
124 Frequency (hz)
125 Output voltage (kV)
126 Total generator mass (t)
127 Overall length
128 Stator housing outside diameter

## D. CONDENSER

129 Number of tubes
130 Heat transfer area
131 Flowrate ($m^3/s$)
132 Pressure (m/bar)
133 Temperature (°C)

## E. CONDENSATE PUMPS

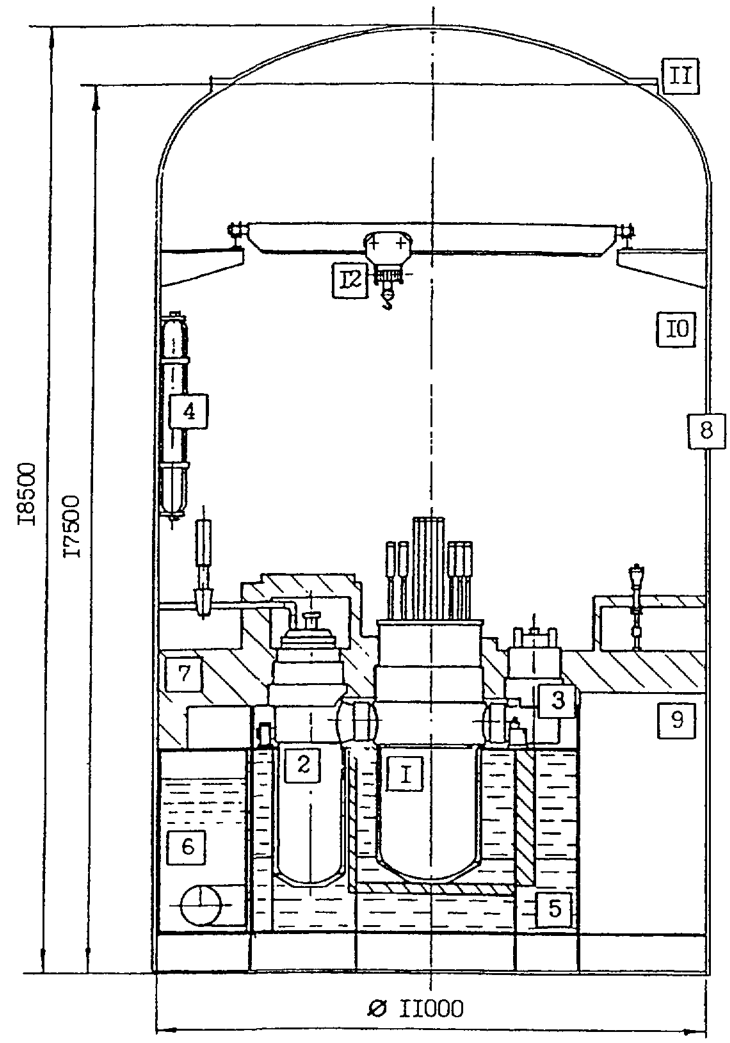
134 Number
135 Flowrate
136 Developed head
137 Temperature
138 Pump speed

## 5.6.5. Project Status

The KLT-40 plant is serially produced and has been operating in nuclear ice-breakers and in the"Sevmorput" lighter-containership with an accumulated experience 140 of reactor-years, therefore no R&D is needed.

## 5.6.6. Project Economics

The cost of the KLT-40 reactor plant including main equipment, metal-water shielding tank, biological shield and protective shell (containment) according to the actual data for Russian-Finland construction of the nuclear ice-breaker "Taimyr" is 30 million US dollars.




1. Reactor
2. Steam generator
3. Electric pump of primary circuit
4. High pressure gas system cylinders
5. Metal-water shielding tank
6. Bubbler tank of the emergency pressure suppression system
7. Biological shield
8. Containment
9. Pipelines cofferdam
10. Mechanisms servicing compartment
11. Hatch cover
12. Crane

*Fig. 5.6.1. Reactor Plant lay-out in the containment*

1. Emergency protection rod
2. Reactor head post
3. Head
4. Wedge sealing gasket
5. Reactor pressure vessel
6. Removable unit
7. Compensating group rod
8. Lateral shield
9. Compensating group plate
10. Core fuel assemblies
11. Compensating group rod
12. Inlet filter

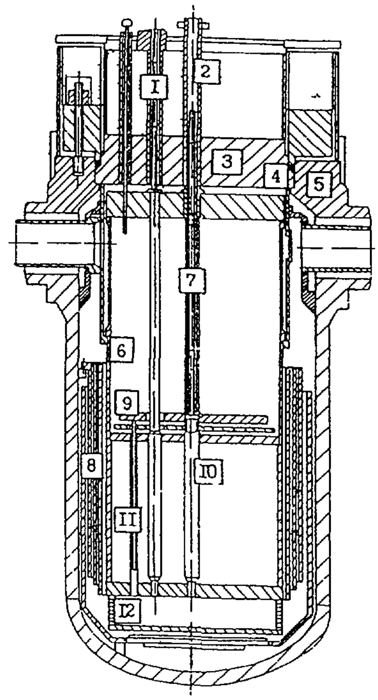



*Fig 5.6.2. Reactor*

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## 5.7. CANDU 6 REACTOR SYSTEM AND DEVELOPMENT STATUS

### 5.7.1. Basic objectives and features

#### *5.7.1.1. Basic objectives*

The basic design objectives of CANDU 6, when developed in the early 1970's, was to create a safe, reliable, economic and robust nuclear power plant with a net electrical output in the range of 600 MW. The performance of the first four CANDU 6 plants, located in two Canadian provinces (Quebec and New Brunswick) and two foreign countries (Argentina and the Republic of South Korea) and which have a total of over forty years of safe and economic reactor operation at average capacity factor of over 84%, confirms that the design objectives were met.

Evolutionary improvement have been incorporated in the CANDU 6 since the first units entered service, with the basic objectives of further enhancing safety, reliability, and overall economics. The CANDU 6 plants now under construction in the Republic of South Korea for example include over one hundred significant improvements in addition to changes to comply with current codes and standards and licensing requirements, relative to the operating CANDU 6 in that country. The net electrical output of the first CANDU 6 units is in the range of 640 MW; this has been increased to about 660 MW in the newer units, largely through improvements made in the turbine generator and Balance of Plant.

Evolutionary improvements continue to be incorporated in the CANDU 6 design, taking advantage of CANDU operating experience, AECL research and development, and technical advances world-wide, in order to further enhance safety, reliability and economics.

#### *5.7.1.2. Basic design features*

The CANDU 6 incorporates all of the basic and well-proven features which are the hallmark of CANDU. These include:

- a reactor consisting of small diameter horizontal pressure tubes housed in a low pressure, low temperature moderator-filled calandria (tank)
- heavy water ($D_2O$) for moderator and reactor coolant
- the standard CANDU 37-element CANDU fuel bundle, and the ability to operate on natural uranium or other low fissile content fuel.
- on-power refuelling, to eliminate the need for refuelling outages
- two diverse, passive, fast-acting and fully capable shutdown systems which are independent of each other, and of the reactor regulating system.
- automated digital control of all key Nuclear Steam Plant and Balance of Plant functions.
- the total absence of all chemicals in the reactor coolant (Heat Transport System) for reactivity control.

### 5.7.2. Design description

#### *5.7.2.1. Nuclear steam supply system*

5.7.2.1.1. General

The CANDU 6 Nuclear Steam Supply System is illustrated in Figure5.7.1. The following sections provide further detail on separate CANDU 6 nuclear steam plant systems.

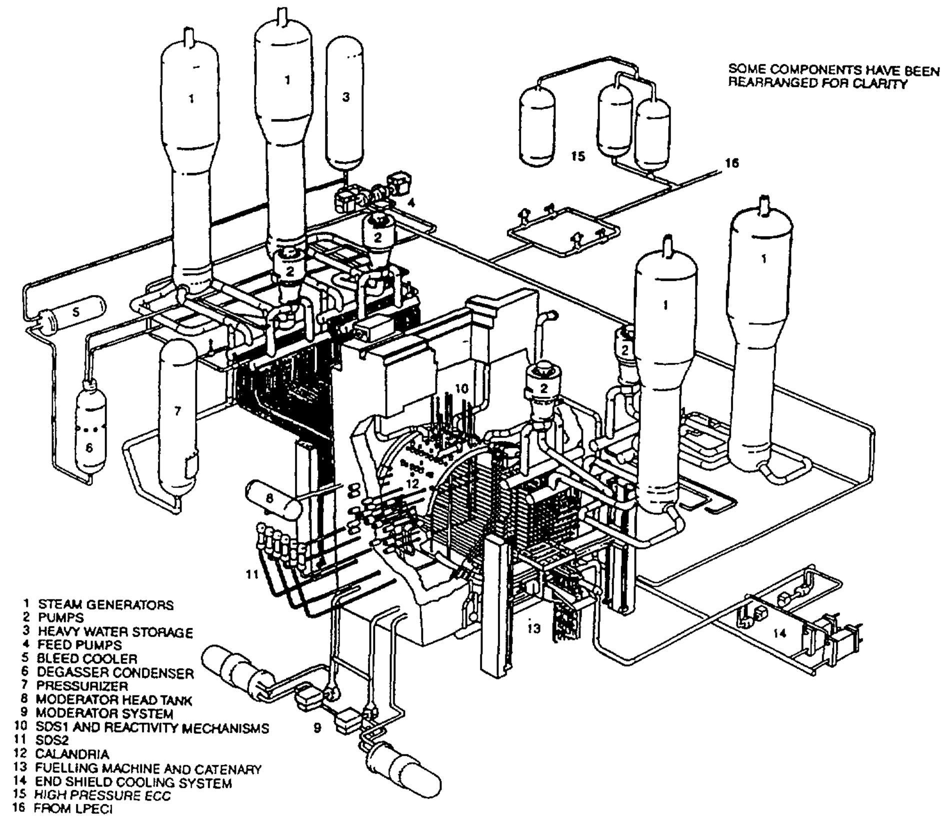



*Fig. 5.7.1. CANDU 6 Nuclear Steam Supply System*

5.7.2.1.2. Fuel

CANDU 6 uses the standard 37-element CANDU fuel bundle (see CANDU 3, Figure 5.8.1B). This fuel bundle is a very simple, easily-fabricated structure about 500 mm long and 100 mm in diameter consisting of 37 elements held together by end plates at each end of the bundle. Current CANDU 6 plants operated with natural uranium fuel (0.7% U235); they can, however, operate on a variety of other low fissile content fuels including slightly enriched uranium, and recovered uranium from PWR fuel reprocessing plants.

The use of natural uranium or other low fissile content fuel in CANDU 6 assures that there is no potential for the fuel (new or irradiated) to achieve criticality outside of the reactor regardless of the storage configuration.

5.7.2.1.3. Fuel channels

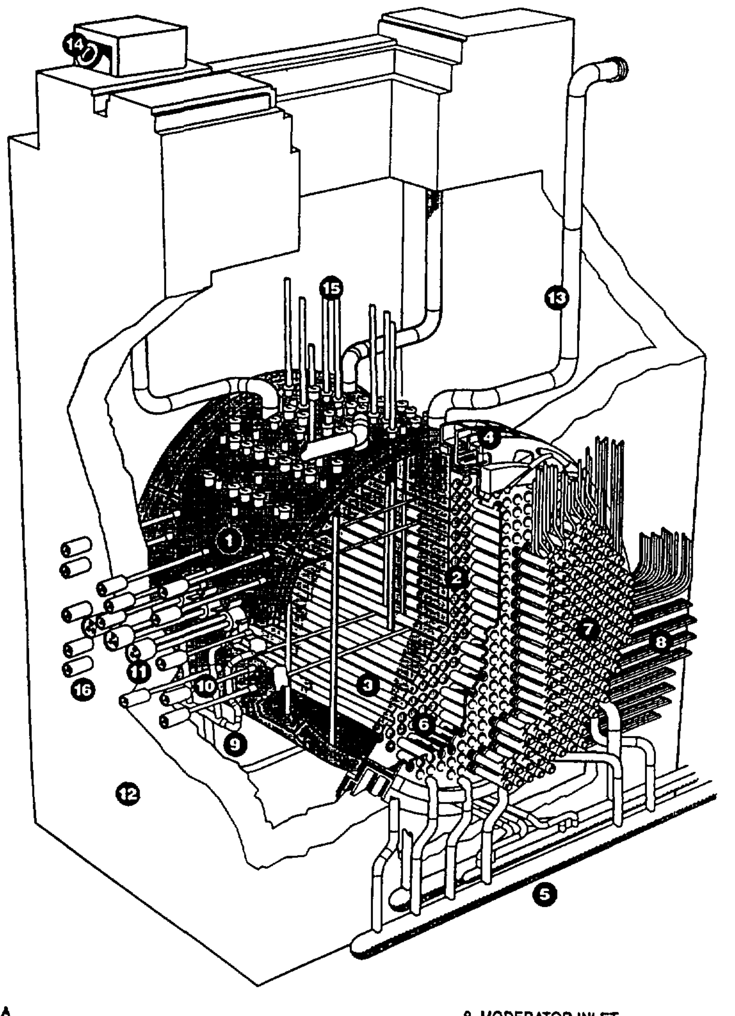
The CANDU 6 reactor includes 380 horizontal fuel channels. Each fuel channel consists of a zirconium-niobium alloy pressure tube about 6 metres long with an inside diameter of about 100 mm, centred in a Zircaloy calandria tube by annular spacers, and expanded into a stainless steel end fitting at both ends. Each channel contains twelve fuel bundles.

The pressure tubes, the only components in CANDU that are subject to high levels of radiation and stress, are readily replaced. However, pressure tube replacement is not required in new CANDU 6 plants for 35 or more years; the design of the fuel channels and the layout within the reactor building facilitate fuel channel replacement.

5.7.2.1.4. Calandria assembly

The reactor core consists of 380 fuel channels, held in a square lattice array by circular end-shields, and contained within a cylindrical low-pressure tank called the calandria (Figure 5.7.2.). The calandria contains the heavy water moderator at low temperature and at near-atmospheric pressure. The calandria is positioned within a low-pressure steel lined concrete vault, filled with ordinary water. This tank provides biological shielding from neutron and gamma radiation from the reactor core (Figure 5.7.2.).

The calandria is penetrated vertically and horizontally by flux measurement and reactivity control devices, and by the in-core components of two safety shutdown systems. All reactivity control devices function in the low-pressure moderator. No reactivity control devices penetrate



1 CALANDRIA
2 CALANDRIA END SHIELD
3 FUEL CHANNEL ASSEMBLIES
4 EMBEDMENT RING
5 END SHIELD COOLING PIPES
6 STEEL BALL SHIELDING
7 END FITTINGS
8 FEEDER PIPES
9 MODERATOR INLET
10 HORIZONTAL FLUX DETECTOR UNIT
11 ION CHAMBER
12 CALANDRIA VAULT
13 PRESSURE RELIEF PIPES
14 RUPTURE DISC
15 SHUTOFF AND CONTROL RODS
16 POISON INJECTION

*Fig 5.7.2. Reactor Assembly*

the heat transport system and no chemicals are added to the heat transport system for reactivity control.

5.7.2.1.5. Heat transport system

The heat transport system is subdivided into two independent circuits or loops, each including half (190) of the 380 fuel channels.

Each fuel channel is connected by individual inlet and outlet feeder pipes to distribution pipes, called headers, at both ends of the reactor. Four heat transport pumps, two in each loop, circulate the heavy water coolant from the inlet header, through the fuel channels, to the outlet header, through the steam generators (where the heat is transferred to ordinary water to generate steam), and back through the reactor again; the coolant flow in adjacent fuel channels is therefore in opposite directions (bi-directional). The arrangement is called a figure-of-eight arrangement, since the fluid flows through the core twice to make a complete circuit. This arrangement places two pumps in series, which limits the coolant flow reduction should one pump seize (Figure 5.7.1.). If all pumps are lost, fuel cooling is maintained by thermosyphoning flow. All heat transport system piping is carbon steel.

5.7.2.1.6. Moderator system

Heat is deposited in the heavy water moderator contained within the calandria during normal operation, principally from direct gamma and neutron interaction. This heat is removed by the moderator system, which circulates and cools the heavy water in an external circuit connected to the calandria; the heat is rejected to the recirculated coolant water system.

5.7.2.1.7. Fuel handling

On-power refuelling is performed by two fuelling machines, located at opposite ends of the reactor. These machines transport new fuel bundles to the reactor fuel channel to be refuelled, and load them into the fuel channel while the reactor is operating, while simultaneously removing used fuel bundles from the fuel channel. The fuelling machines subsequently transport the used fuel to the irradiated fuel transfer system which connects to the irradiated fuel storage bay. The refuelling operation is, once the channel to be refuelled is selected, fully automatic.

In the event that a defect occurs in a fuel bundle during reactor operation, the fuelling machines can be used to remove the defective fuel, thereby limiting the release of fission products to the heat transport system coolant. Systems are provided for the detection and location of defective fuel.

5.7.2.1.8. Reactor control

On-power refuelling provides the principal means for controlling reactivity in the CANDU 6. Additional reactivity control, independent of the safety shutdown systems, is achieved through use of reactivity control mechanisms. These include light-water zone compartments, absorber rods, and adjuster rods; all are located between fuel channels within the low pressure heavy water moderator and do not penetrate the heat transport system pressure boundary. The reactor is controlled by the dual redundant computer control system. The overall station control system is described in Section 5.7.2.3.

5.7.2.1.9. Safety systems

The CANDU 6 incorporates four special safety systems which consist of the two passive, diverse, dedicated reactor shutdown systems; the emergency core cooling system; and the containment system. Each is separated from and is independent of the normally operating plant systems, and of all the other safety systems.

Shutdown system No. 1 utilizes spring-assisted, gravity-drop neutron absorbing rods, which drop into the moderator, between the fuel channels.

Shutdown System No. 2 consists of horizontal perforated tubes through which a liquid neutron absorber is injected into the moderator utilizing high pressure helium.

The Emergency Core Cooling System uses high pressure gas to inject ordinary water into the fuel channels, followed by pumped recirculation and cooling of water within the reactor building.

The containment system includes the reactor building, which encloses the reactor and other nuclear steam supply system components, and the containment isolation system.

The initiation and operation of all special safety systems, if required, is fully automatic, based on diverse and redundant measurements. For example, two independent and diverse reactor trip (shutdown) signals are provided for each of the shutdown systems for every design basis accident requiring reactor shutdown.

*5.7.2.2. Balance of plant*

5.7.2.2.1. Turbine - Generator

The CANDU 6 turbine consists of a double-flow high pressure cylinder and three double flow low pressure cylinders that exhaust to individual condensers. The turbine-generator operates at 1800 rpm.

CANDU plants have operated with turbine-generators supplied by a variety of manufacturers including Parsons, GE, BBC and Hitachi. Turbine generators from different suppliers are readily accommodated by CANDU 6.

In the event of a loss of line, the turbine-generator on CANDU 6 runs back to sustain station loads. In addition, a turbine steam bypass system is provided (100% short-term, 70% long-term capacity) to reject steam directly to the condensers in the event that the turbine is unavailable; this allows reactor power to be sustained during the event.

5.7.2.2.2. Radioactive waste management

The CANDU 6 incorporates comprehensive systems for the management, disposal and storage of solid, liquid, and gaseous radioactive wastes. The irradiated fuel bays, located adjacent to the reactor building have sufficient capacity to accommodate the fuel from ten years of reactor operation; facilities are included for the transfer of irradiated fuel to dry storage. There is no potential for new or irradiated CANDU fuel to achieve criticality in air or ordinary water, regardless of the storage configuration.

*5.7.2.3. Instrumentation, control and electrical systems*

5.7.2.3.1. Information and control system

Plant instrumentation, computer control systems, control room man/machine interface and the plant information system are provided by the ICS 90+ Systems (Information and Control Systems 90+).

The important features of ICS 90+ are the following:

- A Distribution Digital Control System (DCS) that achieves substantial decrease in the quantity and diversity of wiring, terminators and electronic components.
- A Plant Display System (PDS) that will result in a significant reduction in manual surveillance activities in the station.
- Computer-based safety and protection systems that are proven to reduce protection system unavailability and plant power production unavailability.

ICS 90+ evolved from the highly automated CANDU control systems incorporated in the operating CANDU 6 and Darlington stations, and takes advantage of the rapid developments in digital systems and communications systems that have occurred in recent years. The result is significant improvements in safety and reduced operating cost.

The key benefits are:

- A substantial reduction in the number of instrumentation and control components, leading to improved reliability and reduced maintenance and construction costs.
- Increased automation, which frees the operations staff from boring and/or stressful tasks, leading to reduced frequency of operator error.
- Improved operator understanding of the plant operational state at all times.

5.7.2.3.2. Control centre

The CANDU 6 control centre (Figure 5.7.3.) combines the best features of operating CANDU control centres with upgrades made possible by new technology and successful CANDU development programs. These upgrades include the following:

- A large central mimic display, visible to all staff in the control room.
- A substantial reduction in panel complexity.
- One-man operation capability from a sit-down console during normal operation.
- Critical safety parameter monitoring tied back to emergency operating procedures.
- An alarm annunciation system that eliminates operator information overload during major plant transients.
- Automated safety system operational testing.

A secondary control area is provided, remote from the control centre, with the capability of assuring reactor shutdown, fuel cooling and plant monitoring in the event that the control centre is for any reason nonfunctional or noninhabitable.

*Fig 5.7.3. CANDU Control Centre*

5 7 2 3 3. Electrical systems

The CANDU 6 incorporates comprehensive electrical systems that include Class IV, Class III, Class II and Class I systems, and redundancy and diversity to assure that all plant reliability and safety requirements are met. The Class III power system for example, includes four diesel generating sets, each capable of sustaining all necessary plant loads.

5.7.2.3.4. Safety considerations

See section 5 7.3.

*5.7.2.5. Buildings and structures*

The general arrangement of the CANDU 6 nuclear power plant is shown in Figure 5 7 4. The principal structures included the Reactor Building, the Service Building and the Turbine Building. Other structures include the Administration Building, the Main Pumphouse and the Emergency Water Supply Building, and the emergency core cooling system building.

The reactor building is a conventional single unit structure consisting of a post tensioned concrete structure with a non-metalic liner. A passive (gravity) water spray system (refered to as the dousing system), utilizing water supplied from a high level (dousing) tank is used to reduce the pressure in the reactor building following a postulated loss of coolant accident. The reactor building is seismically qualified for the design basis earthquake.

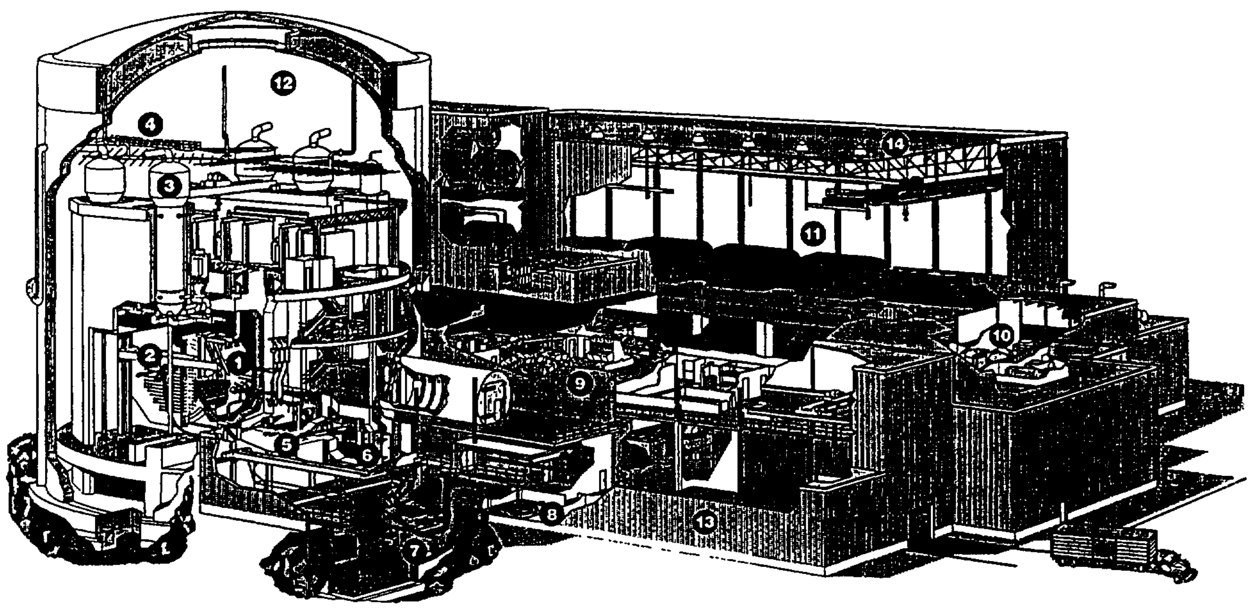



*Fig 5.7.4. CANDU 6 Nuclear Generating Station*

### 5.7.3. Safety concepts

*5.7.3.1. CANDU safety characteristic and origins*

Safety is assured in CANDU 6 through a defence in depth approach that builds on diversity and redundancy, and which takes advantage of the unique CANDU pressure tube reactor concept. Passive systems are used whenever they are shown to be reliable and economic; these systems are complimented by engineered systems. The consistent application of human factors principles, and detailed attention to all aspects of plant design also contributed to CANDU 6 safety.

CANDU design practice places emphasis on the performance of the special safety systems (shutdown systems, emergency core cooling system and containment system).

Accidents must be shown to have acceptable consequences, not only if the safety systems work, but also if any safety system is unavailable or impaired. For example, in most other reactors, a loss-of-coolant accident coupled with prolonged unavailability of the emergency core cooling system, would result in melting of the reactor fuel. In CANDU this sequence would lead to damaged fuel but no meltdown. This overall safety approach is achievable because there are at least two ways of providing the safety functions of shutdown and decay heat removal.

Firstly, there are two independent safety shutdown systems, each equally effective in handling accidents.

Secondly, the heavy water moderator provides an independent source of cooling water surrounding the pressure tubes (Figure 5.7.2.). The moderator cooling system is designed to remove about 5% of the total reactor thermal power during normal operation, a value equivalent to core decay heat shortly after shutdown. In the event of an accident, the pressure tube is close enough to the calandria tube to assure that an emergency heat removal pathway can be passively achieved. For a loss of coolant accompanied by total failure of emergency core cooling system flow, the pressure tube will overheat, then sag or strain into contact with its surrounding calandria tube. Consequently, the fuel decay heat is transferred to the calandria tube through the pressure tube, and then removed by the moderator. Because of the cooling capability of the moderator, damaged fuel would remain within the pressure tubes, without UO2 melting, so that the core geometry would be retained. This is termed the moderator heat sink.

Thirdly, every postulated accident which releases radioactivity must be shown to have acceptable consequences even if any one of the special safety systems fail to function, including the active components in the containment system (e.g., ventilation dampers fail to close).

The special safety systems themselves are:

- Independent of each other and of the normal control and process systems;
- Separated physically from each other, and from the control/process systems, so that common cause events cannot affect more than one safety system;
- Redundant, at both the system and active component level, so that isolated failures, either of active components or of an entire system, cannot disable the safety function;
- Testable during service, to meet a reliability target of 999 times out of 1000 tries;
- Diverse in design and operation, so that a generic fault in design, maintenance or operation cannot affect more than one safety system.

Further safety characteristics include:

The use of natural uranium or other low fissile content fuel in a core lattice arrangement that provides maximum reactivity. This type of fuel requires on-power refuelling, and this in turn means that very little reactivity needs to be held up in movable control devices; no chemicals are added to the reactor coolant for reactivity control. Thus, malfunctions in the control system produce only modest reactivity changes. Further, in a severe accident, a damaged CANDU core would tend to shut down inherently. The injection of ordinary water (e.g. through the emergency core cooling system) inserts negative reactivity in CANDU.

The control and shutdown devices are in the low-pressure moderator, not the high pressure coolant, so are not subject to large hydraulic forces. Natural coolant circulation will remove decay heat from the fuel if pumping power is lost. This is effective even if, following a small loss-of-coolant accident, the heat transport system coolant inventory is somewhat depleted.

For hazards such as earthquakes, fires, floods, etc., the plant is protected through implementation of the two group approach described. This entails a combination of redundancy of systems which protect the plant against frequent events, protection of essential systems so they can withstand the event, and separation of redundant systems so the same event cannot disable more than one.

Digital controllers control the plant routinely, freeing the operator of mundane tasks and reducing the likelihood of operator error. In case of accidents, the safety system responses are automated to the extent that no operator action is needed for several hours following an accident.

Radiation doses to the public and to operators of existing CANDUs during normal operation have generally been less than for other reactor types. This is partly due to the economic need to reduce leaks of heavy water, and to the ability to remove defected fuel during operation.

*5.7.3.2. Passive safety*

CANDU 6 Incorporates a number of passive safety features. These include:

- The two independent, passive safety shutdown systems.
- A gravity feed supply of water to the steam generators (from the dousing tank), available if normal and backup engineered supplies are lost.
- The ability to cool the fuel even if all coolant (normal and emergency supply) are lost, by the rejection of heat from the fuel to the cool, low pressure moderator.

Additional capabilities that contribute to severe accident mitigations are discussed in 5.7.3.4.

*5.7.3.3. Engineered safety*

A range of engineered systems contribute to CANDU 6 safety. These include:

- The shutdown cooling system, consisting of two independent circuits that, utilizing pumps and heat exchangers, can remove decay heat should the steam generator sink be lost. This system can be brought into service at full heat transport system temperature and pressure.
- The connection of all pipes that can potentially discharge reactor coolant from the heat transport system (HTS) to a high pressure tank (the degasser condenser). The relief valves on this tank are set above HTS operating pressure. Hence, failure of any valve in the above lines in the open position does not lead to a loss of reactor coolant.
- The provision of an emergency water supply to the steam generators to backup the normal (high capacity electric main feedwater pumps, and electric and steam turbines auxiliary feedwater pumps). A passive water supply is also provided (see Section 5.7.2.2.).
- The provision of multiple systems and supplies throughout the plant to ensure reliability. For example, four on-site diesel electric generators, two to back up the normal power supplies and two for emergency power.

*5.7.3.4. Severe accident mitigation*

Probabilistic analysis has been a hallmark of CANDU safety philosophy since its inception; Canada remains one of the few countries whose regulations are based largely on probabilistic considerations. Probabilistic safety assessments have been performed for the Darlington station by Ontario Hydro, and for CANDU 6 in a joint study by AECL and the Dutch research organization KEMA. Both studies reached a similar conclusion; e.g. because of the redundant heat removal paths in CANDU, and redundant shutdown systems, the predicted severe core damage frequency is less than about 4x10-6 events/year.

With respect to the redundant heat removal paths, the moderator can act as an emergency heat sink even with no water in the fuel channels. Should the moderator heat removal system subsequently fail, the large water-filled reactor vault surrounding the calandria vessel provides an additional line of defence (Fig. 5.7.2.). Its primary purpose is to provide shielding of the concrete reactor vault from neutrons and gamma rays. However it can also act as a passive emergency water reservoir in case of a severe core damage accident; that is, should the primary coolant system (HTS), the emergency core cooling system, and the moderator heat removal system all fail, the water-filled vault will retain the debris inside the calandria, by keeping the outside of the calandria shell cool for a minimum period of 24 hours. This allows time for fission products to decay further and for decay heat to reduce, and for emergency planning.

*5.7.3.5. Safety related systems and features*

The following tables provide further details on the CANDU 6 safety related systems and safety functions. These include:

Table 5.7.1. Main Safety Related Systems
Table 5.7.2. Main Accident Initiators
Table 5.7.3. Design Features for Prevention
Table 5.7.4. Design Features for Mitigation

# TABLE 5.7.1. MAIN SAFETY RELATED SYSTEMS IN THE CANDU 6

| Name | Safety graded | Main characteristics |
|---|---|---|
| 1 HEAT TRANSPORT SYSTEM | CSA Class 1 DBE | The system safety functions are to facilitate heat removal from the fuel under both normal and accident conditions, and to maintain the integrity of the reactor coolant pressure boundary |
| 2. PRESSURE AND INVETORY CONTROL SYSTEM | CSA Class 1,3 DBE | The system safety functions are to provide overpressure protection of the heat transport system, to maintain sufficient reactor coolant inventory during and after operational states<br>The system consists of the pressurizer, degasser-condenser, heat transport system safety/relief valves, and feed/bleed flow paths with $D_2O$ storage tank and feed pump |
| 3 SHUTDOWN COOLING SYSTEM | CSA Class 1 DBE | The system is designed to provide long-term decay heat removal from the fuel It also has a capability to act as a high-pressure and high-temperature heat sink for events leading to total loss of steam generator feedwater |
| 4 MODERATOR SYSTEM | CSA Class 3 | The system is capable to provide a backup heat sink in the event of a loss of coolant accident coincident with the ECCS unavailable<br>The system, containing the low-pressure cooled $D_2O$ moderator, includes the calandria, pumps and heat exchangers and related piping |
| 5 MAIN STEAM SYSTEM | CSA Class 2 DBE | The safety related portion of the system includes the steam generator secondary side and the main steam safety valves<br>The system provides overpressure protection of the steam generators and provides decay heat removal from the heat transport system. |
| 6 FEEDWATER SYSTEMS | CSA Class 3 | The system includes both the main feedwater pumps and the auxiliary feedwater pumps The system function is to supply feedwater to the steam generators in order to ensure the steam generator heat sink Backed up by emergency and gravity feedwater supply |
| 7 REACTOR SHUTDOWN SYSTEMS (Special Safety Systems) | AECB R-8 CSA N290 1 DBE | There are two passive fully capable, fast-acting independent and diverse shutdown systems independent of the reactor regulating system. |
| 8 REACTOR REGULATING SYSTEM | CSA N290 4 | Provides for control of the reactor power and neutron flux distribution during plant operation It also has the capability to shut down the reactor for anticipated operational occurrences |
| 9 EMERGENCY CORE COOLING SYSTEM (Special Safety System) | CSA Class 2 3 AECB R-9 SDE | The ECC system services to provide cooling to the fuel in a HTS circuit loop that has experienced a LOCA<br>It consists of passive high-pressure injection (pressurized tanks) low pressure recirculation/recovery portion (pumped) |
| 10 CONTAINMENT (Special Safety System) | CSA Class 4 AECB R-7 DBE | The containment system serves as a barrier to the environmental dispersal of radionuclides which have accidentally entered into the containment atmosphere |
| 11 EMERGENCY WATER SUPPLY SYSTEM | CSA Class 6 DBE | The EWS system provides feedwater to any system generator in the event that normal feedwater supplies are lost, and provides backup cooling water to the ECC heat exchangers<br>The emergency water supply to the steam generators is comprised of two water sources the dousing tank at the top of the reactor building (gravity supply) and a long term reservoir (pumped supply) |
| 12 ELECTRIC POWER SYSTEM | Class I,II and III Electric System CSA B290 5 | The Class III system powers designated safety-related and economic equipment protection loads (e g ECCS pump motors, valves etc ) Normally Class IV power supplies the Class III power When the Class IV system fails two redundant standby diesel generators provide Class III power<br>The ac Class II and dc Class I systems supply un-interruptible power to the control and safety systems |
| 13 EMERGENCY POWER SUPPLY SYSTEM | CSA N290 5 DBE | The seismically qualified system is designed to provide a backup source of power for events which may render the Class III Electric System unavailable<br>The system is comprised of two redundant diesel-generator sets completely separated from the standby diesel generators of the Class III system |
| 14 SERVICE WATER SYSTEMS | CSA Class 6 | These systems remove heat from equipment including safety related equipment and transfer this heat to the ultimate heat sink<br>The systems are comprised of a recirculated cooling water system (closed loop) and a raw service water system (once-through) connected to the ultimate heat sink |

**CLARIFICATIONS FOR TABLE 5.7.1.**

Safety graded" is intended to mean the specification of codes, standards, regulatory requirements and other qualification requirements (notably the seismic design level) applicable to a system For each safety related system the table provides

- the CSA class for pressure retaining systems
- the applicable CSA standards for other types of systems
- the applicable regulatory policy statements (Atomic Energy Control Board (AECB) R-7 R-8 and R-9 for the special safety systems)
- the seismic qualification level applicable DBE (Design Basis Earthquake) or SDE (Site Design Earthquake)

## TABLE 5.7.2. MAIN ACCIDENT INITIATORS FOR THE CANDU 6

- LOCA (Primary): Loss of Primary Coolant Accident
- LOCA (Secondary): Secondary Pipe Rupture (feedwater or steam)
- LOCA (Interfacing): e.g.: Steam Generator Tube Rupture
- Primary Transients
- Secondary Transients (turbine trip)
- Loss of electric sources (partial)
- Total loss of the heat sink
- Total loss of the steam generator feedwater
- Station blackout

## CLARIFICATIONS FOR TABLE 5.7.2.

In CANDU, all postulated failures are considered in conjunction with the postulated failure of each of the special safety systems (one at a time), as design basis events.

Anticipated Transients Without Scram (ATWS) are not included in the design basis of CANDU plants, because of the extremely low frequency expected for such events. CANDU plants have three systems with independent and diverse capability for automatic reactor shutdown.

## TABLE 5.7.3. DESIGN FEATURES FOR PREVENTION AND PROTECTION LEVELS THAT REDUCE (R), SUPPRESS (S), THE INITIATORS FREQUENCY OR LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES (L)

| | |
|---|---|
| | PREVENTION LEVEL |
| LOCA (Primary) | - Lifetime monitoring of pressure tubes and heat transport system piping and components<br>- Leak detection in annual gas system |
| LOCA (secondary) | - Lifetime monitoring and inspection of secondary components and piping |
| LOCA (interfacing) | - Lifetime inspection of steam generators<br>- Heavy water appearing in light water system is detected and alarmed |
| Primary Transients | Lower bundle powers increase design margins |
| Secondary Transients | - ASDV's, CSDV's and MSRV's |
| Loss of Electric Sources | Redundant systems provide power to all critical systems |
| Total Loss of the Cold Source (water) | Separate cooling water systems for normal and safety systems, redundancy |
| Station Blackout | All critical loads carried by standby diesel generators or passive systems |
| | PROTECTION LEVEL |
| LOCA (Primary) | - 2 independent and very fast, passive shutdown systems<br>- reliable indications of loss of inventory<br>- Emergency Core Cooling System<br>- Moderator System backup<br>- Shield cooling system backup |
| LOCA (Secondary) | - Steam generator and shut down cooling system heat sinks Note, fast cooldown does not insert positive reactivity in CANDU |
| LOCA (Interfacing) | - Detection ($D_2O$ in steam)<br>- Shutdown cooling system heat sink |
| ATWS | - 2 independent and very fast, passive shutdown systems |
| Primary Transients | - Redundant control and shutdown systems/pressure relief |
| Secondary Transients | - Redundant control and shut down systems/pressure relief |
| Loss of Electric Power | - Fail safe design and design to allow time for operator action |
| Total Loss of Heat Sink | - Not credible as demonstrated by PSA Multiple redundant heat sinks available |
| Total Loss of S G Feedwater | - Actuates shutdown systems, emergency supplies<br>- Shutdown cooling available |
| Station Blackout | - All critical loads carried by standby diesel generators, loss of Class IV and diesel not credible Also passive (gravity) supply to steam generator |

TABLE 5.7.4. DESIGN FEATURES FOR MITIGATION LEVEL OF CANDU 6

| Safety Functions | Systems (cf Tab 5 7 1 ) | Passive/Active | Design features/Remarks |
|---|---|---|---|
| Design Basis | All single failure accidents and selected dual failure accidents | | LOCA + LOECC and LOCA + any safety system failure are included |
| Fission product containment | 1 Fuel clad/sheath<br>2 Heat transport system pressure boundary<br>3 Containment envelope | P | Multiple barriers to release with in-service inspection including activity release from fuel, pressure tube leak-before-break and on-line detection of leak rate from containment |
| Coolant inventory | 1 Feed and bleed system<br>2 Emergency core cooling (ECC)<br>3 Steam generator feed makeup | A | 1 and 2 are on the primary circuit and 3 and 4 on the secondary 1 and 3 are for normal operation and accidents without loss of coolant 2 is for primary LOCA 4 is for earthquakes and secondary side pipe breaks |
| Decay heat removal | 1 Shutdown cooling system<br>2 Emergency core cooling system<br>3 Emergency water system (EWS) | A | System 1 is used on a primary loop without a break System 2 is sued on a primary loop with a pipe break |
| Reactivity control | 1 Reactor regulating system<br>2 Shutdown system #1, shut-off rods<br>3 Shutdown system #2, Gd poison injection<br>4 Loss or dilution of $D_2O$ moderator | 1 A<br>2 P<br>3 P<br>4 P | Power manoeuvres include ramp setback, stepback and trip 2 and 3 are engineered safety systems for trip 1 can serve in an assisting role to shutdown systems No 1 and 2 In 4 $H_2O$ from ECC or leakage or boiling down of moderator leads to subcriticality |
| Primary circuit pressure control | 1 HT pressurizer with steam-bleed valves<br>2 Feed and bleed system<br>3 Automatic liquid relief valves<br>4 Main steam safety valves | A | Systems 1 and 2 provide primary circuit pressure control Systems 3 and 4 prevent overpressure and provide rapid depressurization |

TABLE 5.7.4. DESIGN FEATURES FOR MITIGATION LEVEL OF CANDU 6

| Safety Functions | Systems (Cf Tab 5 7 1 ) | Passive/Active | Design features/Remarks |
|---|---|---|---|
| Severe Accident | Beyond Design Basis Accidents | | Very low probability triple failure accidents |
| Containment temperature and pressure control | 1 Dousing system<br>2 Local area coolers (LACs)<br>3 Other condensation on structures | 1 P<br>2 A<br>3 P | Post-accident, the dousing system and the condensation on structures are available in the early part, the LACs are available in the long term |
| Heat Removal | 1 Moderator cooling the fuel channel or fuel in the calandria<br>2 Shield coolant cooling the calandria and its contents | P | If primary coolant is available, thermosyphoning cools the fuel For a primary loop without coolant the moderator and the radial shield coolant become heat sinks in severe accidents They act in tandem, the heat removal efficiency is improved if electrical power is available |
| Tightness Control | 1 Containment isolation system<br>2 Containment liner | P | The isolation system automatically buttons up the containment ventilation system, wall penetrations of process piping and airlocks and hatches |
| Inflam gas control | Igniters for hydrogen-air-stream mixtures | P | Battery operated glow igniters are automatically heated up on a LOCA signal |
| Fission product containment | 1 From intact pressurized containment leakage over many days.<br>2 Optionally, long term use of filtered air discharge system | 1 P<br>2 A | 1 The containment leakage rate is maintained within acceptable limits,<br>2 The engineered FAD doesn't have the capacity of an Emergency FAD, but can be used at moderate pressure and activity levels |
| Corium management | 1 Primary heat transport cooling within the pressure tubes<br>2 Moderator cooling within the calandria<br>3 Shield coolant within the calandria vault<br>4 Base slab | P | Any corium would have to penetrate the three consecutive boundaries and within each boundary a diverse, independent cooling is provided With natural uranium fuel, $D_2O$ moderator boiling, $H_2O$ ECC coolant and $H_2O$ shield coolant, the disassembled core quickly becomes subcritical and there is no danger of recriticality |

### 5.7.4. Design data questionannaire

## I. GENERAL INFORMATION

1. Design name: CANDU 6
2. Designer/Supplier address: AECL CANDU, Mississauga, Ont. Canada L5K 1B2
3. Reactor Type PHWR Number of modules/per plant: 1
4. Gross thermal power (MW-th) per reactor: 2158 MW-th
5. Net electrical output (MW-e) per reactor: 666 MWe*
6. Heat supply capacity (MW-th): 2064 MW-th

## II. BASIC DESIGN DESCRIPTION

### A. CORE AND REACTIVITY CONTROL

7. Fuel material: Natural $UO_2$
8. Fuel inventory (tons of heavy metal): 88 t of U
9. Average core power density (kW/litre): 8.2 kW/L
10. Average fuel power density (kW/kgU): 23.5 kW/kgU
11. Maximum linear power (W/m): 57 kW/m of fuel rod
12. Average discharge burnup (MWd/t): 7500 MWd/t of U
13. Initial enrichment (wt%): 0.711 wt%
14. Reload enrichment at the equilibrium (wt%): 0.711 wt%
15. Refuelling frequency (months): 10 months gradual changeover of core with ave. refuelling rate of 15 fuel bundles/full power day
16. Type of refuelling (on/off power): On-power
17. Fraction of core withdrawn (%): 0.33 %/day
18. Moderator material and inventory: >99.75 wt.% $D_20$; 240t in calandria
19. Active core horizontal length (m): 5.94 m

* Dependent on cooling water temperature and turbine design.

20. Calandria diameter (m): 6.28 m
21. Number of fuel assemblies: 4560 fuel bundles in core (12 fuel bundles in each of 380 fuel channels)
22. Number of fuel rods per bundle assembly: 37
23. Bundle array: 1+6+12+18 fuel rods on concentric pitch circles
24. Clad material: Zr - 4
25. Clad thickness (mm): 0.4 mm
26. Number of control rods or assemblies: 39 for reactor regulation
27. Type: 4 Control absorber rods; 21 Adjuster rods; 14 Liquid Zone Control Absorbers
28. Additional shutdown systems: Two independent passive safety shutdown systems: 28 shut-off rods and 6 liquid poison injection circuits
29. Control rod neutron absorber material: Cadmium, stainless steel; optional cobalt for Cobalt 60 production
30. Soluble neutron absorber: Boron and gadolinium in moderator only
31. Burnable poison material and form: None (in fuel or HTS)

### B. REACTOR COOLANT SYSTEM

#### B1. Coolant

32. Coolant medium and inventory: 130 t of $D_2O$
33. Design coolant mass flow through core (kg/s): 7700 kg/s
34. Cooling mode (forced/natural): Forced
35. Operating coolant pressure (bar): 100 bar (a)
36. Core inlet temperature (°C): 266°C
37. Core outlet temperature (°C): 310°C

#### B2. Reactor calandria vessel/pressure tube

38. Overall length of assembled vessel/tube (m): 7.77 m over end shields/min. 6.284 m cold installed pressure tube

39. Inside diameter - calandria: 7.59 m
- pressure tube: 103 mm
40. Average thickness (mm) - calandria: 28.6 mm
- pressure tube: 4.19 mm
41. Material - calandria: 304 L stainless steel
- pressure tube: Zr-2.5 wt% Nb
42. Lining material: N/A
43. Design pressure (bar):
- calandria: 0.75 bar(g) at top of calandria
- pressure tube: 113.8 bar(g) at inlet
44. Gross weight (ton/kg): 71 t of empty calandria vessel without 23.4 t of pressure tubes

**B3. Steam generator**

45. Number of steam generators: 4
46. Type: U-tube in shell
47. Configuration (horizontal/vertical): Vertical
48. Tube material: Incoloy-800
49. Shell material: SA 516 carbon steel
50. Heat transfer surface per steam generator (m2): 3190 $m^2$
51. Thermal capacity per steam generator (MW): 516 MW
52. Feed water pressure (bar): 48.5 bar(a)
53. Feed water temperature (°C): 187°C
54. Steam pressure (bar): 47 bar(a)
55. Steam temperature (°C): 260°C

**B4. Pressurizer**

56. Pressurizer total volume ($m^3$): 45 $m^3$
57. Steam volume (full power/zero power, $m^3$):
7.3$m^3$ at 100% power
32.8 m3 at 0% power, hot

**B5. Main coolant pumps**

58. Number of cooling or recirculation pumps: 4
59. Type: Vertical centrifugal pumps with totally enclosed AC induction motor
60. Pump mass flow rate (kg/s): 1925 kg/s
61. Pump design rated head: 215 m
62. Pump nominal power (kW): 6710 kW
63. Mechanical inertia (kg $m^2$): 1300 kg $m^2$

## C. CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS)

There are no chemicals in the reactor coolant for reactivity purposes.

64. Number of purification extraction lines: 2 in each HTS loop
65. Number of purification pumps: 0 (the heat transport pumps also provide the purification flows)
66. Number of injection points: 2 in each HTS loop
67. Feed and bleed connections: 1 feed and 1 bleed connection in each loop

## D. CONTAINMENT

68. Type: Pressure containment with pressure suppression
69. Overall form (spherical/cyl.):
Vertical cylinder with domed top
70. Structural material: Prestressed reinforced concrete
71. Liner material: Non-metallic
72. Single/double wall: Single wall
73. Dimensions (diameter, height) (m): 41.5 m inside dia. x max. 51.2 m inside height above top of base slab
74. Design pressure (bar): 1.24 bar (g)
75. Design temperature (°C): 66°C
76. Design leakage rate (% per day): 0.5% of free volume/day at design pressure

## III. SAFETY RELATED SYSTEMS

## A. DESIGN CONDITIONS

### A1. Fission product retention

77. Containment spray system (Y/N): Y
   a. Duration: min. 200s
   b. Flow rate (m3/h): max. 7.2 $m^3/s$
   c. Mode of operation (active/passive): Passive
   d. Safety graded (Y/N): Y
78. F.P. sparging (Y/N): Y (in long term after a LOCA)
79. Containment tightness control (Y/N): Y
80. Leakage recovery (Y/N): N
81. Guard vessel (Y/N): N

### A2. Reactivity control

82. Absorber injection system (Y/N): Y (Shutdown System #2)
   a. Absorber material: Gadolinium
   b. Mode of operation (active/passive): P
   c. Redundancy: 2 out of 3 trip logic 5 of 6 injection tank
   d. Safety graded: Y
83. Shut-off rods (Y/N): Y (Shutdown System #2)
   a. Maximum rod worth (pcm): 8500 pcm
   b. Mode of operation (active/passive): P
   c. Redundancy: 2 of out 3 trip logic 26 of 28 rods
   d. Safety graded: Y

### A3. Decay heat removal

*A3-1 Primary Side*

84. Water Injection: Emergency Core Cooling (ECC)
   a. Actuation mode (manual/automatic): A
   b. Injection pressure level (bar): Initially 40 bar (g) from tank and later 9.4 bar (g) from pump
   c. Flow rate (kg/s): Max. 600 kg/s
   d. Mode of operation (active/passive): P initially and A in long term
   e. Redundancy: 2 out of 3 trip logic
      2 x 100% heat exchangers and pumps
   f. Safety graded: Y
85. Water recirculation and heat removal: Without LOCA: *Shutdown Cooling System or thermosyphoning through the* steam generators, with LOCA: ECC in failed loop and thermosyphoning in intact loop
   a. Intermediate heat sink: Recirculated cooling water (RCW) and ultimately the service water
   b. Mode of operation (active/passive): A with pumping and P with thermosyphoning
   c. Redundancy: 100% in long term
   d. Self sufficiency (h): 3 months
   e. Safety graded: Y

*A3-2 Secondary Side*

86. Feedwater
   a. Actuation mode (manual/automatic): A
   b. Flow rate (kg/s): 1034 kg/s normal total 41 kg/s auxiliary
   c. Mode of operation (active/passive): A
   d. Redundancy: 3 x 50% pumps + 2 x 4% auxiliary pumps, etc. Emergency backup supply and gravity supply
   e. Self sufficiency (h): Depends on accident type
   f. Safety graded: Y (Emergency and Gravity Supply)

87 Water recirculation and heat removal

- a Ultimate heat sink (cold source) Emergency Water Supply (EWS)
- b Mode of operation (active/passive) P for feed line breaks and steam line break due to earthquake, A for other cases
- c Redundancy 2 x 100% pumps
- d Self sufficiency (h) 1 to 15 days minimum, depending on accident
- e Safety graded Y

*A3-3 Primary pressure control*

88 Implemented System Pressure and Inventory *Pressurizer provides principal control method.* The Liquid Relief Valves provide Control System overpressure protection together with reactor power stepback In accidents with LOCA crash cooling by opening the Main Steam Safety Valves depressurizes the secondary circuit, which cools and depressurizes the primary circuit and helps ECC injection for small breaks

- a Actuation mode (manual/automatic) A
- b Side location (primary/secondary circuit) Pressure and Inventory Control and Liquid Relief Valves in primary, crash cooling via secondary circuit
- c Maximum depressurization rate (bar/s) Hardware limited
- d Safety graded Y

## B: SEVERE ACCIDENT CONDITIONS

### B.1. Fission products retention

89 Containment spray system (Y/N) Y

90 F P sparging (Y/N) Y by ventilation system via HEPA filters and stack

91 Containment tightness control (Y/N) Y

92 Leakage recovery (Y/N) N

93 Risk of recriticality (Y/N) N

### B.2. Recriticality control

94. Encountered design feature Moderator poisoned with boron for guaranteed shutdown state

- a Mode of operation (A/P) A
- b Safety graded Y

### B.3. Debris confining and cooling

95 Core debris configuration (core catcher) Calandria shell is the core catcher, and ultimately the calandria vault

96 Debris cooling system (name) Moderator system and shield cooling system (the latter provides calandria vault cooling as well)

- a Mode of operation (A/P) P in short term, A in long term
- b Self sufficiency min 24 h, but unlimited if service water and electrical power are available
- c Safety graded (Y/N) Classed as safety-related system for reliability and environmental qualification

### B.4. Long term containment heat removal

97 Implemented system Local area coolers

- a Mode of operation (A/P) A
- b Self sufficiency (h) 3 months
- c Safety graded (Y/N) Y

98 Intermediate heat sink Recirculated Water

- a Self sufficiency (h) 3 months
- b Safety graded (Y/N) N

99 External coolant recirculation
- a Implemented components NA
- b Mode of operation (A/P) NA
- c Self sufficiency (h) NA
- d Safety graded (Y/N) NA

100 Ultimate heat sink Service water
- a Self sufficiency (h) Unlimited
- b Safety graded (Y/N) Y (Emergency Water Supply)

**B.5. Combustible gas control**

101 Covered range of gas mixture concentration Achievable mixtures of $H_2$-Air-Steam

102 Modes for the combustible gas control
- a Containment inertation N
- b Gas burning Y
- c Gas recombining N
- d Others Y (forced circulation)

**B.6. Containment pressure control**

103 Filtered vented containment (Y/N) Y in long term only
- a Implemented system Filtered Air Discharge
- b Mode of operation (A/P) A
- c Safety graded N

104 Pressure suppression system (Y/N) Y
- a Implemented system Dousing spray
- b Mode of operation P
- c Safety graded (Y/N) Y

## C. SAFETY RELATED I&C SYSTEM

Automatic load following (Y/N) Y
- range (% power) 100 to 55%
- maximum rate (%/min) 60%/min for steam plant, but may be limited by turbine

Load rejection without reactor trip (Y/N) Y
Full Cathode Ray Tubes (CRT) display (Y/N) Y on control system and alarm displays
Automated start-up procedures (Y/N) Y
Automated normal shutdown procedures (Y/N) Y
Automated off normal shutdown procedures (Y/N) Y
Use of field buses and smart sensors (Y/N) N
Expert systems or artificial intelligence advisors (Y/N) N for safety systems, but Y for some operating systems, e g , refuelling sequencing
Protection system backup (Y/N) Y

## D. EMERGENCY POWER SUPPLY SYSTEM

105 Type (diesel, gas, grid connection) Diesel
106 Number of trains 2 trains DBE qualified, + 2 trains not DBE qualified

## E. AC/DC SUPPLY SYSTEM

107 Type (rectifier, converter, battery) Rectifier, battery, inverter
108 Estimate time reserve (hr) 1hr

# IV. CONVENTIONAL THERMAL CYCLE

## A. TURBINE SYSTEM

109 Type Tandem-Compound-6 Flow, 1 092 m last stage blades, 1800 rpm, 2 stage reheat
110 Overall length (m) 61 m (typical)
111 Width (m) 8 7 m at LP turbine (typical)
112 Number of turbines 1
113 Number of turbine sections per unit (e g HP/LP/LP) 1HP, 3LP
114 Speed (rpm) 1800 rpm

**B. STEAM CHARACTERISTICS**

115. H.P. inlet pressure: 45.1 bar(a)
116. H.P. inlet temperature: 260°C
117. H.P. inlet flowrate: 996 kg/s
118. L.P. inlet pressure: 7.35 bar (a)
119. L.P. inlet temperature: 241°C
120. L.P. inlet flowrate (per section): 249 kg/s

**C. GENERATOR**

121. Type (3-phase synchronous, DC): 3-phase synchronous
122. Apparent power (MVA): 828 MVA
123. Active power (MW): 715 MW
124. Frequency (hz): Either 60 Hz/50 Hz
125. Output voltage (kV): 20 kV
126. Total generator mass (t): 460t (typical)
127. Overall length: 14.5 m (typical)
128. Stator housing outside diameter: 6.1 m (typical)

**D. CONDENSER**

Note: Design and performance depends on cooling water temperature; typical values shown.

129. Number of tubes: 48540
130. Heat transfer area: 49580 $m^2$
131. Flowrate ($m^3/s$): 36 $m^3/s$ cooling water flow
132. Pressure: 6 kPa(a)
133. Temperature (°C): 36°C at condensate outlet

**E. CONDENSATE PUMPS**

134. Number: 2 x 100% + 1 auxiliary
135. Flowrate: 731 kg/s
136. Developed head: 174 m
137. Temperature: 37°C
138. Pump speed: 1190 rpm

### 5.7.5. Design status

There are currently four CANDU 6 plants in operation (two in CANADA, one in the Republic of South Korea, and one in Argentina). In addition, there are eight CANDU 6 plants under construction, five in Romania and three in the Republic of South Korea. The first unit in Romania and the three additional units in the Republic of South Korea enter service in 1995, 1996, 1997, 1998 respectively. Hence, the CANDU is a fully developed and proven design, that can be implemented on a fast schedule.

### 5.7.6. CANDU 6 economics

Economic evaluations show that the cost of energy generation with CANDU 6 is competitive with that from the other reactor types, and with that from coal plants in many areas of the world. Data supplied to OECD by the Republic of South Korea, the only country in the world operating CANDU, PWR and coal generating plants, indicates that CANDU 6 is competitive with both the larger PWR plants and with coal plants in that country.

Specific cost data for single and two unit CANDU 6 stations is presented below; the data assumes Canadian siting, engineering, procurement, construction, and operating environment. All costs are in January 1993 Canadian dollars, assuming 5% real interest rate, 40-year life for economic analysis purposes, and average lifetime capacity factor of 80%.

The cost of capital modifications of 1% overnight capital cost per year and the cost of decommissioning, estimated at 10% of overnight capital cost are included.

| | 1 X CANDU 6 | 2 X CANDU 6 |
|---|---|---|
| Overnight capital - M$ | 2,018 | 3,342 |
| O&M - $/kWe/year | 90 | 71 |
| Fuel - mills/kWh | 2.6 | 2.6 |
| LUEC - mills/kWh | 48 | 40 |

For conversion of above cost data to US dollars, use exchange rate of C$1.30 per US$ for 1993.

Owners cost of 5% of overnight cost are included in the overnight cost. The above costs are based on as 64 month duration from site mobilization to inservice. This can be reduced to less than 54 months through the use of open top construction and a very heavy lift crane without any significant design changes.

## 5.8. CANDU 3 REACTOR SYSTEM AND DEVELOPMENT STATUS

### 5.8.1. Basic objectives and features

#### *5.8.1.1. Basic Objectives*

The principal objective of the CANDU 3 development program was to design a relatively small CANDU power plant (with an electrical output in the range of 400 MW), that is competitive with larger nuclear plants and with coal fired plants in most areas of the world. Sub-objectives directed to achieve the principal economic objective included, simplification, modularization, the use of state of the art technologies, and a 36 month construction schedule.

Other objectives included enhancement of real safety, protection of owner investment, assurance of high capacity factors through the use of proven system components and concepts, and a standard design suitable for most potential sites world wide without significant design modification.

#### *5.8.1.2. Basic design features*

The CANDU 3 incorporates all of the basic and well proven features which are the hallmark of CANDU. These include:

- A reactor consisting of small diameter horizontal pressure tubes housed in a low pressure, low temperature moderator-filled calandria (tank)

- Heavy water ($D_2O$) for moderator and reactor coolant
- The standard CANDU 37-element CANDU fuel bundle, and the ability to operate on natural uranium or other low fissile content fuel
- On-power refuelling, to eliminate the need for refuelling outages
- Two diverse, passive, fast-acting and fully capable shutdown systems which are independent of each other, and of the reactor regulating system.
- Automated digital control of all Key Nuclear Steam Plant and Balance of Plant functions.
- The total absence of all chemicals in the reactor coolant (heat transport system) for reactivity control.

### 5.8.2. Design description

*5.8.2.1. Nuclear steam supply system (NSSS)*

5.8.2.1.1. General

The CANDU 3 nuclear steam supply system includes the reactor, the heat transport system, the moderator system, the shutdown cooling system, the fuel handling system and associated control and service systems located within the reactor building (Figure 5.8.1A) and the emergency core cooling system. The following sections provide detail on specific NSSS system.

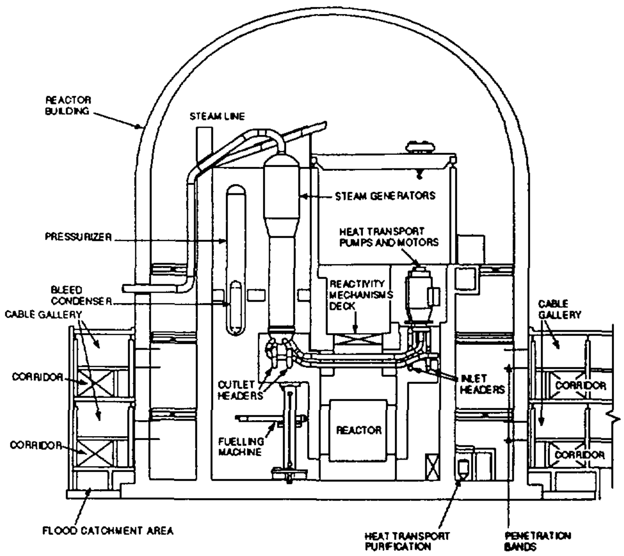



*Fig 5.8.1 Reactor Building*

5.8.2.1.2. Fuel

CANDU 3 uses the standard 37-element CANDU fuel bundle (Figure 5.8.1B). This fuel bundle is a very simple, easily-fabricated structure about 500-mm long and 100 mm in diameter. Current CANDU plants operate with natural uranium fuel (0.7% $U_{235}$); they can, however, operate on a variety of other low fissile content fuels including slightly enriched uranium, and recovered uranium from PWR fuel reprocessing plants.

The use of natural uranium or other low fissile content fuel in CANDU 3 assures that there is no potential for the fuel (new fuel or irradiated) to achieve criticality outside of the reactor regardless of the storage configuration.

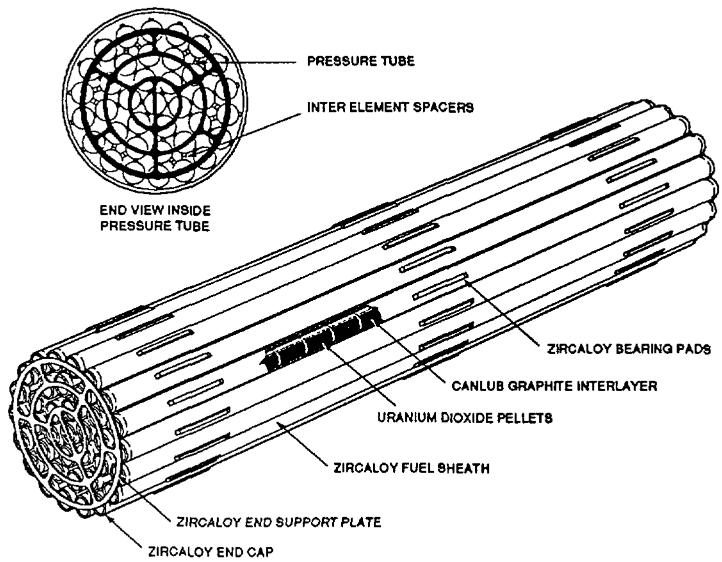



*Fig 5.8.1B. CANDU Fuel Bundle*

5.8.2.1.3. Fuel channels

The CANDU 3 reactor includes 232 horizontal fuel channels. Each fuel channel consists of a zirconium-niobium alloy pressure tube, centred in a Zircaloy calandria tube by annular spacers, and expanded into a stainless steel end fitting at both ends. Each channel contains twelve fuel bundles.

The pressure tubes, the only components in CANDU that are subjected to high levels of radiation and stress, are readily replaceable. However, pressure tube replacement is not required in CANDU 3 plants for 35 or more years. The design of the fuel channels and the layout within the reactor building facilitate fuel channel replacement.

5.8.2.1.4. Calandria assembly

The reactor core consists of 232 fuel channels, held in a square lattice array by circular end-shields, and contained within a cylindrical low-pressure tank called the calandria (Figure 5.8.2.). The calandria contains the heavy water moderator at low temperature and at near-atmospheric pressure.

The calandria is penetrated vertically and horizontally by flux measurement and reactivity control devices, and by the in-core components of two safety shutdown systems. All reactivity control devices function in the low-pressure moderator. No reactivity control devices penetrate the heat transport system and no chemicals are added to the heat transport system for reactivity control.

The calandria is positioned within a low-pressure steel tank, filled with ordinary water. This tank provides biological shielding from neutron and gamma radiation from the reactor core.

5.8.2.1.5. Heat transport system

The heat transport system consists of a single circuit or loop providing coolant to all of the 232 fuel channels.

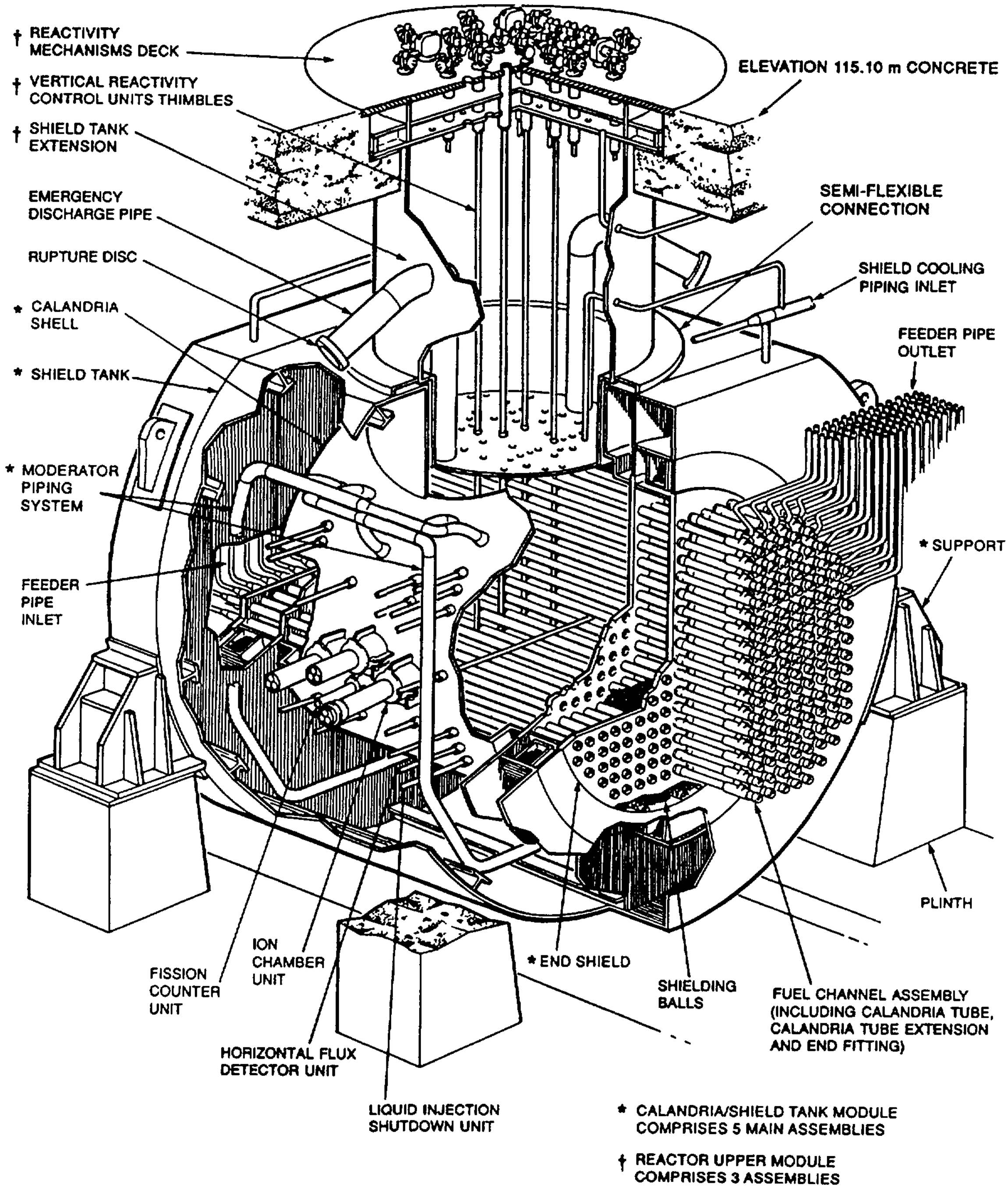



*Fig 5.8.2. CANDU3 Reactor Assembly*

Each fuel channel is connected by individual inlet and outlet feeder pipes to large distribution pipes, called headers, at both ends of the reactor Four heat transport pumps circulate the heavy water coolant from the inlet header, through the fuel channels, to the outlet header, through the steam generators (where the heat is transferred to ordinary water to generate steam), and back through the reactor again The arrangement is called a figure-of-eight arrangement, since the fluid flows through the core twice to make a complete circuit This arrangement places two pumps in series with the other two pumps, which limits the coolant flow reduction should one pump seize

If all pumps are lost, fuel cooling is maintained by thermosyphoning.

The coolant flow through the core is in the same direction for all fuel channels. This permits refuelling operations from a single end (the outlet) of the core employing a single fuelling machine.

5.8.2.1.6. Moderator system

Heat is deposited in the heavy water moderator contained within the calandria during normal operation, from direct gamma and neutron interaction and through thermal conduction from the fuel channels. This heat is removed by the moderator cooling system, which circulates and cools the heavy water in an external circuit connected to the calandria; the heat is rejected to the recirculated coolant water system.

5.8.2.1.7. Fuel handling

On-power refuelling is performed by a single fuelling machine, located at outlet end of the reactor. The machine transports new fuel bundles to the reactor fuel channel to be refuelled. All 12 existing bundles are removed from the channel and then a mixture of used and new bundles, no more than 4 new ones, are placed back into the channel. The fuelling machine subsequently transports the used fuel to the irradiated fuel transfer system which connects to the irradiated fuel storage bay. The refuelling operation is fully automatic, once the channel to be refuelled has been selected.

In the event that a defect occurs in a fuel bundle during reactor operation, the fuelling machine can be used to remove the defective fuel, thereby limiting the release of fission products to the heat transport system coolant. Systems are provided for the detection and location of defective fuel.

5.8.2.1.8. Reactor control

On-power refuelling provides the principal means for controlling reactivity in the CANDU 3. Additional reactivity control, independent of the safety system, is achieved through use of reactivity control mechanisms. These include absorber rods, mechanical zone control units, and adjuster rods; all are located between fuel channels within the low pressure heavy water moderator and do not penetrate the heat transport system pressure boundary. The overall reactor control system is described in Section 5.8.2.3.

5.8.2.1.9. Safety systems

The CANDU 3 incorporates four special safety systems which consist of the two passive, diverse, dedicated reactor shutdown systems; the emergency core cooling system; and the containment system. Each is separated from and is independent of the normally operating plant systems, and of all the other safety systems.

Shutdown system No. 1 utilizes spring-assisted, gravity-drop neutron absorbing rods, which drop into the moderator, between the fuel channels.

Shutdown System No. 2 consists of horizontal perforated tubes through which a liquid neutron absorber is injected into the moderator utilizing high pressure helium.

The Emergency Core Cooling System uses high pressure gas to inject ordinary water into the fuel channels, followed by pumped recirculation and cooling of water within the reactor building. The containment system includes the reactor building and the containment isolation system.

*5.8.2.2. Balance of plant*

5.8.2.2.1. Turbine generator

The CANDU 3 turbine consists of one double flow high pressure and two double flow low pressure turbines, operating at 1800 rpm.

CANDU plants have operated with turbine-generators supplied by a variety of manufacturers including Parsons, GE, BBC and Hitachi. Turbine generators from different suppliers are readily accommodated by CANDU 3.

In the event of a loss of line event, the turbine-generator on CANDU 3 runs back to sustain station loads. In addition, a turbine steam bypass system is provided (100% short-term, 70% long-term capacity) to reject steam directly to the condensers in the event that the turbine is unavailable; this allows reactor power to be sustained during the event.

5.8.2.2.2. Radioactive waste management

The CANDU 3 incorporates comprehensive systems for the management, technical and storage of solid, liquid, and gaseous radioactive wastes.

The irradiated fuel bay, located adjacent to the reactor building has sufficient capacity to accommodate the fuel from ten years of reactor operation; facilities are included for the transfer of irradiated fault to dry storage. There is no potential for CANDU irradiated fuel to achieve criticality, regardless of the storage configuration.

*5.8.2.3. Instrumentation, control and electrical systems*

5.8.2.3.1. Information and control system

Plant instrumentation, computer control systems, control room man/machine interface and the plant information system are provided by the ICS 90+ Systems (Information and Control Systems 90+).

The important features of ICS 90+ are the following:

- A Distribution Digital Control System (DCS) that achieves substantial decrease in the quantity and diversity of wiring, terminators and electronic components.
- A Plant Display System (PDS) that will result in a significant reduction in manual surveillance activities in the station.
- Computer-based safety and protection systems that are proven to reduce protection system unavailability and plant power production unavailability.

ICS 90+ evolved from the highly automated CANDU control systems incorporated in the operating CANDU 6 and Darlington stations, and takes advantage of the rapid developments in digital systems and communications systems that have occurred in recent years. The result is significant improvements in safety and reduced operating cost.

The key benefits are

- A substantial reduction in the number of instrumentation and control components, leading to improved reliability and reduced maintenance and construction costs,
- Increased automation, which frees the operations staff from boring and/or stressful tasks, leading to reduced frequency of operator error,
- Improved operator understanding of the plant operational state at all times

5 8 2 3 2 Control centre

The CANDU 3 control centre (Figure 5 8 3 ) combines the best features of operating CANDU control centres with upgrades made possible by new technology and successful CANDU

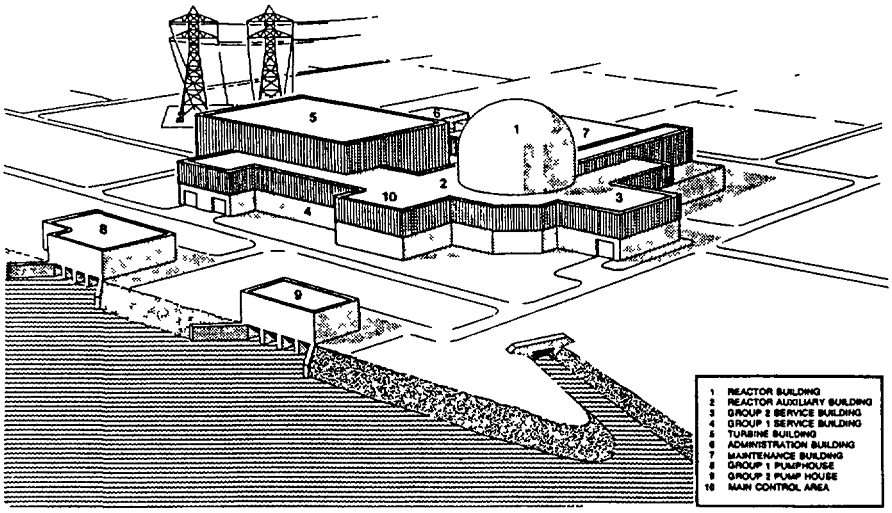



*Fig 5.8.3A CANDU3 Nuclear Generating Station*

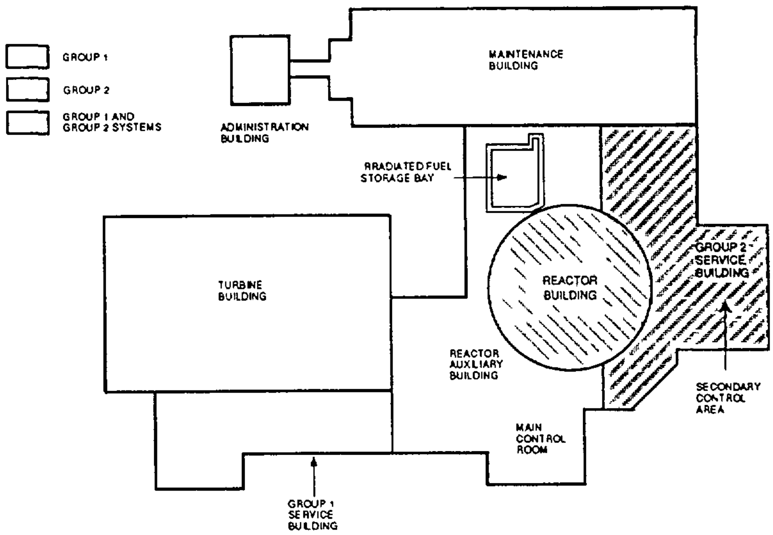



*Fig 5.8.3B CANDU 3 Two group Seperation Philosophy*

development programs. These upgrades include the following:

- A large central mimic display, visible to all staff in the control room.
- A substantial reduction in panel complexity.
- One-man operation capability from a sit-down console during normal operation.
- Critical safety parameter monitoring tied back to emergency operating procedures.
- An alarm annunciation system that eliminates operator information overload during major plant transients.
- Automated safety system operational testing.

A secondary control area is provided, remote from the control centre, with the capability of assuring reactor shutdown, fuel cooling and plant monitoring in the event that the control centre is for any reason nonfunctional or non-inhabitable.

5.8.2.3.3. Electrical systems

The CANDU 3 incorporates comprehensive electrical systems that include Class IV, Class III, Class II and Class I systems, and redundancy and diversity to assure that all plant reliability and safety requirements are met. The Class III power system for example, includes four diesel generating sets, each capable of sustaining all necessary plant loads.

*5.8.2.4. Safety Considerations*

See Section 5.8.3.

5.8.2.5. Buildings and structures

The general arrangement of the CANDU 3 nuclear power plant is shown in Figure 5.8.3A The principal structures included the Reactor Building, two separate Service Buildings and the Turbine Building. Other structures include the Administration Building, two separate Pump houses and the Main Control Area.

The buildings are arranged to provide easy access during construction and to allow separation of systems into 2 groups (Figure 5.8.3B). Group 1 systems provide normal reactor cooling and control, and are necessary for normal electrical production. Group 2 systems provide the safety functions and so are only used in accident situations.

The reactor building (Figure 5.8.1A) is a conventional single unit design consisting of a reinforced concrete structure with a steel liner. The maximum internal pressure that could result from a loss of coolant accident is accommodated by the reactor building design pressure. The reactor building is seismically qualified for the design basis earthquake.

## 5.8.3. Safety concepts

*5.8.3.1. CANDU safety characteristics and origins*

Safety is assured in CANDU 3 through a defence in depth approach that builds on diversity and redundancy, and which takes advantage of the unique CANDU pressure tube reactor concept. Passive systems are used whenever they are shown to be reliable and economic; these systems are complimented by engineered systems. The consistent applicable of human factors principles, and detailed attention to all aspects of plant design also contributed to CANDU 3 safety.

CANDU design practice places emphasis on the performance of the special safety systems (shutdown systems, emergency core cooling system and containment system).

Accidents must be shown to have acceptable consequences, not only if the safety systems work, but also if any safety system is unavailable or impaired. For example, in most other reactors, a loss-of-coolant accident coupled with prolonged unavailability of the emergency core cooling system, would result in melting of the reactor fuel. In CANDU this sequence would lead to damaged fuel but no meltdown. This overall safety approach is achievable because there are at least two ways of providing the safety functions of shutdown and decay heat removal.

Firstly, there are two independent safety shutdown systems, each equally effective in handling accidents.

Secondly, the heavy water moderator provides an independent source of cooling water surrounding the pressure tubes (Figure 5.8.2). The moderator cooling system is designed to remove about 5% of the total reactor thermal power during normal operation, a value equivalent to core decay heat shortly after shutdown. In the event of an accident, the pressure tube is close enough to the calandria tube that an emergency heat removal pathway can be passively achieved. For a loss of coolant accompanied by total failure of emergency core cooling system flow, the pressure tube will overheat, then sag or strain into contact with its surrounding calandria tube. Consequently, the fuel decay heat is transferred to the calandria tube through the pressure tube, and then removed by the moderator. Because of the cooling capability of the moderator, damaged fuel would remain within the pressure tubes, without $UO_2$ melting, so that the core geometry would be retained. This is termed the moderator heat sink.

Thirdly, every postulated accident which releases radioactivity must be shown to have acceptable consequences even if any one of the special safety systems fail to function, including the active components in the containment system (e.g., ventilation dampers fail to close).

The special safety systems themselves are:

- Independent of each other and of the normal Group 1 control and process systems;
- Separated physically from each other, and from the control/process systems, so that common cause events cannot affect more than one safety system;
- Redundant, at both the system and active component level, so that isolated failures, either of active components or of an entire system, cannot disable the safety function;
- Testable during service, to meet a reliability target of 999 times out of 1000 tries;
- Diverse in design and operation, so that a generic fault in design, maintenance or operation cannot affect more than one safety system.

Further safety characteristics include:

The use of natural uranium or other low fissile content fuel in a core lattice arrangement that provides maximum reactivity. This type of fuel requires on-power refuelling, and this in turn means that very little reactivity needs to be held up in movable control devices; no chemicals are added to the reactor coolant for reactivity control. Thus, malfunctions in the control system produce only modest reactivity changes. Further, in a severe accident, a damaged CANDU core would tend to shut down inherently. The injection of ordinary water (e.g. through the emergency core cooling system) inserts negative reactivity in CANDU.

The control and shutdown devices are in the low-pressure moderator, not the high pressure coolant, so are not subject to large hydraulic forces.

Natural coolant circulation will remove decay heat from the fuel if pumping power is lost.

This is effective even if, following a small loss-of-coolant accident, the heat transport system coolant inventory is somewhat depleted.

For hazards such as earthquakes, fires, floods, etc., the plant is protected through implementation of the two group approach described. This entails a combination of redundancy of systems which protect the plant against frequent events, protection of essential systems so they can withstand the event, and separation of redundant systems so the same event cannot disable more than one.

Digital controllers control the plant routinely, freeing the operator of mundane tasks and reducing the likelihood of operator error. In case of accidents, the safety system responses are automated to the extent that no operator action is needed for several hours following an accident.

Radiation doses to the public and to operators of existing CANDUs during normal operation have generally been less than for other reactor types. This is partly due to the economic need to reduce leaks of heavy water, and to the ability to remove defected fuel during operation.

*5.8.3.2. Passive safety*

CANDU 3 Incorporates a number of passive safety features. These include:

- The two independent, passive safety shutdown systems.
- The ability to cool the fuel event if all coolant (normal and emergency supply) are lost by rejection heat to the cool, low pressure moderator.

Additional capabilities that contribute to severe accident mitigations are discussed in 5.8.3.4.

*5.8.3.3. Engineered safety*

A range of engineered systems contribute to CANDU 3 safety. These include:

- The shutdown cooling system utilizing pumps and heat exchangers, can remove decay heat should the steam generator sink be lost. This system can be brought into service at fuel heat transport system temperature and pressure.
- The connection of all pipes that can potentially discharge reactor coolant from the heat transport system (HTS) to a high pressure tank (the bleed cooler). The relief valves on this tank are set above HTS operating pressure. Hence, failure of any valve in the above lines in the open position does not lead to a loss of reactor coolant.
- The provision of an emergency water supply to the steam generators to backup the normal (high capacity electric main feedwater pumps, and electric and steam turbines auxiliary feedwater pumps).
- The provision of multiple systems and supplies throughout the plant to ensure reliability. For example, four on-site diesel electric generators, two to back up the normal power supplies and two for emergency.

### 5.8.3.4. *Severe accident mitigation*

Probabilistic analysis has been a hallmark of CANDU safety philosophy since its inception; Canada remains one of the few countries whose regulations are based largely on probabilistic considerations. Probabilistic safety assessments have been performed for the Darlington station by Ontario Hydro, and for CANDU 3 in a joint study by AECL and the Dutch research organization KEMA. Both studies reached a similar conclusion; e.g. because of the redundant heat removal paths in CANDU, and redundant shutdown systems, the predicted severe core damage frequency is less than about $4x10^{-6}$ events/year.

With respect to the redundant heat removal paths, the moderator can act as an emergency heat sink even with no water in the fuel channels. Should the moderator heat removal system subsequently fail, the large water-filled tank surrounding the calandria vessel provides an additional line of defence (Figure5.8.7.). Its primary purpose is to provide shielding of the concrete reactor vault from neutrons and gamma rays. However it can also act as a passive emergency water reservoir in case of a severe core damage accident; that is, should the primary coolant system (HTS), the emergency core cooling system, and the moderator heat removal system all fail, the water-filled vault will retain the debris inside the calandria, by keeping the outside of the calandria shell cool for a minimum period of 24 hours. This allows time for fission products to decay further and for decay heat to reduce, and for emergency planning.

### 5.8.3.5. *Safety related systems and features*

The following tables provide further detail on the CANDU 3 safety related systems and safety functions. These include:

Table 5.8.1. Main Safety Related Systems
Table 5.8.2. Main Accident Initiators
Table 5.8.3. Design Features for Prevention
Table 5.8.4. Design Features for Mitigation

## TABLE 5.8.1. MAIN SAFETY RELATED SYSTEMS IN THE CANDU 3

| Name | Safety graded | Main characteristics |
|---|---|---|
| 1 HEAT TRANSPORT SYSTEM | CSA Class 1 DBE | The system safety functions are to remove heat from the core and transfer it to the steam generators, and to maintain the integrity of the reactor coolant pressure boundary |
| 2 PRESSURE AND INVENTORY CONTROL SYSTEM | CSA Class 1,3 DBE | The system safety functions are to provide overpressure protection of the heat transport system, to maintain sufficient reactor coolant inventory during and after operational states<br>The system consists of the pressurizer, bleed cooler, heat transport system/relief valves, and feed/bleed flowpaths with $D_2O$ storage tank and feed pumps |
| 3 SHUTDOWN COOLING SYSTEM | CSA Class 1 DBE | The system is designed to provide long-term decay heat removal from the fuel It also has a capability to act as a high-pressure and high-temperature heat sink for events leading to total loss of steam generator feedwater |
| 4 MODERATOR SYSTEM | CSA Class 3 | The system is capable to provide a backup heat sink in the event of a loss of coolant accident coincident with the ECCS unavailable<br>The system, containing the low-pressure cooled $D_2O$ moderator, includes the calandria, pumps and heat exchangers |
| 5 MAIN STEAM SYSTEM | CSA Class 2 DBE | The safety related portion of the system includes the steam generator secondary side and the main steam safety valves.<br>The system provides overpressure protection of the steam generators and provides short term decay heat removal from the heat transport system |
| 6 FEEDWATER SYSTEMS | CSA Class 3 | The system includes both the main feedwater pumps and the auxiliary feedwater pumps The system function is to supply feedwater to the steam generators in order to ensure the steam generator heat sink Back up is by a completely separate emergency supply |
| 7 REACTOR SHUTDOWN SYSTEMS (Special Safety Systems) | AECB R-8 CSA N290 1 DBE | There are two passive fully capable, fast-acting independent and diverse shutdown systems independent of the reactor regulating system |
| 8 REACTOR REGULATING SYSTEM | CSA N290 4 | Provides for control of the reactor power and neutron flux distribution during plant operation It also has the capability to shut down the reactor for anticipated operational occurrences |

## TABLE 5.8.1. (cont.)

| | | |
|---|---|---|
| 9 EMERGENCY CORE COOLING SYSTEM (Special Safety System) | CSA Class 2 3 AECB R-9 SDE | The ECC system services to provide cooling to the fuel in a HTS circuit loop that has experienced a LOCA<br>It consists of passive high-pressure injection (pressurized tanks), low pressure recirculation(pumped)/recovery portion |
| 10 CONTAINMENT (Special Safety System) | CSA Class 4 AECB R-7 DBE | The containment system serves as a barrier to the environmental dispersal of radionuclides which have accidentally entered into the containment atmosphere |
| 11 GROUP 2 FEEDWATER SYSTEM | CSA Class 6 DBE | The G2 feedwater system provides coolant to any steam generator with feedline or feedline plus steamline break due to an earthquake<br>The water supply to the steam generators is a via the Group 2 feedwater storage tank It can provide cooling water for at least 10 hours The supply to the steam generators is through a spray ring above the tube bundle |
| 12 ELECTRIC POWER SYSTEM | Class I, II and III Electric System CSA B290 5 | The Class III system powers designated safety-related and economic equipment protection loads (e g ECCS pump motors, valves etc ) Normally Class IV power supplies the Class III power When the Class IV system fails two redundant standby diesel generators provide Class III power<br>The ac Class II and dc Class I systems supply un-interruptible power to the control and safety systems |
| 13 GROUP 1 SERVICE WATER SYSTEMS | CSA Class 6 | These systems remove heat from equipment including safety related equipment and transfer this heat to the ultimate heat sink<br>The systems are comprised of a recirculated cooling water system (closed loop) and a raw service water system (once-through) connected to the ultimate heat sink |
| 14 GROUP 2 SERVICE WATER SYSTEM | CSA Class 6 DBE | This system provides emergency backup to the Group 1 Recirculated Cooling Water for cooling the ECC heat exchangers it is also seismically qualified to provide cooling to a Shutdown Cooling system heat exchanger after a DBE |

**CLARIFICATIONS FOR TABLE 5.8.1.**

'Safety graded' is intended to mean the specification of codes, standards, regulatory requirements and other qualification requirements (notably the seismic design level) applicable to a system For each safety related system the table provides

- the CSA class for pressure retaining systems
- the applicable CSA standards for other types of systems
- the applicable regulatory policy statements (Atomic Energy Control Board (AECB) R-7, R-8 and R 9 for the special safety systems)
- the seismic qualification level applicable DBE (Design Basis Earthquake) or SDE (Site Design Earthquake)

TABLE 5.8.2. MAIN ACCIDENT INITIATORS FOR THE CANDU 3

- LOCA (Primary): Loss of Primary Coolant Accident
- LOCA (Secondary): Secondary Pipe Rupture (feedwater or steam)
- LOCA (Interfacing) e.g. Steam Generator Tube Rupture
- Primary Transients
- Secondary Transients (turbine trip)
- Loss of electric sources (partial)
- Total loss of the heat sink
- Total loss of the steam generator feedwater
- Station blackout

**CLARIFICATIONS FOR TABLE 5.8.2.**

In CANDU, all postulated failures are considered in conjunction with the postulated failure of each of the special safety systems (one at a time), as design basis events.

Anticipated Transients Without Scram (ATWS) are not included in the design basis of CANDU plants, because of the extremely low frequency expected for such events. CANDU plants have three systems with independent and diverse capability for automatic reactor shutdown

TABLE 5.8.3. DESIGN FEATURES FOR PREVENTION AND PROTECTION LEVELS THAT REDUCE (R), SUPPRESS (S), THE INITIATORS FREQUENCY OR LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES (L)

| | |
|---|---|
| PREVENTION LEVEL | |
| LOCA (Primary) | - Lifetime monitoring of pressure tubes and heat transport system piping and components<br>- Leak detection in annual gas system |
| LOCA (secondary) | Lifetime monitoring and inspection of secondary components and piping |
| LOCA (interfacing) | - Lifetime inspection of steam generators<br>- Heavy water appearing in light water system is detected and alarmed |
| Primary Transients | Lower bundle powers increase design margins |
| Secondary Transients | - ASDV's, CSDV's and MSRV's |
| Loss of Electric Sources | Redundant systems provide power to all critical systems |
| Total Loss of the Cold Source (water) | Separate cooling water systems for normal and safety systems, redundancy |
| Total Loss of the S G Feedwater | Turbine or diesel driven auxiliary feedwater pump, emergency water supply |
| Station Blackout | All critical loads carried by standby diesel generators or passive systems |
| PROTECTION LEVEL | |
| LOCA (Primary) | 2 independent and very fast, passive shutdown systems<br>Reliable indications of loss of inventory<br>Emergency Core Cooling System<br>- Moderator System backup<br>Shield cooling system backup |
| LOCA (Secondary) | - Steam generator and shut down cooling system heat sinks Note, fast cooldown does not insert positive reactivity in CANDU |
| LOCA (Interfacing) | - Detection ($D_2O$ in steam)<br>Shutdown cooling system heat sink |
| ATWS | 2 independent and very fast, passive shutdown systems |
| Primary Transients | - Redundant control and shutdown systems/pressure relief |
| Secondary Transients | - Redundant control and shut down systems/pressure relief |
| Loss of Electric Power | Fail safe design and design to allow time for operator action |
| Total Loss of Heat Sink | - Not credible as demonstrated by PSA Multiple redundant heat sinks available |
| Total Loss of S G Feedwater | Actuates shutdown systems, emergency supplies<br>Shutdown cooling available |
| Station Blackout | - All critical loads carried by standby diesel generators, loss of Class IV and diesel not credible Also passive (gravity) supply to steam generator |

## TABLE 5.8.4. DESIGN FEATURES FOR MITIGATION LEVEL OF CANDU 3

| Safety Functions | Systems (cf Tab 5 8 1 ) | Passive/Active | Design features/Remarks |
|---|---|---|---|
| Design Basis | All single failure accidents and selected dual failure accidents | | LOCA + LOECC and LOCA + any safety system failure are included |
| Fission product containment | 1 Fuel clad/sheath<br>2 Heat transport system pressure boundary<br>3 Containment envelope | P | Multiple barriers to release with in-service inspection including activity release from fuel, pressure tube leak-before-break and on-line detection of leak rate from containment |
| Coolant inventory | 1 Feed and bleed system<br>2 Emergency core cooling (ECC)<br>3 Steam generator feed makeup | A | 1 and 2 are on the primary circuit and 3 and 4 on the secondary 1 and 3 are for normal operation and accidents without loss of coolant 2 is for primary LOCA 4 is for earthquakes and secondary side pipe breaks |
| Decay heat removal | 1 Shutdown cooling system<br>2 Emergency core cooling system<br>3 Emergency water system (EWS) | A | System 1 is used on a primary loop without a break System 2 is used on a primary loop with a pipe break |
| Reactivity control | 1 Reactor regulating system<br>2 Shutdown system #1, shut-off rods<br>3 Shutdown system #2, Gd poison injection<br>4 Loss or dilution of $D_2O$ moderator | 1 A<br>2 P<br>3 P<br>4 P | Power manoeuvres include ramp setback, stepback and trip 2 and 3 are engineered safety systems for trip 1 can serve in an assisting role to shutdown systems No 1 and 2 In 4 $H_2O$ from ECC or leakage or boiling down of moderator leads to subcriticality |
| Primary circuit pressure control | 1 HT pressurizer with steam-bleed valves<br>2 Feed and bleed system<br>3 Automatic liquid relief valves<br>4 Main steam safety valves | A | Systems 1 and 2 provide primary circuit pressure control Systems 3 and 4 prevent overpressure and provide rapid depressurization |

TABLE 5.8.4. (cont.)

| Safety Functions | Systems (Cf Tab 5 8 1 ) | Passive/Active | Design features/Remarks |
|---|---|---|---|
| Severe Accident | Beyond Design Basis Accidents | | Very low probability triple failure accidents |
| Containment temperature and pressure control | 1 Dousing system<br>2 Local area coolers (LACs)<br>3 Other condensation on structure | 1 P<br>2 A<br>3 P | Post-accident, the dousing system and the condensation on structures are available in the early part, the LACs are available in the long term |
| Heat Removal | 1 Moderator cooling the fuel channel or fuel in the calandria<br>2 Shield coolant cooling the calandria and its contents | P | If primary coolant is available, thermosyphoning cools the fuel For a primary loop without coolant the moderator and the radial shield coolant become heat sinks in severe accidents They act in tandem the heat removal efficiency is improved if electrical power is available |
| Tightness Control | 1 Containment isolation system<br>2 Containment liner | P | The isolation system automatically buttons up the containment ventilation system, wall penetrations of process piping and airlocks and hatches |
| Inflam gas control | Igniters for hydrogen-air-stream mixtures | P | Battery operated glow igniters are automatically heated up on a LOCA signal |
| Fission product containment | 1 From intact pressurized containment leakage over many days.<br>2 Optionally, long term use of filtered air discharge system | 1 P<br>2 A | 1 The containment leakage rate is maintained within acceptable limits,<br>2 The engineered FAD doesn't have the capacity of an Emergency FAD, but can be used at moderate pressure and activity levels |
| Corium management | 1 Primary heat transport cooling within the pressure tubes<br>2 Moderator cooling within the calandria<br>3 Shield coolant within the calandria vault<br>4 Base slab | P | Any corium would have to penetrate the three consecutive boundaries and within each boundary a diverse, independent cooling is provided With natural uranium fuel $D_2O$ moderator boiling $H_2O$ ECC coolant and $H_2O$ shield coolant, the disassembled core quickly becomes subcritical and there is no danger of recriticality |

### 5.8.4. Design data

## I. GENERAL INFORMATION

1. Design name: CANDU 3
2. Designer/Supplier address: AECL CANDU, Saskatoon, Sk. Canada S7K 2C3
3. Reactor Type PHWR Number of modules/per plant: 1
4. Gross thermal power (MW-th) per reactor: 1441 MW-th
5. Net electrical output (MW-e) per reactor: 450 MWe*
6. Heat supply capacity (MW-th): 1372 MW-th

## II. BASIC DESIGN DESCRIPTION

### A. CORE AND REACTIVITY CONTROL

7. Fuel material: Natural $UO_2$
8. Fuel inventory (tons of heavy metal): 53 t of U
9. Average core power density (kW/litre): 12.8 kW/L
10. Average fuel power density (kW/kgU): 25.4 kW/kgU
11. Maximum linear power (W/m): 54 kW/m of fuel rod
12. Average discharge burnup (MWd/t): 6167 MWd/t of U
13. Initial enrichment (wt%): 0.711 wt%
14. Reload enrichment at the equilibrium (wt%): 0.711 wt%
15. Refuelling frequency (months): 10 months gradual changeover of core with ave. refuelling rate of 15 fuel bundles/full power day
16. Type of refuelling (on/off power): On-power
17. Fraction of core withdrawn (%): 0.33 %/day
18. Moderator material and inventory: ≥99.75 wt.% D20; 199t in calandria
19. Active core horizontal length (m): 5.94 m
20. Calandria diameter (m): 4.916 m
21. Number of fuel assemblies: 2784 fuel bundles in core 12 fuel bundles in each of 232 fuel channels)
22. Number of fuel rods per bundle assembly: 37
23. Bundle array: 1+6+12+18 fuel rods on concentric pitch circles
24. Clad material: Zr - 4
25. Clad thickness (mm): 0.4 mm
26. Number of control rods or assemblies: 24 for reactor regulation
27. Type: 4 Control absorber rods; 12 Adjuster rods; 8 Mechanical Zone Control Absorbers
28. Additional shutdown systems: Two independent passive safety shut-down systems: 24 shu-off rods and 6 liquid poison injection circuits
29. Control rod neutron absorber material: Cadmium, stainless steel; optional cobalt for Cobalt 60 production
30. Soluble neutron absorber: Boron and gadolinium in moderator only
31. Burnable poison material and form: None (in fuel or HTS)

### B. REACTOR COOLANT SYSTEM

#### B1. Coolant

32. Coolant medium and inventory: 122 t of $D_2O$
33. Design coolant mass flow through core (kg/s): 5336 kg/s
34. Cooling mode (forced/natural): Forced
35. Operating coolant pressure (bar): 100 bar (a)
36. Core inlet temperature (°C): 266°C
37. Core outlet temperature (°C): 310°C

#### B2. Reactor calandria vessel/pressure tube

38. Overall length of assembled vessel/tube (m): 6.35 m
39. Inside diameter: - calandria: 6.32 m
    - pressure tube: 103 mm
40. Average thickness (mm): - calandria: 28.6 mm
    - pressure tube: 4.19 mm

41. Material: - calandria: 304 L stainless steel
    - pressure tube: Zr-2.5 wt% Nb
42. Lining material: N/A
43. Design pressure (bar): - calandria: 0.75 bar(g) at top of calandria
    - pressure tube: 113.8 bar(g) at inlet
44. Gross weight (ton/kg): 308 Mg

**B3. Steam generator**

45. Number of steam generators: 2
46. Type: U-tube in shell
47. Configuration (horizontal/vertical): Vertical
48. Tube material: Incoloy-800
49. Shell material: SA 516 carbon steel
50. Heat transfer surface per steam generator ($m^2$): 4000 $m^2$
51. Thermal capacity per steam generator (MW): 689.7 MW
52. Feed water pressure (bar): 48.5 bar(a)
53. Feed water temperature (°C): 187°C
54. Steam pressure (bar): 47 bar(a)
55. Steam temperature (°C): 260°C

**B4. Pressurizer**

56. Pressurizer total volume ($m^3$): 45 $m^3$
57. Steam volume (full power/zero power, $m^3$): 2 $m^3$ at 100% power

**B5. Main coolant pumps**

58. Number of cooling or recirculation pumps: 4
59. Type: Vertical centrifugal pumps with totally enclosed AC induction motor
60. Pump mass flow rate (kg/s): 1544 l/s of $D_2O$ at 268°C
61. Pump design rated head: 265 m
62. Pump nominal power (kW): 4500 kW
63. Mechanical inertia (kg $m^2$): 1670 kg $m^2$

## C. CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS)

There are no chemicals in the reactor coolant for reactivity purposes.

64. Number of purification extraction lines: 1
65. Number of purification pumps: 2
66. Number of injection points: 1
67. Feed and bleed connections: 1

## D. CONTAINMENT

68. Type: Reinforced concrete with steel liner
69. Overall form (spherical/cyl.): Vertical cylinder with domed top
70. Structural material: Reinforced concrete
71. Liner material: Steel
72. Single/double wall: Single wall
73. Dimensions (diameter, height) (m): 38.6 m inside dia. x max. 46.4 m inside height above top of base slab
74. Design pressure (bar): 3.3 bar (g)
75. Design temperature (°C): 120°C
76. Design leakage rate (% per day): 0.5% of free volume/day at design pressure

# III. <u>SAFETY RELATED SYSTEMS</u>

## A. DESIGN CONDITIONS

**A1. Fission product retention**

77. Containment spray system (Y/N): N
    a. Duration
    b. Flow rate ($m^3$/h)

c. Mode of operation (active/passive)
d. Safety graded (Y/N)

78. F.P. sparging (Y/N): N
79. Containment tightness control (Y/N): Y
80. Leakage recovery (Y/N): N
81. Guard vessel (Y/N): N

**A2. Reactivity control**

82. Absorber injection system (Y/N): Y (Shutdown System #2)
    a. Absorber material: Gadolinium
    b. Mode of operation (active/passive): P
    c. Redundancy: 2 out of 3 trip logic 5 of 6 injection tank
    d. Safety graded: Y
83. Shut-off rods (Y/N): Y (Shutdown System #1)
    a. Maximum rod worth (pcm): 9400 pcm
    b. Mode of operation (active/passive): P
    c. Redundancy: 2 of out 3 trip logic 22 of 24 rods
    d. Safety graded: Y

**A3. Decay heat removal**

*A3-1 Primary side*

84. Water Injection: Emergency Core Cooling (ECC)
    a. Actuation mode (manual/automatic): A
    b. Injection pressure level (bar): Initially 66 bar from tank and later 13 bar from pump
    c. Flow rate (kg/s): Max. 1250 kg/s
    d. Mode of operation (active/passive): P initially and A in long term
    e. Redundancy: 2 out of 3 trip logic
       2 x 100% heat exchangers and pumps
    f. Safety graded: Y
85. Water recirculation and heat removal: Without LOCA: Shutdown Cooling System or thermosyphoning through the steam generators, With LOCA: ECC
    a. Intermediate heat sink: Recirculated cooling water (RCW) and ultimately the service water
    b. Mode of operation (active/passive): A with pumping and P with thermosyphoning
    c. Redundancy: 100% in long term
    d. Self sufficiency (h): 3 months
    e. Safety graded: Y

*A3-2 Secondary side*

86. Feedwater
    a. Actuation mode (manual/automatic): A
    b. Flow rate (kg/s): 700 kg/s normal total 28 kg/s auxiliary
    c. Mode of operation (active/passive): A
    d. Redundancy: 3 x 50% pumps + 1 x 4% auxiliary pump + Emergency backup supply
    e. Self sufficiency (h): Depends on accident type
    f. Safety graded: Y (Emergency Supply)
87. Water recirculation and heat removal
    a. Ultimate heat sink (cold source): Group 2 RSW
    b. Mode of operation (active/passive): A
    c. Redundancy: 2 x 100% pumps
    d. Self sufficiency (h): 10 minimum
    e. Safety graded: Y

*A3-3 Primary pressure control*

*Pressurizer provides principal control*

88. Implemented System Pressure and Inventory Control System: The Liquid Relief Valves provide overpressure protection together with reactor power stepback. In accidents with LOCA crash cooling by opening the Main Steam Safety Valves depressurizes the secondary circuit, which cools and depressurizes the primary circuit and helps ECC injection for small breaks.
    a. Actuation mode (manual/automatic): A

b. Side location (primary/secondary circuit): Pressure and Inventory Control and Liquid Relief Valves in primary; crash cooling via secondary circuit

c. Maximum depressurization rate (bar/s): Hardware limited

d. Safety graded: Y

**B. SEVERE ACCIDENT CONDITIONS**

**B.1. Fission products retention**

89. Containment spray system (Y/N): N
90. F.P. sparging (Y/N): N
91. Containment tightness control (Y/N): Y
92. Leakage recovery (Y/N): N
93. Risk of recriticality (Y/N): N

**B.2. Recriticality control**

94. Encountered design feature: Moderator poisoned with boron for guaranteed shutdown state
    a. Mode of operation (A/P): A
    b. Safety graded: Y

**B.3. Debris confining and cooling**

95. Core debris configuration (core catcher): Calandria shell is the core catcher, and ultimately the calandria vault
96. Debris cooling system (name): Moderator system and shield cooling system (the latter provides calandria vault cooling as well)
    a. Mode of operation (A/P): P in short term, A in long term
    b. Self sufficiency: min. 24 h, but unlimited if service water and electrical power are available
    c. Safety graded (Y/N): Classed as safety-related system for reliability and environmental qualification

**B.4 Long term containment heat removal**

97. Implemented system: Local area coolers
    a. Mode of operation (A/P): A
    b. Self sufficiency (h): 3 months
    c. Safety graded (Y/N): Y
98. Intermediate heat sink: Recirculated Water
    a. Self sufficiency (h): 3 months
    b. Safety graded (Y/N): N
99. External coolant recirculation
    a. Implemented components: NA
    b. Mode of operation (A/P): NA
    c. Self sufficiency (h): NA
    d. Safety graded (Y/N): NA
100. Ultimate heat sink: Service water
    a. Self sufficiency (h): Unlimited
    b. Safety graded (Y/N): Y (Emergency Water Supply)

**B.5. Combustible gas control**

101. Covered range of gas mixture concentration: Achievable mixtures of $H_2$-Air-Steam
102. Modes for the combustible gas control
    a. Containment inertation: N
    b. Gas burning: Y
    c. Gas recombining: N
    d. Others: Y (forced circulation)

**B.6. Containment pressure control**

103. Filtered vented containment (Y/N): Y in long term only
    a. Implemented system: Filtered Air Discharge
    b. Mode of operation (A/P): A

c. Safety graded: N

104. Pressure suppression system (Y/N): N

a. Implemented system

b. Mode of operation

c. Safety graded (Y/N)

## C. SAFETY RELATED I&C SYSTEM

Automatic load following (Y/N): Y

- range (% power): 100 to 55%
- maximum rate (%/min): 60%/min for steam plant, but may be limited by turbine

Load rejection without reactor trip (Y/N): Y

Full Cathode Ray Tubes (CRT) display (Y/N): Y on control system and alarm displays

Automated start-up procedures (Y/N): Y

Automated normal shutdown procedures (Y/N): Y

Automated off normal shutdown procedures (Y/N): Y

Use of field buses and smart sensors (Y/N): N

Expert systems or artificial intelligence advisors (Y/N): N for safety systems, but Y for some operating systems, e.g., refuelling sequencing

Protection system backup (Y/N): Y

## D. EMERGENCY POWER SUPPLY SYSTEM

105. Type (diesel, gas, grid connection): Diesel

106. Number of trains: 2 trains DBE qualified, + 2 trains not DBE qualified

## E. AC/DC SUPPLY SYSTEM

107. Type (rectifier, converter, battery): Rectifier, battery, inverter

108. Estimate time reserve (hr): 1hr

# IV. CONVENTIONAL THERMAL CYCLE

## A. TURBINE SYSTEM

109. Type: Tandem-Compound-4 Flow

110. Overall length (m)

111. Width (m): 3

112. Number of turbines: 1

113. Number of turbine sections per unit (e.g. HP/LP/LP): 1 HP, 2 LP

114. Speed (rpm): 1800 rpm

## B. STEAM CHARACTERISTICS

115. H.P. inlet pressure: 45.1 bar(a)

116. H.P. inlet temperature: 260°C

117. H.P. inlet flowrate: 690 kg/s

118. L.P. inlet pressure

119. L.P. inlet temperature

120. L.P. inlet flowrate (per section)

## C. GENERATOR

121. Type (3-phase synchronous, DC): 3-phase synchronous

122. Apparent power (MVA): 565 MVA

123. Active power (MW): 480 MW

124. Frequency (hz): Either 60 Hz/50 Hz

125. Output voltage (kV): 24 kV

126. Total generator mass (t)

127. Overall length

128. Stator housing outside diameter

## D. CONDENSER

Note: Design and performance depends on cooling water temperature; typical values shown

129. Number of tubes: 19198

130. Heat transfer area: 1962 $m^3$

131. Flowrate ($m^3/s$): 18 $m^3/s$ cooling water flow

132. Pressure: 5 kPa(a)
133. Temperature (°C): 21°C

**E. CONDENSATE PUMPS**

134. Number: 2 x 100%
135. Flowrate: 600 kg/s
136. Developed head: 195 m
137. Temperature: 33°C
138. Pump speed: 1200 rpm

### 5.8.5. CANDU 3 Design status

The CANDU 3 development program was initiated in 1986. The detailed design, approximately 70% complete by early 1994, is continuing.

The CANDU 3 program includes the "Up-Front" licensing process by which the licensing basis is agreed to with the regulatory authority during the detail design process, and in advance of the start of construction.

There are no research or development requirements for CANDU 3 implementation. However, advances made through continuing research and development will be adapted by the CANDU 3 when fully demonstrated.

The CANDU 3 is available now, and can be implemented on a relatively fast schedule.

### 5.8.6. CANDU 3 Economics

Cost data for single and two unit CANDU 3 stations is presented below. The data assumes Canadian siting, engineering, procurement, construction, and operating environment. All costs are in January 1993 Canadian dollars, assuming 5% real interest rate, 40#year life for economic analysis purposes, and average lifetime capacity factor of 80%.

The cost of capital modifications of 1% of overnight capital cost per year and the cost of decommissioning, estimated at 10% of overnight capital cost are included. Owners cost of 5% of overnight cost are included in the overnight cost.

| | 1 x CANDU 3 | 2 x CANDU 3 |
|---|---|---|
| Overnight capital - M$ | 1,337 | 2,223 |
| O & M - $/kWe/year | 110 | 86 |
| Fuel - mills/kWh | 3.1 | 3.1 |
| LUEC - mills/kWh | 51 | 42 |

For conversion of above cost data to US dollars, use exchange rate of C$1.30 per US$ for 1993.

The above costs are based on a 50 month duration from site mobilization to in service, which includes contingency for the first unit. Subsequent units will require 42 months or less from site mobilization to in-service.

## 5.9. PRESSURIZED HEAVY WATER REACTOR PHWR-500 SYSTEM DESCRIPTION AND DEVELOPMENT STATUS

### 5.9.1. Basic objectives and features

Nuclear power plants, mostly of the pressurized heavy water(PHWR) type, are being built in the country to meet the increasing demand for energy. The PHWRs also form the first stage of nuclear power plant types to utilise the existing uranium reserves in such a manner as to lead to the full exploitation of the vast thorium resources available in the country.

The Indian PHWR is a pressure tube type reactor using heavy water moderator, heavy water coolant (in a separate high pressure high temperature system) and natural uranium oxide fuel.

The unique design concepts of PHWR systems offer certain intrinsic advantages with respect to safety, both during normal operations and accident conditions. Some of these inherent features are as follows:

1. Relatively large neutron generation time (about 0.001 second);
2. On-power fuelling and hence minimum requirement for reactivity reserve;
3. Reactor core and steam generator configuration is such as to promote and establish thermasyphon mode of cooling the core;
4. Availability of large body of cool heavy water (moderator) around pressure tubes in the core;
5. Availability of large body of cool light water around the calandria (reactor vessel) in its vault.

Various safety related systems and safety support systems are designed to appropriate safety class specifications depending on the functional importance. In addition, systems are seismically designed for two different intensities of earthquake. The operating basis earthquake (ODE) represents the intensity of earthquake for which the systems are designed to remain functional during and after the event. The safe shutdown earthquake (SSE) considers the maximum earthquake potential of the site and only those systems which are required to:

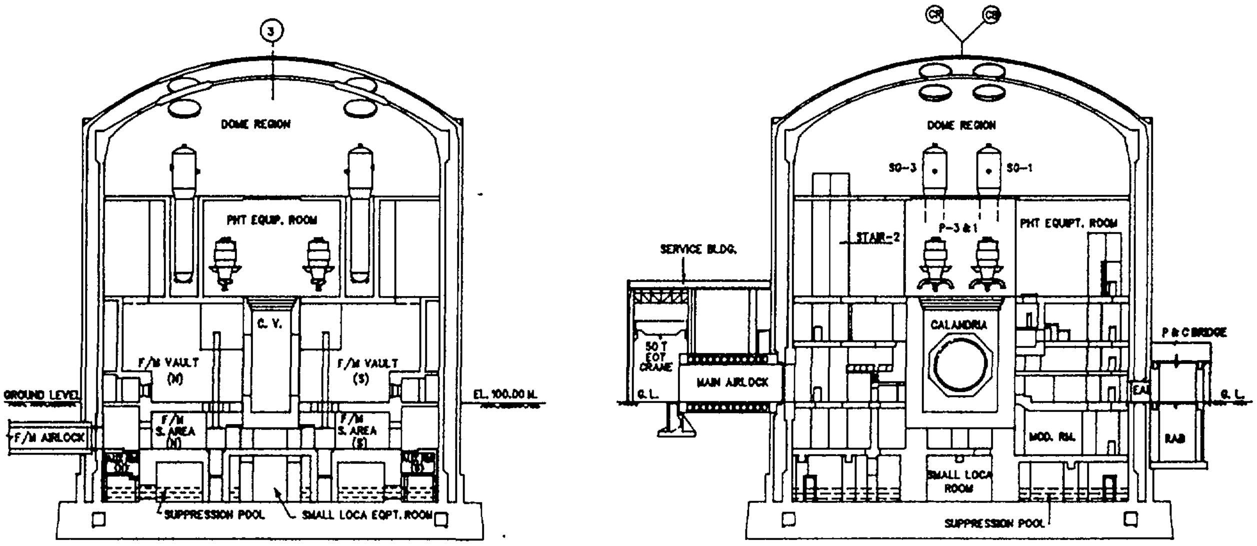
- safely shutdown the reactor and maintain it subcritical;
- remove residual decay heat from the core, and
- mitigate the potential release of radioactivity and to keep them within prescribed limits during operation and within allowable limits during and after accident conditions, are designed to remain functional under SSE condition.

### 5.9.2. Design Description

#### *5.9.2.1 Nuclear Steam Supply System (NSSS)*

The NSSS comprises of various nuclear systems and auxiliaries located in Reactor Building (RD), Reactor Auxiliary Building (RAB), Station Auxiliary Building (SAB), Service

Building (SB) and Control Building (CB). Instrumentation and Control for reactor regulation, process control and reactor protection are also housed in the nuclear island. Typical cross-section of the Reactor Building is shown in Figure 5.9.1.




REACTOR BUILDING SECTION LOOKING EAST

REACTOR BUILDING SECTION LOOKING NORTH

TYPICAL CROSS SECTIONS OF 500 MWe REACTOR BUILDING

*Fig 5.9.1.*

Pressurised Heavy Water Reactors (PHWRS) are horizontal pressure tube reactors using natural uranium oxide fuel in the form of 495 mm clusters. The fuel is cooled by a high-pressure, high temperature circulating heavy water system called the primary heat transport (PHT) system. Heavy Water is also used as moderator in a separate low temperature, low pressure moderator system. Refuelling of the reactor is carried out "on-power" by the fuel handling system. The heat from the reactor is carried away by the heavy water coolant in the PHT system and is given away to the secondary side in the steam generators (SG). The steam from SGs is fed to the turbine-generator in the conventional island for production of electricity.

5.9.2.1.1. Reactor Process System

As mentioned above, the PHT system constitutes part of the nuclear steam supply system (NSSS) and will be briefly described here. The other reactor process systems are:

- moderator system in which heat is dissipated due to slowing down of fast neutrons and absorption of gamma rays;
- reactor shield systems (end shield and calandria vault water) in which heat is dissipated due to attenuation of radiation escaping from the reactor core;
- closed loop process water systems employing demineralised water as a secondary coolant fur the above reactor process system heat exchangers. This process water in the closed loop is in turn cooled by service water in a tertiary circuit.

The PHT system circulates high pressure coolant through the fuel channels to remove the heat generated in fuel. The major components of this system are the reactor fuel channels, feeders, four reactor inlet headers, four reactor outlet headers, four pumps and interconnecting pipes and valves. The headers, steam generators and pumps are located above the reactor and are arranged in two symmetrical banks at both ends of the reactor. The headers are connected to fuel channels through individual feeder pipes. The coolant flow in neighbouring fuel channels is in opposite directions. This arrangement results in a characteristic "figure of 8" circuit layout where coolant in each circuit makes two passes through the core. The PHT system is divided into two independent loops, connected to a common pressuriser. Each loop, with its associated equipment, circulates water through its respective half of the reactor core.

The coolant circulation is maintained at all times during reactor operation, shutdown and maintenance. The PHT pumps are provided with flywheels to provide better flow coast down after pump trip. The system layout as discussed above assures adequate flow for decay heat removal from reactor during shutdown by thermosyphoning action. A separate shutdown cooling system is provided to remove reactor decay heat during cold shutdown conditions. This mode of cooling permits the draining of the steam generators and pumps in the PHT system for maintenance. An emergency core cooling system provides adequate coolant flow to prevent overheating of the fuel in the unlikely event of loss of coolant accident.

PHT system pressure control is achieved through feed and bleed and a pressuriser has been added to improve performance during operational transients. System components are protected from over pressure by instrumented relief valves and suitable regulating system and protective system action. Potential heavy water leak sources are kept to a minimum by using welded construction wherever practicable, and bellow sealed valves. Heavy water leakage collection and recovery systems are connected to the locations where potential leak sources exist.

#### 5.9.2.1.2. On-power Fuelling

On-power fuelling is a feature of all PHWRs which have very low excess reactivity. In this type of reactor, refuelling to compensate for fuel depletion and for overall flux shaping to give optimum power distribution, is carried out with the help of two fuelling machines, which work in unison on the opposite ends of a channel. One of the machines is used to fuel the channel while the other one accepts the spent bundles. In addition, the fuelling machines facilitate on-power removal of failed fuel bundles.

Each fuelling machine is mounted on a bridge and column assembly. Various mechanisms provided allow tri-directional movement of fuelling machine head and make it possible to align it accurately with respect to channels. Various features have been provided which enable clamping of fuelling machine head to the end fitting, opening and closing of the respective seal plugs and perform various fuelling operations.

### *5.9.2.2. Balance of Plant Systems*

#### 5.9.2.2.1. Power Generation System

Turbine building generally houses the power generation system, electrical system and common services system. Other facilities are located in buildings specifically named after ,the facility.

Nearly dry saturated steam is supplied by steam generators to the Turbine. Turbine is a tandem compound machine directly coupled to an electrical generator. The turbine consists of a high pressure, double flow cylinder with external moisture separators and steam reheaters and two double flow low pressure cylinders. Turbine is provided with necessary supervisory and protection instrumentation and devices. All auxiliary systems in respect of turbine-generators viz. turbine bearing oil system, control oil system, condensing system, condensate system, feed water system, feed water heating system, generator stator cooling water system, hydrogen system, heating system, hydrogen system, seal oil system, exciter and excitation system are located in the Turbine Building.

Steam enters the high pressure cylinder and subsequently passes through the moisture separators and reheaters before entering low pressure cylinders. The steam then exhausts to a condenser under vacuum. The condensed steam is extracted from condenser by condensate extraction pumps and the condensate passes through feed water heaters to deaerator. Boiler feed pumps take suction from deaerator and pump feed water via high pressure feed water heaters into steam generators.

Electrical generator directly coupled to the turbine produces electricity and the voltage is stepped up by the generator transformer which in turn is connected to a switch yard. Generated power is thus transmitted to the electrical power grid.

5.9.2.2.2. Auxiliary Feed Water System

Two auxiliary feed pumps (one of them is standby) located in the annex to reactor auxiliary building take suction from condensate storage tanks and supply feed water to the steam generators if the main feed water system is unavailable, say, on failure of class IV power supply.

The system is totally independent from the main feed water system and is capable of supplying feed water at the rated pressure to the steam generator for decay heat removal. The auxiliary feed water system is designed to be operable under safe shutdown earthquake condition.

5.9.2.2.3. Electrical Power System

As explained above, power is supplied to the grid from the power station. Station service power is obtained from station startup transformer (SUT) which is connected to the grid. Similarly generator through unit auxiliary transformers (UT), provides supply to auxiliary power supply buses. Four classes of power are used to supply station requirements. These differ in their nature and consequent security of their supply.

Class I: Class I system is the 220V DC power supply from batteries. This is considered un-interruptible.

Class II: Class II system is 415V A.C. uninterrupted system. It is normally supplied from motor generator (M.G) sets. These M.G. sets are fed from Class I, 220V DC system. 220V DC system is normally supplied by the Automatic Constant Voltage Rectifiers (ACVRs). Thus, the 220V DC batteries connected to Class I system are normally floating but, feed the M.G. sets when power supply from ACVR's is interrupted.

Class III: Class III system is normally fed from Class IV system. But, when supply from Class IV system fails, emergency Diesel Generators (DGs) feed this system. Thus Class III system is an alternative source which assumes loads within one minute of interruption of Class IV.

Class IV: Class IV is the normal source of power to station auxiliary system and supplies loads, which can be dropped on a reactor shutdown.

Auxiliary Power Supply System consists of:

Class IV i) 6.6 kV system (supplied from SUT & UT)
ii) 415V, system from 6.6 kV Class 1V)

Class III i) 6.6 kV system (from Class IV buses with PGs as standby source).
ii) 415V system (supplied from 6.6 kV Class III bus).

Class II 415V system (supplied from UPS with static inverters connected to Class I).
Power:

Class I power: 220V DC (supplied from Class III 415V system through ACVRs with batteries as standby source).

Auxiliary Supply System to Instrumentation and Control (also called Control Power Supplies) are as follows:

Class II Control: 240V DC/AC System (supplied from control converter connected to Class I Power and control batteries).

Class I Control: i) 220V DC System (supplied from Class III, 415V System through ACVR's with 220 V control batteries as standby source).
ii) 24V DC System (supplied from Class III buses through rectifiers with 24 V batteries as standby source).

The station electrical power distribution system has been divided into two physically separate divisions to minimise common cause failures. Division-1 and Division-2 supply the systems of the plant dedicated to normal power production, and safety-related loads. These loads are appropriately shared between the two Divisions of power supply. Class I, II & III supplies, distribution systems and connections between the two Divisions, are seismically qualified to be operational after an earthquake.

5.9.2.2.4. Waste Disposal and Environmental Releases

Facilities are provided at the station for the safe disposal of all radioactive wastes. While all solid wastes are stored at site, releases of liquid and gaseous effluents are so organized that the prescribed dose limits for members of the public are strictly adhered to.

All airborne and gaseous activity from the plant is routed to the atmosphere via a tall stack along with ventilation exhaust air from the Reactor and Service Buildings. Before entry into the stack, the air is passed through high efficiency particulate air filters for removal of most of the particulate radioactivity.

The environmental conditions obtainable at the site determine the quantum of annual dose limit apportionment to each of the air, water and land environments. The apportionment for the air route is then translated into a set of derived working limits taking into account the following:

a) External exposure (due to beta and gamma emitters) from plume;
b) Internal dose due to inhalation;
c) Internal dose due to ingestion of agricultural produce affected by deposition of radioactive material on ground.
d) Internal dose due to ingestion of ground deposited radioactivity by animals and subsequent consumption of milk, meat or eggs.

The above derived working limits are , further apportioned for daily routine releases, and occasional short term high releases, in such a manner that the annual dose limit is not exceeded.

Radioactive effluent released from the stack are continuously monitored for inert gas activity, particulate activity, iodine and tritium, to ensure that releases are within stipulated limits.

The liquid wastes generated at the station are segregated into four categories on the basis of activity level and chemical characteristics of the waste. While no treatment is required for the least active category (potentially active wastes but with no measurable activity above background, viz. those from laundry, showers and washrooms) the other categories are sent to the liquid waste management facility where they are given appropriate treatment like filtration and dilution for active non chemical waste (some time ion exchange is also employed); neutralisation for active chemical waste and solar evaporation for tritiated waste if the site condition are suitable. After treatment, the wastes are diluted with blowdown from the cooling towers for inland sites and CCW for coastal sites to achieve the final stipulated discharge concentration limit.

The diluted wastes are then discharged into the appropriate water body, viz. canal/river/lake/sea. In cases where there is a possibility of water stream in canal/river not being available for some time during the year, adequate hold-up capacity for storing liquid wastes for that period is provided.

The permitted daily average liquid activity discharge rates or concentration limits are based on the quantum, of annual dose limit apportioned to water route. The guiding principle is that there will be no restriction on downstream water utilization. These discharge limits accordingly take into account the nature and methods of water utilisation and biological concentration factors for organisms which may form food for the population. The liquid effluent from the station are discharged on a batch basis. Each batch is sampled and activity determined before release, so as to ensure that stipulated release limits are adhered to.

Solid wastes from the station are generally grouped into three categories, based on the level of activity. The low active wastes (up to 2 mGy/hr) comprise mainly paper trash, mops, glassware, polythene articles etc. which can be either incinerated or baled into small tablets and stored. Medium level wastes (up to a few mGy/hr) consist of process residues and a few highly contaminated articles. These wastes are stored in concrete-lined underground trenches. Highly active wastes (registering dose rates of lGy/hr and above) include filter cartridges, defective fuel bundles, cleanup resins, used sources, etc. Specially constructed tile-holes of RCC construction and steel lining are used for storage of these wastes.

A solid waste management facility within the plant perimeter at a suitable location, provided space for storage of solid wastes of all categories.

5.9.2.2.5. Control of Personnel Exposures

In order to minimise the spread of contamination a system of radiological zoning is adopted. The interiors of the plant building and units have been divided into four zones (1 to 4, according to the contamination potential) in each area. At each interzonal point, foot monitors and friskers (contamination monitors) are located, personnel and equipment are checked for contamination before moving from a higher zone to a lower zone. In addition, rubber areas are set up to check the spread of contamination from locations where floor contamination exists, or is anticipated during work. Protective rubber overshoes are required to be worn when entering such areas.

The shielding in the plant has been designed to reduce the radiation dose rate to below 0.001 milli sievert/hr in all areas intended for normal full time occupancy. Higher levels are permitted in areas where only partial occupancy is envisaged.

*5.9.2.3. Instrumentation, Control and Electrical Systems*

5.9.2.3.1. General

The instrumentation and control systems in the PHWR plant include a variety of equipment intended to perform display, monitoring, control, protection and safety functions. The concepts presented form the basis for the system design and development. General guide-lines followed are:

a) Electrical transmission of signals is preferred to pneumatic, because of better amenability to further processing in addition to inherent fast response etc.;
b) Equipment free from ageing, wear and not needing routine and preventive maintenance is preferred. Microprocessor-based systems, Programmable Logic Controllers (PLCs), solid-state semi-conductor devices are preferred over mechanical systems having moving parts;
c) Micro-processor based control systems and PLC systems are preferred in view of flexibility in subsequent modification of control schemes;
d) Principles of redundancy, diversity, fail-safe, testability and maintainability are extensively employed to maximise availability while ensuring safety. Physical separation of redundant channels is provided;
e) For important control systems, microprocessors with dual processor configuration will be used as controllers for these loops;
f) Important protective and safety functions are based on majority coincidence (2 out of 3) principle. However, some are based on 2 out of 2 scheme;
g) A high degree of automation is aimed at, to eliminate human error affecting availability/reliability. However, some systems, e.g. $D_2O$ leak detection, failed fuel detection, need operator surveillance and action for remedial measures;

h) Introduction of dedicated, distributed microprocessor based data acquisition and analysis systems with appropriate CRT displays and printout facilities for many systems is expected to reduce the number of conventional indicators, recorders and annunciators in the control room and improve operator efficiency;

i) Majority of the controls are situated in the Central Control Room. Controls for systems which do not directly affect the operation of the main plant are preferably situated in local control panels. However, no control panel, which needs operator attention at regular intervals, is situated in the Reactor Building;

j) Uninterruptable power supplies, not influenced by other major electrical loads in the plant, are provided for all the instrumentation and control systems;

k) Simplicity in design, operator acceptance, obsolescence, current trends in technology are given due consideration;

l) Distributed data acquisition and control system incorporating data bus for control and communication with the process is envisaged.

5.9.2.3.2. Reactor Regulating System (RRS)

The main functions of RRS are:

1. Automatic control of reactor power between $10^{-7}$ FP and 100% FP. This covers raising or lowering of reactor power at a desired rate and maintaining it at any desired level within this range (reactor regulation);

2. Maintain the neutron flux profile in the reactor close to its design shape so as to enable operation at the minimum possible power without exceeding the limits on fuel bundle power (flux tilt control);

3. Monitor specified plant parameters and reduce the reactor power, at a predetermined rate up to predetermined low limits, whenever these parameters cross their respective preset limits (reactor setback);

4. Provide limited xenon override capability;

5. Step reduction in reactor power (step back).

The reactivity devices employed for the above functions of RRS are as follows:

| | **Function** | **Devices** |
|---|---|---|
| a) | Reactor Regulation, Flux Tilt Control and reactor setback | Light water zonal, control compartments (14) |
| b) | Xenon Over-ride, reactivity control, additional positive reactivity | Adjuster Rods (17) |
| c) | Additional negative reactivity, reactor setback | Control Rods (4) |
| d) | Reactor step back | Control Rods (4) |
| e) | Automatic liquid poison addition system | Gadolinium solution addition to moderator |

6.5.2.3.3. Reactor Protection System (RPS)

The shutdown function in PHWRs is achieved by the reactor protection system which is capable of completely terminating any of the postulated reactivity transients in the most reactive state of the core. The characteristic design parameters specified for the system are;

a) The allowable delay between detection of the event and initiation of shutdown;
b) The speed of negative reactivity insertion, and
c) The total worth of negative reactivity addition to the core.

The above requirements are specified based on the worst postulated fast reactivity transients.

In PHWRS, the voiding introduced during large break loss of coolant accidents gives rise to the highest rate of positive reactivity addition. The delay in actuation and the rate of insertion of negative reactivity provided by reactor protection system meets the requirements of terminating the effects of the positive reactivity transients caused by large break LOCA. In addition, loss of regulation where many reactivity devices in the core could be moved at their most reactive withdrawal rates also forms the basis for the choice of the design parameters for the reactor protection system. The total worth of reactor protection system is governed by appropriate combinations of events which include voiding in the core; cool down of fuel, coolant and moderator; as well as decay of xenon poison in a prolonged shutdown. In arriving at the worth of the reactor protection system, consideration is given to the core neutron filter pattern under various expected states of the core viz. different levels of boron concentration in the moderator, positions of regulating devices and the fuel burnup levels.

The reactor protection system comprises of two independent, fast acting, diverse, and physically separate shutdown systems SDS-1 and SDS-2, each of which is capable of performing the above functions independently. SDS-1 comprises of 28 cadmium rods. The second system SDS-2 is capable of high speed injection of gadolinium nitrate solution (in heavy water) directly into the moderator through six horizontal nozzles. There are four cadmium control rods which can be dropped or driven into the core to supplement the negative reactivity capability of the protection system in case of prolonged shutdown.

*5.9.2.4. Safety Considerations and Emergency Protection*

5.9.2.4.1. General

The Concept of defence-in-depth is applied in designing safety systems to achieve functional diversity, i.e. by providing diversely functioning systems that can perform same safety function (e.g. two shutdown systems), multi barriers to prevent release of radioactivity, multi-defence system, using physical separation of systems and components which serve as back-up (in safety functions) to each other, and procuring components for different systems from different suppliers, to the extent possible. Such an approach leads to a design of safety systems which will be tolerant to a wide range of human errors and equipment failures.

5.9.2.4.2. Multiple Barriers

There are six barriers in the transmission path of radioactive fission products from the fuel to the environment.

1) Fuel - diffusion resistant ceramic fuel virtually retains all solid fission products even at the operating temperatures;

2) Fuel sheath - sealed to vacuum technology standards;

3) Primary heat transport system - designed and maintained to achieve very low leakages;

4) Containment - double walled and designed to maintain very low leakage under every accident conditions;

5) Exclusion zone - provides atmospheric dilution of any fission to product release. An area encompassed within a radius of 1.6 km from the plant constitutes the exclusion zone;

6) Green belt - consisting of rows of trees around the plant (no credit taken for this in safety analyses).

5.9.2.4.3. Reactor Shutdown Systems

The reactor protection system protects the reactor and associated equipment by tripping the reactor, when certain plant parameters exceed their respective set points. Two independent reactor shutdown systems are provided in PHWRS. These systems introduce a neutron absorbing material into the reactor core to decrease reactivity and therefore shutdown the reactor. The two shutdown systems are physically separate and diverse in function. The features of the shutdown systems are brought out in Section 5.9.2.3.3.

5.9.2.4.4. Containment System

The containment is basically an envelope around the reactor and other nuclear systems. In normal condition, it acts as a housing. In case of an accident involving a break in the primary heat transport system, it acts as a barrier (fourth and final) against the release of fission products to the environment. This is a safety system comprising of many subsystems which are briefly outlined below.

The concept of double containment has been adopted for Indian PHWRS. The containment structure consists of a cylindrical prestressed cement concrete (PCC) primary containment with PCC dome and a secondary containment of reinforced cement concrete (RCC) structure' completely surrounding the primary containment.

Primary containment is designed to withstand the overpressure and higher temperature caused by release of hot fluid from the primary heat transport system under LOCA condition. It is coated on both sides with epoxy paints to further restrict the release of radioactive fission products during the period of pressurised state.

The secondary containment is basically an additional concrete envelope over the primary containment. This reduces the ground level release of radioactivity by way of holdup and dilution of leakage from the inner primary containment, by a purge and maintaining negative pressure.

The Reactor Building primary containment is divided into two accident based zones - the dry well comprising of the Calandria Vault, Fuelling Machine Vaults, PHT Pump Room, Steam Generator Room and dome region; and the wet well comprising of suppression Pool at the basement and rest of the volumes of RB. The lowest floor in wet well filled with water constitutes the suppression pool. This system consists of vent shafts, starting from PHT pump room floor and leading down to the basement and connected distribution headers submerged in the suppression pool. The distribution headers have many vent openings.

In the case of large size pipe break containing high enthalpy water or steam, the drywell will be full of hot flashed steam-air mixture and pressure in dry well will rise. The resulting pressure differential will drive out the water inside the distribution header. Subsequently, hot air-steam mixture will flow through the vent shafts and is released into suppression pool water through the vent openings. In the process, steam will get quenched in the pool water. Hot air will get cooled and released into the air space above water level. This process will limit the peak pressure and also reduce the energy contained in Reactor Building primary containment.

Coolers of adequate capacity are provided in the fuelling machine vaults, PHT pump room and steam generator room in the Reactor Building. These cooldown the areas and thereby bring down the pressure early in the primary containment following an accident.

Depressurisation of the primary containment by RB coolers alone may take a long time. Hence a provision is incorporated to resort to controlled discharge to stack through filters to achieve faster depressurisation of containment.

The annulus between primary and secondary containment is provided with multipass filtration and recirculation within the secondary containment space. A small negative pressure is maintained in it by purging to stack via filters. This effectively stops any ground level release of activity from RB to environment.

The major penetration in the RB containment are the main airlock, emergency airlock and fuelling machine airlock. They provide entry into and exit from the RB for personnel and equipments. Each airlock has two doors with an interlock between them such that only one door can be opened at any time. Two inflatable seals are provided for better reliability for sealing between each door and its frame.

There are a number of other penetrations in: the containment for electrical cabling, process piping and ventilation ducts. All penetrations are specially designed to achieve minimum leakage. The inlet and exhaust ventilation ducts are each provided with two automatic dampers for reliable isolation of the RB in the case of an accident which is, sensed by high RD pressure 18 g/cm"(g) or high activity in the exhaust duct. Process pipe lines penetrating through the containment are provided with valve(s) at appropriate location so that direct communication between containment atmosphere and outside environment can be isolated.

5.9.2.4.5. Emergency Core Cooling System (ECCS)

This is one of the engineered safety features provided to mitigate the consequences of Loss of Coolant Accident (LOCA) in the event of a break in Primary Heat Transport (PHT) system boundary. The ECCS is designed to provide enough coolant to the PHT system and to transport heat from the core to the ultimate heat sink in such a way as to ensure adequate core cooling during all phases of LOCA.

In the case of small breaks in the PHT system (small LOCA), the ECCS system serves to recover the spilled water and transfer it back to the PHT system after cooling it to ambient temperature. A separate heavy water inventory is also provided for making up losses from the PHT system. In this situation where PHT system inventory of heavy water is maintained, the core cooling function is carried out by the shutdown cooling system.

In the event of large breaks, the ECCS operates in two stages. The initial high pressure injection stage uses nitrogen gas pressure to inject light water (from accumulators), into the reactor core through headers. All the spills flow into the suppression pool in the basement of RB. During the second stage, the suppression pool water is recirculated through the core and cooled in heat exchangers. This mode of cooling can continue on a long-term basis.

5.9.2.4.6. Safety Support System

The electrical power supply system in a PHWR station is divided mainly into two sub-systems. They are Class IV, system and on-site emergency power supply, system (section 5.9.2.2.3).

The requirement of compressed air for safety related systems on a continuous basis is supplied by a separate set of dedicated compressors located in seismically designed buildings. The distribution is also done through, separate, dedicated, seismically designed piping system.

In case of loss of main cooling water supply to the steam generators, there are two sources of water which can be utilised in emergency.

The first source is from the auxiliary feed water supply system (AFSS). The AFSS consists of emergency feed storage tank (EFST) and two auxiliary boiler feed water pumps (ABFP). The delivery piping from each ABFP is divided into four branches for feeding individually the four steam generators. The auxiliary feed lines to the steam generators are totally independent of main feed lines. The system is designed to be operable under seismic conditions and the ABFPs are powered by Class III power supply.

The second source is the fire fighting water system which also provides emergency cooling water supply to steam generators, in case of class III power supply failure. The fire water can be injected into the steam generators, after they are sufficiently depressurised, by operating self-powered diesel engine pumps. This source is also designed to be available under seismic conditions.

5.9.2.4.7. Two-Group Concept

In order to protect the plant against "common mode" incidents such as fires and internally generated missiles that could affect many safety systems at the same time, the concept of independence and physical separation is largely followed in the layout of safety systems and their support systems. In addition, in 500 MWe units, these systems have been divided into two physically separate groups. Thus any common mode incident would not disable more than one of these groups. Similarly, it is arranged such that failure of any service system will not impair both the groups simultaneously. The table below shows how the required safety functions are met by the two group independently.

| **Required Safety Function** | **Safety Systems** | |
|---|---|---|
| | **Group I** | **Group 2** |
| Reactor Shutdown | Shutdown System 1 (SDS-1) | Shutdown System 2 (SDS-2) |
| Decay Heat Removal | Steam Generator cooling/shutdown cooling/ECCS | Cooling by fire water through steam generator/cold moderator, calandria vault water |
| Minimising radioactivity release | Timely operation of SDS-1 & ECCS | Containment system |
| Monitoring of the status with regard to the above functions | Main control room | Supplementary control room & local panels/ controls |

*5.9.2.5. Buildings and Structures*

5.9.2.5.1. Station Layout

The overall plant layout is for a twin-unit complex. In addition, Waste Management facility is also located adjacent to the power plant complex, but within the 1.6 km exclusion zone. Other ancillary facilities located within the plant area include Administration Building, Warehouses and, other store yards etc.

The plant layout for Nuclear Power Station consisting of 2 units of 500 MWe is shown in Figure 5.9.2.

The principal features of the layout are:

- The layout is based on the concept of independent operation of each unit. As far as possible each unit is independent and will share only some of the common facilities for reasons of economy.
- Mirror images are avoided to the maximum extent possible to retain uniformity in layout.
- All safety related systems and components are grouped together and placed in separate buildings/structures of appropriate design.
- The buildings have been grouped according to the seismic classification in keeping with the classification of the system/equipment contained.

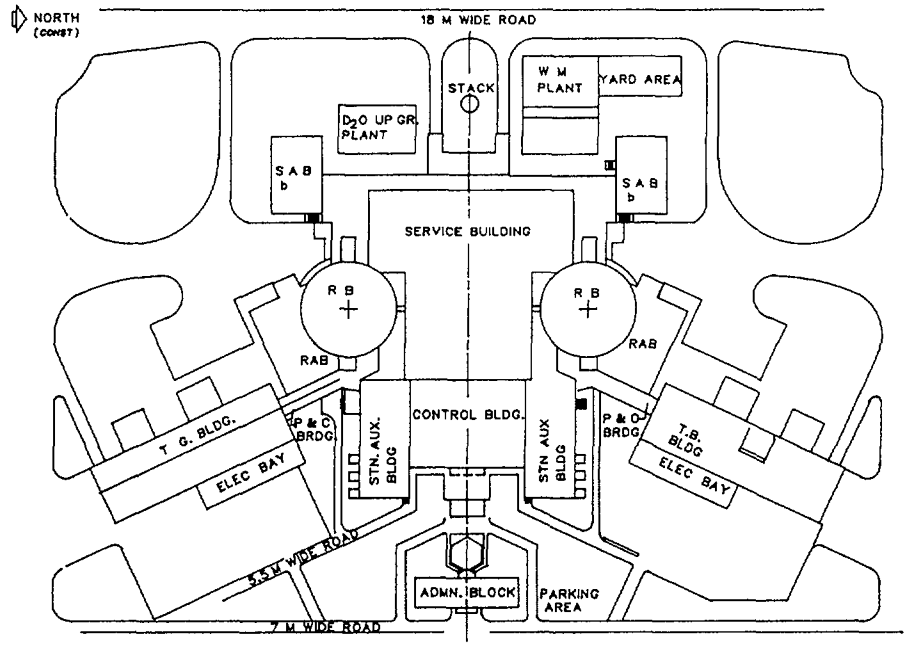



*Fig. 5.9.2. Typical plant layout - (500 MWe).*

- All safety related components and structures are protected from low trajectory missiles emanating from turbine.
- An additional feature comprising of a common fuelling machine calibration and maintenance facility between the two reactor units is provided via fuelling machine airlock and communicating passage leading to each RB. This facility has been located in the Service Building which is common for two units. Transfer ducts for discharge of spent fuel from the RB are kept straight. These transfer ducts from each reactor unit discharge spent fuel into two separate spent fuel storage bays which are located in the Service Building.
- Reactor auxiliary building is located very near to the RB to avoid long piping lengths.
- A separate control building has been provided as a common facility. However, the control room and control equipment rooms in this building are so laid out as to cater for unitised operation.
- Emergency power systems such as diesel generator, motor generators and batteries are provided separately in safety related structures with two such buildings for each unit.
- Space around the RB has also be considered for erection facility for heavy equipment such as steam generators etc. The compactness of the plant layout has been attempted to reduce heavy water inventory, keeping in view ease of operation and movement of the operating staff.
- Physical protection scheme to protect against industrial sabotage and external or internal malevolent action is implemented.

All other buildings have been named after particular facility they house. The containment structure - the ultimate barrier for defence-in-depth is covered in Section 5.9.2.4.1.

A radiological safety programme is established to avoid exposure of persons to radiation due to plant activities and to keep it as low as reasonably achievable (ALARA) where unavoidable. Such unavoidable radiation exposure is further subject to limits specified by the regulatory body. Some aspects of radiation exposure control are covered in Section 5.9.2.2.5.

### 6.9.3. Safety concept

TABLE 5.9.1. MAIN SAFETY RELATED SYSTEMS IN THE PHWR-500

| Name | Safety graded * | Main characteristics |
|---|---|---|
| Primary heat transport (PHT) | class-1 | 392 pressure tubes, configured into two independent loops. Each loop includes reactor headers, feeders, fuel channels, pumps and steam generators |
| Pressure and Inventory control system | class-1,3 | The system maintains reactor coolant inventory during various operational states and performs vital safety function of overpressure protection of PHT system. The major elements of the system are pressurizer, bleed condenser, bleed cooler, relief valves, feed/bleed control valves, storage tank and pumps. |
| Reactor protection system (RPS)<br>(Special Safety System) | class-1&2 | Two independent, diverse systems with redundancy and fail safe design. These are independent of reactor regulating system. Each system, except for actuating devices, is passive and fully capable for keeping the reactor under shutdown state. Primary system adopts gravity-drop. |
| Shutdown cooling system (SCS) | class-2 | Low temperature residual heat removal; can also be used at high temperature for events leading to total unavailability of steam generators. |
| Emergency core cooling System (ECCS)<br>(Special Safety System) | class-2 | Residual heat removal under accident condition. System comprises of high pressure injection followed by low pressure recirculation of cooled suppression pool water. |
| Moderator System (MS) | class-2 | Provides heat sink for loss of coolant accident when ECCS is unavailable |
| Containment System (CS)<br>Containment Pressure Suppression System (CPSS)<br>(Special Safety System) | class-2 | Double containment concept including various sub-systems, namely coolers for energy management; containment isolation for containing radioactivity; controlled discharge and cleanup system for radio-nuclide management; passive suppression pool for peak pressure control. |
| Auxiliary Power Supply (APS) | Safety related as per IEEE requirements | APS comprises of class-I, II & III electrical power systems. Class III power is used for safety related equipment on failure of class-IV system. 4 x 50% standby DG sets are the source of class-III power. The ac class II and dc class I systems provide un-interruptible power to the control and safety systems. |
| Safety related main steam system | class-2 | Secondary side of SG, main steam isolation and safety valves are the main components of the system. The system is designed to provide overpressure protection of SGs and decay heat removal by thermosyphon. |
| Safety Related Feed Water System | class-3 | Main components of the system are emergency feed water storage tanks and auxiliary feed water pumps to supply water to steam generators to ensure heat sink. |
| Safety related service water systems | class-3 | These systems remove heat from reactor components and safety related systems such as shut down cooling and transfer to UHS. A recirculated cooling water system acts as a carrier of heat in a closed loop, for dissipation to UHS by a raw service water system (once-through). |

* Safety class as per *IAEA guide 50-SG-D1*

TABLE 5.9.2. MAIN ACCIDENT INITIATORS FOR THE PHWR-500

- LOCA (primary): Loss of Primary Coolant Accident
- LOCA (Secondary): Secondary Pipe Rupture (water or steam)
- LOCA (interfacing): e.g.: SGTR Steam Generator Tube Rupture
- Primary transients
- Secondary transients (turbine trip)
- Loss of electric sources (partial)
- Total loss of the steam generator feedwater
- Station blackout

Notes

1. Anticipated Transients Without Scram (ATWS) are not included in the design because of extremely low frequency expected for such events
2. In the safety analysis, all postulated failures of PHT pressure boundary are considered in conjunction with the postulated failure of one of the special systems.

TABLE 5.9.3. DESIGN FEATURES FOR PREVENTION AND PROTECTION LEVELS THAT REDUCE (R), SUPPRESS (S), THE INITIATORS FREQUENCY OR LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES (L)

| | |
|---|---|
| **PREVENTION LEVEL** | |
| LOCA (primary) | - "Leak before break" concept and lifetime monitoring of pressure tubes and heat transport system piping and components (L)<br>- Division into two loops (L)<br>- Reactor coolant pumps with multiple mechanical seals (R)<br>- Valveless system (S) |
| LOCA (secondary) | MSIV and ISI of safety related components and piping |
| LOCA Interfacing | $D_2O$ leak detection and ISI of steam generator including tubes |
| Primary transients | Pressurizer, additional inventory (L) |
| Secondary transients | ASDVs, CSDVs, SRVs (R) |
| Loss of electric sources | Two independent, physically separate divisions (R) |
| Total loss of the cold source (water) | More than one source |
| Total loss of the SG feedwater | Fire water fed to SGs |
| Station blackout | Two independent, physically separate divisions (R) |
| **PROTECTION LEVEL** | |
| LOCA (Primary) | - Emergency core cooling system (ECCS)<br>- Automatic actuation of reactor protection system (RPS)<br>- Reliable indication of loss of inventory<br>- Moderator system back-up<br>- Water-filled calandria vault back-up |
| LOCA (Secondary) | MSIV & water stored in emergency water storage tank maintain water level in SG followed by fire water injection |
| LOCA (Interfacing) | Easy isolation of SG |
| ATWS | Not considered, two equally fast acting shutdown systems |
| Primary transients | instrumented relief valves |
| Secondary transients | ASDVs, CSDVs and SRVs |
| Loss of electric sources | Battery power source |
| Total loss of heat sink | Not critical for short-term, due to MS |
| Total loss of SG | Not critical for short term, calandria |
| Feedwater | |
| Station blackout | Not critical for short term water system (CVWS) |

## TABLE 5.9.4. DESIGN FEATURES FOR MITIGATION LEVEL OF PHWR-500

| Safety functions | Systems (Cf Tab 5 9 1 ) | Passive/active | Design features/remarks |
|---|---|---|---|
| **Design basis** | Single and dual failures | | LOCA + any one safety system failure |
| Fission product containment | a) Fuel clad<br>b) Primary heat transport<br>c) Containment | Passive<br>Passive<br>Passive | Multiple barriers to release |
| Coolant inventory | a) Feed and bleed<br>b) ECCS<br>c) Feed water make-up to SG | Active<br>Passive/Active<br>Active | 'a' and 'c' for normal operation and accidents without loss of coolant<br>'b' is for LOCA |
| Decay heat removal | a) SCS<br>b) ECCS<br>c) Emergency feed water to SG | Active<br>Passive/active<br>Active | a) used on primary loop without a break<br>b) used on primary loop with a break<br>c) used on secondary side of SG |
| Reactivity control | a) Reactor regulating system<br>b) Shutoff rods (28)<br>c) Poison injection<br>d) Loss of $D_2O$ moderator | Active<br>Active/Passive<br>Passive<br>Passive | power changes-set back and trip<br>Fail safe principle, redundancy<br>Fail-safe principle, redundancy<br>Loss of D20 moderator leads to subcriticality |
| Primary circuit pressure control | Pressuriser in combination with feed and bleed | Active | Instrumented relief valves in PHT, prevent over pressurization |
| **Severe accident** | Beyond DBA | Passive | Very low probability of triple failure accident |
| Containment temperature and pressure control | CS-Local area coolers<br>CPSS | Passive/active<br>Passive | Air coolers for long term containment cooling<br>For early part of the accident, restricts peak pressure |
| Heat removal | MS<br>CVWS | Active<br>Active | For the primary loop without coolant, MS and shield coolant become heat sinks |
| Tightness control | CS | Passive | Containment isolation system prevents release |
| Inflam gas control | Catalytic recombiners | Passive | System under R&D stage |
| Fission product containment | CS | Passive/ Active | 1 containment leakage rate is maintained within acceptable limits<br>2 Kidney filtration contains Iodine and particulate activity<br>3 Secondary containment at negative pressure through filtered purge |
| Corium management | ECCS/MS/CVWS | Passive | Any corium would have to penetrate four consecutive boundaries-namely, PHT, MS, CVWS and containment pressure suppression pool |
| | CPSS/Base Slab | Passive | Large quantity of stored water in these systems will make disassembled core subcritical without danger of recriticality |

### 5.9.4. Design data questionnaire: PHWR-500

## I. GENERAL INFORMATION

1. Design name: 500 MWe PHWR
2. Designer/Supplier address: Nuclear Power Corporation
3. Reactor type: PHWR Number of modules/per plant: 2
4. Gross thermal power (MW-th) per reactor: 1673
5. Net electrical output (MW-e) per reactor: 500 (gross) nominal
6. Heat supply capacity (MW-th): ---

## II. BASIC DESIGN DESCRIPTION

### A. CORE AND REACTIVITY CONTROL

7. Fuel material: Natural uranium oxide
8. Fuel inventory (tones of heavy metal): 97.8
9. Average core power density (kW/liter): 8.8
10. Average fuel power density (kW/kgU): 17.1
11. Maximum linear power (W/m): 46 700 w/m per rod
12. Average discharge burnup (MWd/t): 7000 MWD/Mt (nominal)
13. Initial enrichment or enrichment range (Wt%): Natural
14. Reload enrichment at the equilibrium (Wt%): Natural
15. Refuelling frequency (months): ---
16. Type of refuelling (on/off power): On-power
17. Fraction of core withdrawn (%)
18. Moderator material and inventory: Heavy water, 290 Mt
19. Active core length (m): 5.94
20. Core diameter (m): 6.39 (active)
21. Number of fuel assemblies: 5096
22. Number of fuel rods per assembly: 37
23. Rod array in assembly: ---
24. Clad material: Zircaloy-4
25. Clad thickness (mm): 0.415
26. Number of control rods or assemblies: 28
27. Type: shutoff rods
28. Additional shutdown systems: Poison injection
29. Control rod neutron absorber material: Cadmium
30. Soluble neutron absorber: Gadolinium nitrate/boric acid
31. Burnable poison material and form: ---

### B. REACTOR COOLANT SYSTEM

**B1. Coolant**

32. Coolant medium and inventory: Heavy water, 165 M
33. Design coolant mass flow through core (kg/s): 7856
34. Cooling mode (forced/natural): Forced
35. Operating coolant pressure (bar): 120 kg/cm$^2$
36. Core inlet temperature (°C): 260
37. Core outlet temperature (°C): 304

**B2. Reactor pressure tube**

38. Overall length of assembled tube (m): 6.3
39. Inside diameter (mm): 103.4
40. Average tube thickness (mm): 4.3
41. Tube material: Zr-2.5% Nb
42. Lining material: ---
43. Design pressure (bar): 123, 126 kg cm$^2$
44. Gross weight: 63

**B3. Steam generator**

45. Number of steam generators: 4
46. Type: U-bundle, with recirculation
47. Configuration (horizontal/vertical): Vertical

48. Tube material: Incoloy-800
49. Shell material: CS
50. Heat transfer surface per steam generator ($m^2$): 3370
51. Thermal capacity per steam generator (MW): 434
52. Feed water pressure (bar): 42.6
53. Feed water temperature (°C): 180
54. Steam pressure (bar): 41.5 $kg/cm^2$
55. Steam temperature (°C): 250

**B4. Pressurizer**

56. Pressurizer total volume ($m^3$): 30
57. Steam volume (full power/zero power, $m^3$): 15/15

**B5. Main coolant pumps**

58. Number of cooling or recirculation pumps: 4
59. Type: Vertical, Centrifugal
60. Pump mass flow rate (kg/s): 1964
61. Pump design rated head: 215.4
62. Pump nominal power (kW): 4900
63. Mechanical inertia (kg $m^2$): 3140

## C. CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS)

64. Number of extraction lines: 2
65. Number of pumps: 2
66. Number of injection points: 2
67. Feed and bleed connections: --

## D. CONTAINMENT

68. Type: Double containment
69. Overall form (spherical/cyl.): Cylindrical with dome
70. Structural material: PCC (inner), RCC (outer)
71. Liner material: --
72. Simple/double wall: --
73. Dimensions (diameter, height) (m): 49.5m/50.55 m
74. Design pressure (bar): 1.44 $kg/cm^2$ (g)
75. Design temperature (°C): 125
76. Design leakage rate (% per day): 0.3% per hour at design pressure

## <u>III. SAFETY RELATED SYSTEMS</u>

## A. DESIGN CONDITIONS

**A1. Fission product retention**

77. Containment spray system (Y/N): --
    a. Duration (h)
    b. Flow rate ($m^3/h$)
    c. Mode of operation (active/passive)
    d. Safety graded (Y/N)
78. F.P. sparging (Y/N): Yes
79. Containment tightness control (Y/N): Yes
80. Leakage recovery (Y/N): Yes
81. Guard vessel (Y/N): No

**A2. Reactivity control**

82. Absorber injection system (Y/N)
    a. Absorber material: Gadolinium
    b. Mode of operation (active/passive): Active (Fail safe)
    c. Redundancy: Yes
    d. Safety graded: Yes
83. Control rods (Y/N)
    a. Maximum control rod worth (pcm)
    b. Mode of operation (active/passive): Active (fail safe)

c. Redundancy
d. Safety graded: Yes

**A3. Decay heat removal**

*A3-1 Primary Side*

84. Water injection
a. Actuation mode (manual/automatic): Automatic
b. Injection pressure level (bar)
c. Flow rate (kg/s)
d. Mode of operation (active/passive): Passive
e. Redundancy: No
f. Safety graded (Y/N): Yes

85. Water recirculation and heat removal
a. Intermediate heat sink (or heat exchanger): Yes
b. Mode of operation (active/passive): Active
c. Redundancy: Yes
d. Self sufficiency (h): Yes
e. Safety graded: Yes

*A3-2 Secondary Side*

86. Feed water
a. Actuation mode (manual/automatic): Automatic
b. Flow rate (kg/s)
c. Mode of operation (active/passive): Active
d. Redundancy: Yes
e. Self sufficiency (h): Yes
f. Safety graded: Yes

87. Water recirculation and heat removal
a. Ultimate heat sink (cold source): Seawater/Atmosphere
b. Mode of operation (active/passive): Active
c. Redundancy: Yes
d. Self sufficiency (h): Yes
e. Safety graded - Yes

*A3-3 Primary Pressure Control*

88. Implemented system (Name): PHT/SG pressure control
a. Actuation mode (manual/automatic): On during operation
b. Side location (primary/secondary circuit): Both sides
c. Maximum depressurization rate (bar/s)
d. Safety graded: Y

## B. SEVERE ACCIDENT CONDITIONS*

**B.1 Fission products retention**

89. Containment spray system (Y/N)
90. F.P. Sparging (Y/N): Yes
91. Containment tightness control (Y/N): Yes
92. Leakage recovery (Y/N): Yes
93. Risk of recriticality (Y/N): No

**B.2 Recriticality control**: Inherently safe

94. Encountered design feature
a. Mode of operation (A/P)
b. Safety graded

**B.3 Debris confining and cooling**

95. Core debris configuration (core catcher): CVWS/CPSS
96. Debris cooling system (name): CVWS/CPSS
a. Mode of operation (A/P): Active
b. Self sufficiency: Yes
c. Safety graded (Y/N): Yes

---

* All systems must be qualified to operate under the accident conditions.

**B.4 Long term containment heat removal**

97. Implemented system
    a. Mode of operation (A/P): Active
    b. Self sufficiency (h)
    c. Safety graded (Y/N): Yes
98. Intermediate heat sink
    a. Self sufficiency (h)
    b. Safety graded (Y/N): Yes
99. External coolant recirculation
    a. Implemented components
    b. Mode of operation (A/P): Active
    c. Self sufficiency (h)
    d. Safety graded (Y/N): Yes
100. Ultimate heat sink
    a. Self sufficiency (h)
    b. Safety graded (Y/N): Yes

**B.5 Combustible gas control**

101. Covered range of gas mixture concentration
102. Modes for the combustible gas control
    a. Containment inertation: ---
    b. Gas burning: ---
    c. Gas recombining: Yes
    d. Others: ---

**B.6 Containment pressure control**

103. Filtered vented containment (Y/N)
    a. Implemented system
    b. Mode of operation (A/P)
    c. Safety graded
104. Pressure suppression system (Y/N): Yes
    a. Implemented system: Suppression P
    b. Mode of operation: Passive
    c. Safety graded (Y/N): Yes

## C. SAFETY RELATED I&C SYSTEM

Automatic load following (Y/N): No
* range (% power)
* maximum rate (%/min)

Load rejection without reactor trip (Y/N): Yes
Full Cathode Ray Tubes (CRT) display (Y/N): Yes
Automated start-up procedures (Y/N): Yes
Automated normal shutdown procedures (Y/N): Yes
Automated off normal shutdown procedures (Y/N): Yes
Use of field buses and smart sensors (Y/N): No
Expert systems or artificial intelligence advisors (Y/N): No
Protection system backup (Y/N): Yes

## D. EMERGENCY POWER SUPPLY SYSTEM

105. Type (diesel, gas, grid connection): Diesel, Battery
106. Number of trains: Two independent

## E. AC/DC SUPPLY SYSTEM

107. Type (rectifier, converter, battery): Converter, battery
108. Estimated time reserve (hr): One

# IV. CONVENTIONAL THERMAL CYCLE

## A. TURBINE SYSTEM

109. Type: Tandem compound impulse reaction type

110. Overall length (m): 35
111. Width (m): 16
112. Number of turbines/reactor: One
113. Number of turbine sections per unit (e.g. HP/LP/LP): HP/LP (2)
114. Speed (rpm): 3000

**B. STEAM CHARACTERISTICS**

115. H.P. inlet pressure: 41 kg $cm^2$ (a)
116. H.P. inlet temperature: 250.6°C
117. H.P. inlet flowrate: 2957 Mt/hr
118. L.P. inlet pressure: 7.2 kg/$cm^2$ (a)
119. L.P. inlet temperature: 231.2°C
120. L.P. inlet flowrate (per section): 1155 Mt/hr

**C. GENERATOR**

121. Type (3-phase synchronous, DC)
122. Apparent power (MVA): 659
123. Active power (MW): 560
124. Frequency (hz): 50
125. Output voltage (kV): 21
126. Total generator mass (t): 428 Mt
127. Overall length: 8.8 m
128. Stator housing outside diameter: 4 m

**D. CONDENSER**

129. Number of tubes: 26064
130. Heat transfer area: ---
131. Flowrate: 1949 T/hr
132. Pressure (m/bar): 0.0863 kg/$cm^2$
133. Temperature (°C): 42.6 (design)

**E. CONDENSATE PUMPS**

134. Number: 3 x 10% capacity
135. Flowrate: 1210 $m^3$/hr
136. Developed head: 274 m
137. Temperature: 49.2°C
138. Pump speed: 1500

### 5.9.5. Project status

The accumulated experience with 220 MWe PHWR Units and the progressively increasing size of the electrical grids has led to the next logical step of undertaking the larger 500 MWe PHWR project.

#### *5.9.5.1. Entities Involvement*

Nuclear Power Corporation of India Ltd., (NPCIL) is responsible for design, engineering, construction, commissioning, operation and decommissioning of nuclear power plants in the country. Complete research and development (R&D) support covering all the above aspects of nuclear power plant is provided by Bhabha Atomic Research Centre (BARC). Various national laboratories and institutes supplement this effort. Technical support, from other consultants is obtained by assigning specific tasks in the areas of design and engineering.

#### *5.9.5.2. Design Status*

At present, conceptual design and basic designs are complete. PSAR is under review by the regulatory body. Detailed design has progressed considerably. Detailed design and engineering respect of most of the systems and major components are complete. Selected long delivery components (like reactor vessel, end shields, PHT pumps, steam generators etc.) are under manufacture as part of advance procurement action.

#### *5.9.5.3. R&D Status*

At present, R&D efforts are mainly related to testing of components (fuel assemblies, coolant channel components, reactivity control devices etc.) and systems in near full-scale facilities to confirm design bases.

#### *5.9.5.4. Licensing Status*

The site at Tarapur, Maharashtra State, for locating the first two units of 500 MWe PHWR has been cleared by the State and Federal Authorities from the point of view of environment and pollution control. The regulatory body has also approved location of the two units at the proposed site. Based on the review of the current level of progress in design, and details of all safety systems and safety support systems, regulatory body has accorded license for construction commencement.

### 5.9.6. Project economics

The 500 MWe PHWR project is a part of the continuing efforts to meet the ever increasing demand for energy in a developing country like India. The unit energy cost for nuclear power compares very favourably to that for a coal fired power plant if the former is located about 1000 km from the location of coal reserves.

## 5.10. PRESSURIZED HEAVY WATER REACTOR PHWR-220 DESCRIPTION AND DEVELOPMENT STATUS

### 5.10.1. Major differences between PHWR-500 and PHWR-220

Pressurized Heavy Water Reactors (PHWRs) form the mainstay of the Indian Nuclear Power Programme. Designs in respect of the 220 MWe PHWRs were mostly standardised by the time the 5th unit was built. Presently seven units of this size are in operation. One is in an advanced stage of commissioning and four are under construction, while four more are in the planning stage. The design of the 500 MWe PHWRs has advanced sufficiently and construction of two units is to commence in the immediate future. The following paragraphs bring out the major differences between the 500 MWe and 220 MWe PHWRs.

1. Primary Heat Transport System (PHT System)

The PHT system of the PHWR-220 is a single "figure of eight" loop with two coolant pumps and steam generators at either end of the reactor. The PHT system of PHWR-500 consists of two figure of eight loops, each loop providing cooling to one half of the core with a coolant pump and steam generator at either end of the reactor. This configuration reduces the severity of the reactivity transient due to void coefficient as only one half of the core experiences voiding during LOCA. The healthy loop is isolated automatically on detection of LOCA.

The PHT pressure control in PHWR-220s is with the help of "Feed and Bleed", whereas it is mainly with the help of the pressurizer in PHWR-500s. In PHWR-500s feed and bleed is also provided for pressure control in the cold condition and it assists the pressurizer in normal operation.

2. Reactor Regulating System (RRS)

Reactor control in PHWR-220s is achieved with the help of control rods, shim rods and adjuster rods. Flux-tilt control is achieved by differential movement of these reactivity controllers.

Reactor control in PHWR-500 is primarily achieved by the light water level control in 14 liquid zonal control compartments. Flux tilt control is achieved by individual control of levels in these compartments. The control functions, including reactor set-back and step-back, are assisted by four mechanical control rods. 17 Adjuster rods provide poison over-ride capability.

3. Reactor Protection Systems (RPS)

In the PHWR 220s two systems, i.e. 14 mechanical shut-off rods which fall freely under gravity into the core and 12 liquid poison stand-pipes into which sodium-penta borate solution is introduced, provide fast acting redundant and diverse shut-down systems. These are reinforced by slow-acting automatic liquid poison addition into the moderator system for prolonged shut-downs.

### 5.10.2. Design data questionnaire: PHWR-220

## I. GENERAL INFORMATION

1. Design name: 220 MWe PHWR
2. Designer/Supplier address: Nuclear Power Corporation of India Ltd.
3. Reactor type: PHWR
   Number of modules/per plant: 2
4. Gross thermal power (MW-th) per reactor: 743
5. Net electrical output (MW-e) per reactor: 194
6. Heat supply capacity (MW-th): ---

## II. BASIC DESIGN DESCRIPTION

### A. CORE AND REACTIVITY CONTROL

7. Fuel material: Sintered $UO_2$ Pellets
8. Fuel inventory (tones of heavy metal): 47 tonnes of $UO_2$ (41.5 tonnes of U metal)
9. Average core power density (kW/liter): 9.38
10. Average fuel power density (kW/kgU): 18.1
11. Maximum linear power (W/m): 8.71 KW/cm
12. Average discharge burnup (MWd/t): 6700
13. Initial enrichment or enrichment range (Wt%): Natural
14. Reload enrichment at the equilibrium (Wt%): Natural
15. Refuelling frequency (months): ---
16. Type of refuelling (on/off power): On-power
17. Fraction of core withdrawn (%): ---
18. Moderator material and inventory: Heavy water, 137 t
19. Active core length (m): 5.005
20. Core diameter (m): 4.51
21. Number of fuel assemblies: 3672 Bundles
22. Number of fuel rods per assembly: 19 elements
23. Rod array in assembly: ---
24. Clad material: Zircaloy-4
25. Clad thickness (mm): 0.41
26. Number of control rods or assemblies: 14
27. Type: shutoff rods
28. Additional shutdown systems: Liquid shut off system
29. Control rod neutron absorber material: Cadmium
30. Soluble neutron absorber: Lithium Penta borate
31. Burnable poison material and form: ---

### B. REACTOR COOLANT SYSTEM

**B1. Coolant**

32. Coolant medium and inventory: Heavy water, 73 Tonnes
33. Design coolant mass flow through core (kg/s): 3528
34. Cooling mode (forced/natural): Forced
35. Operating coolant pressure (bar): 103
36. Core inlet temperature (°C): 249
37. Core outlet temperature (°C): 293.4

**B2. Reactor pressure tube**

38. Overall length of assembled tube (m): 5.33
39. Inside diameter (mm): 82.55 mm
40. Average tube thickness (mm): 3.32 mm
41. Tube material: Zr-2.5% Nb
42. Lining material: ---

43. Design pressure (bar): 109
44. Gross weight: 34

**B3. Steam generator**

45. Number of steam generators: 4
46. Type: U tube budle, with recirculation
47. Configuration (horizontal/vertical): Vertical
48. Tube material: Incoloy-800
49. Shell material: CS
50. Heat transfer surface per steam generator ($m^2$): 2037
51. Thermal capacity per steam generator (MW): ---
52. Feed water pressure (bar): 42
53. Feed water temperature (°C): 170°C
54. Steam pressure (bar): 40.6
55. Steam temperature (°C): 250.7

**B4. Pressurizer**

56. Pressurizer total volume ($m^3$): ---
57. Steam volume (full power/zero power, $m^3$): ---

**B5. Main coolant pumps**

58. Number of cooling or recirculation pumps: 4
59. Type: Vertical, Mech. seal
60. Pump mass flow rate (kg/s): 840
61. Pump design rated head: 184 m
62. Pump nominal power (kW): 2800
63. Mechanical inertia (kg $m^2$): ---

**C. CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS)**

64. Number of extraction lines: 2
65. Number of pumps: 2
66. Number of injection points: 2
67. Feed and bleed connections: --

**D. CONTAINMENT**

68. Type: Double containment
69. Overall form (spherical/cyl.): Cylindrical with dome
70. Structural material: PCC (inner), RCC (outer)
71. Liner material: Epoxy paint
72. Simple/double wall: Double (see 68)
73. Dimensions (diameter, height) (m): 39.6, 53
74. Design pressure (bar): 1.21
75. Design temperature (°C): 120
76. Design leakage rate (% per day): 0.3% per hour at design pressure

**<u>III. SAFETY RELATED SYSTEMS</u>**

**A. DESIGN CONDITIONS**

**A1. Fission product retention**

77. Containment spray system (Y/N): No
    a. Duration (h):
    b. Flow rate ($m^3$/h):

c. Mode of operation (active/passive):
d. Safety graded (Y/N):

78. F.P. sparging (Y/N): Yes
79. Containment tightness control (Y/N): Yes
80. Leakage recovery (Y/N): Yes
81. Guard vessel (Y/N): ---

**A2. Reactivity control**

82. Absorber injection system (Y/N): Y
    a. Absorber material: Boric Anhydride/Boron
    b. Mode of operation (active/passive): Active
    c. Redundancy: Yes
    d. Safety graded: Yes
83. Control rods (Y/N)
    a. Maximum control rod worth (pcm): 31.9 mk
    b. Mode of operation (active/passive): Active
    c. Redundancy: Yes
    d. Safety graded: Yes

**A3. Decay heat removal**

*A3-1 Primary Side*

84. Water injection
    a. Actuation mode (manual/automatic): Automatic
    b. Injection pressure level (bar): ---
    c. Flow rate (kg/s): ---
    d. Mode of operation (active/passive): Active/Passive
    e. Redundancy: No
    f. Safety graded (Y/N): Yes
85. Water recirculation and heat removal
    a. Intermediate heat sink (or heat exchanger): Yes
    b. Mode of operation (active/passive): Active
    c. Redundancy: Yes
    d. Self sufficiency (h): Yes
    e. Safety graded: Yes

*A3-2 Secondary Side*

86. Feed water
    a. Actuation mode (manual/automatic): Automatic
    b. Flow rate (kg/s): ---
    c. Mode of operation (active/passive): Active
    d. Redundancy: Yes
    e. Self sufficiency (h): Yes
    f. Safety graded: Yes
87. Water recirculation and heat removal
    a. Ultimate heat sink (cold source): Seawater/Atmosphere
    b. Mode of operation (active/passive): Active
    c. Redundancy: Yes
    d. Self sufficiency (h): Yes
    e. Safety graded - Yes

*A3-3 Primary Pressure Control*

88. Implemented system (Name): PHT/SG pressure control
    a. Actuation mode (manual/automatic): Automatic
    b. Side location (primary/secondary circuit): Both sides
    c. Maximum depressurization rate (bar/s): ---
    d. Safety graded: Y

## B. SEVERE ACCIDENT CONDITIONS*

### B.1 Fission products retention

89. Containment spray system (Y/N): No
90. F.P. Sparging (Y/N): Yes
91. Containment tightness control (Y/N): Yes
92. Leakage recovery (Y/N): Yes
93. Risk of recriticality (Y/N): No

### B.2 Recriticality control: Inherently safe

94. Encountered design feature
    a. Mode of operation (A/P): ---
    b. Safety graded: ---

### B.3 Debris confining and cooling

95. Core debris configuration (core catcher): CVWS/CPSS
96. Debris cooling system (name): CVWS/CPSS
    a. Mode of operation (A/P): Active/Passive
    b. Self sufficiency: Yes
    c. Safety graded (Y/N): Yes

### B.4 Long term containment heat removal

97. Implemented system
    a. Mode of operation (A/P): Active
    b. Self sufficiency (h): Yes
    c. Safety graded (Y/N): Yes
98. Intermediate heat sink
    a. Self sufficiency (h): Yes
    b. Safety graded (Y/N): Yes
99. External coolant recirculation
    a. Implemented components: ---
    b. Mode of operation (A/P): Active
    c. Self sufficiency (h): Yes
    d. Safety graded (Y/N): Yes
100. Ultimate heat sink
    a. Self sufficiency (h): Yes
    b. Safety graded (Y/N): Yes

### B.5 Combustible gas control

101. Covered range of gas mixture concentration
102. Modes for the combustible gas control
    a. Containment inertation: ---
    b. Gas burning: ---
    c. Gas recombining: Yes
    d. Others: Mixing and venting

### B.6 Containment pressure control

103. Filtered vented containment (Y/N):
    a. Implemented system: Controlled discharge
    b. Mode of operation (A/P): Active
    c. Safety graded: Yes
104. Pressure suppression system (Y/N): Yes
    a. Implemented system: Yes
    b. Mode of operation: Passive
    c. Safety graded (Y/N): Yes

* All systems must be qualified to operate under the accident conditions

### C. SAFETY RELATED I&C SYSTEM

Automatic load following (Y/N): No
* range (% power)
* maximum rate (%/min)

Load rejection without reactor trip (Y/N): Yes
Full Cathode Ray Tubes (CRT) display (Y/N): Yes
Automated start-up procedures (Y/N): Yes
Automated normal shutdown procedures (Y/N): Yes
Automated off normal shutdown procedures (Y/N): Yes
Use of field buses and smart sensors (Y/N): No
Expert systems or artificial intelligence advisors (Y/N): No
Protection system backup (Y/N): Yes

### D. EMERGENCY POWER SUPPLY SYSTEM

105. Type (diesel, gas, grid connection): Diesel, Battery
106. Number of trains: Two independent

### E. AC/DC SUPPLY SYSTEM

107. Type (rectifier, converter, battery): Converter, battery
108. Estimated time reserve (hr): One (min.)

## IV. CONVENTIONAL THERMAL CYCLE

### A. TURBINE SYSTEM

109. Type: Tender Compound Impulse
110. Overall length (m): 15
111. Width (m): 15.8
112. Number of turbines/reactor: One
113. Number of turbine sections per unit (e.g. HP/LP/LP): HP (1) / LP (1)
114. Speed (rpm): 3000

### B. STEAM CHARACTERISTICS

115. H.P. inlet pressure: 38.55 kg cm$^2$ (g)
116. H.P. inlet temperature: 250°C
117. H.P. inlet flowrate: 1323 Mt/hr
118. L.P. inlet pressure: 5.7 kg/cm$^2$ (g)
119. L.P. inlet temperature: 233.3°C
120. L.P. inlet flowrate (per section): 1010.5 Mt/hr

### C. GENERATOR

121. Type (3-phase synchronous, DC): 3-phase synchronous
122. Apparent power (MVA): 264
123. Active power (MW): 235
124. Frequency (hz): 50
125. Output voltage (kV): 16.5
126. Total generator mass (t): 253 Mt
127. Overall length: 4.2 m
128. Stator housing outside diameter: 4 m

### D. CONDENSER

129. Number of tubes: 20896
130. Heat transfer area: 19500 m$^2$

131. Flowrate: 857.2 T/hr
132. Pressure (m/bar): 0.0835 $kg/cm^2$
133. Temperature (°C): 42.6 (design)

**E1 CONDENSATE PUMPS**

134. Number: 2 x 100% capacity
135. Flowrate: 1100 T/hr
136. Developed head: 125 M
137. Temperature: 44°C
138. Pump speed: 1500

**E2 BOILER FEED PUMPS**

139. Number: 3 x 50%
140. Flow rate: 630 T/hr
141. Developed Head: 579 M
142. Temperature: 133 °C
143. Pump Speed (rpm): 3000

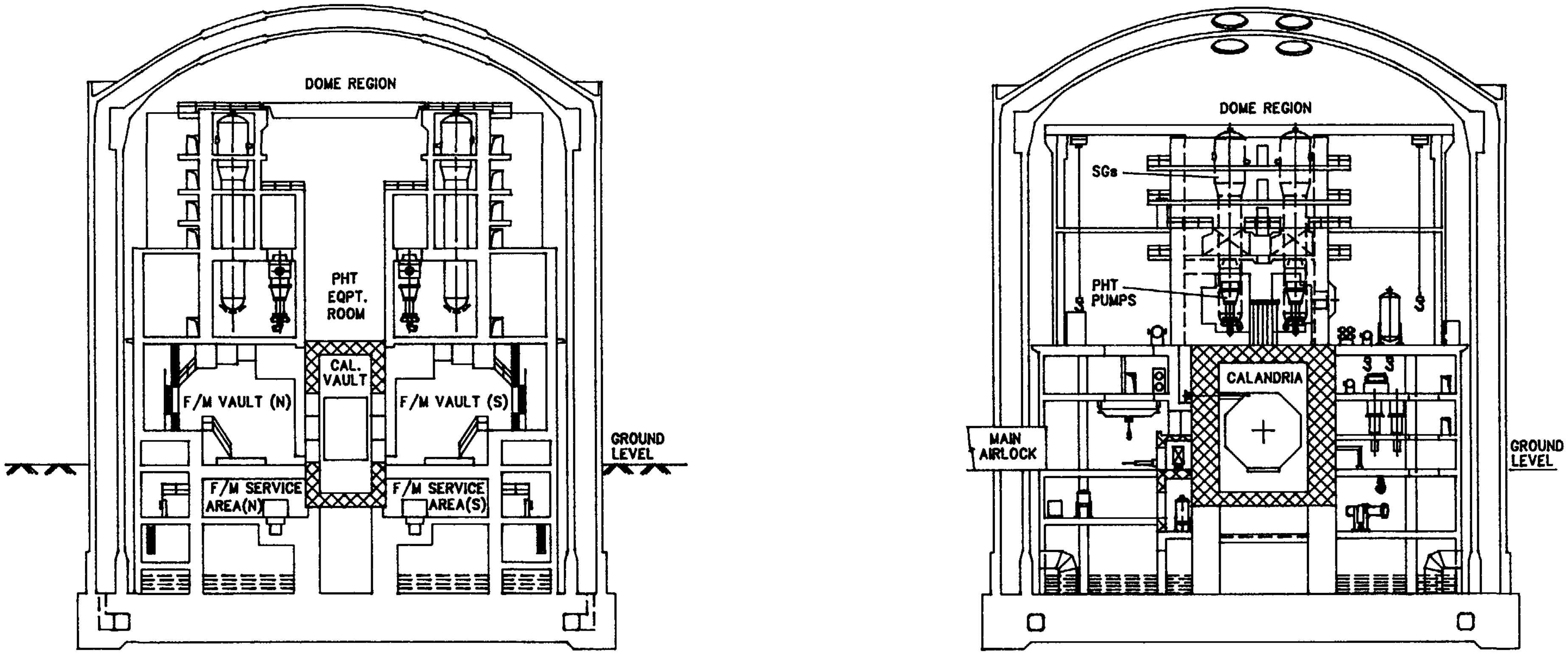
### 5.10.3. Project status

Nuclear Power Corporation of India Ltd. (NPCIL) is responsible for design, engineering, construction, commissioning, operation and decommissioning of nuclear power plants in the country. Complete research and development (R&D) support covering all the above aspects of nuclear power plant is provided by Bhabha Atomic Research Centre (BARC). Various national laboratories and institutes supplemented this effort. Technical support from other consultants is obtained by assigning specific tasks in the areas of design and engineering.

The first two units of PHWR-220 were constructed in Rajasthan as a collaborative venture with Atomic Energy of Canada Ltd. Work on 2 units of Madras Atomic Power Stations was taken up in 1967 as a totally indigenous effort including engineering, procurement, construction, commissioning and operation. Current designs are largely standardized. Seven units are in operation, five are under construction/commissioning with four units in the planning stage.

### 5.10.4. Project economics

The installation of nuclear power plants of PHWR-220 type is a part of the continuing efforts to meet the ever increasing demand for energy in a developing country like India. The unit energy cost for nuclear power compares very favourably to that for a coal fired power plant if the former is located about 1000 km from the location of coal reserves. For PHWR-220 units very recently connected to the grid, the unit energy cost is about 6.9 cents per unit.




REACTOR BUILDING SECTION LOOKING EAST

REACTOR BUILDING SECTION LOOKING NORTH

*Fig. 5.10.1 Typical cross sections of 220 MWe reactor building.*

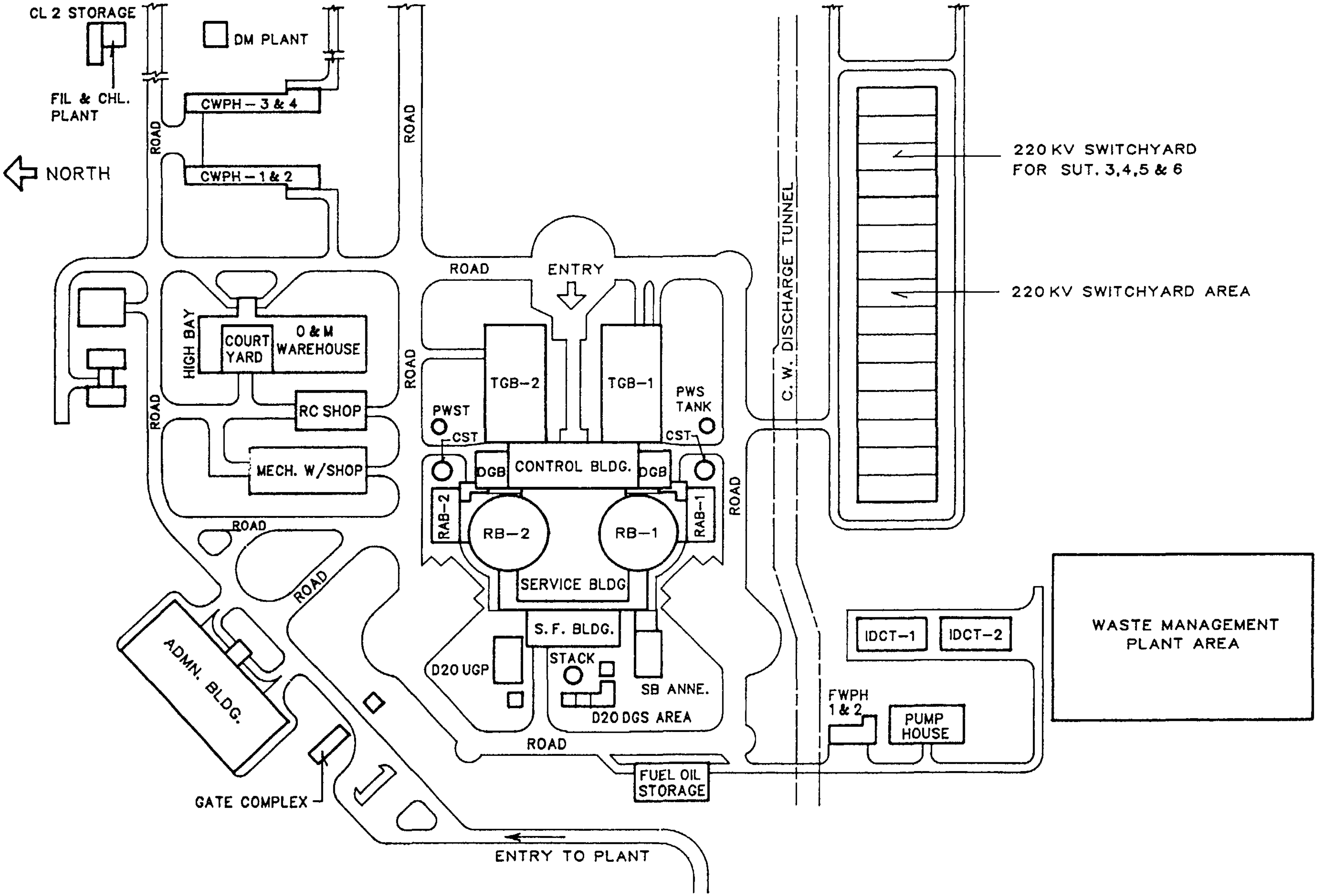



*Fig 5.10.2 Typical Plant Layout of 220 Mwe PHWR.*

# 6. DESIGN DESCRIPTIONS FOR REACTORS IN THE BASIC DESIGN STAGE

## 6.1. PIUS REACTOR DESCRIPTION AND DESIGN STATUS

### 6.1.1. Basic objectives and features

The PIUS concept 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 by ABB Atom at the start of the development work on SECURE, its heat only reactor design, 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**, ie., the "human factor" as a risk element should be largely eliminated by design;
- It should be **safe enough to be located almost anywhere**, even in densely populated areas (from a technical point of view);
- 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 the power reactor PIUS, 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.

### 6.1.2. Design description

#### *6.1.2.1. Nuclear steam supply system*

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. The modified arrangement reflects the 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 Figure6.1.1. and a main flow diagram for the power plant is presented in Figure 6.1.2. Main plant parameters are provided in Section 6.1.4.

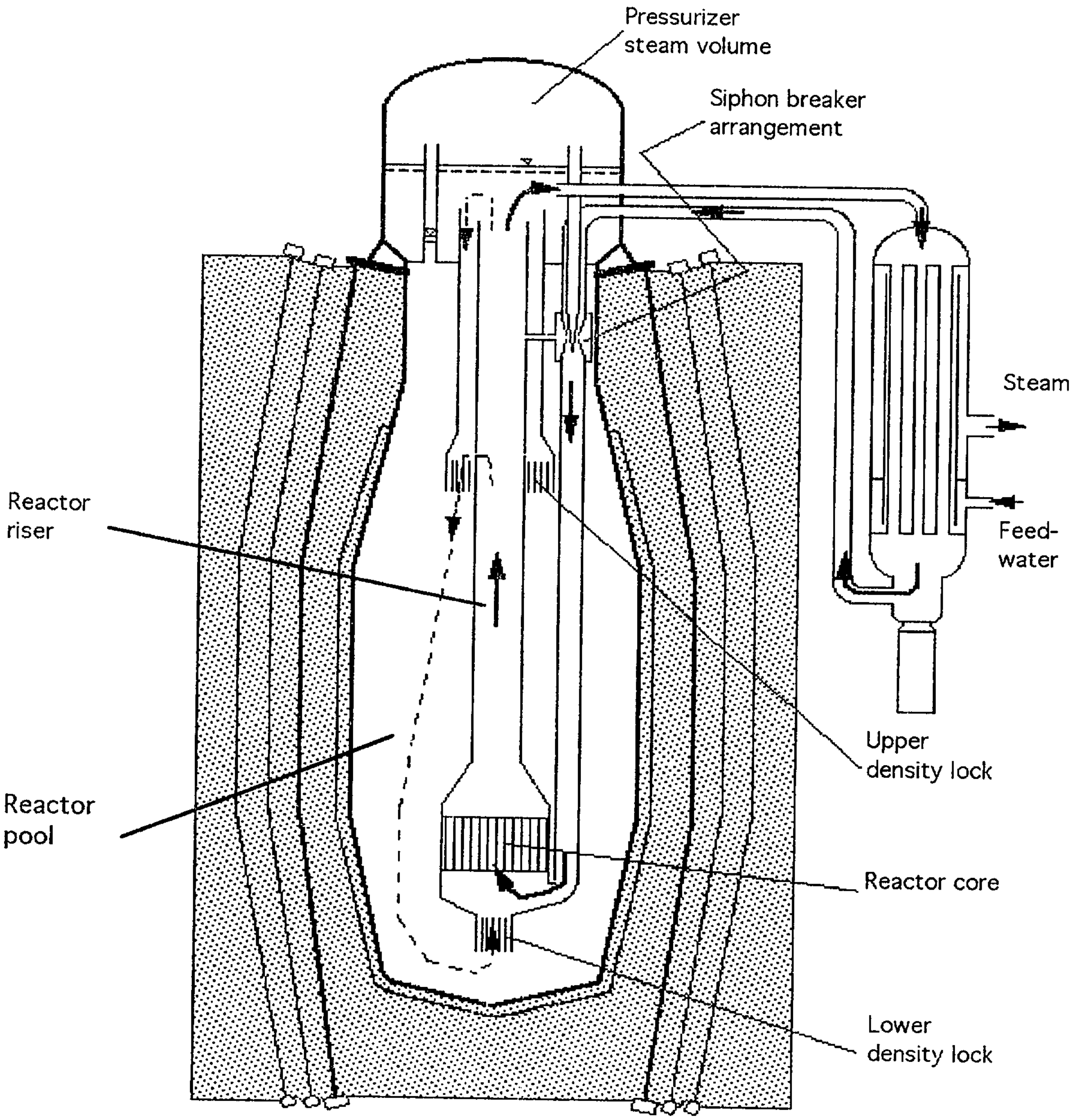



*Fig 6.1.1. PIUS Basic Arrangement Principle*

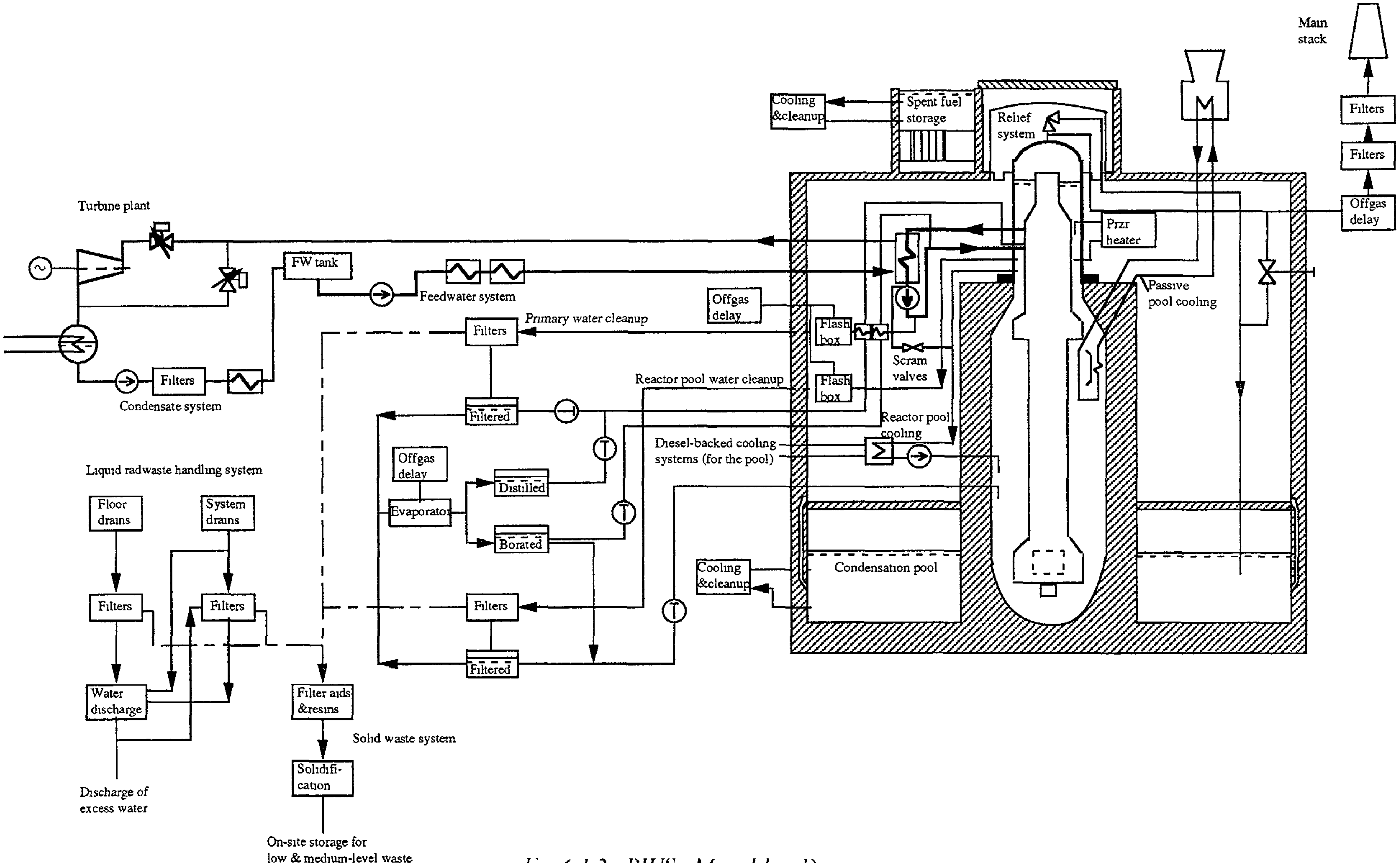



*Fig 6.1.2. PIUS - Main Flow Diagram*

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.

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 down- comer. 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 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 main coolant pumps' speed (or flow rate) 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 the 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 thermal conditions 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 temperature measurements for the interface level in the upper lock are basically used for reactor pool volume control purposes. (The reactor primary loop volume control utilizes 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).

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.

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 cannot 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.

The reactor system is pressurized by means of steam supplied from an electrically heated recirculation boiler, drawing water from the water volume of the pressurizer. The steam volume of the pressurizer is comparatively large and, together with its volume of saturated water, the reactor system can accommodate pressure and level variations that may occur during operational transients and accident situations. 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 pressurizer "pool".

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.

The nuclear steam supply system (ie., 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 PIUS plant is also 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; 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, allowing major simplification of the plant.

The remaining safety-grade functions are performed by the reactor protection system (it initiates opening of the scram valves to achieve a reactor scram), the containment isolation system (it initiates isolation of the containment by closing isolation valves), the reactor vessel safety valves (based on pressure-activated components), and the passive reactor pool cooling function. These functions are not needed for the protection of the core, however.

As a result, the plant should be simpler to operate and maintain than present day LWR plants, and the elimination of severe reactor accidents as a practical concern should also contribute to simplified operation, making it possible for the plant operator(s) to concentrate on producing electric energy. It can also be noted that PIUS compares favourably against the top-tier requirements of the ALWR Requirements Documents developed by the Electric Power Research Institute (EPRI) in the US.

*6.1.2.2. Balance of plant systems*

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 turbine 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 turbines and, on startup, hot standby and turbine trip, flow from the main steam and bypass system.

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.

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., 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 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".

*6.1.2.3. Instrumentation, control and electrical systems*

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, eg., 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 of the control room is based on the assumption 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.

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. Separated batteries and DC distributions are maintained for power supplies to the "remaining safety-grade" equipment in the reactor building, however.

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.

*6.1.2.4. Safety considerations and emergency protection*

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.

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
- Negative power coefficient throughout operating cycle
- Lowered reactor pressure and temperature as compared to present-day PWR plants

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 protection against core degradation accidents is ensured by the laws of physics alone. 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.

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 its 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.

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.

In PIUS, 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. It 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.

*6.1.2.5. Buildings and structures*

The plant layout 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.

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 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, ie., 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.

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 switch gear 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.

### 6.1.3. Safety concept

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 - 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

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 entities ENEA-DISP/ENEL (ENEA-DISP is the state regulator and ENEL is the (former) state utility) 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 $10^{-7}$

A summary of safety features etc is presented in the Tables 6 1 1 to 6 1 4

TABLE 6 1 1 MAIN SAFETY RELATED SYSTEMS IN THE PIUS PLANT

| Name | Safety | Main characteristics |
|---|---|---|
| All structures in RCPB | X | Reactor vessel (PCRV + steel top), piping up to isolation valves |
| Containment | X | |
| Primary loop | | |
| - external loops, integrity | X | 4 x (hot let, SG, pump, cold leg) |
| - in-vessel internals | X | Flow guiding structure, upper and lower density locks |
| Reactivity control system | X | Highly borated reactor pool (in PCRV cavity) & scram valve system (4x2 valves) |
| Passive core cooling system | X | Natural circulation loop from pool, through core & riser, to pool |
| Long-term passive RHR system | X | 8 coolers submerged in reactor pool, 8 natural circulation loops to natural-draft cooling towers |
| Main control room ventilation system | X | Coolers, filters, fans |

TABLE 6.1. 2: MAIN ACCIDENT INITIATORS FOR THE PIUS PLANT

| % contribution | Initiating event |
|---|---|
| 60 | LOCA (primary): Loss of Primary Coolant Accident |
| -- | LOCA (secondary): Secondary (Side) Pipe Rupture (water or steam) |
| -- | LOCA (interfacing: eg. SGTR (Steam Generator Tube Rupture) |
| -- | ATWS: Anticipated Transients Without Scram |
| 30 | Primary Transients |
| -- | Secondary Transients (turbine trip) |
| -- | Loss of electric sources (partial) |
| 1 | Total loss of the heat sink |
| 6,5 | Total loss of the steam generator feedwater |
| 2,5 | Station blackout |

TABLE 6.1.3: DESIGN FEATURES FOR PREVENTION AND PROTECTION LEVELS: THAT REDUCE (R), SUPPRESS (S) THE INITIATORS FREQUENCY OR LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES (L)

| Initiating event | Prevention level features |
|---|---|
| LOCA (primary): | - Multi-redundant load-bearing members in PCRV; reduce initiating frequency for severe vessel leaks and ruptures;<br>- primary loop largely integrated in reactor pool; limits LOCA effects on core and fuel;<br>- wet motor RCPs; eliminates small LOCAs due to shaft seal failure<br>- not important<br>- simplifications and improved materials; reduce initiator frequency |
| LOCA (secondary): | - increased design margins; reduce initiator frequency; |
| LOCA (interfacing): | |
| Primary Transients: | - not important |
| Secondary Transients: | |
| Loss of electric sources: | - not important |
| Total loss of heat sink: | |
| Total loss of SG feedwater: | - passive long-term RHR to ambiant air; largely suppresses initiator<br>- not important |
| Station blackout: | - not important |

TABLE 6.1.3 (cont.)

| Initiating event | Protection level features |
|---|---|
| LOCA (primary): | - primary loop integration; limits leakages;<br>- increased water inventory; increases time constants;<br>- automatic and passive (degree C) boron insertion and core coolant supply |
| LOCA (secondary) | - not important |
| LOCA (interfacing): | - easy isolation of secondary side; |
| ATWS | - not important, due to "inherent" self-protective functions and strong negative MTC |
| Primary transients. | - reduced power density and increased primary system thermal inertia; |
| Secondary transients: | - as for primary transients (but not important events) |
| Loss of electric sources: | - as for primary transients (but not important events)<br>(NB. Battery power is not a passive system!) |
| Total loss of heat sink:<br>Total loss of SG feedwater.<br>Station blackout: | these three events are not critical in the short term due to the passive decay heat removal (by natural circulation following automatic (fully passive) ingress of pool water and the large primary system inertia. |

In this context, it should be noted that:
the self-protective functions (yielding reactor pool water ingress and reactor shutdown, followed by continuous decay heat removal) and the passive long-term RHR to the ambient air provide efficient protection against serious consequences in all abnormal conditions.

## TABLE 6.1.4. DESIGN FEATURES FOR MITIGATION LEVEL OF PIUS PLANT

| Safety functions | Systems (Cf Tab 6 1 1 ) | Passive/ Active | Design features / Remarks |
|---|---|---|---|
| **Design Basis**<br>Fission product containment | Containment of PS type, inherent shutdown and reactor pool cooling | passive<br>passive<br>passive | Following blowdown after LOCA, negligible steam flow to containment. Iodine spike remains in pool water, since there will be no boiling |
| Coolant inventory | Water losses limited by siphon breaker function (primary side) | passive | Condensation pool water can be replenished from outside, reactor pool water also |
| Decay heat removal | Forced circ through SG or natural circ to pool | active<br>passive | SG heat removal via turbine condenser to heat sink Reactor pool heat removal to heat sink or to ambient air |
| Reactivity control | Injection of boron<br>Ingress of pool water | active<br>passive | via CVCS<br>Backup, ever-present |
| **Severe Accident**<br>Containment temperature and pressure control | As for Design Basis containment atmosphere cooling is available, but not credited for design basis events. | active | High containment temperature and pressure may occur only in a postulated "shutdown failure" situation following a large series of different failures If deemed necessary, small venting system can be added for overpressure protection |
| Heat removal | See design basis above | | In addition, the water inventories in the high located pools (see "others" below) can be used for makeup in a feed and bleed mode |
| Tightness control | | | The surrounding reactor building serves as a secondary containment, kept at a below-atmospheric pressure, ventilation exhaust air passes through filters If over pressure protection of containment is found necessary, a small vent system can be added |
| Inflam. gas control | | | No accident sequence leading to core melt has been identified, and there is no motivation for further countermeasures |
| Fission product containment | See design basis above | | In addition, the surrounding reactor building serves as a secondary containment, kept at a below atmospheric pressure, ventilation exhaust air passes through filters |
| Corium management | A possible mass of corium will collect at the reactor pool bottom - in a high boron content sump | passive | No accident sequence leading to core melt has been identified and there is no motivation for further countermeasures |
| Others | High boron content water inventories of the pools at the reactor service room level for water makeup | active | |

### 6.1.4. Design data questionnaire (for Water-Cooled Reactors PIUS)

## I. GENERAL INFORMATION

1. Design name: PIUS
2. Designer/Supplier address: ABB Atom AB Sweden
3. Reactor type: PWR    Number of modules/plant: 1
4. Gross thermal power (MWth)/reactor: 2000 MWth
5. Net electrical output (MWe)/reactor: 610-640 MWe
6. Heat supply capacity (MWth): 100-500 MWth

## II. BASIC DESIGN DESCRIPTION

## A. CORE AND REACTIVITY CONTROL

7. Fuel material: $UO_2$
8. Fuel inventory (tonnes of heavy metal): 91,3
9. Average core power density: 72 kW/l
10. Average fuel power density: 24,8 kW/kgU
11. Maximum linear power: 29,8 kW/m
12. Average discharge burnup: 45 500 MWd/t
13. Initial enrichment or enrichment range: 2,0 Wt%
14. Reload enrichment at equilibrium: 3,5 Wt%
15. Refuelling frequency: 12 months
16. Type of refuelling (on/off power): off
17. Fraction of core withdrawn: 1/6
18. Moderator material and inventory: $H_2O$
19. Active core height: 2,5 m
20. Equivalent core diameter: 3,75 m
21. Number of fuel assemblies: 213
22. Number of fuel rods/assembly: 312
23. Rod array in assembly: 18x18
24. Cladding material: Zr 4
25. Clad thickness: --
26. Number of control rods or assemblies: N/A
27. Type: N/A
28. Additional shutdown systems: N/A
29. Control rod neutron absorber material: N/A
30. Soluble neutron absorber: Boron acid
31. Burnable poison material and form: $Gd_2O_3$ in $UO_2$

## B. REACTOR COOLANT SYSTEM

### B1. Coolant

32. Coolant medium and inventory: $H_2O$/630 $m^2$
33. Design coolant mass flow through core: 13 000 kg/s
34. Cooling mode (forced/natural): forced
35. Operating coolant pressure: 9,0 MPa
36. Core inlet temperature: 260°C
37. Core outlet temperature: 290°C

### B2. Reactor pressure vessel (refers to PCRV+steel top)

38. Overall length of assembled vessel: 58 m
39. Inside vessel diameter: 12,2 m (PCRV)/≈ 7 m (top)
40. Average vessel thickness: 7-10 m (PCRV)/0,2 m (top)
41. Vessel material: concrete/ASTM steel
42. Lining material: stainless steel
43. Design pressure: 9,5 MPa
44. Gross weight: 63,5 $10^6$kgs

### B3. Steam generator

45. Number of steam generators: 4
46. Type: Once-through
47. Configuration (horizontal/vertical): vertical
48. Tube material: Inconel 600
49. Shell material: SA 533, CL.2, Gr.B
50. Heat transfer surface/SG: 20 950 $m^2$
51. Thermal capacity/SG: 500 MW
52. Feedwater pressure: 4,4 MPa
53. Feedwater temperature: 210°C
54. Steam pressure: 4,0 MPa
55. Steam temperature: 270°C

### B4. Pressurizer

56. Pressurizer total volume: 150 $m^3$
57. Steam volume: 30 $m^3$

### B5. Main coolant pumps

58. Number of cooling or recirc. pumps: 4
59. Type: wet, glandless, asynchronous motor
60. Pump mass flow rate: 13 200 kg/s
61. Pump design rated head: 80 mWG
62. Pump nominal power: 3 500 kW
63. Mechanical inertia: low

## C. CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS)

64. Number of extraction lines: 4
65. Number of pumps: 3x2
66. Number of injection points: 2x4
67. Feed and bleed connections: see above

## D. CONTAINMENT

68. Type: Pressure-suppression
69. Overall form: cylindrical
70. Structural material: concrete
71. Liner materialcarbon steel (stainless steel in pools)
72. Single/double wall: single
73. Dimensions (diameter/height): D=63/H=61 m
74. Design pressure: 0,2 MPa
75. Design temperature: 175°C (drywell)
76. Design leakage rate: 1% per day

# III. SAFETY RELATED SYSTEMS

## A. DESIGN CONDITIONS

### A1. Fission products retention

77. Containment spray system (Y/N): No
    a. Duration (h)
    b. Flow rate ($m^3$/h)
    c. Mode of operation (active/passive)
    d. Safety-graded (Y/N)
78. F.P. sparging (Y/N): Y
79. Containment tightness control (Y/N): periodical insp. and testing
80. Leakage recovery (Y/N): Y
81. Guard vessel (Y/N): N

**A2. Reactivity control**

82. Absorber injection system (Y/N): Yes
   a. Absorber material: boron
   b. Mode of operation: active & passive
   c. Redundancy: Yes
   d. Safety-graded: passive injection from pool is safety-graded; active inj. via scram valves only partly SG. (CVCS inj. is NS)

83. Control rods (Y/N): No
   a. Maximum control rod worth (pcm): N/A
   b. Mode of operation (active/passive): N/A
   c. Redundancy: N/A
   d. Safety-graded: N/A

**A3. Decay heat removal**

*A3-1 Primary side*

84. Water injection
   a. Actuation mode (manual/automatic): N/A
   b. Injection pressure level: N/A
   c. Flow rate (kg/s): N/A
   d. Mode of operation (active/passive): N/A
   e. Redundancy: N/A
   f. Safety-graded (Y/N): N/A

85. Water recirculation and heat removal
   a. Intermediate heat sink: Reactor pool
   b. Mode of operation (active/passive): Passive
   c. Redundancy: Multiple
   d. Self-sufficiency (h): 7 days
   e. Safety-graded (Y/N): Yes

85b. Water recirculation and heat removal: from the pool
   a. Implemented components: submerged pool coolers, cooling towers, and connecting pipes, ultimate heat sink (cold source) ambient air
   b. Mode of operation (active/passive): Passive
   c. Redundancy: Multiple
   d. Self-sufficiency (h): weeks
   e. Safety-graded (Y/N): Yes

*A3-2 Secondary side*

86. Feedwater
   a. Actuation mode (manual/automatic): Automatic
   b. Flow rate (kg/s): as needed
   c. Mode of operation (active/passive): Active
   d. Redundancy: 2x
   e. Self-sufficiency (h): N/A
   f. Safety-graded (Y/N): No

87. Water recirculation and heat removal
   a. Ultimate heat sink (cold source): Sea or lake
   b. Mode of operation (active/passive): Active
   c. Redundancy: 150%
   d. Self-sufficiency (h): -
   e. Safety-graded (Y/N): No

*A3-3 Primary pressure control*

88. Implemented system (Name): Reactor pressure relief system
   a. Actuation mode (manual/automatic): 4 safety valves: automatically (on pressure directly or via measuring channels); 2 relief/2 emergency blowdown valves: manually.
   b. Side location (primary/secondary): primary, przr
   c. Max. depressurization rate: 100 kg/s
   d. Safety-graded (Y/N): safety valves: Yes
      relief valves: no

## B. SEVERE ACCIDENT CONDITIONS

[A severe accident is any beyond DBE accident.]

**B1. Fission products retention**

89. Containment spray system (Y/N): Y
90. F.P. sparging (Y/N): Not applicable
91. Containment tightness control (Y/N): Y, see item 79
92. Leakage recovery (Y/N): Filters
93. Risk of recriticality (Y/N): N

**B2. Recriticality control**

94. Encountered design feature
    - a. Mode of operation (active/passive): N/A
    - b. Safety-graded (Y/N): N/A

**B3. Debris confining and cooling**

95. Core debris configuration (core catcher): N/A
96. Debris cooling system (name): N/A
    - a. Mode of operation (active/passive): N/A
    - b. Self-sufficiency (h): N/A
    - c. Safety-graded (Y/N): N/A

**B4. Long term containment heat removal**

97. Implemented system: Pumps, HXs, valves
    - a. Mode of operation (active/passive): active
    - b. Self-sufficiency (h): --
    - c. Safety-graded (Y/N): No
98. Intermediate heat sink
    - a. Self-sufficiency (h): --
    - b. Safety-graded (Y/N): No
99. External coolant recirculation
    - a. Implemented components: Pumps, HXs, valves
    - b. Mode of operation (active/passive): Active
    - c. Self-sufficiency (h): --
    - d. Safety-graded (Y/N): No
100. Ultimate heat sink
    - a. Self-sufficiency (h): --
    - b. Safety-graded (Y/N): No

**B5. Combustible gas control**

101. Covered range of gas mixture concentration: N/A
102. Modes for the combustible gas control
    - a. Containment inertation: N/A
    - b. Gas burning: N/A
    - c. Gas recombining: N/A
    - d. Others: N/A

**B6. Containment pressure control**

103. Filtered vented containment (Y/N): No
    - a. Implemented system: N/A
    - b. Mode of operation (active/passive): N/A
    - c. Safety-graded (Y/N): N/A
104. Pressure-suppression system: (Y/N)
    - a. Implemented system: Yes
    - b. Mode of operation (active/passive): Passive
    - c. Safety-graded (Y/N): Yes

## C. [SAFETY-RELATED] I&C SYSTEM

Automatic load following (Y/N): Yes
* range: 30% power
* maximum rate: 20%/min

Load rejection without reactor trip: Yes
Full Cathode Ray Tubes (CRT) display: Yes
Automated startup procedures: Yes, largely
Automated offnormal shutdown procedures: Yes
Use of field buses and smart sensors: Yes
Expert systems or artificial intelligence advisors: Yes
Protection system backup: Yes

## D. EMERGENCY POWER SUPPLY SYSTEM
***[rather backup power supply]***

105. Type (diesel, gas, grid conn.): conn. to 2nd grid + 2 dg units
106. Number of trains: 2

## E. AC/DC SUPPLY SYSTEM

107. Type (rectifier/converter/battery): AC from rectifier/battery/converter trains; DC by local rectifiers
108. Estimated time reserve: 1 hour
[Separate battery system with 2 hours capacity is provided for safety systems.]

# IV. CONVENTIONAL THERMAL CYCLE

## A. TURBINE SYSTEM

109. Type
110. Overall length (m): depends on make
111. Width (m): depends on make
112. Number of turbines/reactor: 1 (or 2)
113. Number of turbine sections per unit: 1 HP/ 3LP
114. Speed (rpm) (- will depend on site): 3 000 rpm

## B. STEAM CHARACTERISTICS

115. HP inlet pressure: 3,87 MPa
116. HP inlet temperature: ≈269°C
117. HP inlet flow rate: 1 012 kg/s
118. LP inlet pressure: ≈0,9 MPa
119. LP inlet temperature: ≈234°C
120. LP inlet flow rate (per section): ≈255 kg/s

## C. GENERATOR

121. Type: 3-phase, synchr.
122. Apparent power: 785 MVA
123. Active power: 667 MW
124. Frequency (depends on site): 50 Hz
125. Output voltage: 20 kV
126. Total generator mass (tonnes): depends on make
127. Overall length: depends on make
128. Stator housing outside diameter: depends on make

## D. CONDENSER

129. Number of tubes: depends on make
130. Heat transfer area: depends on make
131. Flow rate: ≈32 $m^3/s$
132. Pressure: ≈5 kPa
133. Temperature of circulating water: 15 °C

## E. CONDENSATE PUMPS

134. Number: 3
135. Flow rate: ≈320 kg/s
136. Developed head: ≈1,43 MPa
137. Temperature: ≈ 33 °C
138. Pump speed: depends on make

### 6.1.5. Project status

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. Some additional testing is planned, 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, and evaluation studies are underway in other countries. 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, in accordance with the general stipulations presented by the organizers, represents a very difficult task; the status of the PIUS project lies somewhere between D1 and D2 - pretty close to D2 even though all D2 activities have not been completed.

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, and "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 Reaktorsicherheit), a German Advisory Institute to the Government.

### 6.1.6. Project economics

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 show 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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## 6.2. REACTOR SYSTEM DESCRIPTION AND DEVELOPMENT STATUS OF THE NUCLEAR HEATING REACTOR (HR-200)

### 6.2.1. Basic objectives and features

The HR-200 is a Nuclear Heating Reactor (NHR) with 200MW of thermal power. It is specially designed for district heating by the Institute of Nuclear Energy Technology (INET), Tsinghua University, PR CHINA. It can be also used in refrigeration, desalination and so on.

The basic HR-200 design objectives are as follows:

A. Safety

- Reactor core is cooled by natural circulation
- Reactor core is prevented from uncovery under any accidents
- Integral primary circuit arrangement, self-pressurization. Dual pressure vessel structure and all the penetrations located on the upper part of the reactor pressure vessel (RPV)
- Use of simple, dedicated, independent, passive safety systems that require no operator action in the event of accidents, and maintain core cooling indefinitely without AC power
- Predicted core damage frequency $<10^{-8}$ per year; a significant release frequency $<10^{-9}$ per year
- Reliable reactivity control and shutdown system
- Large safety margin.

B. Reliability

- To simplify design, operation and maintenance
- Public radiation exposure within a range of 80Km from the DaQing plant site $<5\times10^{-4}$ man-Sv per year
- Overall plant availability goal greater than 95% considering forced and planned outages.

C. Economics

- Net heating output 196 MWth
- The capital cost of a HR-200 is about 110.0 Million $US in 1991 dollars and the heat generation cost is competitive with coal (or oil) on an average site in China.

The major innovating features are as follows:

- Integrated arrangement, self-pressurization
- Dual pressure vessel structure
- Low operating temperature, pressure and low power density reactor design which provides increased operating margins and improved fuel economy
- Reactor core cooling with simple, passive means which use "natural" driving forces only
- Adopting innovative hydraulic system to drive the control rods.
- State-of-the-art digital instrumentation and control systems and an advanced man-machine interface control room, console-type work stations, soft controls and integrated, prioritized alarms and procedures
- Innovative design of the plant arrangement and advanced construction concept adopted to minimize cost, to shorten the construction schedule and to meet safety, operational and maintenance criteria
- Multi-purpose applications.

### 6.2.2. Design description

The main features of HR-200 are summarized:

#### *6.2.2.1. Nuclear heat supply system*

6.2.2.1.1. Primary circuit

The primary coolant absorbs the heat from the reactor core, then passes through the riser and enters the primary heat exchangers, where the heat carried is transferred to the intermediate circuit. An integrated arrangement is adopted to decrease the possibility of LOCA. All main parts of the primary circuit are contained in the RPV.

(1) The RPV

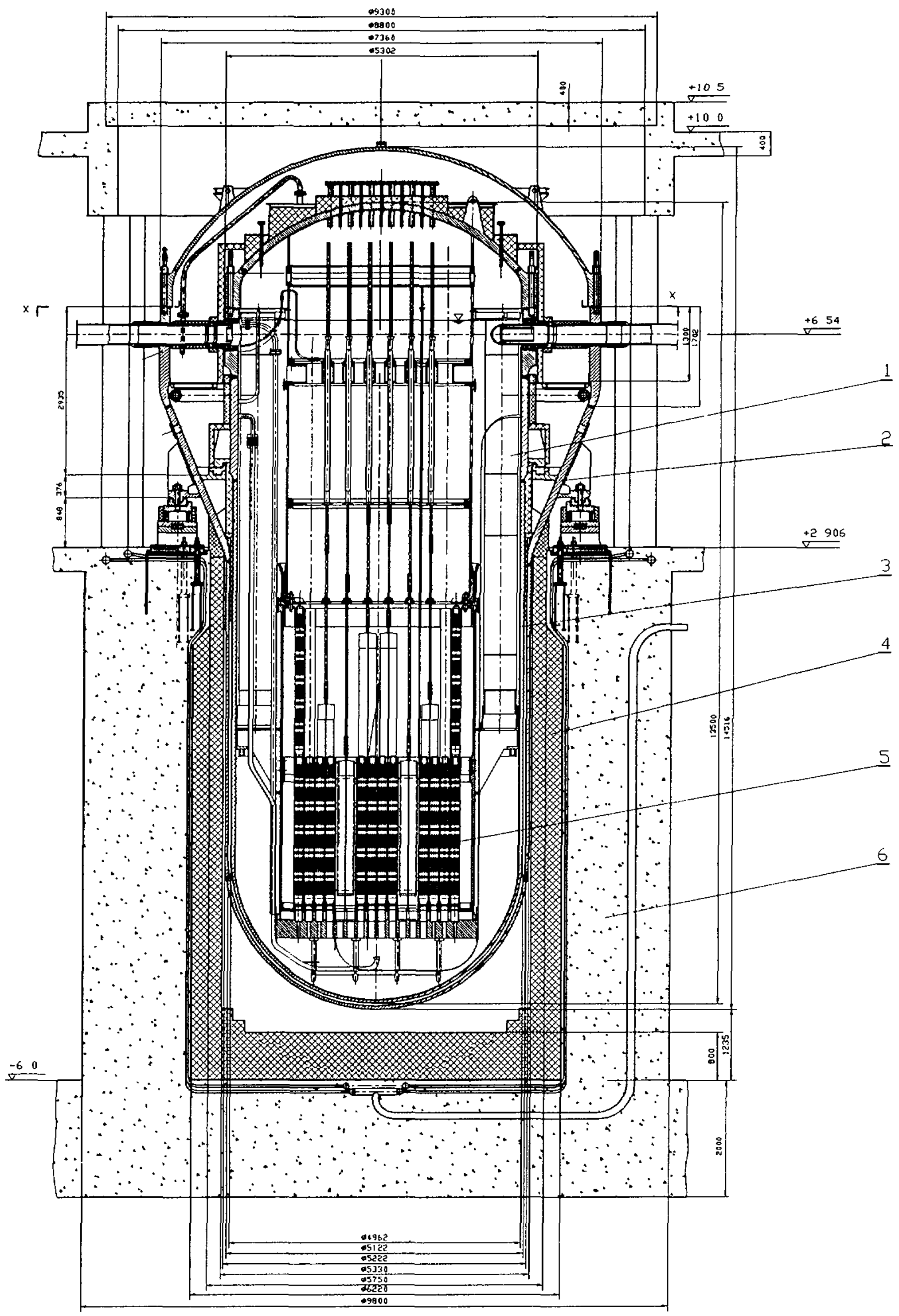
The HR-200 is designed with an integral, vessel-type structure. The RPV is the pressure boundary of the reactor coolant. It is 13.62m inside height, 5.00m inside diameter, (see figure6.2. 1.)

The design also features a large coolant inventory inside the RPV (about 200T) which provide for a lower total neutron fluence to the RPV ($<10^{16}$n/cm$^2$ for 40Y lifetime of the reactor).

All vessel penetrations (only with small diameter) are located on the upper part of the RPV.

(2) Reactor core

The reactor core of HR-200 consists of 96 assemblies (fuel ducts) and 32 control rods. A long riser is located above the core to enhance the capability for natural circulation. The reactor core stands on the lattice-support structure, which is fixed on the RPV.




1.Main heat exchanger 2.Containment 3.RPV
4.Thermal isolation 5.Reactor core 6.Biological shield

*Fig 6.2.1. Vertical Section of HR-200*

The fuel bundle arranged on a 12x12 lattice with an active length of 1.9m is contained in a duct (or box). The cruciform control rods are placed in the gaps between the square ducts. There are 3 enrichments in the initial loading: 1.8%, 2.4% and 3.0% as uranium dioxide. The discharge burn up is about 30000 MWd/tU.

The equivalent core diameter is 2.3m. Spent fuel assemblies are stored in the rack around the active core (eq. dia. ~1.9m).

Burnable poison ($Gd_2O_3$) is used to partly compensate the fuel burn up reactivity, and soluble boron is utilized for reactor shutdown only. This results in a negative temperature coefficient of reactivity over the complete core life.

A low core power density (~36kw/l) provides thermal reliability during normal and accidental operating conditions.

(3) Control rod and control rod drive mechanism

A new type of hydraulic drive mechanism is used to drive the control rods in HR-200. In the drive system the reactor coolant (water) is the actual medium. The water is pumped into step-cylinders of which the movable parts contain the neutron absorber. A pulsed flow, generated by a controlling magnetic valve in the control unit moves the movable part of the step-cylinder step by step. The drive system is very simple in its structure and is designed on the "fail-safe" principle, i.e. all control rods will drop into the reactor core by gravity under loss of electric power, depressurization, postulated breaks in its piping systems and pump shut down events.

(4) Primary heat exchanger (PHE)

On the periphery of the RPV upper part the 6 sets of PHE are located. Triangular-pitch, U-tube-shaped and vertically placed bundles are adopted for easy on site repair. The coolant enters the upper plenum of the exchangers, then divides two ways to flow downward through the tubes. Flow distribution baffles are installed to give optimum heat transfer efficiency. The total heat transfer area is approximately 2982m$^2$. The operating temperature of each PHE is 210°C and the operating pressure is 3.0MPa.

(5) Fuel handling and storage

The initial core is divided into 4 fuel regions and contains 96 fuel assemblies. The 24 assemblies of each region will be refuelled together. The spent fuel will be moved into the spent region in the RPV and stored in it. This solution greatly simplifies the refuelling equipment and reduces the necessary space for fuel storage in the reactor building.

6.2.2.1.2. Guard vessel (containment)

A steel containment is fitted tightly around the RPV as a guard vessel so that the core will not be uncovered under postulated coolant leakage from the RPV.

6.2.2.1.3. Intermediate circuit

To keep the heating grid free of radioactivity, an intermediate circuit is needed in the HR-200 and its operating pressure is higher than that of the primary side.

*6.2.2.2. Balance of plant systems*

(1) Heating grid

A conventional heating network is supplied to distribute heat to the households.

(2) Radioactive waste management

The radioactive wastes are processed within the plant with equipment and systems that have been designed for easy maintenance. The waste processing system has been tailored according to waste category to produce effluent suitable for reuse, discharge or final disposal.

The management is divided into 3 categories:

- Liquid Waste Processing System (LWPS)

The LWPS limits the release of liquid rad waste to the environment to meet the as low as reasonably achievable (ALARA) criteria. The design release limit is 7.4 Bq/l in normal operation and the release is about 200 $M^3/Y$.

- Gaseous Waste Management System (GWMS)

All gaseous wastes generated in operation are treated by the waste gas processing system and/or ventilation system before their release.

- Solid Waste Processing System (SWPS)

The objective of the SWPS is to convert radioactive waste into an acceptable packaged form for off site disposal as solid waste. About $16M^3/Y$ of solid waste is generated in HR-200.

*6.2.2.3. Instrumentation, control and electrical systems*

The design of the instrumentation, control and electrical systems corresponds with the safety concept for safe reactor operation, and advanced electronic and information processing technology has been incorporated in the design. The plant is automated to a high degree, and all safety precautions are taken into account. The plant control scheme is based on "the reactor follows plant loads principle".

The computer system is intensively used in plant control and data acquisition and takes the place of hardware analogue control. This results in a significant reduction in the scope of cabling.

In the case of unsafe conditions the reactor protection system can automatically scram the reactor and actuate the relevant safety system.

*6.2.2.4. Safety considerations and emergency protection*

The safety of HR-200 is effected through the two key trends:

1) development of the plant self-protection feature,
2) creation of a multi-barrier system of functional and physical protection (defense in depth).

A number of advanced features have been incorporated into NHR to achieve its safety goal which can be summarized as follows:

(1) Large negative reactivity coefficient at all times.
(2) Integrated arrangement, self-pressurization, minimization of in-vessel penetrations and their location at the upper part of the RPV.
(3) Natural coolant circulation in the primary circuit under all conditions.
(4) Low power density and large coolant inventory.
(5) Use of guard vessel to keep the core under coolant and reduce radioactivity release from the primary circuit.
(6) Use of unique hydraulic control rod drive mechanism to exclude rod ejection.
(7) Spent fuel in-RPV storage.
(8) Passive safety system

- Residual heat removal system (RHRS)

RHRS is connected to the intermediate circuit and consists of two independent trains, each of which is able to disperse core decay heat into atmosphere properly by means of natural circulation.

- Boron acid injection system

A boron acid injection system, as a secondary reactor shutdown system, is designed to function by gravity.

*6.2.2.5. Buildings and structures*

The HR-200 plant is divided into the main building region and the auxiliary region. The reactor and almost all of its auxiliary system are housed in the main building, which is 72m long, 51m wide and 28.5m high above ground and 9m under ground. The total construction area is about 8900m$^2$. See figure 6.2.2. for schematic elevation drawing.

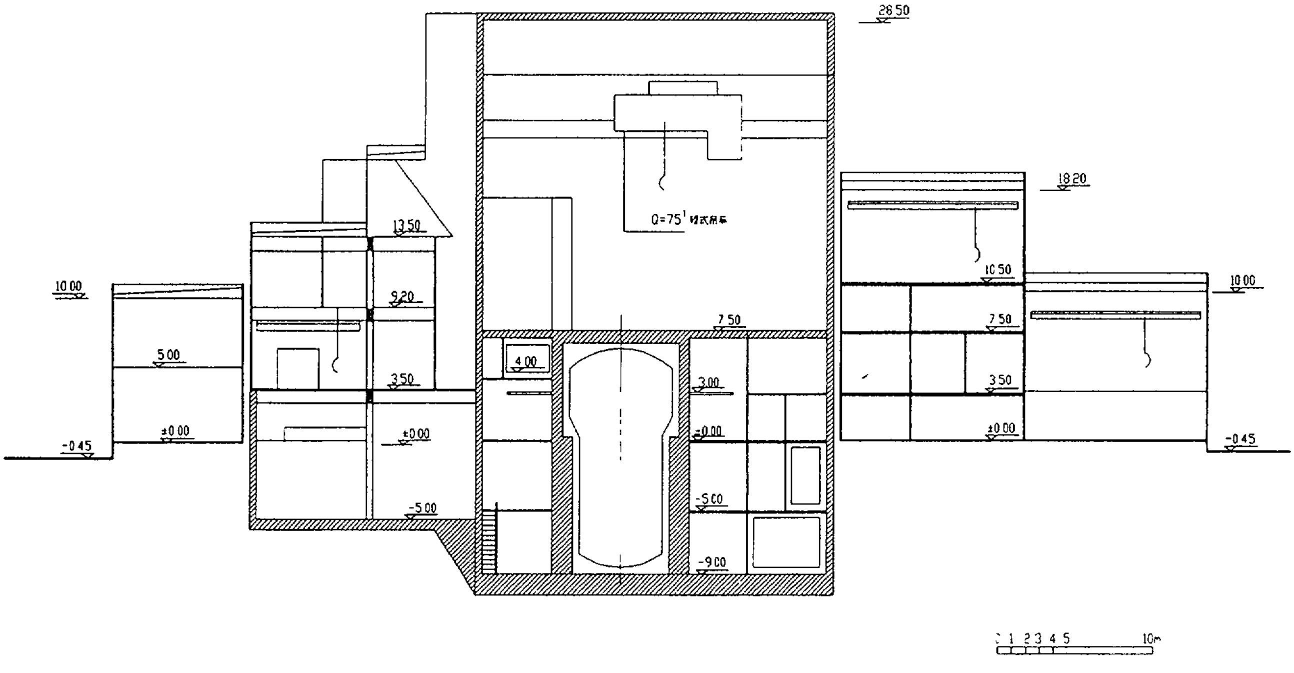



*Fig 6.2.2. The General Plant Layout*

### 6.2.3. Safety concepts

The safety concept of NHR is based to a high degree on its inherent safety instead of on the engineered safety features.

The NHR is operated under low pressure, low temperature, low power density and low radioactivity content in the primary coolant. The large sub-cooled water inventory results in a high thermal inertia in the primary system. A large negative temperature reactivity coefficient has been achieved in the core nuclear design, so any uncontrolled reactivity addition and anticipated transients without scram (ATWS) will be counteracted very well. The core decay heat is transferred into atmosphere by natural circulation. Moreover, loss of primary coolant is limited to the extent that the core will never be uncovered, therefore, an emergency core cooling system is not necessary for the NHR. Even in the worst case of the beyond design accident, the grace period is long enough for operator to take corrective action to mitigate the consequences of the accident. As a result, core melt-down can be excluded.

TABLE 6.2.1. MAIN SAFETY RELATED SYSTEMS IN THE HR-200 CONCEPT

| Name | Safety graded | Main characteristics |
|---|---|---|
| Primary circuit | X | reactor vessel<br>6 primary heat exchangers |
| Control rods | X | 32 crucitype rods as the primary shutdown system |
| Diverse reactivity control | X | Passive boric acid water injection |
| Passive residual heat removal | X | Two independent systems transfer core decay heat to atmosphere by natural circulation |
| Containment (guard vessel) | X | Prevent core from uncovery under failure of coolant pressure boundary |
| Passive containment cooling system | X | Heat transportation by natural circulation |

TABLE 6.2.2. MAIN ACCIDENT INITIATORS FOR THE HR-200

- LOCA (primary) : Loss of Primary Coolant Accident, including a crack of the RPV
- LOCA (secondary) : Secondary Pipe Rupture (water)
- LOCA (Interfacing) : e.g. Primary Heat Exchanger Tube Rupture
- ATWS: Anticipated Transients Without Scram
- Primary Transients,
- Secondary Transient,
- Loss of Electric Sources (all AC sources),
- Total Loss of the Heat Sink,
- Station Blackout

TABLE 6.2.3. DESIGN FEATURES FOR PREVENTION AND PROTECTION LEVELS THAT REDUCE (R), SUPPRESS (S), THE INITIATORS FREQUENCY OR LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES (L)

| |
|---|
| **Prevention level** |
| LOCA (primary)<br>- Reduced vessel fluence: reduce initiator frequency<br>- Primary circuit integration: reduces initiator frequency and limits accident consequences<br>- Guard vessel (containment): limits accident consequences<br>- All the penetrations at the upper of the RPV: limits accident consequences<br>LOCA (Secondary)<br>LOCA (Interfacing)<br>- Higher pressure in secondary circuit than in primary circuit<br>- Isolation valves : limits accident consequences |
| Primary Transients:<br>- Negative reactivity coefficient: limits accident consequences<br>- Large thermal inertia: limits accident consequences |
| Secondary Transients:<br>- triple loop and greater thermal inertia: limits accident consequence |
| Loss of electric sources:<br>- Passive RHRS: limits accident consequences<br>- All control rods drop into reactor core: limits accident consequences |
| Total loss of the heat sink:<br>- Passive RHRS: limits accident consequences<br>- Large thermal inertia: limits accident consequences |
| Station blackout: Natural circulation |
| **Protection level** |
| LOCA (Primary and including crack at the bottom of RPV)<br>- Primary circuit integration<br>- Increased water inventory around the core<br>- Guard Vessel<br>LOCA (Secondary): still to be defined<br>LOCA (Interfacing): - Easy intermediate circuit isolation |
| ATWS: - Strong negative temperature coefficient |
| Primary transients: Large primary inertia (inventory) |
| Secondary transients: - Large primary inertia (inventory) |
| Loss of electric sources: - Implementation of passive systems (battery power sources) |
| Total loss of heat sink: not critical |
| Total loss of PHE feedwater: not critical |
| Station blackout: not critical |

TABLE 6.2.4. DESIGN FEATURES FOR MITIGATION LEVEL OF HR-200

| Safety functions | System (cf Tab 6 2 1 ) | Passive/active | Design features/remark |
|---|---|---|---|
| **Design Basis**<br>Fission product<br>Containment | Cladding<br>RPV<br>Containment<br>Intermediate circuit | passive<br>passive<br>passive<br>passive | Integrated primary circuit<br>Compact containment design<br>Pressure in intermediate is higher than in primary LOOP |
| Decay heat removal | Residual heat<br>Removal system (RHRS)<br>passive containment cooling system (PCCS) | passive<br>passive | Natural circulation |
| Coolant inventory | Containment<br>Primary isolation valves<br>Primary coolant injection | passive<br>passive<br>active | Large volume of coolant inventory<br>Containment design |
| Reactivity control | Control rods<br>Boron acid injection | active/passive | In-core hydro-driven control<br>Rods system/gravity injection |
| Primary circuit pressure control | Self pressurized performance<br>Safety valves (SV) at RPV | passive<br>passive | Integrated primary circuit<br>Steam volume in PRV |
| **Severe accident**<br>Containment<br>Temperature and pressure control | Containment/PCCS/RHRS | passive | Steam ejected to sink<br>Natural circulation RHRS |
| Heat removal | PCCS/Containment | passive | Natural circulation |
| Tightness control | Containment | passive | |
| Inflame gas - control | Inter gas | passive | |
| Fission product containment | Containment | passive | Inclosed in Secondary containments |
| Corium management | Containment | passive | In containments |

### 6.2.4. Design Data Questionnaire

## I. GENERAL INFORMATION

1. Design Name: HR-200
2. Design/Supplier address: INET,Tsinghua University
3. Reactor type/Number of modules/per plant: 1 (or 2)
4. Gross thermal power (MW-th) per reactor: 200MW
5. Net electrical out put (MW-e) per reactor
6. Heat supply capacity (MW-th): 196MW per reactor

## II. BASIC DESIGN DESCRIPTION

### A. Core and reactivity control

7. Fuel material: $UO_2$
8. Fuel inventory (tons of heavy metal): 13.09
9. Average fuel power density (kW/liter): 36.23
10. Average fuel power density (kW/kgU): 15.28
11. Maximum linear power (W/m): 33870
12. Average discharge burnup (MWd/t): 30000 MWd/tu
13. Initial enrichment or enrichment range (Wt%): 1.8%,2.4%,3.0%
14. Reload enrichment at the equilibrium (Wt%): 3.0%
15. Refuelling frequency (months): ~ 36
16. Type of refuelling (on/off power): off power
17. Fraction of core withdrawn (%): 25%
18. Moderator material and inventory: $H_2O$, ~200t
19. Active core height (m): 1.9
20. Core diameter (m): ~ 2.3 (~1.9m initial core)
21. Number of fuel assemblies: 96
22. Number of fuel rods per assembly: 142
23. Rod array in assembly: 12 x 12
24. Clad material: Zr-4
25. Cladding thickness (mm): 0.7
26. Number of control or assemblies: 32
27. Type: crucitype
28. Additional shutdown systems: Boron acid injection system
29. Control rod neutron absorber material: $B_4C$
30. Soluble neutron absorber: /
31. Burnable poison material and form: $Gd_2O_3$, Integral

### B. REACTOR COOLANT SYSTEM

#### B1. Coolant

32. Coolant medium and inventory: demineralized water/200T
33. Design coolant mass flow through core: ~0.64
34. Cooling mode (forced/natural): natural
35. Operating coolant pressure (MPa): 2.5
36. Core inlet temperature (°C): 140
37. Core outlet temperature (°C): 210

#### B2. Reactor pressure vessel/tube

38. Overall length of assembled vessel/(m): 13.62
39. Inside vessel/diameter (m/mm): 5m
40. Average vessel/tube thickness (mm): 65
41. Vessel/tube material: SA516 (Carbon Steel)
42. lining material: Stainless steel
43. Design pressure (MPa): 3.1
44. Gross weight (ton/kg): ~ 260t

#### B3. Steam generator (Main heat exchanger)

45. Number of steam generators (heat exchanger): 6

46. Type: U
47. Configuration (horizontal/vertical): Vertical
48. Tube material: 0Cr18Ni11Ti
49. Shell material: 0Cr18Ni11Ti
50. Heat transfer surface per heat exchanger ($m^2$): 497
51. Thermal capacity per heat exchanger (MW): ~38
52. Feed water pressure (MPa): 3.0
53. Feed water temperature (°C): 95
54. System pressure (MPa): 3.0
55. Outlet temperature (°C): 145

**B4. Pressurizer**

56. Pressurizer total volume ($m^3$): ~24.5 (integral)
57. Steam volume (full power/zero power, $m^3$)

**B5. Main coolant pumps**

58. Number of cooling or recirculation pumps
59. Type
60. Pump mass flow rate (kg/s)
61. Pump design rate head
62. pump nominal power (kW)
63. Mechanical inertia (kg $m^2$)

## C. CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS)

64. Number of extraction lines: 1
65. Number of pumps: 2
66. Number of injection points: 2
67. Feed and bleed connections: 1

## D. CONTAINMENT (Guard Vessel)

68. Type: hang up
69. Overall form (spherical/cyl.): cylindrical
70. Structural material: carbon steel
71. Liner material: /
72. Single/double wall: single
73. Dimensions (diameter, height) (m): D: 5.84m, H: 15.1m
74. Design pressure (MPa): 1.9
75. Design temperature (°C): 200
76. Design leakage rate (% per day): <1% per day

# III. SAFETY RELATED SYSTEMS

## A. DESIGN CONDITIONS

A1. Fission product retention

77. Containment spray system (Y/N)
    a. Duration (h)
    b. Flow rate ($m^3$/h)
    c. Mode of operation (active/passive)
    d. Safety graded (Y/N)
78. F.P.Sparging (Y/N): N
79. Containment tightness control (Y/N): Y
80. Leakage recovery (Y/N): N
81. Guard vessel (Y/N): Y

A2. Reactivity control

82. Absorber injection system (Y/N): Y
    a. Absorber material: $Na_2B_5O_7$
    b. Model of operation (active/passive): active/passive
    c. Redundancy: Y
    d. Safety graded: Y

83. Control rods (Y/N): Y
   a. Maximum control rod worth (pcm): 14220
   b. Mode of operation (active/passive): active/passive
   c. Redundancy: Y
   d. Safety graded: Y

A3. Decay heat removal

*A3-1. Primary side*

84. Water injection
   a. Actuation mode (manual/automatic)
   b. Injection pressure level (bar)
   c. Flow rate (kg/s)
   d. Mode of operation (active/passive)
   e. Redundancy
   f. Safety graded (Y/N)
85. Water recirculation and heat removal
   a. Intermediate heat sink (or heat exchanger)
   b. Mode of operation (active/passive): Passive
   c. Redundancy: Y
   d. Self sufficiency (h): indefinitely
   e. Safety graded: Y
86. Feed water
   a. Actuation mode (manual/automatic)
   b. Flow rate (kg/s)
   c. Mode of operation (active/passive)
   d. Redundancy
   e. Self sufficiency
   f. Safety graded
87. Water recirculation and heat removal
   a. Ultimate heat sink (cold source): atmosphere
   b. Mode of operation (active/passive): passive
   c. Redundancy: Y
   d. Self sufficiency (h): indefinitely
   e. Safety graded: Y
88. Implemented system (Name)
   a. Actuation mode (manual/automatic)
   b. Side location (primary/secondary circuit)
   c. Maximum depressurization
   d. Safety graded

**F. SEVERE ACCIDENT CONDITIONS**

B.1. Fission products retention

89. Containment spray system (Y/N): N
90. F.P. Sparging (Y/N): N
91. Containment tightness control (Y/N): Y
92. Leakage recovery (Y/N): N
93. Risk of recriticality (Y/N): N

B.2. Recriticality control

94. Encountered design feature
   a. Mode of operation (A/P)
   b. Safety graded

B.3. Debris confining and cooling

95. Core debris configuration (Core catcher)
96. Debris cooling system (name)
   a. Mode of operation (A/P)
   b. Self sufficiency
   c. Safety graded (Y/N)
97. Implemented system
   a. Mode of operation (A/P)
   b. Self sufficiency (h)

c. Safety graded (Y/N)

98. Intermediate heat sink
- a. Actuation mode (manual/automatic)
- b. Side location (primary/secondary circuit)
- c. Maximum depressurization rate (bar/s)

99. External coolant recirculation
- a. Implemented components
- b. Mode of operation (A/P)
- c. Self sufficiency (h)
- d. Safety graded (Y/N)

100. Ultimate heat sink
- a. Self sufficiency (h): atmosphere
- b. Safety graded (Y/N): (natural circulation)

B.5 Combustible gas control

101. Covered range of gas mixture concentration

102. Modes for the combustible gas control
- a. Containment inner: Y
- b. Gas burring: /
- c. Gas recombining: /
- d. Others: /

B.6 Containment pressure control

103. Filtered vented containment (Y/N)
- a. Implemented system
- b. Mode of operation (A/P)
- c. Safety graded)

104. Pressure suppression system (Y/N)
- a. Implemented system
- b. Mode of operation
- c. Safety graded (Y/N)

SAFETY RELATED I AND C SYSTEM

Automatic load following (Y/N): Y
I. range (% power): 10-100%
II maximum rate (%/min): approx. 2%/min
Load rejection without reactor trip (Y/N): Y
Full Cathode Ray Tubes (CRT) display (Y/N)
Automated normal shutdown procedures (Y/N): not yet
Use of field buses and smart sensors (Y/N): fixed in detail
Expert system or artificial intelligence advisors (Y/N)
Protection system backup (Y/N)

EMERGENCY POWER SUPPLY SYSTEM

105. Type (diesel, gas, grid connection): diesels, batteries

106. Number of trains: 2 trains

AC/DC SUPPLY SYSTEM

107. Type (rectifier, converter, battery): Y
1. AC un-interruptible Power Supply (2 trains): Y
2. diesel

108. Estimated time reserve (hr): 200hr

### 6.2.5. Project status

*6.2.5.1. Entities involved:*

Under the financial support of the Chinese Government, INET has been in charge of the R&D of NHR in China.

*6.2.5.2. Design status*

The NHR development program was initiated in 1980s. It has been one of the national key research projects in China. In 1989 the HR-5 was completed and has operated successfully for district heating since then. It has been decided to build the first commercial reactor- HR-200 in northeast China. The basic design of HR-200 has been finished in INET and the detail design of HR-200 is being undertaken.

*6.2.5.3. R and D work*

A. Executed R and D work

Since 1984 a series of research and development work has been made in order to design and build the NHR:

- stability study of the natural circulation,
- hydraulic driving facility for the control rod,
- hydraulic experiment of the primary heat-exchanger,
- reactor physical experiment on the critical facility,
- boric acid injection experiment,
- LOCA modelling experiment,
- sealing experiment of the pressure vessel,
- isolating valve test,
- penetration on the containment test,
- experiment on the new type of position indicator for control rod, etc.

At the same time necessary computer analysis models in the fields of physics, thermo-hydraulics, transient and accident analysis have been developed for the reactor design.

B. Ongoing and planned R and D

- Hydraulic control rod drive mechanism test
  The full scale (1:1) test of a hydraulic control rod drive mechanism is being made for HR-200.
- Thermal hydraulic test of reactor core
  The thermal hydraulic test of HR-200 reactor core is carried out in order to study the flow resistance, stability at operation conditions and operation parameters of reactor.
- Primary heat exchangers test
  The flow resistance and velocity distribution in HR-200 PHE is investigated.

C. R and D work needed

All known R and D required has been included in the tests completed, or is part of the ongoing and planned test program. However, until the licensing process has been completed need for additional tests may raise.

*6.2.5.4. Licensing status*

During HR-5 construction, main design features and safety concepts of the NHR have been reviewed and approved by the Chinese National Nuclear Safety Administration (NNSA). The HR-200 design is based on the HR-5 practice. Their main design features are the same. Therefore the HR-200 will be licensable. Up to now the first HR-200 plant siting report and environmental impact assessment report have been reviewed and approved, the design certification of INET for NHR has been also released by National Nuclear Industrial Co. (NNIC), and the review on the preliminary safety analysis report is under way.

**6.2.6. Project economic**

*6.2.6.1. Multi-Purpose Application*

The HR-200 is a kind of small sized reactor for heating supply initially and it is used also for multi-purpose applications. Some other work have been or will be finished by INET in order to improve NHR's economic benefits and load factor. These studies mainly include as the follows:

a. Refrigeration for air conditioning by heating LiBr with hot water or low pressure steam: This means that the NHR could be introduced in the summer so that the NHR could serve as district heating in winter and air conditioning in the summer in order to increase its utilization rate. The experiment on refrigeration using nuclear heat has been completed at the HR-5 in 1992.
b. Power generation at low-temperature for special consumers or the plant-own: The co-generation experiment has been finished at the HR-5 in 1990. To improve its economic efficiency, the NHR could be operated under co-generation mode by changing some of its system parameters.
c. Sea water desalination: The feasibility study on the desalination using the NHR has been finished and the experiment on this direction has being carried out at the HR-5.
d. Production of low pressure steam for industrial process heavy oil recovery and other applications.

*6.2.6.2. Economics of HR-200*

According to the primary economic analysis, the capital cost for HR-200 is about US$ 110 Million estimated in 1991. The heat generation cost of HR-200 is competitive with that of coal-fired plant under the Chinese condition. Considering the NHR multi-purpose applications, nuclear heating could be more competitive in economic.

## 6.3. CAREM REACTOR SYSTEM DESCRIPTION AND DEVELOPMENT STATUS

### 6.3.1. Basic Objectives and Features

#### *6.3.1.1. Basic objectives*

The main purpose of the CAREM project is to employ, in a safe way, nuclear energy in power ranges much lower than those used nowadays.

The CAREM project is based on a module of around 100 thermal Megawatt. A reactor of such power might be used to various ends :

- Electric energy production
- Industrial steam production
- Water desalting
- Domestic heating

#### *6.3.1.2. Main features of the CAREM reactor*

The reactor module may be defined according to the following characteristics:

- Light water and enriched uranium
- Integrated primary system
- Self-pressurized primary system
- Cooling by natural circulation
- Passive heat removal systems

Light water and enriched uranium

The choice is a light water and enriched uranium reactor. Enrichment was determined at a maximum value of 5%. The CAREM-25 (C-25) nucleo-electric module has an enrichment of 3.4%.

Integrated primary system

This feature implies that the steam generators are included in the pressure vessel (see fig.6.3.1). In this way, the primary system is made up by only one component: the pressure vessel.

For comparison, it may be outlined that the primary system of a conventional pressurized water reactor (PWR) has a pressure vessel, two to four steam generators, a pressurizer, two to four pumps and all the corresponding connection pipes.

Self-pressurized primary system

This means that the reactor generates its own operating pressure. The operating pressure is the steam pressure corresponding to the highest temperature, which is the temperature of the coolant at the core exit. However, some net steam is generated in the hot channel, but this steam is condensed in the riser, where it is mixed with the flow coming from the outer channels. The core is an open core and some mixing between channels takes also place inside the core. A steam

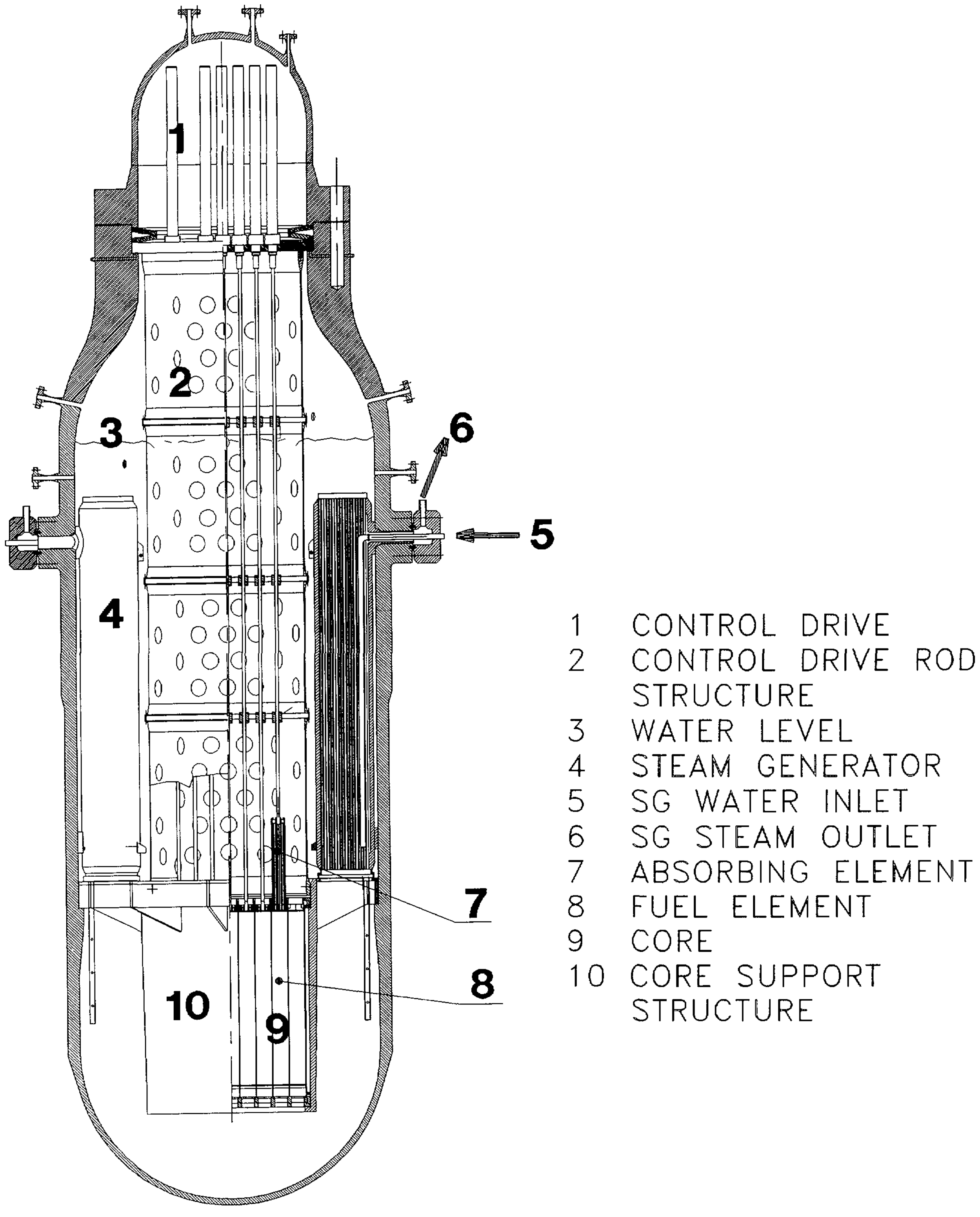



*Fig 6.3.1. Pressure Vessel and Primary System*

chamber located in the pressure vessel upper part absorbs the coolant volume changes due to temperature variations.

Cooling by natural circulation

The difference in temperature and therefore in density between cold and hot water, together with the difference in height between steam generator (heat sink) and core (heat source) centres, produce a given water circulation pattern inside the pressure vessel. The proper design adjustment of these parameters produces the required flow for core cooling.

Natural circulation is possible due to the very low head loss of the integrated system. Pumps in the primary system become unnecessary. This feature has many advantages, such as operation and maintenance simplicity, higher safety and costs reduction.

Passive heat removal systems

Cooling systems are fundamental to guarantee the safety of PWR's during reactor shutdown and emergencies.

The criterion adopted in the CAREM is to make these systems as simple and reliable as possible. Reactor design makes it possible to have very simple safety systems. The small power to be removed during shutdown and emergencies, allows systems designed to operate in a passive way (no pumps required).

### 6.3.2. Design Description

#### *6.3.2.1. Nuclear steam supply system*

6.3.2.1.1. Core design

To obtain a core design that is both safe and economically sound the most important neutronic, thermohydraulic and material science parameters have to be optimized. For a given power level, the basic parameters from the economy viewpoint are: degree of initial enrichment and discharge fuel burnup. From the viewpoint of reactor safety, important parameters are: cladding diameter, ratio between cladding diameter and pitch between fuel rods, distribution of absorbing rods, and the establishment of areas with different degrees of enrichment.

These parameters are not independent from each other. Optimum conditions are achieved through detailed consideration of their mutual interrelationship. Some of the established design goals are:

- Thermal power: 100 Mwt (approx. 25 Mwe)
- Minimum equilibrium cycle: 300 days at full power
- Highly negative temperature and density feedback coefficients
- Minimum fuel enrichment
- No boron for reactor control, except at start up sequences.
- Homogenous burnable poisons in the core.
- Good reactor thermal margins
- Low fuel centre-line temperature

The fuel element design also imposes some constrains:

- It has to withstand fast power increases, so that reactor power may follow demand.
- Fuel elements must accept neutron absorbing rods and burnable poisons.
- Fuel element flow impedance has to be as low as possible to ease natural circulation. This condition affects the design of support structures, upper and lower fuel element grids and fuel rod spacers.

6.3.2.1.2. Primary circuit

The pressure vessel contains the reactor core. It is the third barrier (after fuel matrix and fuel cladding) against dispersion of radioactive products generated in the core, and ensures the availability of water for core cooling.

Its design is based on ASME Pressure Vessel Code, Section III for Class 1 Pressure Vessels. Besides the application of pertinent international standards and codified procedures, the following design conditions were met in the design of the pressure vessel:

- Pipes and connections emerge above core level
- Pipework penetrations are reduced to less than 2"
- Water stagnation zones within the vessel are avoided.
- Safety valves are used to limit pressure surges
- Circulation of primary coolant is by natural convection. Therefore, undue pressure drops and flow direction changes are minimized.

Steam generators are contained inside the pressure vessel. The steam generators are twelve shell and tube once-through heat exchangers. The primary coolant flows downwards through the tube side, while the secondary coolant flows upwards through the shell side, where it is preheated, evaporated and superheated.

In order to minimize penetrations to the pressure vessel, feedwater inlet and steam outlet pipes for each steam generator share the same penetration.

*6.3.2.2. Balance of plant systems*

6.3.2.2.1. Secondary circuit

The secondary circuit comprises the steam generators, turbine, condenser, condensate pumps, deaerator, feedwater pump, feedwater heaters, bypass line, connecting pipes and control valves.

The once-through steam generators allow production of superheated steam.

Turbine inlet pressure has been set to 4.7 MPa. From this pressure upwards, gain in efficiency is not enough to compensate for the corresponding increase in steam generator area and pressure vessel volume.

Preheating temperature does not crucially affect operation. It is adjusted to ensure that the heat necessary to warm the condensate to boiling temperature is homogeneously distributed among preheaters and steam generator preheating zone.

Condensation pressure, on the other hand, has a strong effect on plant efficiency, and should be as low as possible.

6.3.2.2.2. Auxiliary systems

The plant has all the necessary and sufficient auxiliary systems usual in nuclear reactors, including :

- Primary and secondary water supply and treatment systems
- Chemical and volume control system
- Water supply and treatment for auxiliary systems
- Component cooling system
- Refuelling system
- Spent fuel element pool
- Radioactive waste treatment system
- In-plant electricity generation system
- Electricity distribution system
- Heating, ventilation and air conditioning system
- Bridge-crane and other lifting systems
- Gas, compressed air, fire-fighting and services in general

*6.3.2.3. Control systems*

The CAREM reactor is designed with a distributed digital control system. Sensor signals are digitalized and fed to micro-processors. These microprocessors are placed as close to the field as possible. Microprocessors communicate with each other through a redundant network. The man-machine interface is connected to the same network.

The reactor protection system (that is the instrumentation and control corresponding to the engineered safety systems), is also digitalized and based in microprocessors. The protection system has a high degree of redundancy and diversification. The control room is based mainly on screen data presentation, with some mimic diagrams projected on the wall. a secondary control point is available for monitoring and controlling the safety systems.

*6.3.2.4. Safety systems*

The CAREM reactor has five main safety systems :

- First shutdown system: absorbing elements
- Second shutdown system: boron injection
- Passive heat removal system
- Emergency injection system
- Containment system

They are designed to provide adequate protection under different circumstances: The two shutdown systems are responsible for stopping the chain reaction under requirement. The passive heat removal system is in charge of removing decay heat after reactor shutdown in case of power failure. The emergency injection system is responsible for keeping the core covered with water in case of a LOCA. The containment system prevents radioactive components from reaching the environment, upon occurrence of a serious accident. A great effort has been made to keep safety systems as simple, safe and reliable as possible. Simplicity and reliability are not unrelated concepts: the simpler the system, the more accurately its reliability can be assessed and enhanced.

One of the outstanding features of the CAREM reactor is that its safety systems are passive, which means that they have no moving parts except for valves and switches. This characteristic has as a consequence a qualitative improvement in reliability. Another of its consequences is that the safety systems do not require electric power, thus eliminating the need for safety grade Diesel generators.

6.3.2.4.1. First shutdown system: absorbing elements

The First or Rapid Shutdown System uses neutron absorbing rods which, when introduced in the core at high speed, produce the immediate stop of the fission chain. The absorbing material is a Ag-In-Cd alloy.

Each fuel element has 18 positions not occupied by fuel rods, that were designed to fit absorbing rods. Absorbing rods corresponding to the same fuel element are held together and move as a unit, called absorbing element. There are twenty four absorbing elements in the first shutdown system.

Sixteen absorbing elements of the system plus one absorbing element associated to the central fuel element are used to regulate and control reactivity during normal operation of the reactor. Each absorbing element has its own drive. The drives are of the hydraulic type, another innovation of the CAREM concept. This type of drive is fully included in the pressure vessel. They are designed so as to have inherent safety features. Absorbing elements are maintained at fixed positions while flow is being circulated through the drive. Controlled changes of rod positions are achieved by means of flow pulses. Should flow be interrupted, the rods will fall.

6.3.2.4.2. Second shutdown system: Boron injection

It is a back-up of the first shutdown system. It is capable of causing reactor shutdown by injecting borated water into the primary circuit. Two tanks filled with borated water are connected to the primary circuit during normal operation, with the same operative pressure as the primary system. When the Boron injection system is triggered by the protection system, a valve opens and borated water, driven by gravity, floods the primary circuit. The system has two redundant trains, with two redundant triggering valves in each train.

6.3.2.4.3. Passive heat removal system

It must remove heat generated by the core after a reactor shutdown in case of a power failure. Core cooling after shutdown is normally achieved by means of steam generators. Steam is dumped into the condenser. This operation requires electricity for the secondary and tertiary pumps. Its functioning cannot be guaranteed under all circumstances, making it unfit as a safety system for the CAREM reactor. Passive heat removal is achieved through a system driven by gravity, without moving parts, simple and highly reliable. It consists of two tanks that inject water in the steam generators. The system has two redundant trains, with two redundant triggering valves in each train.

6.3.2.4.4. Safety injection system

The core of a light water reactor must always be kept covered with water, to prevent a melt down and the consequent release of radioactive products. The safety injection system task is to ensure that the core will remain covered with water in the event of a Loss of Coolant Accident (LOCA). In this event, the pressure vessel water level will begin to decrease. Before the water level is low enough to endanger the core, the injection system is triggered and floods the pressure vessel with water, preventing core uncovering.

In the CAREM reactor only small LOCAs are possible: penetrations in the vessel have diameters smaller than 50mm. Furthermore, there is a large quantity of water in the pressure vessel above the core. This feature means that the time elapsed between the start of the accident (e.g. pipe rupture) and the beginning of core uncovering is long. During this time span the system depressurizes down to 15 bar with the core still covered. So the system must inject water into the pressure vessel at low pressure.

The system consists of a water filled tank that can be pressurized by gas. The tank is connected by a pipe with a valve to the pressure vessel. In the event the LOCA signal is triggered, the tank is pressurized by gas and the valve is opened allowing the water to flow into the pressure vessel. The tank volume and the water flow are calculated to ensure that the core will be kept covered for as long as necessary. The system has two redundant trains, with two triggering valves in each train.

6.3.2.4.5. Containment

The CAREM reactor containment is of the pressure suppression type. It has two regions: the dry region and the wet region. The dry region surrounds the pressure vessel. The wet region, that occupies most of the containment, has a pool filled with water in its lower part. The pool and the dry region are connected by pipes. The wet region outer wall transfers the energy to the atmosphere: the greater the wet region temperature the greater the heat transfer process efficiency.

*6.3.2.5. Buildings and structures*

The general lay-out of the plant is shown in figure 6.3.2. The reactor pressure vessel is almost completely under floor level. The upper part of the containment (the wet region) is metallic, the lower part of it (the dry region, containing the pressure vessel) is made of concrete, with a steel liner. The buildings, structures and foundations have been designed for a safety earthquake of 0.4 g horizontal acceleration.

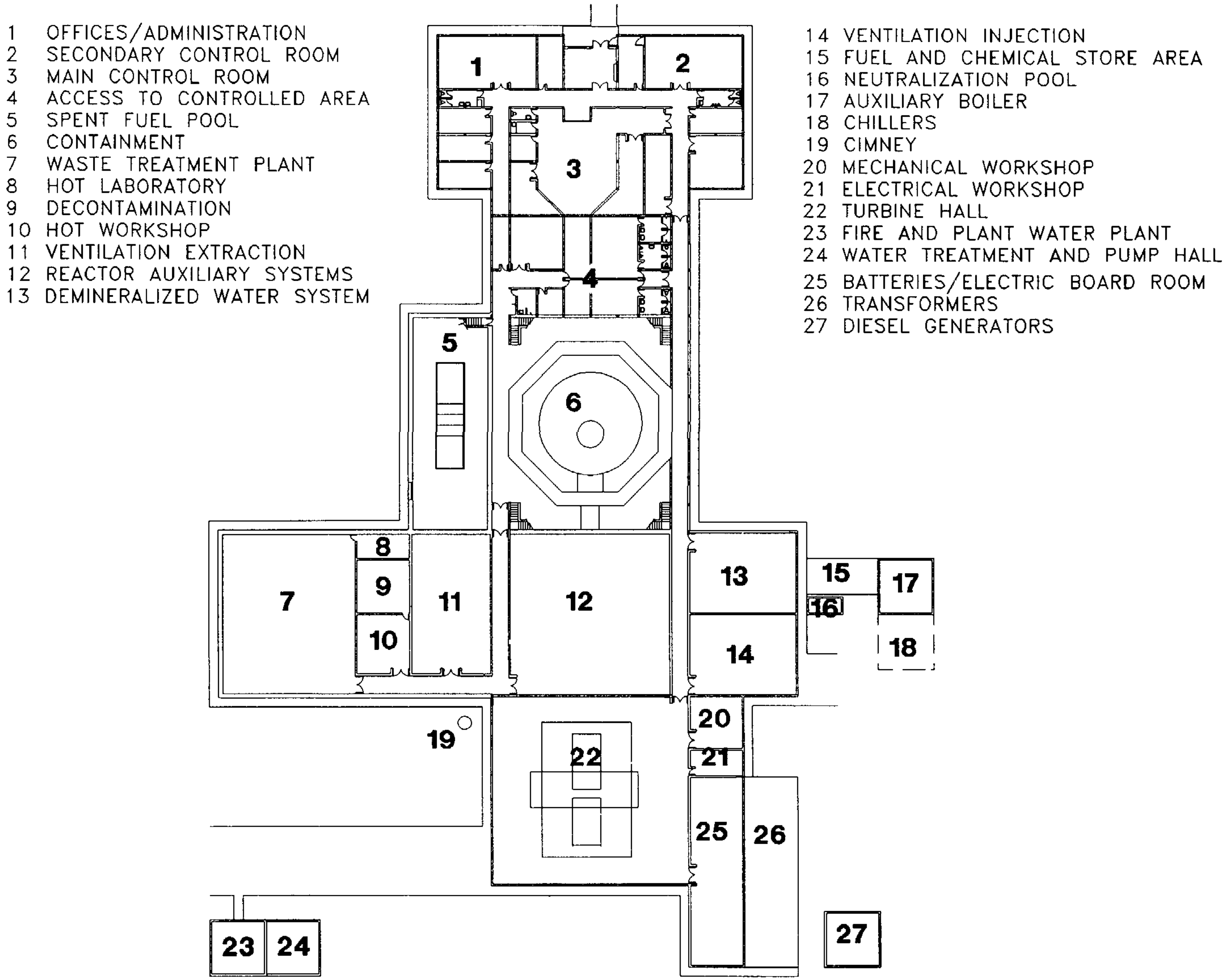



*Fig. 6.3.2. Plant Layout*

### 6.3.3. Safety concepts

The main safety concepts, criteria, design basis and systems are shown in the Tables 6.3.1. to 6.3.4.

TABLE 6.3.1. MAIN SAFETY RELATED SYSTEMS IN THE CAREM CONCEPT

| Name | Safety graded | Main characteristics |
|---|---|---|
| Primary Circuit | X | Integrated Natural circulation<br>12 steam generators |
| First shutdown system | X | 25 control rods |
| Second shutdown system | X | Passive boron injection |
| Containment | X | Pressure suppression type |
| Passive residual heat removal system | X | 2 trains, using the steam generators |
| Passive low pressure injection system | X | 2 trains |
| Passive containment cooling system | X | Air cooling through containment wall |

TABLE 6.3.2. MAIN ACCIDENT INITIATORS FOR THE CAREM

| |
|---|
| LOCA (Primary): Loss of Primary Coolant Accident<br>LOCA (Secondary): Secondary Pipe Rupture (water or steam)<br>LOCA (Interfacing): Steam Generator Tube Rupture<br>ATWS: Anticipated Transients Without Scram<br>Transients<br>Loss of main heat sink<br>Loss of safety heat sink<br>Station blackout |

TABLE 6.3.3. DESIGN FEATURES FOR PREVENTION AND PROTECTION LEVELS THAT REDUCE (R), SUPPRESS (S), THE INITIATORS FREQUENCY OR LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES (L)

**PREVENTION LEVEL**

LOCA (Primary)

- Reduced vessel fluence: reduces initiator frequency
- Integrated primary system: reduces initiator frequency and limits consequences
- Large primary system thermal inertia: reduces the number of relief valve actuation which reduces LOCA frequency and limits consequences of accidents
- Limitation of vessel penetrations to positions more than 3 meters above the top of the core: suppress initiators
- Small pipes diameters connecting to the primary system: supress initiators and limits consequences

LOCA (Secondary):

- Full condensate polishing: reduces secondary circuit corrosion and hence, reduces initiator frequency

LOCA (Interfacing)

- Soluble boron suppression: reduces corrosion and therefore initiator frequency
- Full condensate polishing: reduces steam generator corrosion and hence, reduces initiator frequency

Transients: Soluble boron suppression: reduces initiator frequency
Natural convection cooling: supress initiator
Self-pressurized primary: supress initiator
ATWS: Anticipated Transients Without Scram
Strong negative temperature coefficient: reduces frequency of problems caused by transients
Large primary system thermal inertia: reduces the number of relief valve actuation which reduces LOCA frequency.
Loss of main heat sink:
Loss of safety heat sink:
Air cooling of containment: supress initiator Station blackout:
Passive safety systems: supress initiators

**PROTECTION LEVEL**

LOCA (Primary)
Passive safety injection system: reduces core damage probability
Pressure suppression containment: high retention factor and so limits consequences
LOCA (Secondary): Low steam generator secondary side water inventory: limits consequences
LOCA (Interfacing)
Transients
Strong negative temperature coefficient: limits consequences
High primary cooling system thermal inertia: limits consequences
ATWS: Anticipated Transients Without Scram Strong negative temperature coefficient: limits consequences
Loss of main heat sink:
Strong negative temperature coefficient: Limits consequences Passive residual heat removal: Limits consequences
Station blackout:
Natural convection cooling: Limits consequences
Passive safety systems: Reduce core damage probability

TABLE 6.3.4. DESIGN FEATURES FOR MITIGATION LEVEL OF CAREM

| Safety functions | Systems | Passive/Active |
|---|---|---|
| **Design basis** | | |
| Fission product containment | Fuel rod<br>Integrated primary circuit | Passive<br>Passive |
| Coolant inventory | Integrated primary circuit<br>Low pressure injection system<br>Passive residual heat removal | Passive<br>Passive<br>Passive |
| Decay heat removal | Passive heat removal system<br>Passive containment cooling | Passive<br>Passive |
| Reactor shutdown | Shutdown rods<br>Boron injection | Passive<br>Passive |
| Pressure control | Self-pressurization | Passive |
| **Severe Accident** | | |
| Containment pres. & temperature control | Containment cooling to environment through containment wall | Passive |
| Heat removal | Low pressure injection system<br>Decay heat removal<br>Containment cooling system<br>Pressure suppression pool | Passive<br>Passive<br>Passive<br>Passive |
| Decay heat removal | Passive heat removal system<br>Passive containment cooling | Passive<br>Passive |
| Tightness control | | |
| Inflam. gas control | Igniters | Passive |
| Fission product containment | Pressure suppression pool<br>Containment | Passive<br>Passive |
| Corium management | | |

### 6.3.4. Design data sheet of CAREM

n.a = not applicable; (blank) = data not available to this date

## I. GENERAL INFORMATION

1. Design name: CAREM 25
2. Designer/Supplier address: CNEA/INVAP
3. Reactor Type: Integrated PWR / Number of modules/per plant: 1
4. Gross thermal power (MW-th): 100
5. Net Electrical output (MW-e): 27
6. Heat supply capacity (MW-th): n.a.

## II. BASIC DESIGN DESCRIPTION

## A. CORE AND REACTIVITY CONTROL

7. Fuel Material: $UO_2$
8. Fuel Inventory (tones of heavy metal): 3.721
9. Average core power density (kw/liter): 55
10. Average fuel power density (kW/kU): 26.87
11. Maximum linear power (W/m): 36000
12. Average discharge burnup (MWd/t): 22000
13. Initial enrichment (%): 3.4
14. Reload enrichment at equilibrium (%): 3.4
15. Refuelling frequency (months): 13
16. Type of refueling: Off Power
17. Fraction of core withdrawn (%): 50
18. Moderator material: Water
19. Active core height (m): 1.40
20. Core diameter (m): 1.29
21. Number of fuel assemblies: 61
22. Number of fuel rods per assembly: 108
23. Rod array in assembly: Triangular
24. Clad material: Zircaloy-4
25. Clad thickness (mm): 0.65
26. Number of control assemblies: 25
27. Type: Cluster
28. Additional shutdown systems: Boron-Injection
29. Control rod neutron absorber: Ag-In-Cd
30. Soluble neutron absorber: Boric acid
31. Burnable poison material and form: $Gd_2O_3$-$UO_2$

## B. REACTOR COOLANT SYSTEM

### B1. Coolant

32. Coolant medium and inventory: Water/39 $m^3$
33. Design coolant mass flow though core (Kg/s) 410.
34. Cooling mode: natural
35. Operating coolant pressure (bar): 122.5
36. Core Inlet temperature (°C): 284
37. Core Outlet temperature (°C): 326

### B2. Reactor pressure vessel

38. Overall length of assembled vessel (m): 11
39. Inside vessel diameter (m): 2.84
40. Average vessel thickness (mm): 120
41. Vessel material: SA-533B
42. Lining material: SS-304L
43. Design pressure (bar): 145
44. Gross weight (tone): 120

### B3. Steam Generator

45. Number of steam generator: 12
46. Type: Once through
47. Configuration: Integrated Vertical
48. Tube Material: Incolloy 800
49. Shell Material: SA-533B
50. Heat transfer surface per steam gen. ($m^2$): 73
51. Thermal Capacity per Steam Gem (MW): 8.4

52 Feed water pressure (bar) 50
53 Feed water temperature (°C) 200
54 Steam pressure (bar) 47
55 Steam temperature (°C) 290

**B4. Pressurizer**

56 Type As free volume in pressure vessel
57 Steam volume (full power) ($m^3$) 8

**B5. Main coolant pumps**

58 Number of cooling or recirculation pumps n a
59 Type n a
60 Pump mass flow rate (Kg/s) n a
61 Pump design rated head n a
62 Pump nominal power (kW) n a
63 Mechanical inertia (Kg $m^2$) n a

## C. CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS)

64 Number of extraction lines 1
65 Number of Pumps 3
66 Number of injection points 2
67 Feed and bleed connections n a

## D. CONTAINMENT

68 Type Pressure suppression
69 Overall form Cylinder
70 Structural material Steel
71 Liner Material n a
72 Simple/double wall Simple wall
73 Dimensions (diameter, height)(m) 11 5-15 8
74 Design pressure (bar) 5 5
75 Design temperature (°C) 120
76 Design leakage rate (% per day) 0 5

# III. SAFETY RELATED SYSTEMS

## A. DESIGN CONDITIONS

**A1. Fission product retention**

77 Containment spray system N
78 F P sparging N
79 Containment tightness control Y
80 Leakage recovery· N
81 Guard vessel N

**A2. Reactivity control**

82 Absorber injection system Y
- a Absorber material Boron
- b Mode of Operation Passive
- c Redundancy Tanks 2*100% Valves 4*100%
- d Safety graded Y

83 Control rods Y
- a Maximum control rod worth (pcm) 2200
- b Mode of operation Active
- c Redundancy Stuck rod criteria
- d Safety graded Y

**A3. Decay heat removal**

*A3-1 Primary side*

84 Water injection
- a Actuation mode Automatic
- b Injection pressure level (bar) 15
- c Flow rate (kg/seg) 0 92
- d Mode of Operation Passive
- e Redundancy Tanks 2*100% Valves 4*100%
- f Safety graded Y

85 Water Recirculation and heat removal
- a Intermediate heat sink n a
- b Mode of operation Passive

c. Redundancy Tanks: 2*100% Valves: 4*100%
d. Self sufficiency (h): 72
e. Safety graded: Y

*A3-2 Secondary side*

86. Feed water
- a. Actuation mode: Automatic
- b. Flow rate (kg/seg): 1.0
- c. Mode of operation: Active
- d. Redundancy Pumps: 2*100%
- e. Self sufficiency (h): none
- f. Safety graded: N

87. Water recirculation and heat removal
- a. Ultimate heat sink (cold source) Atmosphere
- b. Mode of operation: Passive
- c. Redundancy: n.a.
- d. Self sufficiency: Unlimited
- e. Safety graded: n.a.

*A3-3 Primary pressure control*

88. Implemented System: Relief Valves
- a. Actuation mode: Automatic
- b. Side location: Top of the pressure vessel
- c. Maximum depressurization rate (bar/s) n.a.
- d. Safety graded: Y

## B. SEVERE ACCIDENT CONDITIONS*

**B1. Fission product retention**

89. Containment spray system: N
90. F.P. sparging: N
91. Containment tightness control: Y
92. Leakage recovery: N
93. Risk of recriticality: N

* All systems must be qualified to operate under the accident conditions

**B2. Recriticality control**

94. Encountered design feature: N
- a. Mode of Operation: n.a.
- b. Safety Graded: n.a.

**B3. Debris confining and cooling**

95. Core debris configuration (core catcher): N
96. Debris cooling system: N
- a. Mode of operation: n.a.
- b. Self sufficiency: n.a.
- c. Safety graded: n.a.

**B4. Long term containment heat removal**

97. Implemented system: Natural convection
- a. Mode of operation: Passive
- b. Self sufficiency (h): Unlimited
- c. Safety graded: Y

98. Intermediate heat sink: N
- a. Implemented system: n.a.
- b. Safety graded: n.a.

99. External coolant recirculation: N
- a. Implemented components: n.a.
- b. Mode of operation: n.a.
- c. Self sufficiency (h): n.a.
- d. Safety graded: n.a.

100. Ultimate heat sink: Atmosphere
- a. Self sufficiency (h): Unlimited
- b. Safety graded: n.a.

**B.5. Combustible gas control**

101. Covered range of gas mixture concentration
102. Modes for the combustible gas control
- a. Containment inertation: N
- b. Gas burning: Y
- c. Gas recombining: N
- d. Others: N

**B.6. Containment pressure control**

103. Filtered vented containment: N

a Implemented system n a
b Mode of operation n a
c Safety graded n a

104 Pressure suppression system Yes
a Implemented Condensing with water
b Mode of operation Passive
c Safety graded Y

**C. SAFETY RELATED I&C SYSTEM**

Automatic load following system Y
* range(% power) 50
* maximum rate(%/min)

Load rejection without reactor trip Y
Full Cathode Ray Tubes (CRT) display Y
Automated start-up procedures Y
Automated normal shutdown procedures Y
Automated off normal shutdown procedures N
Use of field buses and smart sensors Y
Expert systems or artificial
Intelligent advisors Y
Protection system backup N

**D. EMERGENCY POWER SUPPLY SYSTEM**

105 Type n a
106 Number of trains n a

**E. AC/DC SUPPLY SYSTEM**

107 Type Batteries
108 Estimated time reserve (hr)

**IV. <u>CONVENTIONAL THERMAL CYCLE</u>**

**A. TURBINE SYSTEM**

109 Type Condensing
110 Overall length (m) 5 3
111 Width (m) 3 7
112 Number of turbines/reactor 1
113 Number of turbine sections per unit 1
114 Speed (rpm) 3000

**B. STEAM CHARACTERISTICS**

115 H P inlet pressure (bar) 45
116 H P inlet temperature (°C) 290
117 H P inlet flowrate (t/h) 175 32
118 L P inlet pressure n a
119 L P inlet temperature n a
120 L P inlet flowrate (per section ) n a

**C. GENERATOR**

121 Type (3-phase synchronous)
122 Apparent power (MVA) 37 5
123 Active Power (MW) 30 0
124 Frequency (hz) 50
125 Output voltage (kV) 13 2
126 Total generator mass (t)
127 Overall length (m) 6 2
128 Stator housing outside diameter

**D. CONDENSER**

129 Number of tubes
130 Heat transfer area ($m^2$) 2116
131 Flowrate (secondary/tertiary) ($m^3$/h) 130/8400
132 Pressure (secondary side) (mbar) 75
133 Temperature (°C) (tertiary inlet/outlet) 28/35

**E. CONDENSATE PUMPS**

134 Number 2
135 Flowrate ($m^3$/h) 130/180
136 Developed head (bar) 10
137 Temperature (°C) 40
138 Pump speed

### 6.3.5. Project status

#### *6.3.5.1. Entities involved*

The Carem reactor is being developed by the Argentine "Comisión Nacional de Energía Atómica" (CNEA). CNEA has subcontracted the design and development of the reactor to INVAP. The design and development of the fuel element are carried on by CNEA.

#### *6.3.5.2. Design status*

The basic design of the CAREM reactor is already completed. This means a qualification of 2 in the corresponding scale.

#### *6.3.5.3. Research and development status*

There is a comprehensive research and development effort going on. This effort is oriented to validate the design assumptions made for completing the basic engineering. The main areas and installations involved in the development are:

- Critical facility: A critical facility has been built and will be operated during 1994/95, for adjusting the core neutronic calculations.
- Natural convection loop: A natural convection loop is in operation, its main purpose being the modelling of the behaviour of CAREM primary circuit and the validation of the correlations used for DNB calculations. The loop is a full temperature and pressure model, with an electrical heater of 200 Kw.
- Hydraulic control rod drives: A full scale prototype of the control rod drive is under construction and will be tested during 1994.
- Steam generators: A full scale model of the steam generator will be constructed and tested during 1994/95.
- Instrumentation and control: During 1994 and 95, tests will be carried out to demonstrate the reliability of the distributed control system.
- Protection system: A full scope simulation of the protection system will be tested during 1994 and 1995.
- Fuel element development: A comprehensive program, aim at resolving every detailed design and manufacturing aspect of the fuel element is going on.

In 1995, the reactor will be ready for construction.

#### *6.3.5.4. Licensing status*

A Preliminary safety analysis report is available and has been sent to the Regulatory authority, but no formal licensing application has been submitted yet. This means a qualification of 1.5 in the corresponding scale.

### 6.3.6. Project economics

Feasibility studies have been carried out in two states in Argentina, showing CAREM to be competitive with alternative sources in these regions, in the same range of power.

## 6 4. MRX REACTOR SYSTEM DESCRIPTION AND DEVELOPMENT STATUS

### 6.4.1. Basic objectives and features

MRX (marine Reactor X) is an advanced marine reactor which adopts the following new concepts to satisfy the requirement of the nuclear ships for the next generation. MRX is designed at a power of 100MWth for the reactor plant of an icebreaker scientific observation ship, but the concept could be applied to general commercial nuclear ships with an output range from 50 to 300MWth depending on the application and on type, size and velocity of ship.

The following specific features of the MRX design can be identified:

* Integral type PWR
  - eliminates the possibility of a large LOCA, simplifies the engineered safety system and makes for a compact reactor plant,
* In-vessel type control rod drive mechanisms
  - eliminate the possibility of "rod ejection" accidents, simplify the engineered safety system and also makes for a compact reactor plant.
* Water-filled containment vessel
  - passively maintains core flooding.
  - is of great help towards compactness of reactor plant.
* Passive decay heat removal system
  - simplifies engineered safety system.

MRX is significantly smaller and lighter compared with previous marine reactors such as those of the Nuclear Ship MUTSU, OTTO HAHN and SAVANNAH. Figures 6.4.1. and 6.4.2. show the conceptual view of MRX and an example of a two-MRX equipped icebreaker for scientific observations.

### 6.4.2. Design description

#### *6.4.2.1. Nuclear steam supply system*

Reactor pressure vessel

The steam generator and pressurizer of MRX are installed in the reactor pressure vessel (RPV). Welds are made only at the boundary between the cylindrical and bottom shells as shown in Fig 6.4.1. Twelve pipes penetrate the RPV, of which four pipes are for main steam and feed water in the secondary system and eight pipes ($\leq$50mm ID) of the primary coolant system are for the volume control system, the residual heat removal system, the pressurizer relief valve and water spray. All the penetrations are incorporated into the upper cylindrical region of the reactor vessel. The RPV is relatively larger in size because it is an integral type PWR. This provides a larger primary water inventory, increases the distance between reactor core and RPV, and reduces the neutron fluence at the RPV. The calculated value of the fluence of fast neutrons is below $8\times10^{15}$n/cm$^2$ (E$\geq$1.11MeV) at the inner-surface of RPV for full power reactor operation of 20 continuous years. This value is sufficiently low compared with the design criteria of $10^{19}$n/cm$^2$(E$\geq$1MeV) for radiation damage of the RPV. Specific design characteristics for the RPV are given in the general design data (in section 6.4.4.).

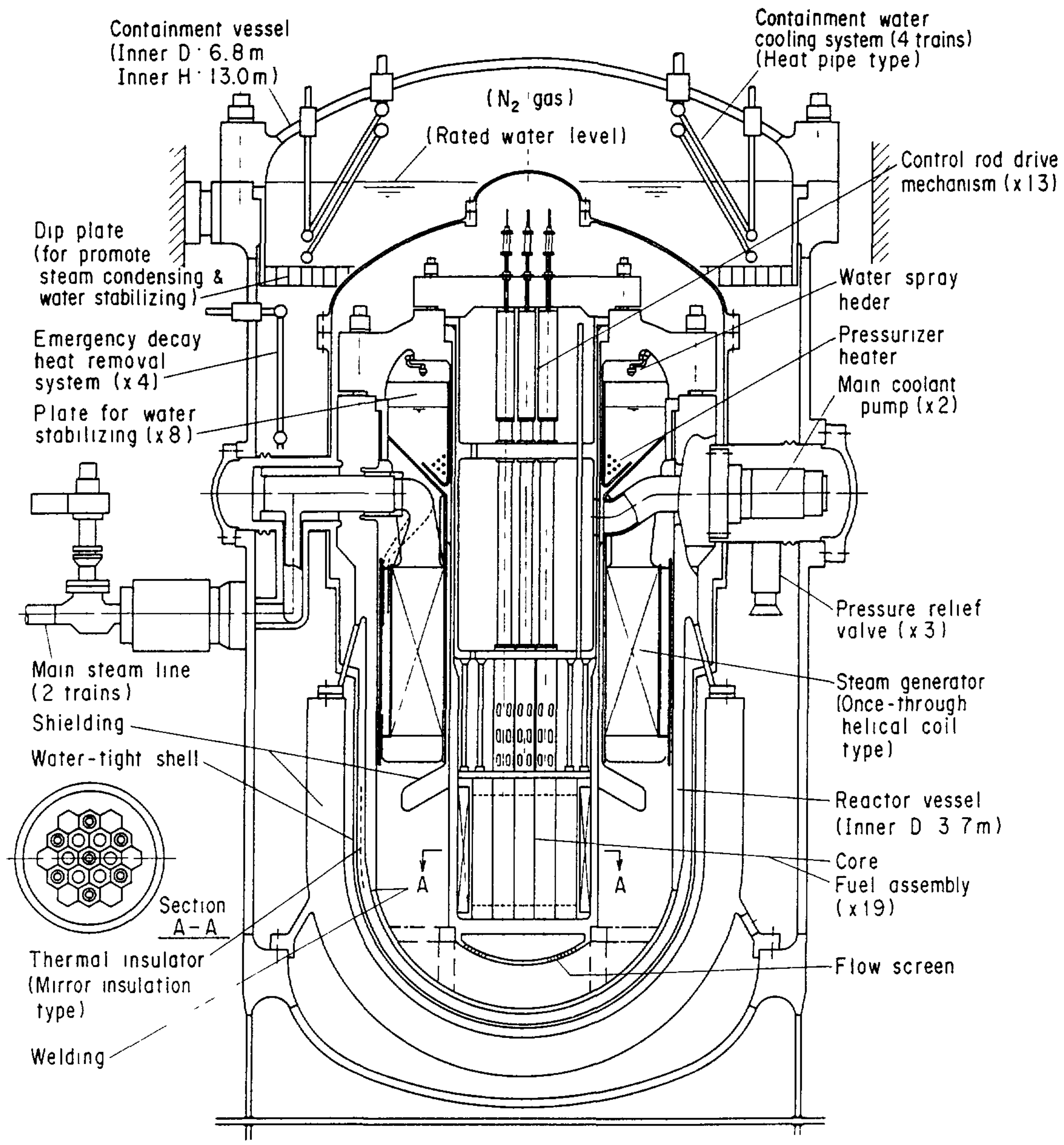



*Fig 6.4.1. Concept of MRX Plant*

Reactor core

Some of the fuel pins contain burnable poison to reduce the peaking factor. The design of fuel element is based on well developed PWR fuel technology. The core consists of 19 hexagonal fuel assemblies. Thirteen assemblies contain the control rod clusters, where six are used for reactivity control and the other seven for reactor shutdown. The average power density is sufficiently low (42kW/l) for the core to have enough margin for thermal reliability. The fuel life time and refuelling frequency are 8 and 4 years, respectively. The fuel handling system is installed in land facilities.

The major features are as follows:

- to maintain non-criticality without use of soluble poison even under the condition that the control rod cluster which has the largest reactivity worth is withdrawn from the core.
- to operate the reactor with a sufficient power level for steerageway even in the case that the control rod cluster which has the largest worth cannot be withdrawn from the core.
- to keep enough residual reactivity for overriding Xe poisoning at the EOL (end of life).
- to maintain a negative temperature coefficient of reactivity.

Control rod drive mechanism

The control rod drive mechanisms are placed in the upper region inside the RPV to enhance the safety by excluding the "Rod Ejection Accident" and to achieve a compact reactor size.

Steam generator

The steam generator, of the once-through helical coil type, is positioned inside RPV. Two trains are adopted for the main steam and feed water lines.

Primary circuit

Most of the primary circuit is incorporated inside the RPV including the core, the steam generator and a pressurizer installed in the upper region of RPV. Main coolant pumps, a volume control system and a residual heat removal system are outside the RPV. Two main coolant pumps are placed in the hot leg of the upper cylindrical region the RPV as shown in Fig 6.4.1.

Containment vessel

A water-filled containment system is adopted to maintain core flooding passively and to realize a compact reactor plant. The design pressure of the containment vessel (CV) is 4MPa to withstand the pressure from a LOCA. The RPV is surrounded with a water tight shell for thermal insulation.

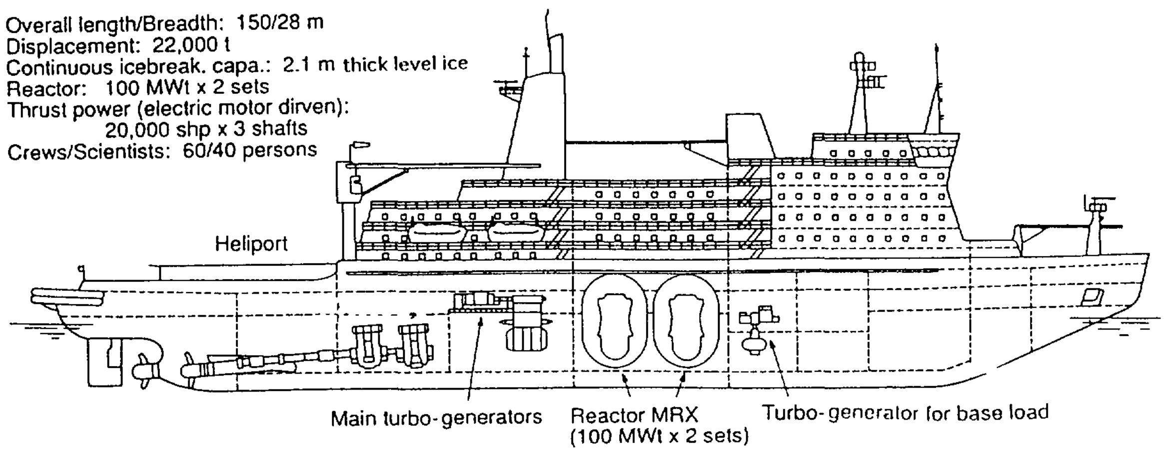



*Fig 6.4.2. Two MRX equipped icebreakers for scientific oberservations*

Thermal insulation for RPV

The water-tight shell surrounding the RPV contains a thermal insulator. Heat loss from the RPV is estimated to be a few 100kW during reactor operation. The shell can withstand the normal static pressure and the additional pressure from anticipated transients. Pressure relief valves are installed in the CV to mitigate the rise of pressure in the space between the RPV and the water tight shell due to pipe break accidents. Leakage of water into the thermal insulation space can be detected by a moisture detector placed outside the CV using a nitrogen gas flow through the insulation space. Leak of the primary coolant can be identified from that of the containment water by its radioactivity.

#### *6.4.2.2. Balance of plant*

- Main Steam System

The main steam system of MRX consists of a main generator turbine for propulsion, two hotel service generator turbines, three feed water pump turbines, a low pressure steam generator, etc. The main generator turbine is a double compound articulated type and produces 30MWe. The output is used for the propulsion motors. The electric power for auxiliary machines and accommodations is supplied by two hotel service generator turbines.

A moisture separator is installed in the exhaust steam line from the high pressure turbine of the main generator turbine. The low pressure turbine is mounted on the condenser.

The design pressure between the steam generator and the isolation valves in the secondary circuit is to equal with that of primary circuit. In the case of a steam generator tube rupture, the isolation valves in the secondary circuit are closed immediately to prevent the release of primary coolant to the environment.

- Feed Water System

The feed water system consists of main water pumps, auxiliary feed water pumps, feed water heaters, etc. The feed water heaters consist of a low pressure feed water heater, a dearator, and two stage high pressure feed water heaters which raise the temperature of feed water to 185°C.

Three feed water pumps are installed which are driven by steam turbines. Two pumps are used in normal operation. Each has a flow capacity of 50% of full reactor power operation and one feed water pump trip does not cause any reactor scram.

The two auxiliary feed water pumps are driven electrically. They are used to start up and shut down the reactor, and to remove the decay heat.

#### *6.4.2.3. Instrumentation, control and electrical system*

The primary purpose of the instrumentation, control and electrical system is to provide proper reactor operation during steady state and transient power situations, and to maintain the safety automatically.

Control rod motion is required to follow large and rapid load changes due to ship propulsion. The main load conditions are as follows:

- Ramp load increase and decrease of 3% per second over the load rang of 70% to 100% of full power.
- Rapid load reduction from 100% to 70% of full power in one second.

The primary coolant system is controlled to maintain the average temperature and pressure. Feed water flow to the steam generator is controlled to maintain steam pressure within a predetermined limit during normal operating transients. Supervision of both nuclear and turbine propulsion plants is carried out in the control room.

### *6.4.2.4. Safety considerations*

A large LOCA cannot occur in MRX, since only small size pipes ($\leq$ 50mm) exist in the primary system  In a small LOCA, the engineered safety system of MRX keeps the core flooded and removes the decay heat without emergency water injection  Figure 6 4 3  shows a schematic of the MRX engineered safety  system  The engineered safety is accomplished by passive  and active systems

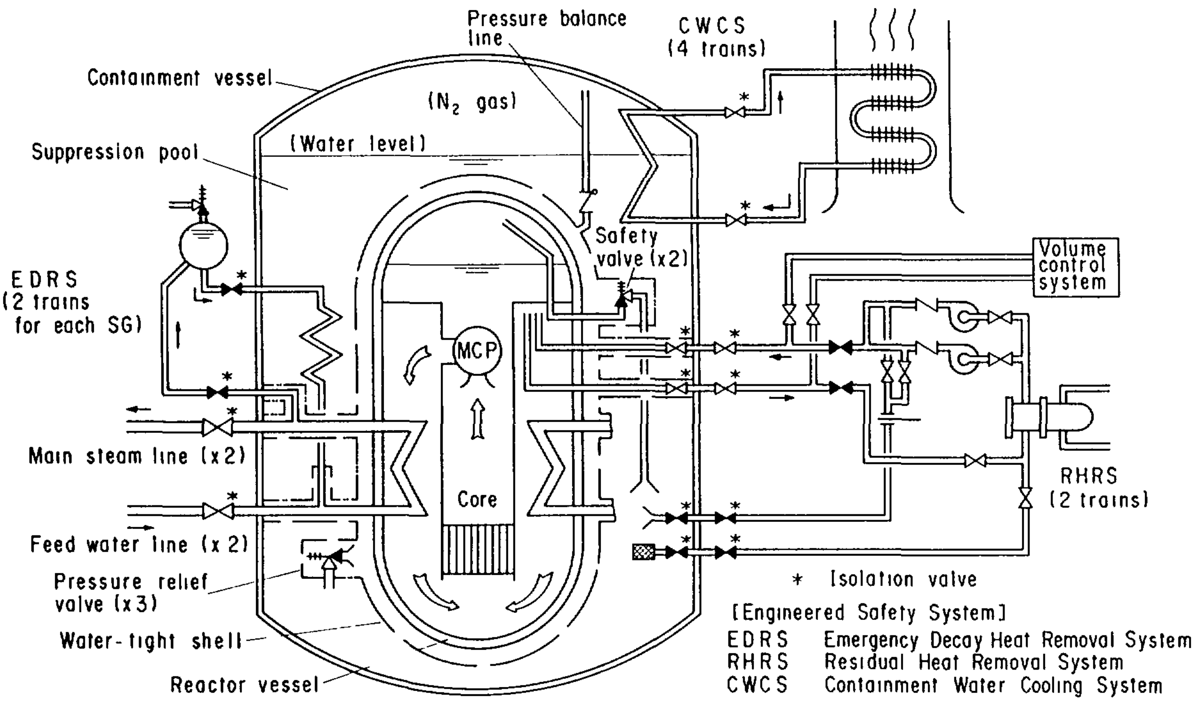



*Fig 6.4.3    Concept of MRX engineered safety system*

- Emergency decay heat removal system

The emergency decay heat removal system (EDRS) is a closed system which transfer decay heat from the core to the containment water  It includes four trains each of which has the ability to remove the core decay heat  Each train consists of a water reservoir tank, a cooler,  two valves and piping  In the case of accidents, the valves are opened after  the main feed water/steam isolation valves are closed  Coolant will be circulated by natural convection, heated by the steam generator and cooled by  the cooler   An alternative EDRS is also being studied

- Containment water cooling system

The containment water cooling system (CWCS) is a heat pipe  system for long term decay heat removal transferring the heat in the containment water to the atmosphere   For its working gas,  an  anti-freezing  gas  such  as  R22  (CHCIF2)  will  be  used  to  allow  for  possible  low temperature  conditions at sea

- Residual heat removal system

The residual heat removal system (RHRS) is used for controlling the water temperatures in both the RPV and CV for long term cooling after LOCA and is also used for long term decay heat  removal  during  scheduled  reactor shutdown

- LOCA analysis

Figure 6.4.4. shows results of LOCA analysis obtained by RELAP5/Mod2 calculation assuming 50 mm dia. pipe break (double ended Guillotine break) in the space between the RPV and the water tight shell. The diameter of the pressure relief valves (see Fig.6.4.1.) is 300 mm. The initial free volume in the containment vessel is 27m$^3$ filled with nitrogen gas at 60°C and 0.1MPa. Cooling by the EDRS is taken into account, but the functions of the CWCS and RHRS are neglected. The analysis shows that the lowest water level in the RPV is about 1m above the upper edge of the core, which is enough to keep the core flooded even taking into account ship inclination and oscillation. The analysis also shows that the pressure in the CV during accidents is below 1.5MPa, which is sufficiently small compared with the design value of 4MPa.

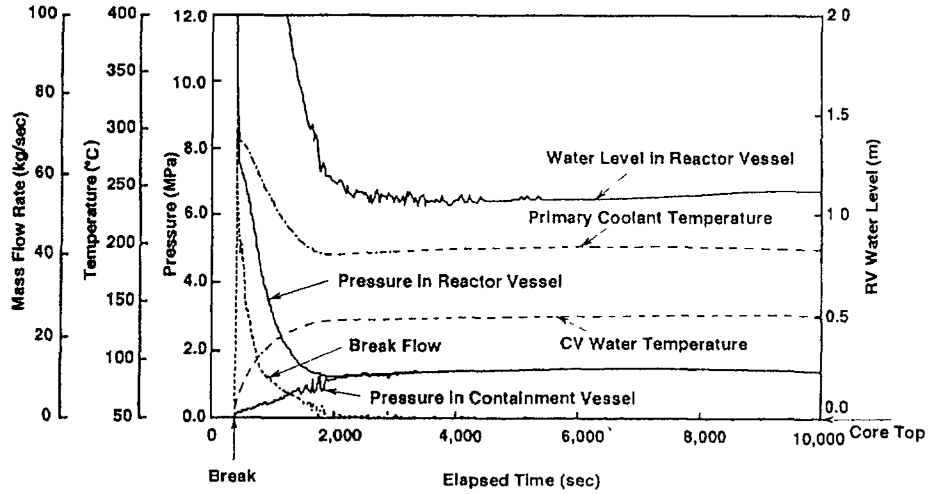



*Fig 6.4.4. Transient responses in LOCA ibtained by RELAP5/Mod2 (50mm dia. pipe break)*

*6.4.2.5. Buildings and structure*

The nuclear icebreaker is considered for use as a scientific observation ship with two MRX power units, as shown in Fig 6.4.2.

The principal particulars of the ships are as follows:

| | | | |
|---|---|---|---|
| Overall length (m) | 150 | Reactor type | MRX (integral type PWR) |
| Max. breadth (m) | 30 | Reactor output | 100MWt x 2 units |
| Moulded depth (m) | 15 | Shaft horsepower (ps) | 20,000 x 3 shafts |
| Max. draught (m) | 9 | Main turbo-gene.output(kW) | 24,000 x 2 sets |
| Displacement (t) | 22,000 | Speed in open water (kn) | 20 |

Continuous ice breaking ability 2.1m thick ice at a speed of 2kn
Complement (persons) 100 (officers & crews: 60, scientists: 40)

The most severe ships motion conditions should be determined in the future by conducting long term analysis of acceleration, inclination and oscillation due to weather and marine conditions according to IMO nuclear ship safety standards. However, the following values are assumed at present:

| | | | |
|---|---|---|---|
| Acceleration (vertical and horizontal) 1g | | Heel angle | 30 degree |
| Pitching angle 10 degree | | Rolling angle | 45 degree |

TABLE 6.4.1. MAIN SAFETY RELATED SYSTEMS IN THE MRX CONCEPT

| Name | Safety graded | Main characteristics |
|---|---|---|
| Primary Circuit | X | Integral type PWR eliminating large size primary coolant piping<br>2 main coolant pumps<br>2 residual heat removal systems (RHRS) |
| Control Element Absorber | X | 13 control rod drives with built-in type CRDM eliminating Control Rod Ejection Accident |
| Diverse Reactivity Control System | X | Boron injection |
| Water filled Containment System | X | Passive maintenance of core flooding by limiting primary coolant flow out |
| Passive Residual Heat Removal System | X | Heat transfer from primary coolant to CV water by natural circulation of coolant in S.G. |
| Passive Containment Cooling System | X | Cooling of CV water by heat pipe type cooler |
| Auxiliary Feed Water System | X | 2 pumps |

TABLE 6.4.2. MAIN ACCIDENT INITIATORS FOR THE MRX

- LOCA (Primary): Loss of Primary Coolant Accident (small pipe break)
- LOCA (Secondary): Secondary Pipe Rupture (water or steam)
- LOCA (Interfacing): Steam Generator Tube Rupture
- ATWS: Anticipated Transients Without Scram
- Primary Transients
- Secondary Transients: Turbine Trip
- Total loss of Load
- Total loss of the Steam Generator Feed Water
- Total loss of Electricity

TABLE 6.4.3. DESIGN FEATURES FOR PREVENTION AND PROTECTION LEVELS THAT REDUCE (R), SUPPRESS (S), THE INITIATORS FREQUENCY OR LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES (L)

| PREVENTION LEVEL | |
|---|---|
| LOCA (Primary) | |
| - Reduced vessel fluence | reduce initiator frequency |
| - Primary circuit integration | avoid large size LOCA |
| - Canned pumps | suppress initiator (seal caused small LOCA) |
| - Adaption of the leak before break | limits accident consequence |
| LOCA (Secondary) | |
| - | |
| LOCA (Interfacing) | |
| - Tough structure of S G tubes | reduce initiator frequency |
| Primary transients | |
| - Built-in type CRDM | eliminate CR ejection accident |
| Secondary transients | |
| **PROTECTION LEVEL** | |
| LOCA (Primary) | |
| - Water filled containment | keep core flooding passively |
| - Passive cooling system | avoid operator's mis-operation |
| LOCA (Secondary) | |
| - | |
| LOCA (Interfacing) | |
| - Isolation valves | easy S G isolation |
| ATWS | |
| - Strong negative temperature coefficient | |
| Primary transients | |
| - Low power density | |
| - Large primary inertia | |
| Secondary transients | |
| - Low power density | |
| - Large primary inertia | |
| Total loss of load | not critical * |
| Total loss of the S G feed water | not critical * |
| Total loss of electricity | not critical * |

* Decay Heat Removal System by natural convection, low power density and large primary inertia contribute to make these three situations not significant

TABLE 6.4.4. DESIGN FEATURES FOR MITIGATION LEVEL

| Safety Functions | Systems (Cf Tab 6 4 1 ) | Passive / Active | Design features / Remarks |
|---|---|---|---|
| **DESIGN BASIS**<br>Fission product containment | Cladding<br>Primary circuit<br>Containment | passive<br>passive<br>passive | |
| Coolant inventory | Water filled containment | passive | Limitation of coolant flow out |
| Decay heat removal | Emergency decay heat removal system<br>Containment water cooling system | passive<br>passive | Heat pipe type cooling system |
| Reactivity control | Control rod<br>Boron injection | active<br>active | |
| Primary circuit pressure control | not necessary | | Emergency injection is not necessary |
| **SEVERE ACCIDENT**<br>Containment temperature and Pressure control | Water filled containment<br>Containment water cooling system | passive<br>passive | |
| Heat removal | Water filled containment | passive | |
| Tightness control | Water filled containment | passive | |
| Inflam gas control | not necessary | passive | Inert gas environment ($N_2$) |
| Fission product containment | Water filled containment | passive | Flood containment |
| Corium management | Core catcher (shielding outside RPV) | passive | |

### 6.4.4. Design Date Questionnaire

## I. GENERAL INFORMATION

1. Design name: MRX
2. Designer/Supplier address: Japan Atomic Energy Research Institute, Tokai-mura, Ibaraki-ken, 319-11, Japan
3. Reactor type: Integral type PWR
   Number of modules/per plant: 2
4. Gross thermal power (MW-th) per reactor: 100
5. Net electrical output (MW-e) per reactor: 30
6. Heat supply capacity (MW-th): --

## II. BASIC DESIGN DESCRIPTION

### A. CORE AND REACTIVITY CONTROL

7. Fuel material: $UO_2$
8. Fuel inventory (tones of heavy metal): 6.3
9. Average core power density (kW/liter): 42
10. Average fuel power density (kW/kgU): 16
11. Maximum linear power (W/m): 30,000
12. Average discharge burnup (MWd/t): 23,000
13. Initial enrichment or enrichment range (Wt%): 4.3
14. Reload enrichment at the equilibrium (Wt%): 4.3
15. Refuelling frequency (months): 44
16. Type of refuelling (on/off power): Off power
17. Fraction of core withdrawn (%): 52.6
18. Moderator material and inventory: $H_2O$
19. Active core height (m): 1.4
20. Core diameter (m): 1.49
21. Number of fuel assemblies: 19
22. Number of fuel rods per assembly: 493
23. Rod array in assembly: Triangle
24. Clad material: Zircaloy 4
25. Clad thickness (mm): 0.57
26. Number of control rods or assemblies: 13
27. Type: Rod cluster
28. Additional shutdown systems: see No. 82
29. Control rod neutron absorber material: $B_4C$
30. Soluble neutron absorber: see No. 82
31. Burnable poison material and form: Fuel rod with $Gd_2O_3$ and burnable poison rod of borosilicate glass

### B. REACTOR COOLANT SYSTEM

#### B1. Coolant

32. Coolant medium and inventory: $H_2O$ (41t)
33. Design coolant mass flow through core (kg/s): 1,250
34. Cooling mode (forced/natural): Forced
35. Operating coolant pressure (bar): 120
36. Core inlet temperature (°C): 282.5
37. Core outlet temperature (°C): 297.5

#### B2. Reactor pressure vessel

38. Overall length of assembled vessel (m): 9.4
39. Inside vessel diameter (m): 3.7
40. Average vessel thickness (mm): 150
41. Vessel material: JIS G 3204 SFVQ1A
42. Lining material: Stainless steel
43. Design pressure (bar): 137
44. Gross weight (tone): 305

**B3. Steam generator**

45. Number of steam generators: 1 (2 trains)
46. Type: Once-through helical coil type
47. Configuration (horizontal/vertical): Vertical
48. Tube material: Incoloy 800
49. Shell material: --
50. Heat transfer surface per steam generator ($m^2$): 754
51. Thermal capacity per steam generator (MW): 100
52. Feed water pressure (bar): 58
53. Feed water temperature (°C): 185
54. Steam pressure (bar): 40
55. Steam temperature (°C): 289

**B4. Pressurizer**

56. Pressurizer total volume ($m^3$): 8.2
57. Steam volume (full power/zero power, $m^3$): 4.0

**B5. Main coolant pumps**

58. Number of cooling or recirculation pumps: 2
59. Type: Horizontal axial flow canned motor type
60. Pump mass flow rate (kg/s): 640
61. Pump design rated head (m): 12
62. Pump nominal power (kW): 145
63. Mechanical inertia (kg $m^2$): --

## C. CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS)

64. Number of extraction lines: 1
65. Number of pumps: 2
66. Number of injection points: 1
67. Feed and bleed connections: --

## D. CONTAINMENT

68. Type: Water filled
69. Overall form (spherical/cyl.): Cylindrical
70. Structural material: JIS G 3115 SPV50
71. Liner material: --
72. Simple/double wall: Simple
73. Dimensions (diameter, height) (m): Inner diameter 6.8m, Inner height 13m
74. Design pressure (bar): 40
75. Design temperature (°C): 200
76. Design leakage rate (% per day): --

# III. SAFETY RELATED SYSTEMS

## A. DESIGN CONDITIONS

**A1. Fission product retention**

77. Containment spray system (Y/N): N
    a. Duration (h)
    b. Flow rate ($m^3$/h)
    c. Mode of operation (active/passive)
    d. Safety graded (Y/N)
78. F.P. sparging (Y/N): N
79. Containment tightness control (Y/N): Y
80. Leakage recovery (Y/N): Y
81. Guard vessel (Y/N): N

**A2. Reactivity control**

82. Absorber injection system (Y/N): Y
    a. Absorber material: Boron
    b. Mode of operation (active/passive): Active

c Redundancy N
d Safety graded N

83 Control rods (Y/N) Y
a Maximum control rod worth (pcm) 1,440
b Mode of operation (active/passive) Active
c Redundancy Y
d Safety graded Y

**A3. Decay heat removal**

*A3-1 Primary side*

84 Water injection N
a Actuation mode (manual/automatic)
b Injection pressure level (bar)
c Flow rate (kg/s)
d Mode of operation (active/passive)
e Redundancy
f Safety graded (Y/N)

85 Water recirculation and heat removal N
a Intermediate heat sink (or heat exchanger)
b Mode of operation (active/passive)
c Redundancy
d Self sufficiency (h)
e Safety graded

*A3-2 Secondary side*

86 Feed water Y (EDRS)
a Actuation mode (manual/automatic) Automatic
b Flow rate (kg/s)
c Mode of operation (active/passive) Passive
d Redundancy Y
e Self sufficiency (h)
f Safety graded Y

87 Water recirculation and heat removal N
a Ultimate heat sink (cold source)
b Mode of operation (active/passive)
c Redundancy
d Self sufficiency (h)
e Safety graded

*A3-3 Primary pressure control*

88 Implemented system (Name) N
a Actuation mode (manual/automatic)
b Side location (primary/secondary circuit)
c Maximum depressurization rate (bar/s)
d Safety graded

## B. SEVERE ACCIDENT CONDITIONS*

**B.1 Fission products retention**

89 Containment spray system (Y/N) N
90 F P Sparging (Y/N) N
91 Containment tightness control (Y/N) Y
92 Leakage recovery (Y/N) Y
93 Risk of recriticality (Y/N) N

**B.2 Recriticality control**

94 Encountered design feature N
a Mode of operation (A/P)
b Safety graded

**B.3 Debris confining and cooling**

95 Core debris configuration (core catcher) Y

---

* All systems must be qualified to operate under the accident conditions

96. Debris cooling system (name): Containment water cooling system
    a. Mode of operation (A/P): Passive
    b. Self sufficiency
    c. Safety graded (Y/N): Y

**B.4 Long term containment heat removal**

97. Implemented system: Containment water cooling system
    a. Mode of operation (A/P): Passive
    b. Self sufficiency (h)
    c. Safety graded (Y/N): Y
98. Intermediate heat sink: N
    a. Self sufficiency (h)
    b. Safety graded (Y/N)
99. External coolant recirculation: N
    a. Implemented components
    b. Mode of operation (A/P)
    c. Self sufficiency (h)
    d. Safety graded (Y/N)
100. Ultimate heat sink: Atmosphere
    a. Self sufficiency (h)
    b. Safety graded (Y/N)

**B.5 Combustible gas control**

101. Covered range of gas mixture concentration
102. Modes for the combustible gas control
    a. Containment inertation: Y ($N_2$ gas)
    b. Gas burning
    c. Gas recombining
    d. Others

**B.6 Containment pressure control**

103. Filtered vented containment (Y/N): Under consideration
    a. Implemented system
    b. Mode of operation (A/P)
    c. Safety graded
104. Pressure suppression system (Y/N): Y
    a. Implemented system: Suppression pool type
    b. Mode of operation (A/P): Passive
    c. Safety graded (Y/N): Y

## C. SAFETY RELATED I&C SYSTEM

Automatic load following (Y/N): Y
* range (% power): 10%-100%
* maximum rate (%/min): 3%/sec

Load rejection without reactor trip (Y/N): Y
Full Cathode Ray Tubes (CRT) display (Y/N): Y
Automated start-up procedures (Y/N): Y
Automated normal shutdown procedures (Y/N): Y
Automated off normal shutdown procedures (Y/N): Y
Use of field buses and smart sensors (Y/N): Y
Expert systems or artificial intelligence advisors (Y/N): Y
Protection system backup (Y/N): Y

## D. EMERGENCY POWER SUPPLY SYSTEM

105. Type (diesel, gas, grid connection): Diesel
106. Number of trains: 2

## E. AC/DC SUPPLY SYSTEM

107. Type (rectifier, converter, battery)
108. Estimated time reserve (hr)

## IV. CONVENTIONAL THERMAL CYCLE

### A. TURBINE SYSTEM

109. Type: Steam turbine
110. Overall length (m)
111. Width (m)
112. Number of turbines/reactor: 1
113. Number of turbine sections per unit (e.g. HP/LP/LP): HP/LP
114. Speed (rpm)

### B. STEAM CHARACTERISTICS

115. H.P. inlet pressure: 38.7 bar
116. H.P. inlet temperature: 285.7 °C
117. H.P. inlet flowrate: 161 t/h
118. L.P. inlet pressure: 3.2 bar
119. L.P. inlet temperature: 136°C
120. L.P. inlet flowrate (per section): 126 t/h

### C. GENERATOR

121. Type (3-phase synchronous, DC): 3-phase synchronous
122. Apparent power (MVA)
123. Active power (MW)
124. Frequency (Hz)
125. Output voltage (kV)
126. Total generator mass (t)
127. Overall length
128. Stator housing outside diameter

### D. CONDENSER

129. Number of tubes
130. Heat transfer area
131. Flowrate ($m^3/s$)
132. Pressure (m/bar)
133. Temperature (°C)

### E. CONDENSATE PUMPS

134. Number
135. Flowrate
136. Developed head
137. Temperature
138. Pump speed

### 6.4.5. Project Status

#### *6.4.5.1. Entities involved*

The design study has been performed by the Japan Atomic Energy Research Institute (JAERI) continuously since 1983 co-operating with Ishikawajima-Harima Heavy Industries, Co. Ltd. and Mitsubishi Heavy Industries, Ltd.

#### *6.4.5.2. Design status*

The conceptual design of MRX has been already been established by JAERI design evaluation research. At present the design research is being conducted to particularize all systems related to reactor plant so that the safety, economy and reliability of MRX can be evaluated in detail with the basic design level.

#### *6.4.5.3. Research and development work*

Synthetic hydrothermal dynamics experiments

The operational and safety performance of the advanced marine reactor MRX, which is based on the features such as an integral type PWR, a water filled containment vessel and a passive decay heat removal system, depends strongly on the characteristics of the hydrothermal dynamics. JAERI has a plan to study experimentally the hydrothermal characteristics of MRX under oscillating conditions by setting a synthetic hydrothermal dynamics test facility (SHTF) on board. This facility uses electric heaters and a core characteristics simulator instead of nuclear fuel.

Development of advanced design tool

Nuclear Ship Engineering Simulation System (NESSY) is being developed as an advanced design tool by which characteristic analysis of marine reactors and nuclear ships can be easily performed in each design stage. NESSY will be particularly useful as a tool for the development of the advanced automatic control system and for research on the human interface. The calibration and validation of NESSY was made with the use of data obtained from the Nuclear ship MUTSU. NESSY is being improved to enable it to simulate advanced marine reactors by adding basic software peculiar to new concepts.

Development of built-in type control rod drive mechanism

The built-in type control rod drive mechanism (CRDM) is one of the special features of MRX. Manufacturing and performance tests of the main components such as driving motor and latch magnet are almost finished, confirming its feasibility. At present, preparation for a trial manufacture of CRDM is completed.

Fundamental tests related to passive core cooling

Passive core cooling is achieved by adopting a water filled containment vessel, emergency decay heat removal systems and containment water coolers. To evaluate the hydrothermal characteristics related to these systems, experiments are being performed by JAERI using a fundamental test facility related to passive core cooling which was manufactured for this purpose.

*6.4.5.4. Licensing status*

A licence application is planned for the prototype of MRX. The prototype reactor will be built on land, not in a ship. The hydrothermal characteristics of MRX under oscillating conditions are being examined by setting the synthetic hydrothermal dynamics test facility on board a ship, using electric heaters instead of nuclear fuel as mentioned in 6.5.3.. The licence application will be made soon after 1998.

### 6.4.6. Project Economics

A marine reactor should be compact and lightweight since it has to be installed in the narrow and limited space in a ship, and also for ship economics.

The marine reactors equipped in previously constructed nuclear ships have needed a secondary shield which is installed outside the containment vessel. Most of the weight and volume of the reactor plants are occupied by this secondary shield. MRX is a new type of marine reactor which is an integral type PWR adopting a water-filled containment vessel and a new shielding design method with no need for a secondary shield. As a result, MRX is considerably lighter in weight, more compact in size and economical as compared with the reactors equipped in previously constructed nuclear ships. For instance, the plant weight and volume of the containment vessel of MRX are 50% and 70% of those of the Nuclear Ship MUTSU, in spite of the larger power of MRX by a factor of 2.8.

## REFERENCES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## 6.5.1. Basic objectives and features

The small power reactor plant ABV is designed as a unified steam supply system for autonomous nuclear power sources intended for electricity, steam and fresh water production in little-developed regions, as well as for heating industrial plants and settlements

There is a possibility to use the ABV reactor plant for the combined production of electricity and sea water desalination

The ABV plant is designed on the basis of technically proven design decisions and technologies of marine reactors whose excellent operational properties have been confirmed by long-term successful operation of nuclear powered ships in Russia

## 6.5.2. Design description

*6.5.2 1 Nuclear Steam Supply System (see Fig 6.5.1.)*

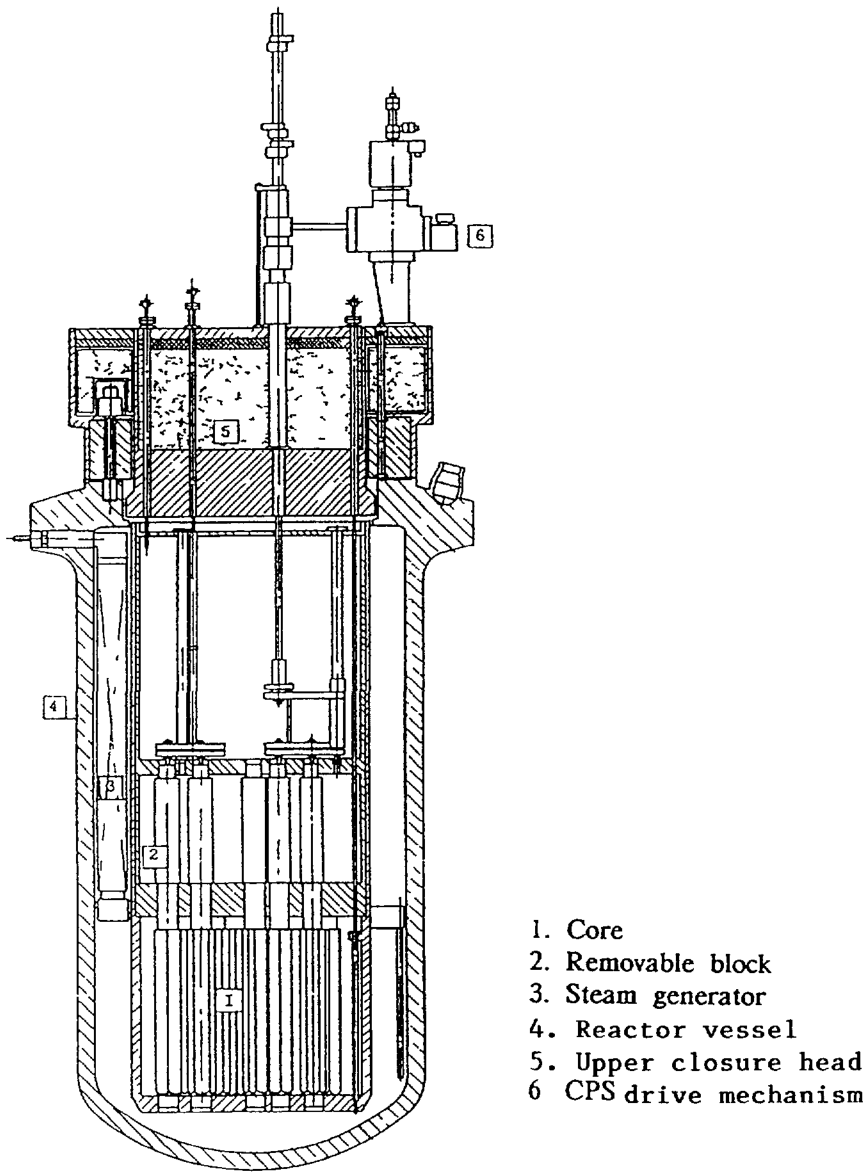



*Fig 6 5 1 ABV Reactor*

An integral PWR is used in the NSSS

Reactor vessel

The reactor vessel Fig 6 5 2 is a welded cylindrical vessel with elliptical bottom In the upper part of the vessel the following nozzles are located

1 One nozzle for the pressurizing system
2 Two nozzles for the purification and residual heat removal system
3 One nozzle for gas removal

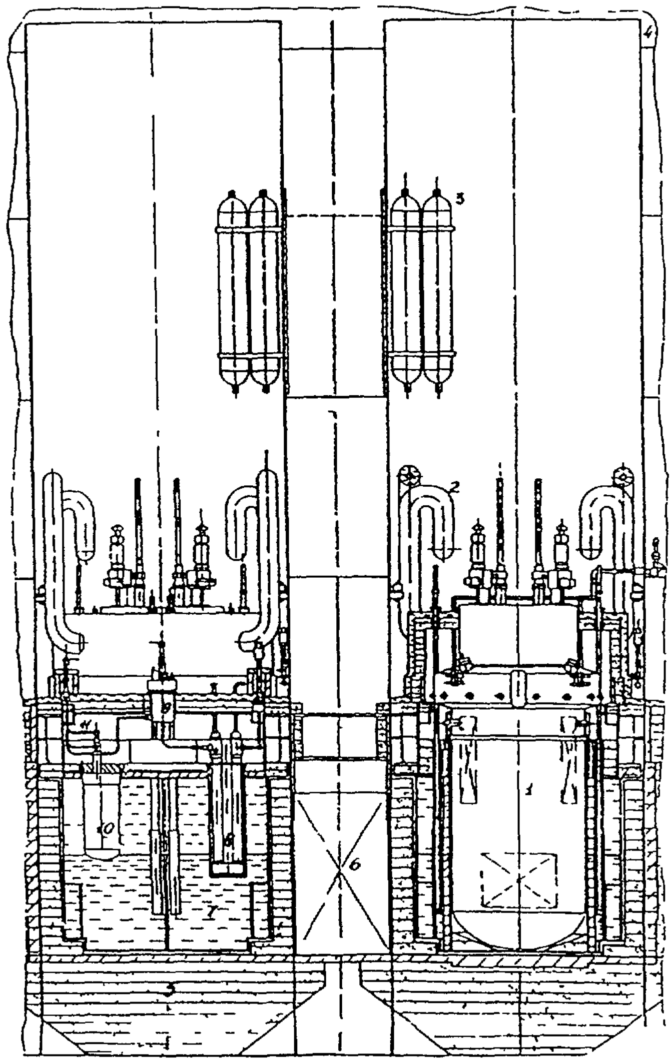


*Fig 6.5.2. Twin ABV-Reactor for floating NPP*

1 Reactor
2 Secondary circuit pipelines
3 High pressure gas flasks
4 Protective shell - containment
5 Biological shield
6 Bubbler tank
7 Metal-water shielding tank
8 Cooler of purification and decay heat removal system
9 Canned circulating pump
10 Primary circuit filter
11 Primary circuit pipelines

In addition, 4 nozzles and 4 headers for the feed water supply and steam removal are located in the flange section of the vessel.

Vessel height is 4.8 m
Vessel diameter is 2.6 m.

Core

A heterogeneous core based on thermal neutrons is used. It consists of 55 hexagonal FAs of 144.2 mm width across flats, located in the in-vessel barrel on a 147.5 mm pitch. The FAs are spaced by the lower plate of the core barrel and by an upper plate, which serves to connect with guide tubes, containing orifices into of which FA end fittings are inserted.

Fuel rods are spaced in the FAs by grids.

Primary circuit

Natural coolant circulation is used in the primary circuit. The primary circuit is a completely leak-tight system. The main circulation circuit is enclosed in the reactor vessel and includes: core-upstream flow/section, steam generator, downstream flow section. There is an external pressurizing system and the reactor coolant purification/residual heat removal system. Nitrogen is used for primary circuit pressurization.

The volume of primary coolant is 12 $m^3$.

Secondary circuit

The secondary circuit includes:

feed water and steam pipelines, valves;
instrumentation;
four built-in steam generators (SGs).

In case of a leak each of the four SG sections can be isolated on both the steam and feed water sides by double shut-off valves. The pipelines before the shut-off valves are designed for primary circuit pressure.

To prevent an isolated SG section from becoming over pressurized, automatic safety valves are used on the secondary circuit side.

*6.5.2.2. Balance of plant system*

The steam-turbine plant includes:

- turbogenerator;
- main condenser of the surface type;
- main condensate pumps;
- feed turbine pumps (3 in number) for feed water supply to SG;
- process condenser for residual heat removal, as well as for emergency make-up of feed water.

The ABV steam-turbine plant includes a specially designed turbogenerator. The distinctive feature of the turbogenerator is controlled steam extraction maintaining constant pressure of extracted steam at any power level. The extracted steam is used as the heating medium for feed water reheaters and for grid heaters heating the water up to 120°C.

*6.5.2.3. Instrumentation, control and electrical systems*

The I&C system provides:

- automatic and remote control of the plant equipment and systems at normal operation and in case of emergencies;
- protection of the plant in emergencies;
- control of the plant parameters, equipment and systems status;
- recording of information under normal operation and emergency conditions;
- information aids for the operator;
- radiation conditions monitoring.

*6.5.2.4. Safety considerations and emergency protection*

The main features providing for the ABV enhanced safety are as follows:

1. Developed inherent self-protective properties of the reactor

The plant's self-protection is provided by the application of integral reactor design, natural convection of primary coolant, by self-regulation and self-limitation of the reactor power due to negative reactivity coefficients (power, void, temperature for fuel and coolant) over the whole range of the reactor parameter variation and by the high heat capacity of the plant circuits. The increased grace period is sufficient for correct action in most cases.

2. In-depth protection and redundancy and diversity of safety systems

The reactor shut-down system includes 6 independent control rod groups (CGs) which can be inserted into the core by electric drives or under gravity in the case of de-energization of the drives. The drives are also equipped with a special mechanism for hand-operated insertion of each CG. In addition, a liquid absorber injection system is provided.

The decay heat is removed by two channels via the steam turbine plant and by one channel via the purification and residual heat removal system consisting of two passive trains. The water inventory in one storage tank of the passive system provides residual heat removal for 4 hours. After that, the reactor is maintained in a safe state for an unlimited period of time due to residual heat dissipation to the environment.

Core cooling in the case of LOCA accidents with primary circuit leaks includes a two-channel system for water make-up and injection with three feed water pumps, two standby pumps and one recirculation system.

Regarding the possible release of radioactive materials and their propagation, there is a series of independent physical barriers: fuel matrix, fuel element cladding, leak-tight integral reactor and primary circuit systems, leak-tight containment with the bubbler and pressure suppression system.

3. Application of self-actuated devices and safety systems, which do not require either external power sources or personnel intervention for their operation.

In the event of accidents with unauthorized addition of reactivity or loss of heat removal, complete failure of protective systems and operator inaction on primary circuit pressure increase, the plant will be tripped by a direct-acting pressure-actuated power breaker cutting supplies to the control rod drives.

4. Availability of engineered systems with ample time for control of severe accidents.

In the event of reactivity accidents, even in the case of postulated sticking of all CGs following on emergency protection signal, core damage is excluded by virtue of the inherent self-protective properties of the reactor.

For heat-removal accidents, with complete loss of all means for heat removal the primary circuit remains leak-tight for approximately 5 hours (reactor scrammed). For loss-of-integrity accidents, with complete failure of water injection, there is a time reserve of approx.. 3 hours. To prevent these accidents the systems described above in item 2 may be used, as well as a special water supply system to the reactor cavity thus providing for cooling the reactor vessel from outside.

Reactor Shutdown Features

1. Insertion of the absorber rods into the core, by the electromechanical drives;
2. Insertion of the absorber rods into the core under gravity on de-energization of the drives. De-energization of the drives is achieved automatically and remotely by the protective system, by the operator or due to actuation of direct-acting devices following pressure rise in the primary circuit;
3. Hand-driven insertion of the CGs;
4. Injection of the liquid absorber (boric acid solution) on multiple failures of the drives;
5. Power self-limitation of the "hot" reactor at the balance of the reactor thermal power and the emergency heat removal channels" capacity in beyond design accidents;
6. Power self-reduction during emergencies owing to the negative void reactivity effect in beyond design accidents.

Means for Residual Heat Removal

The residual heat is removed by the following channels:

1. Through SGs by supplying feed water to them and dumping heat to the condenser of the steam-turbine plant.
2. Through SGs by supplying feed water to them and dumping heat to the process condenser of the steam-turbine plant.
3. Through SGs by supplying feed water to them and dumping heat to the grid heaters (into the heating system).
4. Through the primary coolant purification system heat exchanger.
5. Through SGs following failure of the residual heat removal system.
6. Through the reactor vessel cooling system providing dissipation of heat from the reactor to the environment.

One of four SG sections or one of two sections of the reactor coolant purification system heat exchanger is sufficient to remove residual heat.

Means for Confinement of Radioactivity

1. Matrix fuel with reduced working temperature
2. Leak-tight fuel element cladding (its integrity is continuously monitored).
3. Leak-tight integral reactor and primary circuit.
4. Limitation of all primary pipelines diameter in the reactor pressure vessel penetrations (ND 25).
5. Quick-acting valves in primary circuit pipelines in combination with self-actuated localizing devices.
6. Leak-tight containment designed for excess pressure - with filters, bubbler of the emergency pressure suppression system and isolation valves in pipeline penetrations through the containment.

Insensitivity of the plant to operator errors and CPS failures is attained owing to the passive principle for reactor shutdown and residual heat removal in combination with primary coolant natural circulation, the reactor power self-regulation capability and availability of self-actuated devices. The reactor will be shutdown and brought into a safe state irrespectively of the estimation of the plant state and action by the operator.

The multi-barrier functional and physical protection of the plant, its self-protection properties and passive redundant safety systems exclude accidents with core melt and severe radiological consequences.

The estimated core damage probability does not exceed $10^{-7}$ per reactor-year.

Taking account of such accident does not introduce any noticeable contribution to the risk to the population and excludes completely the necessity for evacuation of the local population.

*6.5.2.5. Buildings and structures*

The ABV reactor plant consists of compact units, which allow the plant to be used for both land based and floating NPPs.

The ABV reactor plant containment is a steel leak-tight shell designed for the ultimate pressure and temperature arising in beyond design accidents. Protection against external impacts (e.g. air crash) is provided by a special protective enclosure compartment in a floating NPP, or by the reactor building structure in a land-based NPP.

The estimated time for floating NPP construction is approximately 3 years, while that for land based NPP is 5 years if the site is ready for building activity.

### 6.5.3. Safety concepts

TABLE 6.5.1. MAIN SAFETY RELATED SYSTEMS IN THE ABV

| Name | Safety graded | Main characteristics |
|---|---|---|
| Primary Circuit (PC) | X | Integral reactor, all primary components are leak-tight |
| Emergency water injection system (EWIS) | X | Two channels for water injection to the reactor by pumps from the water storage tank |
| Active residual heat removal system (ARHRS) | X | Two heat removal channels through a process condenser and one back-up channel through reactor coolant purification system |
| Passive residual heat removal system (PRHRS) | X | Two channels for water supply from hydroaccumulator to SG |
| Control and protection system (CPS) | X | Six control rod groups (10-16 rods in each fuel assembly), two 3-train sets in protective subsystem |
| Liquid absorber injection system (LAIS) | X | Two channels for boric acid solution injection by electric pumps from acid solution storage tank |
| Pressure suppression / rad mitigation system (PSRMS) | X | System of leak-tight rooms adjacent to containment equipped with bubbler and filters, interconnected by ducts with safety valves and membranes |
| Reactor vessel cooling system (RVCS) | X | Water supply from RVCS hydroaccumulator to the reactor cavity in metal-water shielding tank |
| Containment | X | Prefabricated leak-tight steel shell |

TABLE 6.5.2. MAIN ACCIDENT INITIATORS FOR THE ABV

- LOCA (primary Loss of Primary Coolant Accident
- LOCA (Secondary) Secondary Pipe Rupture (water or steam)
- LOCA (Interfacing) e g SGTR Steam Generator Tube Rupture
- ATWS Anticipated Transients Without Scram,
- Primary Transients,
- Secondary Transients (turbine trip),
- Loss of electric sources (all AC sources),
- Total loss of the cold sources,
- Total loss of the steam generator feedwater,
- Station blackout

TABLE 6.5.3. DESIGN FEATURES FOR PREVENTION AND PROTECTION LEVELS THAT REDUCE (R), SUPPRESS (S), THE INITIATORS FREQUENCY OR LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES (L)

| |
|---|
| **PREVENTION LEVEL** |
| LOCA (Primary) |
| - Fluence decrease R |
| - Leak before break S |
| - Decrease of number and extent of primary pipelines R |
| - Decrease of primary pipeline diameters R,S |
| - Coolant pressure control using passive means S |
| - Decrease of thermocyclic stresses in primary pipelines by design modification R |
| LOCA (Secondary) |
| - A part of secondary pipelines is designed for primary circuit pressure R |
| - System for secondary pipeline over pressure prevention R |
| LOCA (Interfacing) |
| - Corrosion-resistant alloys for steam generator R |
| - Decreased temperatures of equipment R |
| Primary transient - Increased design margin reduced initiator frequency R |
| Secondary transients - Increased design margin reduced initiator frequency R |
| Total loss of the source (Water) |
| - Passive ultimate decay heat removal (air) suppress initiator S |
| Total loss of the SG feedwater |
| - Redundancy of feed water supply means R |
| Station blackout Availability of auxiliary power sources R |
| **PROTECTION LEVEL** |
| LOCA (Primary) |
| - Leak limitation by passive flow restrictors L |
| - Increased water inventory in reactor vessel L |
| - Multichannel system for emergency water injection L |
| LOCA (Secondary) |
| - Possibility and simplicity of SG leaky section isolation L |
| LOCA (Interfacing) |
| - Possibility and simplicity of SG leaky section isolation L |
| ATWS Increased values of negative reactivity coefficients, passive ERHR systems L |
| Primary transients Greater primary inertia L |
| Secondary transient Multichannel shutdown and RHR systems L |
| Loss of electric sources - Implementation of passive systems (battery power sources) L |
| Total loss of heat sink Passive heat removal to ambient air, non-critical L |
| Total loss of S G feedwater non-critical - Passive residual heat removal L |
| Station Blackout non-critical - Passive shutdown and RHR systems L |

TABLE 6.5.4. DESIGN FEATURES FOR MITIGATION LEVEL OF ABV

| Safety Functions | Systems (Cf Tab 6 5 1 ) | Passive/Active | Design features/Remarks |
|---|---|---|---|
| **Design Basis**<br>Fission Product Containment | Primary Circuit<br>PSRMS<br>Containment | Passive<br>Passive<br>Passive | <br>Self actuates at containment pressure of 0 4 MPa<br>Bubbler volume is 25 $m^3$ |
| Coolant inventory | Emergency water injection system | Active | |
| Decay Heat Removal | Passive heat removal system, PRHRS<br>ARHRS<br>RVCS | Passive<br>Active<br>Passive | Self-actuates at $P_{sec}$ = 0 6 MPa, 2x2 5 $m^3$<br>Water storage tank of 40 $m^3$ capacity<br>P = 12 MPa hydroaccumulator |
| Reactivity control | Control rods, CPS<br>Liquid absorber injection system | Active/Passive<br>Active | |
| Primary Circuit Pressure Control | Active residual heat removal system ARHRS<br>Passive heat removal system, PRHRS | Active<br>Passive | <br>Primary pressure self-actuated system |
| **Severe Accident**<br>Containment temperature and pressure control | PSRMS<br>RVCS | Passive<br>Passive | <br>Operator-actuated system |
| Heat Removal | EWIS/ARHRS/PRHRS<br>Reactor vessel cooling system | Active/Passive<br>Passive | Final heat removal is completely passive |
| Tightness control | Containment<br>PSRMS | Passive<br>Passive | |
| Inflam gas control | Igniters | Passive | |
| Fission product containment | PSRMS/Containment | Passive | |
| Corium management<br><br>Others | Reactor vessel cooling system | Passive | Need of manual actuation |

**6.5.4. Design Data Questionnaire (Water-Cooled Reactors for ABV)**

## I. GENERAL INFORMATION

1. Design name - ABV
2. Designer/Supplier address - OKBM
3. Reactor type - Integral PWR
   Number of modules/per plant - 2 (or more)
4. Gross thermal power (MW-th) per reactor - 38
5. Net electrical output (MW-e) per reactor - 6
6. Heat supply capacity (MW-th) - Up to 12 G cal/hr

## II. BASIC DESIGN DESCRIPTION

### A. CORE AND REACTIVITY CONTROL

7. Fuel material - Uranium-aluminium alloy
8. Fuel inventory (tones of heavy metal) - 81.7 Kg, U235
9. Average core power density (kW/liter)- 43
10. Average fuel power density (kW/kgU) - 80
11. Maximum linear power (W/m)
12. Average discharge burnup (MWd/t) - 72,5
13. Initial enrichment or enrichment range (Wt%) - 17
14. Reload enrichment at the equilibrium (Wt%) - 11
15. Refuelling frequency (months) - 48-60
16. Type of refuelling (on/off power) - Off power
17. Fraction of core withdrawn (%) - 100
18. Moderator material and inventory - Water, 8 $m^3$
19. Active core height (m) - 0.85
20. Core diameter (m) - 1.225
21. Number of fuel assemblies - 55
22. Number of fuel rods per assembly - 99
23. Rod array in assembly - Triangular
24. Clad material - Zirconium alloy
25. Clad thickness (mm) - 2.1
26. Number of control rods or assemblies - 6 groups
27. Type - Cluster
28. Additional shutdown systems - Boric acid
29. Control rod neutron absorber material - $B_4C$
30. Soluble neutron absorber - Boric acid
31. Burnable poison material and form - Rods on gadolinium basis

### B. REACTOR COOLANT SYSTEM

#### B1. Coolant

32. Coolant medium and inventory - Water
33. Design coolant mass flow through core (kg/s) - 85
34. Cooling mode (forced/natural) - Natural circulation
35. Operating coolant pressure (bar)- 15.4 MPa
36. Core inlet temperature (°C) - 245
37. Core outlet temperature (°C) - 327

#### B2. Reactor pressure vessel/tube

38. Overall length of assembled vessel/tube (m) - 4.8
39. Inside vessel/diameter (m/mm) - 2.6
40. Average vessel/tube thickness (mm) - 103.5
41. Vessel/tube material - Heat resistant steel
42. Lining material - Austenitic steel
43. Design pressure (bar) - 17.6 MPa
44. Gross weight (tone/kg) - 86 t

#### B3. Steam generator

45. Number of steam generators - 4
46. Type - Once-through

47. Configuration (horizontal/vertical) - Vertical
48. Tube material - Titanium alloy
49. Shell material
50. Heat transfer surface per steam generator ($m^2$) - 101.25
51. Thermal capacity per steam generator (MW) - 9.5
52. Feed water pressure (bar) - 3.87 MPa
53. Feed water temperature (°C) - 106
54. Steam pressure (bar) - 3.14 MPa
55. Steam temperature (°C) - 290

**B4. Pressurizer** (gas type)

56. Pressurizer total volume ($m^3$) - 1.5
57. Steam volume (full power/zero power, $m^3$) - 0.3/1.2 ($N_2$)

**B5. Main coolant pumps: No**

58. Number of cooling or recirculation pumps
59. Type
60. Pump mass flow rate (kg/s)
61. Pump design rated head
62. Pump nominal power (kW)
63. Mechanical inertia (kg $m^2$)

## C. CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS)

64. Number of extraction lines - 1
65. Number of pumps - 1
66. Number of injection points - 1
67. Feed and bleed connections

## D. CONTAINMENT

68. Type - Welded steel
69. Overall form (spherical/cyl.) - Rectangular
70. Structural material - Steel
71. Liner material
72. Simple/double wall - Simple
73. Dimensions (diameter, height) (m) - 5.1 x 4 x 14
    (length x width x height)
74. Design pressure (bar) - 0.8 MPa
75. Design temperature (°C) - 150
76. Design leakage rate (% per day) - 1%

# III. SAFETY RELATED SYSTEMS

## A. DESIGN CONDITIONS

**A1. Fission product retention**

77. Containment spray system (Y/N) - No
    a. Duration (h)
    b. Flow rate ($m^3$/h)
    c. Mode of operation (active/passive)
    d. Safety graded (Y/N)
78. F.P. sparging (Y/N) - Yes
79. Containment tightness control (Y/N) - Yes
80. Leakage recovery (Y/N) - Yes
81. Guard vessel (Y/N) - No

**A2. Reactivity control**

82. Absorber injection system (Y/N) - Yes
    a. Absorber material - $H_3BO_3$ Solution
    b. Mode of operation (active/passive) - Active
    c. Redundancy - Two channels
    d. Safety graded - Yes
83. Control rods (Y/N) - Yes
    a. Maximum control rod worth (pcm)
    b. Mode of operation (active/passive) - Active/Passive
    c. Redundancy - Cold shut down is provided without one most effective control group

d Safety graded - Yes

**A3. Decay heat removal**

*A3-1 Prımary Sıde*

84 Water injection

a Actuation mode (manual/automatic) - Automatic

b Injection pressure level (bar) - 17 5

c Flow rate (kg/s) - 0 67

d Mode of operation (active/passive) - Active

e Redundancy - Two channels, 3 pumps

f Safety graded (Y/N) - Yes

85 Water recirculation and heat removal

a Intermediate heat sink (or heat exchanger) - Bubbler

b Mode of operation (active/passive) - Active

c Redundancy - Two channels

d Self sufficiency (h) - unlimited

e Safety graded - Yes

*A3-2 Secondary sıde*

86 Feed water (Active/Passive)

a Actuation mode (manual/automatic) - Automatic

b Flow rate (kg/s) - 0 42/variable

c Mode of operation (active/passive) - Active/Passive

d Redundancy - Two channels

e Self sufficiency (h) - 26/4

f Safety graded - Yes

87 Water recirculation and heat removal

a Ultimate heat sink (cold source) - Atmosphere

b Mode of operation (active/passive) - Passive

c Redundancy

d Self sufficiency (h) - unlimited

e Safety graded - Yes

*A3-3 Prımary pressure control*

88 Implemented system (Name) - ARHRS, PRHRS

a Actuation mode (manual/automatic) - manual/automatic

b Side location (primary/secondary circuit) - secondary circ

c Maximum depressurization rate (bar/s)

d Safety graded - Yes

## B. SEVERE ACCIDENT CONDITIONS*

**B.1 Fission products retention**

89 Containment spray system (Y/N) - No

90 F P Sparging (Y/N) - No

91 Containment tightness control (Y/N) - Yes

92 Leakage recovery (Y/N) - Yes

93 Risk of recriticality (Y/N) - No

**B.2 Recriticality control**

94 Encountered design feature

a Mode of operation (A/P)

b Safety graded

**B.3 Debris confining and cooling**

95 Core debris configuration (core catcher) - Localization in reactor vessel

96 Debris cooling system (name) - Reactor vessel cooling system

a Mode of operation (A/P) - Passive, operator-activated

b Self sufficiency - Unlimited

c Safety graded (Y/N) - Yes

**B.4 Long term containment heat removal**

97 Implemented system - Natural heat dissipation

a Mode of operation (A/P) - Passive

* All systems must be qualıfied to operate under the accıdent condıtıons

b. Self sufficiency (h) - Unlimited
c. Safety graded (Y/N) - Yes

98. Intermediate heat sink - Air in containment, shielding water tank, bubbler
a. Self sufficiency (h) - unlimited
b. Safety graded (Y/N) - Yes

99. External coolant recirculation - No
a. Implemented components
b. Mode of operation (A/P)
c. Self sufficiency (h)
d. Safety graded (Y/N)

100. Ultimate heat sink - Atmosphere
a. Self sufficiency (h) - Unlimited
b. Safety graded (Y/N) - No

**B.5 Combustible gas control**

101. Covered range of gas mixture concentration
102. Modes for the combustible gas control
a. Containment inertation - Yes ($N_2$)
b. Gas burning - Yes
c. Gas recombining
d. Others

**B.6 Containment pressure control**

103. Filtered vented containment (Y/N) - Yes
a. Implemented system - PSRMS
b. Mode of operation (A/P) - Passive
c. Safety graded - Yes

104. Pressure suppression system (Y/N) - Yes
a. Implemented system - PSRMS
b. Mode of operation - Passive
c. Safety graded (Y/N) - Yes

## C. SAFETY RELATED I&C SYSTEM

Automatic load following (Y/N) - Yes
* range (% power) - 20-100
* maximum rate (%/min) - 30

Load rejection without reactor trip (Y/N) - Yes
Full Cathode Ray Tubes (CRT) display (Y/N) - Yes
Automated start-up procedures (Y/N) - Yes
Automated normal shutdown procedures (Y/N) - Yes
Automated off normal shutdown procedures (Y/N) - Yes
Use of field buses and smart sensors (Y/N)
Expert systems or artificial intelligence advisors (Y/N) - Yes
Protection system backup (Y/N) - Yes

## D. EMERGENCY POWER SUPPLY SYSTEM

105. Type (diesel, gas, grid connection) - Diesel
106. Number of trains - 4

## E. AC/DC SUPPLY SYSTEM

107. Type (rectifier, converter, battery) - Converter, battery
108. Estimated time reserve (hr) - 1 month for DG

# IV. CONVENTIONAL THERMAL CYCLE

## A. TURBINE SYSTEM

109. Type - With regulated steam extraction for heating
110. Overall length (m)
111. Width (m)
112. Number of turbines/reactor - 1
113. Number of turbine sections per unit (e.g. HP/LP/LP) - 1

114. Speed (rpm) - 3000

## B. STEAM CHARACTERISTICS

115. H.P. inlet pressure - 3 MPa
116. H.P. inlet temperature - 285°C
117. H.P. inlet flowrate - 13 Kg/S
118. L.P. inlet pressure
119. L.P. inlet temperature
120. L.P. inlet flowrate (per section)

## C. GENERATOR

121. Type (3 phase synchronous, DC) - Asynchronous three phase
122. Apparent power (MVA)
123. Active power (MW) - 6 MW
124. Frequency (hz) - 50
125. Output voltage (kV) - 6.3
126. Total generator mass (t)
127. Overall length
128. Stator housing outside diameter

## D. CONDENSER

129. Number of tubes
130. Heat transfer area
131. Flowrate ($m^3/s$) - Up to 14.7 KG/S
132. Pressure (m/bar) - 0.1 bar
133. Temperature (°C) - 160

## E. CONDENSATE PUMPS

134. Number - 2
135. Flowrate - 22.6 kG/S
136. Developed head - 75 m of water column
137. Temperature
138. Pump speed

### 6.5.5. Project status and licensing

Project status (D-2).

The basic design has been developed for the floating NPP and detailed design is under development now. The basic design is also developed for the NSSS of land-based NPP.

The main R&D work has been performed. The safety assessment report and quality assurance programme have been developed.

An estimation of the project economics has been performed and competitiveness of the project on the market investigated (D=2).

Licensing

The design is being licensed at present.

### 6.5.6. Project economics

Comparison of economic indicators for ABV NPP with that for fossil fuelled plants has proved the competitiveness of the NPPs for regions with expensive imported fuel. In addition , NPPs with an ABV reactor have indisputable advantages from the environmental point of view.

Evaluation of the construction cost of a floating NPP with the ABV reactor gives approximately US$85 million for a pilot plant.

## 6.6. GAS TURBINE MODULAR HELIUM REACTOR (GT-MHR) POWER PLANT

### 6.6.1. Basic objectives and features

The Gas Turbine - Modular Helium Reactor (GT-MHR) is a power unit for the production of electricity or the co-generation of electricity and process heat. The GT-MHR power unit couples the passively safe Modular Helium Reactor (MHR) with a highly efficient energy conversion system. The energy conversion system is based on an intercooled and recuperated closed Brayton (gas turbine) cycle that has a projected efficiency of approximately 48%. The GT-MHR produces electricity directly with the reactor primary helium coolant driving a gas turbine generator derived from technologies developed in the gas turbine and aerospace industries.

Each GT-MHR power unit has a design rating of 600 MWt/286 MWe. (The rating of initial units is planned to be limited to 550 MWt/262 MWe to provide design margin). One or more standardized power units are used to form plants with ratings up to 1150 MWe while maintaining the passive safety features of the MHR.

- Coated Particle Fuel - The multiple ceramic coatings surrounding the fuel kernels constitute tiny independent pressure vessels which retain fission products. These coatings are capable of maintaining their integrity and fission product retention at temperatures much higher than those imposed during postulated extreme accident conditions.

- Helium Coolant - The inert and single phase helium coolant has several advantages: no flashing or boiling of coolant is possible, pressure measurements are certain, and pump cavitation cannot occur. Further, there are no reactivity or corrosive effects associated with helium and no potential chemical or energetic reactions between coolant and fuel is possible.

- Graphite Core - The strength of the graphite core at high temperatures results in a wide margin between operating temperatures and temperatures that would result in core damage. Further, the high heat capacity and low power density of the core result in very slow and predictable temperature transients.

- Core Configuration - Selection of an annular core geometry, low core power density, and core power level assures that fuel temperatures remain hundreds of degrees below the integrity limit of the coated particle fuel even if all active coolant circulation fails and even if the coolant were lost.

- Passive Decay Heat Removal System - In addition to the normally operating power conversion system and an independently powered shutdown cooling system, a completely passive, safety-grade reactor cavity cooling system is provided.

The selection of helium as the coolant, graphite for the core structure, and ceramic fuel sets the MHR apart from other power reactors and is the cornerstone of its high temperature capability. This unique heat source enables high power conversion efficiency and a range of energy conversion alternatives.

### 6.6.2. Design description

#### *6.6.2.1. Nuclear energy supply system*

6.6.2.1.1. Main construction and design features

A GT-MHR plant consists of one or more identical GT-MHR power units where each unit consists of a GT-MHR power module (Figure 6.6.1.) plus its support systems. Each power module is housed in a below grade reinforced concrete structure which serves as an independent, vented, low pressure confinement.

The GT-MHR power module components are contained within three steel pressure vessels: a reactor vessel, a power conversion vessel and a connecting concentric cross vessel. The reactor vessel contains the annular reactor core, control rod drives, refuelling access penetrations, and a shutdown cooling system. The power conversion vessel, contains the entire power conversion system which is comprised of a turbo-machine and three heat exchangers. The turbo-machine consists of a generator, turbine and two compressor sections mounted on a single shaft suspended by magnetic bearings. The heat exchangers consist of a plate-fin recuperator, a helical flow precooler, and a helical flow intercooler.

Pressurized helium enters the core from a plenum located above the core at a temperature of about 490°C. The coolant flows downward through cooling channels and exits into an outlet plenum at approximately 850°C. The heated helium is directed through the inner part of the concentric cross vessel to the power conversion vessel where it enters the turbine. Upon expanding through the turbine, helium at 510°C flows through the hot side of the recuperator, configured of six parallel units. Heat is transferred to the helium on the cold recuperator side on the return path to the reactor core. This heat is recovered and not lost from the cycle. The helium is cooled in the precooler and directed to low- and high-pressure compressor sections where the pressure is raised from 2.6 MPa to 7.2 MPa. Precooling reduces the work required to compress the helium by the compressor. An intercooler is located between the low- and high pressure compressor section to further reduce the required compressor work. From the compressor section, the helium enters the cold, high pressure side of the recuperator where it recovers thermal energy before it is directed through the outer portion of the cross vessel back to the reactor vessel.

The safety grade reactor cavity cooling system (RCCS) is located in the concrete structure external to the reactor vessel to remove heat radiated from the vessel during normal operation as well as all events when forced core cooling is unavailable. The RCCS naturally convects outside air down through enclosed ducts and panels that surround the below-grade core cavity before returning the warmed air through above-grade outflow structures. The core heat is transferred by conduction, convection, and radiation to the RCCS. This system is completely passive and has no controls, valves, circulating fans, or other active components.

6.6.2.1.2. Fuel element and reactor core

The ceramic fuel of the GT-MHR is the key to the unique capability for the high passive safety level of this reactor concept. The multiple ceramic coatings of pyrolytic carbon and silicon carbide surrounding the fuel kernels constitute tiny independent pressure vessels which retain fission products at their source, in the centre of the fuel particle. These coatings are capable of maintaining their integrity and fission product retention capability at temperatures very much higher than those imposed during extreme accident conditions.

The active region of the core consists of fuel blocks arranged in a ring. The centre and outer portions of the core consist of unfuelled, solid, removable reflector blocks. The core assembly is surrounded by a steel core barrel and contained inside an uninsulated reactor vessel.

The MHR fuel consists of two types of coated particles: fissile particles containing 19.8% enriched uranium oxycarbide kernels and fertile particles containing natural uranium oxycarbide kernels. The concept also has the capability to use highly enriched uranium or plutonium from retired nuclear weapons as fuel. The fuel particles are bonded together to form fuel rods which are inserted into blind vertical holes in the graphite fuel blocks. Helium flowing through adjacent vertical coolant holes in the fuel blocks removes the fission heat. The fuel cycle is a one-through, 3-year cycle with one-half of the core refuelled every 18 months.

The core reactivity is controlled by control rods in the core and reflectors. A completely independent and redundant reserve shutdown system provides a diverse reactivity control capability using boron pellets stored in hoppers above special channels in the core. The inherent features that control reactivity and thus heat generation, include a strong negative temperature coefficient, and the single phase, neutronically inert coolant.

The innovative approach taken in the design of the MHR is to utilize the high temperature capability of the coated fuel particles to meet the US-10CFR50 Appendix I and the US-10CFR100 dose limits with large margin and to rely on additional, largely passive retention barriers such as the vented low pressure confinement to meet the more restrictive PAG doses for individuals located at the site boundary for any licensing basis event.

6.6.2.1.3. Reactor auxiliary and ancillary systems

Key reactor auxiliary and ancillary systems required for the GT-MHR, in addition to those described above, are as follows:

- Helium purification system
- Shutdown cooling system
- Fuel handling equipment and spent fuel storage

The helium purification system purifies the helium primary coolant, removing fission products released from defective fuel particles, tritium and chemical impurities ($H_2O$, CO, $CO_2$, $N_2$, $CH_4$). Additionally, the helium purification system provided for helium supply (to the power module) and for helium pressure control. Two-fold redundant helium purification systems are provided for each power module.

The shutdown cooling system removes decay heat during power module shutdown periods. An independent shutdown system is provided for each power module and consists of a helium-to-water heat exchanger, a helium circulator, a cooling water cooling loop and the associated controls. The heat exchanger and circulator are located in the bottom of the reactor vessel. The shutdown system rejects core decay heat to the atmosphere by means of air-blast exchangers.

The fuel handling equipment is used to carry out the periodic, remote replacement of core fuel and reflector elements. Water cooled spent fuel storage pools are provided for spent fuel elements until their heat generation rate decreases sufficiently to allow for transfer to spent fuel casks for on-site storage or shipment off-site.

*6.6.2.2. Balance of plant systems*

The GT-MHR uses a direct Brayton (gas turbine) cycle for the generation of electricity. As a result, the complex and expensive balance of plant systems required by steam cycle plants are not needed for the GT-MHR.

The GT-MHR gas turbine uses the same technology as the modern jet engine. However, in the case of the GT-MHR, its design requirements are less demanding. Temperatures, stresses and blade tip speeds are all far below those proven in millions of hours of aircraft engine operation. A large helium gas turbine has been built and very successfully tested in Germany as part of the HHV test loop. Plate-fin recuperators are highly efficient and compact heat exchangers. The GT-MHR recuperators draw on extensive experience from the fossil-fuel industry, including the construction of 2.5 million units, sixty of which have been for large gas turbine plants.

A summary of the Turbo-Machine/Generator design parameters are listed below:

**Summary of plant conceptual design parameters**

| | |
|---|---|
| **Reactor** | |
| Power | 600 MW(t) |
| Power Density | 6.6 W/cc |
| Core Inlet/Outlet Temperatures | 490°C/850°C |
| **Turbo-machine Thermodynamic Parameters** | |
| Working Fluid | Helium |
| Compressor Overall Pressure Ratio | 2.82 |
| Turbine Overall Pressure Ratio | 2.65 |
| Compressor Polytropic Efficiency | 0.90 |
| Turbine Polytropic Efficiency | 0.92 |
| Helium Mass Flow Rate | 320 kg/sec |
| Turbine Inlet Temperature | 850°C |
| Compressor Inlet Temperature | 33°C |
| Turbine Inlet Pressure | 7.02 MPa |

| | |
|---|---|
| **Turbo-machine Design Parameters** | |
| Net Shaft Power | 298 MW |
| Rotational Speed | 3600 RPM |
| LP Compressor Stages | 14 |
| HP Compressor Stages | 19 |
| Turbine Tip Speed | 322 m/sec |
| Bearings | Active magnetic, 3 radial, 1 thrust |
| **Generator Design Parameters** | |
| Active Rotor Volume Power Density | 100 KW/L |
| Power Output at Generator | 298 MW(e) |
| Net Module Electrical Output | 286 MW(e) |
| Plant Net Efficiency | 47 6% |
| **Recuperator** | 6 compact plate-fin modules |
| **Precooler and Intercooler** | Finned tube helical bundle |

*6.6.2.3. Instrumentation, Control and Electrical Systems*

The large thermal capacity of the nuclear core, and a strong negative temperature coefficient of reactivity provide the GT-MHR a unique ability to follow load with minimal thermal variations Abundant thermal energy stored in the high-thermal-capacity core is always available to meet fast changes in load demand with minimum changes in core temperatures. The GT-MHR instrumentation and control systems are designed to take advantage of these characteristics.

Output power is varied by adjusting the mass flow of the primary coolant in the core-turbine path. Lowering this flow causes a reduction in electric output, whereas increasing it causes the reverse effect Not only do core temperature variations tend to be of a small amplitude, but they take place at very slow rates, which allows the control rods to make early adjustments to reactor power and further reduce the amplitude of core thermal changes.

The plant control systems are non-safety-related. However, for high reliability, three channels are used in all control loops that are directly involved in the generation of power. The median of the three is used for control, and the other two are used to identify channel failures with no interruptions in the power generation process Two channels are used to control auxiliary systems that are needed frequently Single channels are used for auxiliary systems that see infrequent use

All control systems are implemented on industry proven distributed computer control platforms, linked together via redundant data highways A dedicated work station control console is provided for each power module in a central control room Each power module control console is manned by a dedicated operator A separate supervisor's console is provided for monitoring overall plant operations

From the safety standpoint, the thermal capacity and strong negative temperature coefficient of reactivity also work to passively mitigate reactivity and loss of coolant accidents. Nevertheless, a safety-related reactor trip and safety features monitoring systems are included

The safety-related reactor trip is implemented using dedicated microprocessors on a two out of four configuration. The safety features monitoring systems utilize dedicated redundant data highways operating strictly in a prescheduled (deterministic mode).

The simplicity of the GT-MHR is reflected in its instrumentation systems: Total visibility of plant conditions, and all control and protection actions are provided with a relatively low number of instrumentation channels. Clear and concise operator information is provided. The passive safety of the reactor, its slow response, and the neutronic transparency and the absence of phase changes in the gas coolant eliminate many of the human factors complications found in other reactors.

*6.6.2.4. Safety Considerations and Emergency Protection*

1. General

The combination of the passively safe MHR and the gas turbine power conversion system represents the ultimate in simplicity and safety. The reactor coolant directly drives the turbine which turns the generator. This allows costly and failure prone steam generating equipment to be eliminated. Advantages include:

- no steam generator leaks with water/steam ingress into primary circuit (leaks in the low-pressure and low-temperature precooler and intercooler are very improbable)
- no corrosion-caused leaks
- no corrosion-caused reduction in operating life
- no stress corrosion-caused structural failures

2. Environmental protection; dose rates for operators; non-proliferation

- MHR thermal discharge to the environment is low, due to the system's high efficiency
- The GT-MHR is free of the emissions associated with burning fossil fuels
- Radioactive emissions from helium-cooled reactor plants are lower than those from comparably sized coal-fired plants
- Worker radiation doses are only a fraction of those from today's nuclear power plants
- The MHR spent fuel characteristics result in substantially reduced proliferation risks.

3. Levels of Safety

a. Definition

Level 0:

No hazardous materials or confined energy sources

Level 1:

No need for active systems in event of subsystem failure.
Immune to major structural failure or operator error.

Level 2:

No need for active systems in event of subsystem failure.
No immunity to major structural failure or operator error.

Level 3:

Positive response required to subsystem malfunction or operator error.
Defense in depth. No immunity to major structural failure.

b. GT-MHR

- The MHR is the only reactor that meets the criterion of Level 1 safety. Its design is derived from natural properties of materials and optimum choice of reactor size, geometry and power density. It can withstand the total loss of coolant without the possibility of a meltdown.

- The GT-MHR has a large negative temperature co-efficient

- The Modular Helium Reactor's decay heat will not cause a meltdown even if the coolant is lost. The reactor's low power density and geometry assure that decay heat will be dissipated passively by conduction and radiation without ever reaching a temperature that can threaten the integrity of the ceramically-coated fuel particles, even under the most severe accident conditions.

*6.6.2.5. Buildings and structures*

6.6.2.5.1. Buildings and arrangement

The plant building arrangement is based on housing each reactor module in a below grade Reactor Building (RB), Figure 6.6.1. Each RB consists of a multicell, reinforced concrete structure. While the passive safety characteristics of the GT-MHR make a pressure retaining, leak tight containment unnecessary, the RB does serve as a vented low pressure containment that provides an additional barrier to limit the release of small leaks of radioactivity to the atmosphere during normal operation.

The RB lower portion, from elevation -39 m to elevation -9.4 m, is configured as a cylindrical silo 24.9 m in diameter, measured at the silo shell midwall. At elevation -9.4, the shape of the RB changes to a rectangular structure. The cylindrical portion of the below grade cavity is subdivided into a number of vertical cells which house the reactor module equipment and serve as general access ways. The rectangular portion of the building is divided into several compartments. A portion of this area is occupied by the RCCS ducting and the cavity vent path. Other spaces house modularized protection and instrumentation system equipment and other nuclear auxiliaries dedicated to each reactor module.

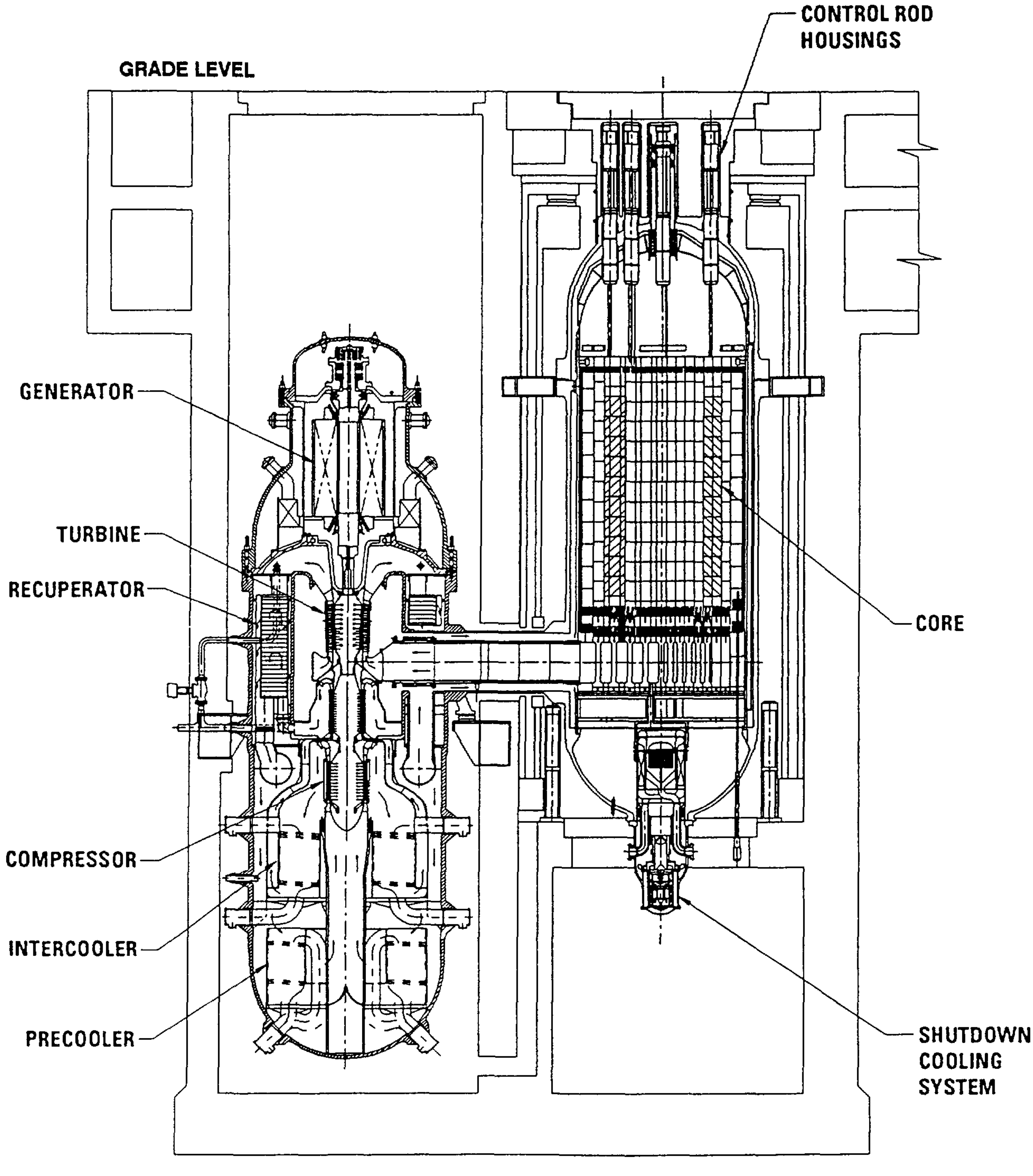



*Fig 6.6.1. GT-MHR Power Module Arrangement*

In the reference GT-MHR plant design, four RBs and a Reactor Service Building are set in a row and are interconnected by a continuous operating floor at grade level. The entire operating floor is covered with a maintenance enclosure. A Personnel Services Building, and Operations Centre, a Radioactive Waste Management Building, and a Cask Washdown Bay are located adjacent to the Reactor Service Building as shown in Figure 6.6.2. The only other major plant buildings are a nuclear Island Warehouse and a Turbo Machinery Maintenance Facility.

6 6.2 5.2. Ventilation, access

A Heating, Ventilation, and Air Conditioning (HVAC) system supplies conditioned outside air to the plant buildings. The air flows from clean spaces to potentially contaminated spaces before being exhausted. Air exhausted from potentially radioactive spaces is continuously monitored, and if necessary, filtered through HEPA filters to remove airborne particles

During reactor operations, the ventilation ducts serving the reactor and power conversion system cavities in the Reactor Buildings are sealed by cavity isolation valves. To prevent condensation on the RCCS panels, a small amount of air is purged from the cavities, filtered and monitored prior to discharge to the atmosphere. This purge air flows through bypass pipes, with filters and flow control valves, that connect to the exhaust duct upstream of exhaust fans Before personnel enter a reactor or power conversion system cavity during shutdown, the isolation valves are opened and the cavity is purged with filtered, tempered outdoor air which is filtered and monitored before release to the atmosphere

The vent paths from the reactor and power conversion system cavities, as well as the RCCS ducts, follow tortuous routes to limit neutron streaming from the Reactor Buildings The vent paths are fitted with dampers that open on overpressure and then reshut. The dampers are protected from external hazards and discharge through louvred openings located greater than 6 m above grade.

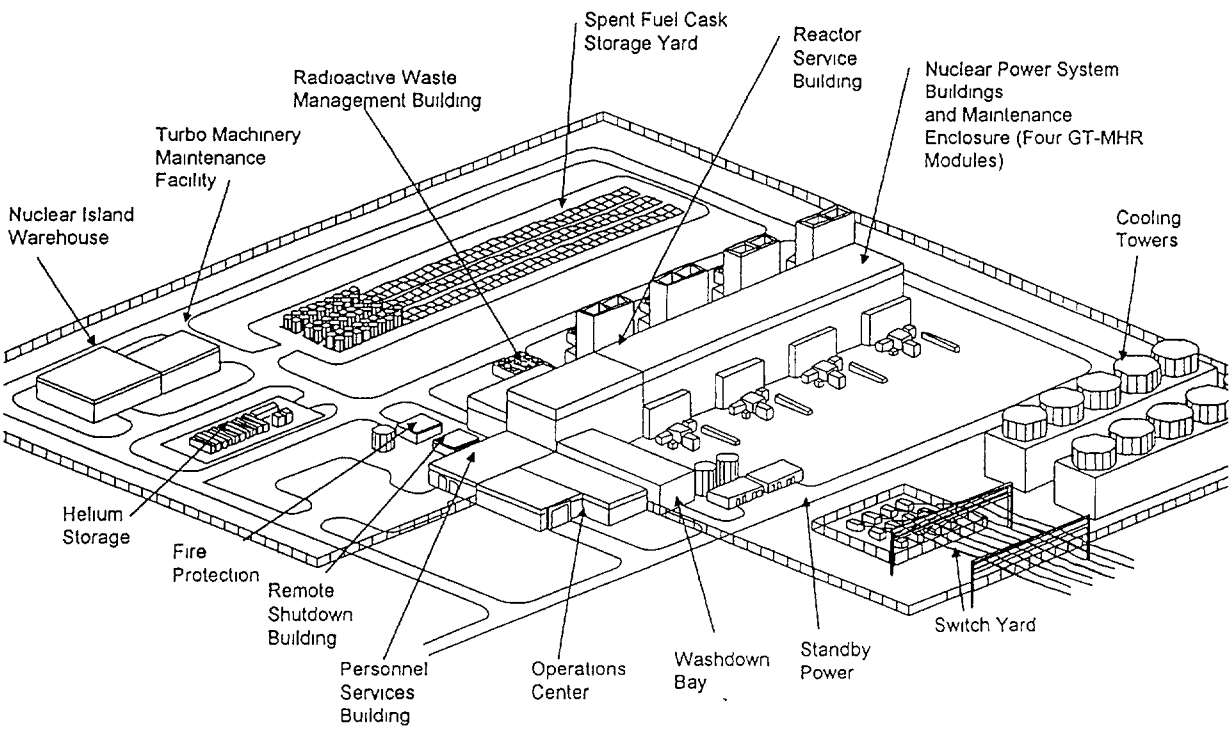



*Fig 6.6.2. GT MHR Plant Layout*

### 6.6.3. Safety concepts

TABLE 6.6.1. MAIN SAFETY RELATED SYSTEMS IN THE GT-MHR

| Safety function: name of system | Safety graded | Main characteristics |
|---|---|---|
| Fission product retention: | | |
| • fuel element | X | C-SiC-C-coated fuel particles, maintaining their integrity and fission product retention for long times at temperatures higher than those imposed during postulated extreme accidents |
| • pressure vessel unit | X | burst-proof; diameter of connecting pipes limited; isolation valves provided |
| Decay heat removal: | | |
| • reactor cavity cooling system | X | passive; rejects decay heat to the environment by radiation conduction and convection. |
| Reactivity Control: | | |
| • control rods into core/reflector | X | failsafe initiation; gravity-driven |
| • reserve shutdown pellets into reflector/core | X | gravity-driven |

TABLE 6.6.2. MAIN ACCIDENT INITIATORS FOR THE GT-MHR

| |
|---|
| - Depressurization (Primary): Core Heat-up, air ingress, |
| - Over-speed/failure of turbo machine/compressor; failure of magnetic bearings |
| - ATWS: Reactivity insertion |
| - Station blackout. |
| - Heat exchanger tube leak: moisture ingress, reactivity insertion |

TABLE 6.6.3. DESIGN FEATURES FOR PREVENTION AND PROTECTION LEVELS THAT REDUCE (R), SUPPRESS (S), THE INITIATORS FREQUENCY OR LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES (L)

| | |
|---|---|
| | **PREVENTION LEVEL** |
| Reactivity Insertion (including ATWS) - | Minimizing excess reactivity by provision of burnable poison in the core and by reshuffling of fuel elements (L)<br>- Limitation of control rod speed (L)<br>- Single phase coolant (S) |
| Disturbances - | High level of quality and redundancy (R)<br>- Careful Inspections and maintenance (R) |
| Turbo machine - | Very high level of quality (R)<br>- Magnetic bearings (S) |
| Leakages, Ruptures (primary) | - Very high quality level for pressure vessel (burst proof) (S)<br>- Limitation of connecting pipe diameters (L)<br>- Provision of isolation valves to be closed automatically (L) |
| Leakages/Ruptures - (auxiliaries arranged or connected in/to primary)- | Low-pressure and low-temperature water-cooled precooler and intercooler<br>High quality of shut down cooling system (R) |
| Total loss of power supply (station blackout) - | Emergency power supply (gas turbines, batteries) (R)<br>- Adequate design and supply of/for magnetic bearings at compressor and helium turbine (R) |
| Reactor cavity cooling system | - Passive (S) |
| | **PROTECTION LEVEL** |
| Reactivity Insertion (incl ATWS) - | Strong negative temperature coefficient, avoiding a high heatup of fuel (L)<br>- Provision of two diverse neutron control systems, control rods and shutdown pellets (R)<br>- Provision of startup control rods in addition to operating control rods (L) |
| Disturbances - | In case of failed shut down cooling system, heat dissipation via reactor cavity cooling system (S)<br>- In case of (partial or complete) cavity cooling system failure, heat dissipation to ultimate heat sink by heating up concrete, steel structures and surrounding environment (L) |
| Leakages/Ruptures - | Re-buildup of pressure from He-storage system (R)<br>- Passive heat dissipation from core (L) |
| Leakages/Ruptures - (outside primary) | Automatic closure of isolation valves (L) |
| Loss of Power Supply - (Station blackout) - | Passive heat dissipation from core (L)<br>Emergency power supply (gas turbines, batteries) (R) |

## TABLE 6.6.4. DESIGN FEATURES FOR MITIGATION LEVEL OF GT-MHR

| Safety functions | Systems (Cf Tab 6 6 1 ) | Passive/active | Design features/remarks |
|---|---|---|---|
| **Design Basis Accidents**<br>Fission product retention | Coated fuel particles<br>Pressure vessel unit | Passive<br>Passive/Active | - Inherently safe fission product retention<br>- Automatic closure fail-safe isolation valves<br>- Unfiltered release below permissible limits (normal and disturbed operation)<br>- Further reduction by filtered venting of reactor building in case of disturbed operation |
| Coolant inventory loss | Helium storage | Active | - Passive core heat removal from core even if down to atmospheric pressure |
| Decay heat removal | Shutdown cooling system<br>Reactor cavity cooling system | Active<br>Passive | <br>- If cavity cooling fails, heatup of environment without overheating fuel |
| Reactivity control | Operating control rods<br>Diverse reserve shut down system<br>Negative temperature coefficient<br>Startup control rods | Active/Passive<br>Active/Passive<br>Passive<br>Active/Passive | Automatic activation, gravity drop<br>Automatic activation, gravity drop<br><br>Automatic activation, gravity drop |
| Water Air Ingress<br><br>**Severe Accidents**<br>Fission product retention | Cooling water systems<br><br><br>Same systems and design features as for Design Basis Accidents | Active/Passive<br><br><br>Passive/Active | - Automatic isolation - valves<br>- Low-temperature, low-pressure coolers<br><br>- See design Basis Accidents<br>- No accident with fuel heating above permissible limits |
| Decay heat removal | Reactor Cavity Cooling System | Passive | - See design basis accidents |
| Large air ingress | No systems installed | ----- | - Graphite corrosion limited by naturally limited air flow through core |
| Inflam gas control in case of large air ingress | Ventilation (exhaust) | Active | |
| Extreme reactivity insertion | Negative temperature coefficient<br>Insertion of all control rods and reserve shutdown devices | Passive<br>Active/Passive | - Limitation of peak fuel temperature due to negative temperature coefficient<br>- Rod ejection prevention by design |
| Water air ingress | Same systems and design features as for Design Basis Accidents | Active/passive | - See design basis accidents |

### 6.6.4. Design data questionnaire

## I. GENERAL INFORMATION

1. Design Name: Gas Turbine Modular Helium Reactor (GT-MHR)
2. Designer/Supplier address: General Atomics, San Diego, USA
3. Reactor type: GT-MHR
   Number of modules/per plant: 1 or more
4. Gross thermal power (MW-th) per module: 600
5. Net electrical output (MW-e) per module: 286 (if no process heat used)
6. Heat Supply capacity (MW-th): Dependent on process applied (between 0 and appro. 300 MWt per unit)

## II. BASIC DESIGN DESCRIPTION

### A. CORE AND REACTIVITY CONTROL

7. Fuel material and chemical form: UCO ($UO_2$/UC)
8. Fuel inventory (tones of heavy metal): 4.67 tonnes (initial core)
9. Average core power density (kW/liter): 6.6 W/cc
10. Average discharge burnup (MWd/t): 121,000
11. Initial enrichment or enrichment range (Wt%): 15.5% (AVG)
12. Reload enrichment at the equilibrium (Wt%): 19.8% natural uranium (2 types of particles)
    AVG = 15.5%
13. Refuelling frequency (months): 15.8 months, 87% capacity factor
14. Type of refuelling (on/off power): Off power
15. Fraction of core withdrawn (%): 50
16. Moderator material: Graphite
17. Active core dimensions (m): 7.93 high; 4.83 DD; 2.96 ID
18. Type of fuel element: Graphite Block with fuel pins
19. Number of fuel elements: 102 columns, 10 blocks/column
20. Number of control rods: 48
21. Control rod material: Natural boron ($B_4C$) in graphite compacts
22. Additional shutdown systems: Reserve shutdown system
23. Neutron absorber material: $B_4C$
24. Burnable poison material and form: $B_4C$ granules dispersed in graphite compacts

### B. REACTOR COOLANT SYSTEM

#### B1. Coolant

25. Coolant medium and inventory: Helium
26. Design coolant mass flow through core (kg/s) at 100% power: 320 Kg/sec
27. Cooling mode (forced/natural): Forced (for power operation)
28. Operating coolant pressure (bar): 7.02 MPa at turbine inlet
29. Core inlet temperature (°C): 490
30. Core outlet temperature (°C): 850

#### B2. Reactor pressure vessel

31. Overall height of assembled vessel (m): 31.4
32. Vessel diameter (m): 8.4
33. Vessel thickness (mm): 204
34. Vessel material: 9 Cr - 1 Mo - V
35. Design pressure (bar): 80
36. Gross weight (tonne): 1252

**B3. Steam generator**

| | | |
|---|---|---|
| 37 | Number of steam generators | Not relevant |
| 38 | Type | |
| 39 | Configuration (horizontal/vertical) | |
| 40 | Tube material | |
| 41 | Heat transfer surface per steam generator ($m^2$) | |
| 42 | Thermal capacity per steam generator (MW) | |
| 43 | Feed water pressure (bar) | |
| 44 | Feed water temperature (°C) | |
| 45 | Steam pressure (bar) | |
| 46 | Steam temperature (°C) | |

**B4. Main coolant compressor**

47 Number 1
48 Type Axial flow
49 Pump mass flow rate (kg/s) 320 Kg/sec
50 Pump design rated head not relevant
51 Pump nominal power (kW) not relevant

## C. CONFINEMENT

52 Type vented, low pressure containment
53 Overall form cylindrical/rectangular
54 Structural material reinforced concrete
55 Liner material None
56 Simple/double wall Simple
57 Dimensions (diameter, height) (m) 29 m high, 7400 $m^3$ volume
58 Design pressure (bar) 0 69
59 Design temperature (°C) not specified
60 Design leakage rate (% per day) not relevant (vented)

# III. SAFETY RELATED SYSTEMS

## A. DESIGN CONDITIONS

**A1. Reactivity control**

61 Control rods
- a Maximum control rod worth (pcm) 1 50% $\Delta\ \rho$ (group worth)
- b Mode of operation (active/passive) Active withdrawal, gravity insertion
- c Redundancy Yes
- d Safety graded Yes
- e 3rd shutdown system negative coefficient of reactivity

62 Reserve shutdown system
- a Absorber material boronated graphite pellets
- b Mode of operation (active/passive) active
- c Redundancy No
- d Safety graded Yes

**A2. Decay heat removal**

*A2-1 Primary side Shutdown cooling system* (watercooled)
- a Actuation mode (manual/automatic) automatic
- b Pressure level (bar) 1 to 70 bar
- c Flow rate (kg/s) 2 to 80 KG/S Helium flow
- d Mode of operation (active/passive) Active
- e Redundancy Redundant cooling water train
- f Safety graded (Y/N) No

*A2-2 Cooling system for precooler and intercooler*

63 Feed water
- a Actuation mode (manual/automatic) manual start, automatic control during operation
- b Flow rate (kg/s) 322

c Mode of operation (active/passive) Active
d Redundancy No
e Safety graded No

*A2-3 Reactor Cavity Cooling System (Air cooled)*
a Activation mode continuous passive operation
b Heat transfer mode natural convection air flow
c Mode of operation Passive
d Redundancy Yes
e Safety graded Yes

*A2-4 Primary pressure control*
64 He - purification/storage system
a Actuation mode (manual/automatic) Runs continuously under automatic control
b Side location (primary/secondary circuit) Primary
c Maximum depressurization rate (bar/s) > 3 bar
d Safety graded No

## B. SEVERE ACCIDENT CONDITIONS*

### B.1 Fission products retention
65 Confinement filter system (Y/N) Yes, for minimizing purposes

### B.2 Recriticality control
66 Encountered design feature } 
a Mode of operation (A/P) } Not relevant
b Safety graded }

### B.3 Long term heat removal
67 Implemented system Shutdown cooling system or reactor cavity cooling system
a Mode of operation (A/P) Active/passive (shutdown cooling system is active, reactor cavity cooling system is passive)
b Self sufficiency (h) Not limited
c Safety graded (Y/N) No/Yes (shutdown cooling system is non-safety grade, reactor cavity cooling is safety grade)
68 Ultimate heat sink Heat dissipation from via conduction, convection, radiation to environment
a Self sufficiency (h) Not limited
b Safety graded (Y/N) Yes

### B.4 Combustible gas control (in case of large air ingress into primary system)
69 Covered range of gas mixture concentration
70 Modes for the combustible gas control
a Containment inertation }
b Gas burning } Not relevant
c Gas recombining }
d Others isolation of primary system, venting via filters

### B.5 Confinement pressure control
71 Filtered vented confinement (Y/N) Yes, for minimizing purposes
a Implemented system venting/ filtering
b Mode of operation (A/P) Active
c Safety graded No

---

* All systems must be qualified to operate under the accident conditions

## C. SAFETY RELATED I&C SYSTEM

Automatic load following (Y/N): Yes
- * range (% power): 50 - 100%
- * maximum rate (%/min): 0.5%/min. (2%/min. for step load changes)

Load rejection without reactor trip (Y/N): Yes
Full Cathode Ray Tubes (CRT) display (Y/N): Yes
Automated start-up procedures (Y/N): Yes
Automated normal shutdown procedures (Y/N): Yes
Automated off normal shutdown procedures (Y/N): Yes
Use of field buses and smart sensors (Y/N): Yes
Expert systems or artificial intelligence advisors (Y/N): Yes
Protection system backup (Y/N): Yes

## D. EMERGENCY POWER SUPPLY SYSTEM

72. Type (diesel, gas, grid connection): Gas turbines, Batteries, auxiliary grid
73. Number of trains: 2
    Mobile power units: Connectable

## E. AC/DC SUPPLY SYSTEM

74. Type (rectifier, converter, battery): Rectifier, convertor, inverter batteries
75. Estimated time reserve (hr): 1

# IV. CONVENTIONAL THERMAL CYCLE

## A. TURBINE SYSTEM

76. Type: Helium Gas Turbine
77. Overall length (m): ~ 13
78. Width (m): ~ 2.7
79. Number of turbines/reactor: 1
80. Number of turbine sections per unit (e.g. HP/LIP/LP): 1 (HP)
81. Speed (rpm): 3600

## B. HELIUM GAS TURBINE CHARACTERISTICS

82. Turbine inlet pressure: 7.02 MPa
83. Turbine inlet temperature: 850°C
84. Turbine inlet flowrate: 320 Kg/Sec
85. Turbine outlet pressure: 2.65 MPa
86. Turbine outlet temperature: 510°C
87. Turbine outlet flowrate: 320 Kg/sec

## C. GENERATOR

88. Type (3-phase synchronous, DC): 3 phase synchronous
89. Apparent power (MVA): 325
90. Active power (MW): 294 (power output at generator) (net Module Electrical output: 286 Mwe)
91. Frequency (hz): 50 to 60
92. Output voltage (kV):
93. Total generator mass (t) }
94. Overall length } Not yet fixed in detail
95. Stator housing outside diameter }

## D. CONDENSER

96. Number of tubes
97. Heat transfer area
98. Flowrate ($m^3/s$) — Not relevant
99. Pressure (m/bar)
100. Temperature (°C)

E. CONDENSATE PUMPS

| | | |
|---|---|---|
| 101. | Number | Not relevant |
| 102. | Flowrate | |
| 103. | Developed head | |
| 104. | Temperature | |
| 105. | Pump speed | |

## 6.6.5. Project status

Up to 1991 General Atomics was developing a Modular High Temperature Gas-cooled Reactor (MHTGR) with a side-by-side arrangement of reactor and steam generator, having a reference power of 350 MWt per module with an option for 450 MWt. A reference plant consisting of 4 modules, had been extensively designed. Details of the plant were provided in a preliminary safety report prepared in 1988. This documentation was submitted to the NRC, Washington for a Safety Evaluation.

The NRC involved numerous experts to evaluate the safety concept and issued, in 1990, an unofficial Safety Evaluation Report (SER) with a rather positive statement on the licenseability of this concept. In 1988 GA and Bechtel had already performed preliminary analyses for design basis and beyond design basis accidents. The work for the structural, mechanical and electrical engineering was well beyond the conceptual design stage.

In 1991 GA decided to take a further development step in order to improve the economics but also the safety of such a Modular High Temperature Reactor Plant. This step lead to the introduction of the Helium Gas Turbine, directly combined with the Modular Helium Reactor.

Since the main-design features and safety objectives of the reactor have not been changed, a large part of the former work can be used directly for the reference design of the GT-MHR producing a power of 600 MWt.

### *6.6.5.1. Entities involved*

The development of the GT-MHR is led by General Atomics. Substantial advisory work has been done by the MIT, Boston and very substantial R& D work has been contributed by ORNL. The entire program is financially supported by DOE.

### *6.6.5.2. Design Status*

Since much R & D work, design work and safety/radiology analyses has already been done for the MHTGR, and a very large part of this fundamental work can be applied to the GT-MHR, the status of the design is considered to have reached an interim status between D1 and D2.

### *6.6.5.3. R & D Status*

The GT-MHR can be deployed in the near term without the need for fundamental research because of the considerable gas-cooled reactor technology base available. This technology has been demonstrated by more than 50 gas-cooled reactors built and operated in the United Kingdom, Germany, France, Italy, Spain, Japan and the United States since

1956 including the peach Bottom and Fort St. Vrain reactors. More recently, the Japanese are constructing a helium-cooled High Temperature Test Reactor (HTTR), scheduled to go on line in 1998.

The engineering development needed for the GT-MHR falls basically into two categories 1) technology development which provided data for design methods and validation of computer codes, and 2) component or process verification including prototypical component testing. In the latter category, a key verification test to be performed is an integrated test of the energy conversion system.

#### *6.6.5.4. Licensing Status*

In view of the fact, that an unofficial SER has been released by NRC for the MHTGR and in view of the fact that a large part of the safety engineering for the MHTGR can be used for the GT-MHR, it is estimated, that a L1 - Status has been reached.

#### *6.6.5.5. Conclusions*

The future acceptance of nuclear power will be very dependent on safety and economic considerations, but for it to fulfill a role as a "green technology" it must meet the needs of a spectrum of energy users, broader than just electricity production. The MHR is the only type of reactor that can realize this goal. The high temperature capability of this reactor is key to its versatility and will gain in recognition as natural resources become more precious and concerns for our environment become more sensitive. Not only is the MHR, coupled to a GT, one of the most efficient electricity producers, but its high temperature capability makes it an ideal heat source for providing the thermal energy for a variety of industry applications such as the production of clean burning transportation fuels. Its simplicity and inherent safety characteristics eliminate the need for a complex infrastructure to assure safe and reliable operation, which make it attractive for deployment in developing countries. The technology for the MHR has a sound historical basis, and no fundamental breakthroughs are necessary in terms of component design/performance or materials advancements.

### 6.6.6. Economics

The investigations of General Atomics have shown, that the GT-MHR results in essential improvements of the economics:

> The GT-MHR design directly couples the reactor with a turbogenerator in a closed helium Brayton cycle to produce electricity with 48% net plant efficiency. This high efficiency and the expansion of the power output to 600 MW(t) within the existing GT-MHR physical envelope results in a substantial reduction in the busbar power costs compared to the steam cycle modular helium reactors. The power generation costs are further reduced by the simplified operation and maintenance required of the gas turbine plant, as compared to the steam cycle plant with its much more complicated balance of plant.
>
> Preliminary economic assessments indicate the projected levelized busbar generation cost for the four module "equilibrium" GT-MHR plant at 600 MW(t) per module is about 35 mills/kWh (in 1994). The "equilibrium" plant reflects learning cost reductions from construction of several previous plants. This is a decrease of over 35% in busbar costs compared to the early 1980s designs of the 350 MW(t) steam

cycle MHR, and this cost challenges combined cycle combustion gas turbines coming into service at the same time even if only modest real escalation in natural gas prices is assumed.

The high efficiency of the GT-MHR has a number of environmental advantages as well. The GT-MHR produces 75% less high level radioactive waste and 50% less thermal discharge to the environment than competing light water reactor (LWR) designs.

## REFERENCES

The following High Temperature Reactors have already been operated or are now under construction:

England- Dragon - 1964 to 1976 - This helium-cooled test reactor provided early successful demonstration of the high temperature gas-cooled reactor.

Germany - AVR - 1966 to 1988 - This prototype helium reactor operated successfully for over 20 years and provided demonstration of 1740°F gas outlet temperature and key safety features, including safe shutdown with total loss of coolant circulation and without control rod insertion.

U.S. - Peach Bottom - 1967 to 1974 - This prototype helium reactor achieved a remarkable 86% availability during the electricity production phase.

U.S. - Fort St. Vrain - 1979 to 1989 - This reactor used water-lubricated circulator bearings which resulted in frequent water ingress into the reactor system and caused significant down time. In spite of a poor operating record, the Fort St. Vrain coated fuel and reactor core worked extremely well. Because of the non-corrosive nature of helium, workers were exposed to radiation doses only about 1% that of average water reactors. Fort St. Vrain generated about 5 billion kWH.

Germany - Oberhausen 2 - 1975 - 1987 - This 50 MW electric turbine plant represented the evolutionary step from fossil-fired gas turbines with air as the working fluid towards the realization of nuclear powered helium gas turbines. Helium was used as the working fluid in a closed-cycle process for electricity and heat production. The plant incorporated heat exchangers (recuperator, precooler, intercooler) of comparable size to those required for a 600 MW thermal GT-MHR.

Germany - THTR - 1985 to 1988 - This helium-cooled nuclear power plant generated about 3 billion kWh. Political resistance in the post-Chernobyl era precipitated early shutdown.

Russia - Various successful demonstrations of fuel fabrication and fuel irradiation performance.

Japan - A high temperature helium-cooled test reactor is now under construction.

Germany - A further reference should be made to the German Modular High Temperature Reactor (MHTR) which is also included in this report.

China - A 10 MWt test reactor (MHTR) is under construction.

## 6.7. MODULAR HIGH TEMPERATURE REACTOR (MHTR)

### 6.7.1. Basic objectives and features

The modular HTR power plant is an energy source applicable for the co-generation of electricity, process steam or district heating. Standardized reactor units with power ratings of 200 MW (per module) and a core outlet temperature of 700 °C can be combined to form power plants with any rating. The provision of process heat at a helium temperature up to 950 °C for coal gasification or for the aluminum industry is also technically possible. The special safety features of small high temperature reactors are maintained for all applications.

Due to its universal applicability and excellent safety features, the modular HTR power plant is suitable for erection on any site, but particularly on sites near other industrial plants or in densely populated areas. The principal safety feature of the HTR-Module is based on the fact that, even in the case of failure of all active cooling systems and complete loss of coolant, the fuel element temperatures remain within limits at which there is virtually no release of radioactive fission products from the fuel elements, due to:

- The use of spherical fuel elements, which are capable of retaining all radiologically relevant fission products up to fuel element temperatures of approx. 1600°C with an appropriately designed core.
- Active core cooling is not necessary for decay heat removal during accidents since passive heat transport mechanisms to a simple cooler outside the reactor pressure vessel are available.
- Reactor shutdown is carried out solely by absorber elements, which, on demand, can drop freely into bore holes of the reflector.
- Due to the high activity retention of the fuel elements, a pressure-tight reactor building is not necessary.
- Reactor core and steam generator are installed in separate steel pressure vessels in such a way that there is no danger of components overheating in the case of failure of the primary circuit cooling. This chosen arrangement also increases the accessibility of the components for maintenance and repair.

### 6.7.2. Design description

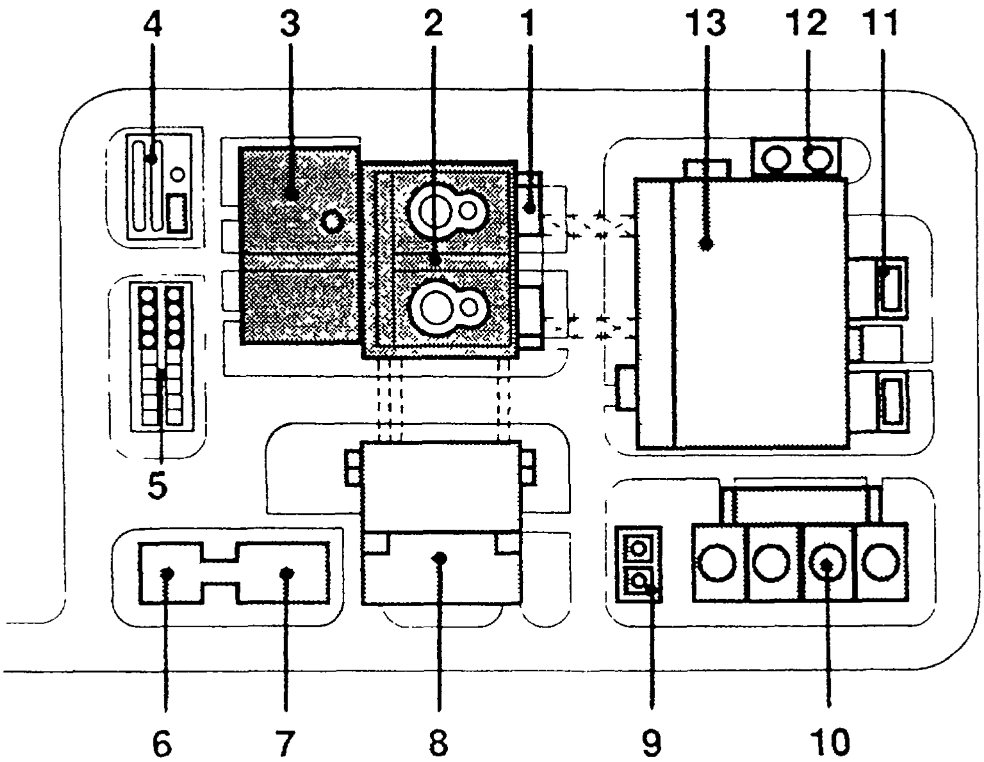
#### *6.7.2.1. Nuclear steam supply system*

6.7.2.1.1. Construction and design features

The nuclear steam generating system (HTR-module) basically consists of:

- The reactor pressure vessel with core, core internals, shutdown systems, and facilities for the charging and discharging of fuel elements,
- The connecting pressure vessel with a hot gas duct,
- The steam generator with the heating tube bundle and the primary circuit blower.

Each HTR-Module is installed in a primary cell, the concrete walls of which support the weight of the reactor pressure vessel, the connecting pressure vessel, and the steam generator pressure vessel and their internals (fig 6.7.1.). Pipes with water flowing through them (surface coolers) are installed on the inside wall of the primary cavity. These coolers discharge dissipated heat during normal operation and decay heat during reactor shut-down. The reactor and the steam generator are positioned beside each other with the following substantial advantages:




1 Reactor building annex
2 Reactor building
3 Reactor auxiliary building
4 Helium storage
5 Spent fuel element storage
6 Gate house
7 Office and staff building
8 Central control building
9 Wet-cooling-cells
10 Cooling tower
11 Transformer
12 Deionat tanks
13 Turbine building

*Fig 6.7.1. Cross Section through the Primary System*

- After a reactor shutdown harmful natural circulation of hot helium through the primary circuit is prevented, thanks to the thermohydraulic decoupling of heat source and heat sink. Consequently, there is no need to cool the steam generator after shutdown. Thus it can be shut off and left in a hot condition.
- The positioning of the steam generator beside and lower than the reactor permits simple and operationally favorable upward evaporation.
- The substantial separation of the reactor core and the steam generator by the shielding concrete walls of the reactor cell makes it possible to repair all defects in the steam generator bundle or the primary circuit blower without accessibility problems.
- The staggered arrangement leads to a favorable height-to-width ratio of the reactor building. This simplifies verification of its stability in the case of loads resulting from external impacts (e.g. earth-quake, explosion blast wave).

6.7.2.1.2. Fuel element and reactor core

Fissionable uranium is present in the HTR-Module in spherical fuel elements. Each fuel element has a diameter of 6 cm and contains approximately 11 600 coated particles within the inner graphite matrix. Each of these coated particles consists of a fuel kernel (uranium dioxide, $UO_2$) with a diameter of about 0.5 mm, which is coated with layers of pyrocarbon (PyC) and of silicon carbide (SiC). These layers enclose the fuel kernel, thus preventing a fission product release from the fuel element to the highest degree. One fuel element contains a total of 7 g of uranium (equivalent to a moderation ratio of approx. 6900) with a 235 U enrichment of 7.8%.

A fuel-free, 5-mm thick graphite shell is pressed on to the inner fuel-bearing zone of the fuel element which has a diameter of approximately 5 cm. This fuel-free shell prevents the coatings of the particles from being damaged by mechanical or chemical impacts, e.g. during transport by the charging facility or in the event of water ingress into the core area as the result of an accident.

During normal operation the fuel elements attain a maximum temperature of approximately 850°C. The evaluation of numerous heat-up experiments has shown that the coatings of the particles are capable of almost completely retaining all radiologically relevant fission products in the intact coated particles up to fuel element temperatures of around 1600°C. Numerous experiments have shown that the number of defective coated particles to be expected is very low (average coated particle failure fraction of $3 \text{ X } 10^{-5}$ during normal operation, $4 \text{ X } 10^{-5}$ during core heat-up events resp. leading to low coolant activity and the low environmental exposures after loss of coolant accidents.

The reactor core, which is located inside the reactor pressure vessel (fig. 6.7.1.) consists of approx. 360 000 spherical fuel elements in a loose pebble bed with a diameter of approx. 3 m and an average height of approx. 9.4 m. It is cooled by helium. The mean power density of the core is limited to 3 $MW/m^3$ and the mean core outlet temperature to 700°C.

The design of the reactor core is based on the following principles:

- In all accidents and accident combinations, a maximum fuel element temperature of approx. 1600°C is not exceeded even without active removal of the decay heat from the core. Decay heat removal can be effected solely by heat conduction, heat radiation, and natural convection to the cavity coolers positioned outside the reactor pressure vessel.
  The reactor can be shut down purely by dropping the absorbers into the reflector bore holes. Additional shutdown rods, which have to be inserted into the pebble bed using external energy, are not necessary.
- The uranium loading of the fuel elements amounting to 7g of uranium is selected such that water ingress into the primary circuit as the result of an accident will cause a reactivity increase not higher than the accidental withdrawal of all absorbers during normal operation.
- The faulty withdrawal of all reflector rods as a covering accident is therefore controlled by simply switching off the primary circuit blower, whereby the permissible fuel element temperature of approx. 1600°C is not exceeded.
- In order to obtain the most uniform possible power density distribution, the spherical fuel elements pass through the core approximately 15 times before reaching their final burn-up.

The ceramic core structure, which essentially consists of side, bottom and top reflectors, forms the cylindrical vessel which holds the pebble bed. The ring shaped side reflector consists of 24 individual columns made up of single graphite and carbon blocks. The top reflector consists of several layers of graphite and carbon segments. The cold-gas plenum is located in the top reflector. The bottom reflector, which also consists of several layers, contains the hot gas plenum.

A metallic core vessel, which is supported in the lower part of the reactor pressure vessel, acts as supporting structure for the ceramic core internals. The cover of the core vessel is constructed as a radiation shield (so-called top thermal shield) to provide access to the area above the core vessel for maintenance work. This area is filled with stagnant helium during operation.

The drives for the shutdown and control systems are mounted on the thermal shield. These consist of the reflector rods and the small sphere shutdown units. The 6 reflector rods are for the reactor control and hot shutdown of the core. Each rod consists of several individual sections holding the absorber material $B_4C$.

In order to initiate a reactor scram, the power supply to the drive is interrupted, thus causing the rod to drop freely into its lowest position in the bore holes of the side reflector due to gravity. Eighteen small sphere shutdown units serve to compensate the reactivity increase due to a cold, unpoisoned core. Graphite spheres with a 10% $B_4C$ content and a diameter of approx. 10 mm are used as shutdown elements. The spheres, which are stored in storage containers located above the top thermal shield and over the side reflector, drop freely into the reflector bore holes on demand.

During normal operation the pebble bed is cooled by the primary coolant flow. After flowing upwards through bore holes in the side reflector, the primary coolant is collected and deflected in the top reflector. It then flows downward through the top reflector, the pebble bed and the bottom of the core, whereby it removes the heat generated in core, before being collected in the hot gas plenum and conveyed to the steam generator via a penetration in the side reflector. The selected primary coolant guiding system ensures that the pressure vessels only come into contact with cold gas (250°C).

6.7.2.1.3. Pressure vessel unit

The reactor core is located in the reactor pressure vessel which, together with the connecting pressure vessel and the steam generator pressure vessel, form the so-called pressure vessel unit (fig 6.7.1). The reactor pressure vessel is approx. 25 m high with an internal diameter of approx. 6 m and an empty weight of approx. 760 tons. The fuel element discharge tube passes centrally through the bottom of the reactor pressure vessel. Four other connecting pipes supply the fuel elements and the conveying gas for the small sphere shutdown units. The top cover, which can be removed for inspection or maintenance purposes, is bolted down with prestressed bolts and is sealed with two metallic sealing rings. The reactor pressure vessel is supported in the reactor cell by 3 brackets on the level of the connecting pressure vessel.

The connecting pressure vessel, together with the hot gas duct, serves to guide the primary gas between the reactor pressure vessel and the steam generator pressure vessel. It consists of a forged ring which is welded to the reactor and the steam generator pressure vessels. The connecting pressure vessel contains the hot gas duct comprising the metallic gas liner (through which the hot gas flows from the reactor to the steam generator), a fibre insulation and the outer metallic support tube. The cold gas flows back from the steam generator to the reactor between the support tube and the connecting pressure vessel, thus cooling the metallic components.

The steam generator pressure vessel is positioned in the steam generator cell which is beside and slightly lower than the reactor pressure vessel. It consists of the steam generator pressure vessel section and the blower pressure vessel section screwed on top of this. The upper part of the steam generator pressure vessel is a reinforced jacket ring holding the nozzle to the connecting pressure vessel, the live-steam nozzle, consisting entirely of Incoloy 800, and the brackets to support the pressure vessel in the steam generator cell. The inlet nozzle for the feedwater is welded into the lower part of the vessel.

6.7.2.1.4. Steam generator and primary circuit blower

The steam generator is designed as a one-through helical tube steam generator. The hot helium is fed into the steam generator above the heating tube bundle. While flowing around the steam generator heating tubes, the helium releases its heat to the water/steam side whereby it cools down from approx. 700°C to approx. 250°C. On leaving the heating tube bundle, the flow is deflected through 180°. The cold helium then flows upwards between the steam generator shroud and the inner wall of the pressure vessel so that the pressure vessel is cooled by the cold gas. The primary circuit blower returns the helium to the reactor through the annular gap between the connecting pressure vessel and the hot gas duct.

The feedwater enters the feedwater nozzle with a temperature of 170°C. The water flows through the helical tubes from the bottom to the top. During this process it evaporates and is subsequently superheating. Via a compensating bundle above the heating tube bundle, the steam, which is 530°C hot enters the live-steam nozzle and passes through the live-steam line to the machine hall.

The primary circuit blower consists of a single-stage radial compressor. A speed-controlled asynchronous motor is provided as drive. The motor is cooled with water. The blower with motor is installed in the blower pressure vessel section as a slide-in unit with vertical shaft and hanging impeller. A blower flap is located on the suction side of the blower. The blower increases the pressure of the helium sucked in from the steam generator outlet side by approx. 1.5 bar.

6.7.2.1.5. Reactor auxiliary and ancillary systems

Some auxiliary and ancillary systems are assigned to the nuclear steam generating systems:

- Pressurization and cleaning of the primary circuit,
- Continuous charging and removal of fuel elements,
- Removal of dissipated heat and decay heat from the primary cell,
- Conditioning of the air in the rooms.

6.7.2.1.6. Helium purification facility

The helium purification system is installed for the purification of the primary coolant to specified values by removing chemical impurities ($H_20$, CO, $CO_2$, $N_2$, $CH_4$), the supply of helium to the helium-filled systems (pressure regulation) and the drainage for maintenance work and removal of tritium from the primary circuit.

One line of the helium purification facility is directly assigned to each of the two HTR-modules. A third purification line is available in the case of fault-incurred failure, maintenance work and regeneration operation of the directly allocated line. The through-put of a purification line amounts to 5% of the primary circuit inventory per hour.

Each purification line must be regenerated after about 1000 operating hours. During this regeneration period the third purification line is connected to the respective HTR-Module. This third purification line is additionally equipped with a water separator which can be included in the circuit when required.

6.7.2.1.7. Fuel element handling

The fuel elements are continuously fed to and removed from the core by means of the refuelling and discharge facility. After passing through the core, the fuel elements are removed via the fuel element discharge tube. Broken spheres or fuel elements which are not geometrically satisfactory drop into a fragments casket, while the intact fuel elements are transported to a burn-up measurement facility. This determines whether the fuel element has reached its final burn-up and has to be discarded or whether it can be returned to the core. The 360 spent fuel elements obtained on average per day are conveyed to the fuel element transport container via the outward transfer buffer line.

The re-usable fuel elements and the new fuel elements fed in by the input station are pneumatically transported to the core via the charge tube. Primary coolant at cold-gas temperature acts as conveying gas.

To facilitate the performance of unexpected repairs to the primary circuit, it is possible to remove all fuel elements from the core, to keep these in intermediate storage, and to return them to the reactor core on completion of the repairs.

6.7.2.1.8. Cavity cooler

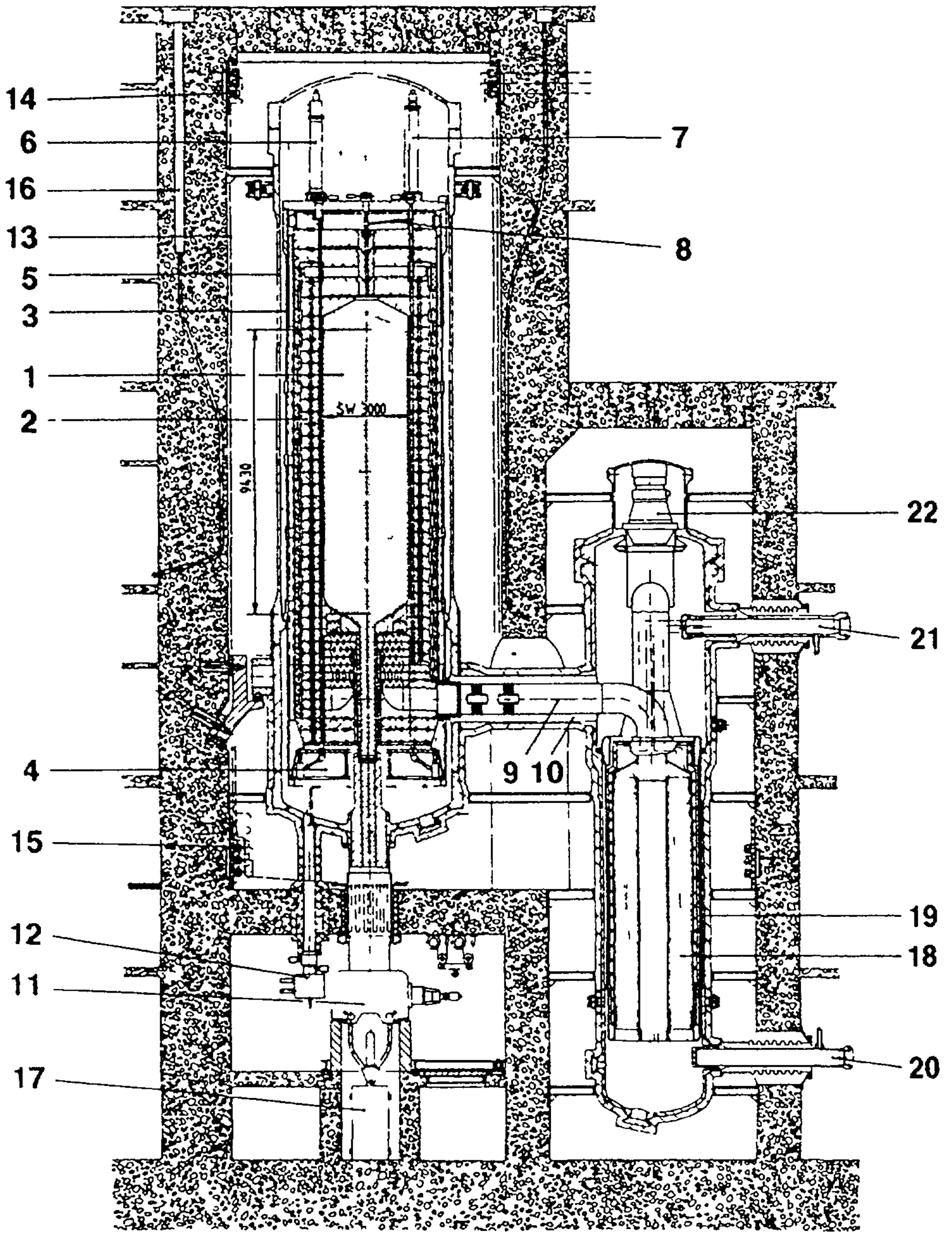
The cavity cooler surrounds the reactor pressure vessel at a distance of about 1.5m. It is installed as a closed tube wall in front of the concrete walls of the reactor cavity with a clearance distance of approximately 10 cm. Water at a temperature of about 40°C flows through the cavity cooler at low pressure. Its task is to remove the dissipated heat amounting to approx. 400 kW to the connected intermediate cooling systems during normal operation and to remove the decay heat from the reactor of up to approx. 850 kW to these systems in the case of failure of the main heat sink.

*6.7.2.2. Balance of plant systems*

In the HTR application described here the balance of plant consists in essence of a standard superheated steam turbine circuit (fig 6.7.2.) with wet cooling.

*6.7.2.3. Instrumentation, control and electrical systems*

The power plant generators feed into the 110 kV high-voltage network via one generator transformer per generator and cover the power requirement for operation of the




1 Pebble bed reactor core
2 Graphite reflector
3 Core vessel
4 Core support
5 Reactor pressure vessel
6 Small absorber spheres unit
7 Reflector rod
8 Fuel element
9 Hot gas duct
10 Cold gas channel
11 Valve bank for fuel element discharge
12 Valve bank for small absorber spheres system
13 Cavity cooler
14 Return lines cavity cooler
15 Forward lines cavity cooler
16 Neutron flux instrumentation
17 Failed fuel cask
18 Tube bundle
19 Steam generator pressure vessel
20 Feed water line
21 Live steam line
22 Circulator

*Fig 6.7.2. General Plan of the HTR-Module Power Plant*

plant via an auxiliary power transformer. A further power supply possibility via an external network transfer is also provided.

In the event of failure of the existing auxiliary power (station service) supply a double-line network is available for the emergency power supply of some important consumers. The emergency power supply is provided by two diesel generators providing 750 kW each. Consumer groups which are to continue uninterrupted operation in the case of failure on the normal auxiliary power supply (e.g. reactor protection system, measurements, monitoring) are supplied by 220 V (resp. 24V) batteries. The emergency power supply is located in the central control building.

The monitoring instrumentation of the emergency control room which is situated inside the reactor buidling is equipped with its own battery system for the improbable failure of the emergency power supply. All vital systems and components of the reactor plant can be monitored in the emergency control room for at least 15 h before the batteries have to be charged again from outside. Operation of the power plant is controlled and monitored from a control room in the central control building.

### *6.7.2.4. Safety considerations and emergency protection*

#### 6.7.2.4.1. Essential Safety-related properties

The aim of the safety design of a nuclear power plant is the in-plant retention of the radioactive materials, which unavoidably form due to nuclear fission, such that any danger to the environment can be excluded both during normal operation and after accidents.

The modular HTR power plant is designed in such a way that all accidents based on physically and technically plausible assumptions do not lead to inadmissible radioactivity releases. This is primarily due to the optimum exploitation of the favourable inherent features of small high-temperature reactors.

Inherent safety features

"Inherent (i.e. intrinsic) safety feature" is a specialized term used in this case to describe the fact that the reactor itself reacts to certain malfunctions without the actuation of active systems or external controlling interventions in such a way that no inadmissible or even dangerous situations can be reached. These reactions are governed by the laws of nature, i.e. they always function regardless of the condition of active systems. This means that they cannot malfunction or fail. The technical and nuclear physical design of the HTR-Module is such that the maximum fuel element temperature always stabilizes itself below 1600°C even in the cäse of assumed failure of all active shutdown and decay heat removal systems.

On the one hand, this is achieved by the fact that there is a temperature span of approx. 700°C between the maximum permissible fuel element temperature of 1600°C and the maximum operating temperature of the fuel elements. This temperature span ensures that the reactor core shuts itself down via the negative temperature coefficients of reactivity, even after accident-incurred introduction of any existing surplus reactivity.

On the other hand, the selection of a low mean power density in the reactor core, the selection of a suitable geometry for the reactor core and the surrounding core internals, and the use of appropriate materials ensure that the decay heat can escape from the reactor core to the surrounding components and structure solely by means of physical processes (heat conduction, radiation, convection) before the above limit temperature of 1600°C is reached.

Hence, the major two dangerous situations of any nuclear reactor,

- an inadvertent power surge due to a reactivity insertion, and
- the inability to get rid of the decay heat,

are none existent in the HTR-Module due to the exploitation of inherent design properties.

Activity release barriers

A characteristic safety feature of the HTR-Module is to be found in the fact that, in all operating and accident situations, the radioactive materials formed during nuclear fission are enclosed in the coated particles in such a way that a significant activity release from the coated particles can be excluded. This safe activity enclosure is guaranteed by the design of the fuel particle coatings and the inherent limitation of the maximum possible accident temperature of the fuel elements to approx. 1600°C.

The radioactive materials escaping from the few defective particles are partly retained in the fuel element matrix. The non-retained fraction is transferred to the primary coolant and is distributed in the primary circuit.

The gas-borne activity in the primary circuit is reduced by radioactive decay, by extraction through the helium purification facility, and by deposition on the surfaces of the primary circuit. The primary circuit therefore acts as a further barrier to prevent a release of radioactive materials.

The components of the pressure vessel unit are designed such that failure can be excluded. Leakages in the connecting pipes which cannot be isolated are very improbable due to the planned quality assurance measures. In a nonetheless postulated leakage, in principle only the very low gas-borne primary coolant activity and a small fraction of the activity deposited on the primary circuit surfaces could be released into the reactor building. For this reason, and basically because of the high retention capability of the coated particles, no tightness requirements are imposed on the reactor building of the HTR-Module for adherence to the permissible accident dose limit values of Article 28 of the German Radiation Protection Ordinance, i.e. the released activity can be discharged to the environment unfiltered without any danger.

6.7.2.4.2. Emergency protection

Even though the nuclear power plant is designed and constructed in accordance with the highest quality requirements, it is impossible to completely exclude failures of technical systems.

Due to the selected design of the HTR-Module, only a few protective actions have to be taken after accidents to shut the reactor down and to keep it in a safe condition.

Reactor protective actions

Accidents cause characteristic changes in the process variables which are detected by the reactor protection system and which in turn initiate shutdown of the modular HTR plant if the specified limit values are exceeded. Regardless of the type of accident, the same three protective actions are always performed for plant shutdown:

- Dropping of the reflector rods,
- Shutdown of the primary circuit blower, and
- Isolation of the steam generator.

The first two actions are for nuclear shutdown purposes, while isolation of the steam generator (i.e. closure of the feedwater and live-steam valves) separates the reactor plant from the steam power plant.

In the case of a loss of pressure accident and a steam generator heating tube rupture, the primary circuit isolation valves are additionally closed (resp. the steam generator is quickly drained).

Accident control measures

After being shut down, the HTR-Module can be left standing in a hot condition as, in this condition, the maximum fuel element temperatures only increase to approx. 1100°C at a normal operating pressure even without active core cooling, and to approx. 1600°C after loss of pressure. The steam generator need not be cooled due to the thermohydraulic decoupling of reactor and steam generator after shutdown. The HTR-Module can be left standing in a hot condition until the causes and consequences of the accident have been repaired. As there is no need to remove the decay heat by means of forced circulation within the primary circuit, no separate decay heat removal within circuits are necessary.

In the case of an accident, the decay heat is removed from the HTR-Module via heat conduction and heat radiation to the cavity cooler installed outside the reactor pressure vessel. This cooler is capable of protecting the core internals, the shutdown systems, the reactor pressure vessel and the concrete structure of the reactor cavity from inadmissible temperatures during all accidents.

During normal operation, the cavity cooler removes the dissipated heat from the reactor cell. It is therefore continuously in operation and need not be specially started up when an accident occurs.

The water throughput and water temperature of the cooler are so low (9 l/s for one train) that, after an assumed failure of all pumps or recooling chains, water supplied by a host pipe after approx. 15h (e.g. by the fire department) would suffice to ensure adequate cooling of the reactor cavity and the reactor pressure vessel without any damage having been caused in the plant that would not allow further operation.

In principle, in the case of a postulated primary circuit leakage, it is possible to directly discharge the primary coolant unfiltered to the environment. Nevertheless, in the case of depressurization accidents with leakage cross-sections of up to approx. 1 $cm^2$, it is planned to pass the leaking primary coolant through the subpressure maintenance system and to discharge it filtered via the stack in order to minimize the radiological exposure of the environment. In the case of larger assumed leakage cross-sections which would cause a notable pressure increase in the reactor building, the building is quickly depressurized with direct discharge to the environment via the stack.

Design basis accident analyses

In order to assess the efficiency of the safety design and the planned protective measures, the occurrence of accidents is assumed in analyses and the resulting plant responses are examined. The analysed accidents are characteristic with respect to the possible release of radioactive materials and to the loads they impose on components and systems.

As design basis accidents are defined and analysed

- Reactivity accidents (covering ATWS)
- Disturbances in the main heat transfer system
- Leakages and ruptures on the secondary side
- Steam generator leakages (water ingress)
- Loss of pressure in the primary circuit with subsequent core heat-up
- Station black out

The list of accidents to be analysed is established in accordance with - but under modification of - the "Accident Guidelines for Pressurized Water Reactors" published by the German Federal Minister for the Interior.

The design basis accident analyses are to prove the accident consequences within the radiological limits which are prescribed by Art. 28.3 of the German Radiation Protection Ordinance (Strahlenschutzverordnung StrSchV).

External events

In addition to accident events, which are caused by operation of the reactor itself, a damage of the plant can also be caused by the external events

- Earthquake
- Aircraft crash
- Explosion blast wave

The reactor building is designed to withstand all external events. The pressure vessel unit and the inside located parts of the cavity cooling system are designed against all transferred vibrations in consequence. In case of destruction of the outer parts of the cavity cooling system the supply via hose pipe connected inside the reactor building is possible.

The central control building (switch gear building) is designed to withstand earthquake only. In the case of its destruction the reactor is safely shut down due to the fail safe principle. The reactor then is monitored from the emergency control room inside the protected reactor building.

### 6.7.2.5. *Buildings and structures*

#### 6.7.2.5.1. Buildings and cooling tower

The reactor building is in the center of the plant. In addition to two HTR modules, it contains some auxiliary and ancillary systems. Components of the start-up and cool-down systems, the steam generator fast discharge systems, and the intermediate cooling systems are installed in the annex to the reactor building. The two modular units are separated from each other by a central service area. An outer protective shell encloses the inner building structure. It fulfills the requirements for protecting the reactor plant from external impacts (e.g. aircraft crash).

The reactor auxiliary building directly adjoins the reactor building. It accommodates the facilities for helium purification and for the treatment and storage of radioactive waste, the ventilation systems, the store for new fuel elements, the sanitary rooms and other rooms such as laboratories. The central, monitored entrance to the reactor building is through the reactor auxiliary building.

The central control building contains the control room, the switching, control and closed-loop control systems, the reactor protection systems, the emergency power supply, and further installations such as process computers, air-conditioning and ventilation systems.

The machine hall basically holds the components of the water/steam circuit with the turbo sets and the components for process steam generation. The cooling tower (hybrid design) is positioned at a distance from the machine hall. An office and staff building, an intermediate fuel element store, and a gate house are provided as auxiliary buildings.

#### 6.7.2.5.2. Ventilation; Access

In the reactor building and reactor auxiliary building of the modular HTR power plant, all rooms are accessible during normal operation, with the exception of the primary cavity. Treated outside air is fed into the rooms to remove the dissipated heat and to retain the necessary quality of air in the rooms. The desired subpressure is maintained by regulating the supply of air flow while keeping constant the exhaust air quantity for each module and the service area. During normal operation the vented air is released unfiltered to the environment via the vent stack.

Dust-bound radioactive materials may enter the vent air during repair or maintenance work on the primary circuit or connected systems. For this reason, the vent air filter system (aerosol filter) is switched on during such work as precautionary measure.

If, as the result of leakages in components containing primary coolant, gas-borne radioactivity is released into the reactor building or the reactor auxiliary building, the facility is automatically switched over to the subpressure maintenance system with its activated carbon and aerosol filters. It has to be noted that these action are not necessary in order to meet allowable dose limits, but they are carried out as a matter of precaution due to the principle of minimization.

### 6.7.3. Safety concept

TABLE 6.7.1. MAIN SAFETY RELATED SYSTEMS IN THE MHTR CONCEPT

| Safety function: Name of system | Safety graded [1] | Main characteristics |
|---|---|---|
| Fission Product Retention | | |
| - Fuel element | X | C-SiC-C coated particle<br>Leak-tight up to 1600°C |
| - Pressure vessel unit | X | All connecting pipes with diameters below 65 mm<br>Most of connecting pipes with two isolation valves in series |
| Residual heat removal: | (X) | Active |
| - Main heat transfer system | (X) | Only the accessible parts are active |
| - Cavity cooling system | | |
| Reactivity control: | (X) | |
| - 6 Control and Shutdown rods | (X) | } Fail safe initiation, driven by gravity |
| - 18 Small absorber sphere columns (KLAK) | | |

TABLE 6.7.2. MAIN ACCIDENT INITIATORS FOR THE MHTR

| |
|---|
| - Reactivity insertion (including ATWS) |
| - Disturbances in the main heat transfer system (MHTS) |
| - Leakages and ruptures in the MHTS on secondary side |
| - Steam generator leakages (water ingress) |
| - Loss of pressure in the primary circuit (core heat-up, air ingress) |
| - Total loss of power supply (station black out) |

1) X: Highest quality level
(X): Lower quality level system, not necessary to meet prescribed safety limits for fission product release to the environment

TABLE 6.7.3. DESIGN FEATURES FOR PREVENTION AND PROTECTION LEVELS THAT REDUCE (R), SUPPRESS (S), THE INITIATORS FREQUENCY OR LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES (L)

| PREVENTION LEVEL | |
|---|---|
| Reactivity insertion (including ATWS) | - By on-line refuelling and by limiting of reactor power change capability minimizing of excess reactivity (L)<br>- By reduced heavy metal loading limitation of reactivity effect due to water ingress (L)<br>- Limitation of control rod speed and of number of KLAK columns to be discharged simultaneously (L) |
| Disturbances MHTS: | - Level of quality and redundancy to ensure high level of availability, no safety relation (R) |
| Leakages/ruptures MHTS (secondary): | - Level of quality to ensure high level of availability, no safety relation (R) |
| Loss of primary pressure: | - Very high quality level for pressure vessel unit (PWR-quality) (R)<br>- All connecting pipes with diameters below 65 mm (L)<br>- Most of them with two isolation valves in series (one automatically closed by safety system) (S)<br>- Filtered venting of the reactor building (L) |
| Steam generator leakages: | - High quality level and repeated inspection of steam generator bundle (R)<br>- Automatic isolation and discharge (L)<br>- Evacuation of water entered in the primary system by the water separator of the helium purification system (L) |
| Total loss of power supply (station black out): | - Auxiliary Grid (L)<br>- Diesels (L)<br>- Batteries (L) |
| **PROTECTION LEVEL** | |
| The following three protection actions are performed in all cases of disturbances<br>• Dropping of reflector rods<br>• Shut down of the primary circuit blower<br>• Steam generator isolation | |
| Reactivity insertion: | - No additional protection measures required, cavity cooling system running on-line (L)<br>- In case of reflector rods failure inherent reactor shut down by negative temperature coefficient for all physically possible reactivity insertions without fission product release and component damages (L) |
| Disturbances MHTS: | - No additional protection measures required, cavity cooling system running on-line (L) |
| Leakages/ruptures MHTS (Secondary): | - No additional protection measures required cavity cooling system running on-line (L) |
| Loss of primary pressure: | - Primary circuit isolation valves additionally closed, cavity cooling system running on-line (L) |
| Steam generator leakages: | - Steam generator additionally quickly drained, cavity cooling system running on-line (L) |
| Total loss of power supply (Station black out): | - No additional protection measures required,<br>Passive decay heat removal by naturally governed heat transfer,<br>No fission product release from the fuel elements (max. fuel temp. below 1600°C ) (L)<br>- Connection of mobile water and power supply facilities possible, if performed within 15h no component damages (L) |

TABLE 6.7.4. DESIGN FEATURES FOR MITIGATION LEVEL OF THE MHTR

| **Design Basis Accidents** | Systems/Design features | Passive/active | Remarks |
|---|---|---|---|
| Fission Product Retention | • Coated particle<br>• Pressure vessel unit | Passive<br>Passive/active | Inherently safe fission product retention within the fuel element<br>- For minimizing purposes<br>- Valves fail safe<br>- Further mitigation by filtered venting of the reactor building<br>- In case of non-isolatable leakages unfiltered release of coolant to environment via stack, no exceeding of limiting values of Art. 28.3 StrSchV. |
| Reactivity control | • Reflector rods<br>• Small absorber spheres (KLAK)<br>• Negative temperature coefficient | Active<br>Active<br>Passive | Fail safe (gravitationally driven)<br>Fail safe (gravitationally driven)<br>Inherent |
| Decay heat removal | • Main heat transfer system<br>• Cavity cooling system | Active<br>Active/passive | |
| Primary coolant inventory | Pressure vessel unit | Passive/active | - Valves fail safe<br>- The system presents air ingress |
| Steam generator, tube rupture, water ingress | • Staggered arrangement of SGPV and RPV<br>• Steam generator isolation and relief<br>• Throttle in steam generator bundle entry | <br><br>Active<br>Passive | No overflow of water from SGPV to RPV possible<br><br>Limitation of water/steam ingress<br>Limitation of Water ingress<br>Graphite corrosion negligible small |
| **Severe Accidents** | | | |
| Fission product retention | Same systems and design features as for design basis accidents | Passive/active | Same safety features as for design basis accidents |

TABLE 6.7.4. (cont.)

| | | | |
|---|---|---|---|
| Extreme insertion of reactivity | Negative temperature coefficient | Passive | - Perturbation initiated by the malfunction of the reflector rods or of the small absorber spheres' system (KLAK)<br>- Inherent reactor shut down by negative temperature coefficient limits the maximum core temperature to < 1300°C<br>- Ejection of the shut down devices not possible due to their integration within the pressure vessel |
| Decay heat removal without CCS | • Naturally governed heat transfer | Passive | Inherent decay heat removal from RPV to RB structures finally to the environment via conduction, convection and radiation |
| | • Provision of water and power supply to CCS from outside (mobile devices) | Active | - If performed within 15h no component damages, if not performed within 15h partial exceeding of concrete design temperatures possible but no danger of loss of structural load capacity<br>- In all cases maximum fuel temperature below 1600°C |
| Important air ingress (i.e. rupture of CPV) | No systems installed | | - Graphite corrosion conditioned by the naturally limited air flow through the core<br>- Maximum fuel temperature keeps below 1600°C<br>- No significant fission product release<br>- Access to the reactor building possible<br>- Air ingress can be stopped by simple accident management measures, available time at least some days |
| Important water ingress (i.e. multiple ruptures of steam generator bundle tubes) | No additional systems installed<br>Staggered arrangement of SGPV and RPV | | - No transfer of water from the steam generator to the higher located core possible<br>- Amount of steam transferred to the core is limited to about 3300 Kg due to the maximum possible evaporation by the heat stored in the steam generator structures<br>- The total reaction of the maximum possible steam with core graphite does not lead to an inadmissable fission product release |

Abbreviations to Table 6.7.4.: CCS-Cavity Cooling System, CPV-Connecting Pressure Vessel
KLAK-Small absorber spheres shut down system, RB-Reactor Building
RPV-Reactor Pressure Vessel, SGPV-Steam Generator Pressure Vessel
StrSchV - Strahlenschutzverordnung (German Radiation Protection Ordinance)

### 6.7.4. Design Data Questionnaire

## I. GENERAL INFORMATION

1. Design name: Modular High Temperature Reactor
2. Designer/Supplier address:
   HTR-GmbH (ABB-AG/Siemens - AG(KWU))
3. Reactor type: MHTR Number of modules/per plant: 2 or more
4. Gross thermal power (MW-th) per reactor: 200 MW(th)
5. Net electrical output (MW-e) per reactor: 85.5 MW(e)
6. Heat supply capacity (MW-th): 202 MW(th)

## II. BASIC DESIGN DESCRIPTION

## A. CORE AND REACTIVITY CONTROL (1 Reactor)

7. Fuel material and chemical form: $UO_2$
8. Fuel inventory (tones of heavy metal): 2396 t
9. Average core power density (kW/liter): 3 kW/lt
10. Average discharge burnup (MWd/t): 80 000 MWd/t
11. Enrichment at the equilibrium (Wt%): 8%
12. Refueling frequency (months): Continuous fuel recycling (15 times)
13. Type of refueling (on/off power): On power
14. Fraction of core withdrawn (%): Recycled: 1.4%/d, discharged: 0,1%/d
15. Moderator material: Graphite
16. Active core dimensions (m): Diam.: 3 m, Height.: 9.4 m
17. Type of fuel element: Spherical (Ø 6 cm)
18. Number of fuel elements: 360 000
19. Number of control rods: 6 (moved in lateral reflector)
20. Control rod material: Stainless steel
21. Additional shutdown systems: Small absorber spheres in lat. refl. (KLAK)
22. Control rod neutron absorber material: B4C
23. Burnable poison material and form: No

## B. REACTOR COOLANT SYSTEM

### B1. Coolant

24. Coolant medium and inventory: Helium 2700 kg
25. Design coolant mass flow through core (kg/s) at 100% power: 85 kg/s
26. Cooling mode (forced/natural): Forced
27. Operating coolant pressure (bar): 60 bar
28. Core inlet temperature (°C): 250°C
29. Core outlet temperature (°C): 700°C

### B2. Reactor pressure vessel

30. Overall height of assembled vessel (m): 25 m
31. Vessel diameter (m): 6.1 m
32. Vessel thickness (mm): 118 mm
33. Vessel material: 20 MnMoNi5 5
34. Design pressure (bar): 70 bar
35. Gross weight (tonne/kg): 790 t

### B3. Steam generator

36. Number of steam generators: 1
37. Type: Helical coil
38. Configuration (horizontal/vertical): Vertical
39. Tube material: X10 Ni Cr Al Ti 3220 (Incoloy 800)
40. Heat transfer surface per steam generator ($m^2$): 2100 $m^2$
41. Thermal capacity per steam generator (MW): 202 MW
    Steam Production (t/h): 277 t/h
42. Feed water pressure (bar): 210 bar

43. Feed water temperature (°C): 170°C
44. Steam pressure (bar): 190 bar
45. Steam temperature (°C): 530°C

**B4. Main coolant pumps/circulators**

46. Number of pumps: 1
47. Type: 1 stage blower
48. Pump mass flow rate (kg/s): 85 kg/s
49. Pump design rated head: 1.5 bar
50. Pump nominal power (kW): 2950 kW

## C. CONFINEMENT

51. Type: Vented reactor building
52. Overall form: Cubic
53. Structural material: Concrete
54. Liner: No
55. Simple/double wall: Simple
56. Dimensions (length, width, height) (m): 46 m, 36 m, 51 m
57. Design pressure (bar): 1.3. bar
58. Design temperature (°C): 30°C
59. Design leakage rate (% per day): 50 vol.-%/d

## III. SAFETY RELATED SYSTEMS

## A. DESIGN CONDITIONS

**A1. Fission product retention**

61. Coated particle: Leak-tight up to 1600°C
62. Pressure vessel unit: With isolation valves, for minimizing purpose filtered venting of the reactor building

**A2. Reactivity control**

63. Normal shut down system (Y/N): Yes (Control rods moved in lateral reflector)
    a. Maximum control rod worth (pcm): 3.2. + 0.4%
    b. Mode of operation (active/passive): Shut down by gravitation
    c. Redundancy: Yes
    d. Safety graded: Yes
64. Longterm shut down system (Y/N): Yes
    Small absorber spheres dropped in lateral reflector (KLAK)
    a. Absorber material: Boron
    b. Mode of operation: By gravity (pass.)
    c. Redundancy (Y/N): Yes
    d. Safety graded (Y/N): Yes

65. **A3. Decay heat removal**
    *A3-1 MHTS Primary side at normal operation*
    a. Actuation mode (manual/automatic): Manual or automatic
    b. Pressure level (bar): 60 bar
    c. Flow rate (kg/s): about 10-20 kg/s
    d. Mode of operation (active/passive): Active
    e. Redundancy: No
    f. Safety graded: Yes

    *A3-2 MHTS Secondary side at normal operation*
66. Feed water
    a. Actuation mode (manual/automatic): Manual or automatic
    b. Flow rate (kg/s): Variable
    c. Mode of operation (active/passive): Active
    d. Redundancy: Yes
    e. Self sufficiency (h): Unlimited
    f. Safety graded: Yes

*A3-3 Second Decay Heat Removal System*

67. Implemented System (Name): Cavity cooling system/Surface coolers
Actuation mode: Permanently on-line
Pressure level (bar): 5 bar
Flow rate (kG/s): 27 kg/s
Mode of operation: Active
Redundancy: 3 X 100%
Safety graded: Yes

**B. SEVERE ACCIDENT CONDITIONS**

**B.1 Fission products retention**

68. Confinement Filter System (Y/N): Not required
Reason:
- Coated fuel particles leak-tight up to 1600°C
- Pressure vessel unit provided with isolation valves (retention of fission product inventory in primary helium at normal operation) and with safety valves (for overpressure protection, release into confinement)
- For minimizing purposes non-safety graded filtered venting of the reactor building confinement provided

**B.2 Recriticality control:**
Not relevant

**B.3 Long term heat removal**

69. Implemented system: Cavity cooling system/surface coolers
   a. Mode of operation (A/P): Active on secondary side/passive on primary side
   b. Self sufficiency (h): Unlimited
   c. Safety graded (Y/N): Yes
70. Ultimate heat sink in case of non-availability of the cavity cooling system: Heat transfer from RPV to cavity and reactor building structures and finally to environment by conduction, convection and radiation.
   a. Self sufficiency (h): Unlimited
   b. Safety graded (Y/N): Inherent

**B.4 Combustible gas control**: Not relevant

**B.5 Confinement pressure control**: Not relevant (Confinement with non-safety-graded filtered venting for minimizing purposes)

C. SAFETY RELATED I&C SYSTEM

71. Automatic load following (Y/N): Yes
   * range (% power): 100% - 50% - 100%
   * maximum rate (%/min)

   Load rejection without reactor trip (Y/N): Yes
   Full Cathode Ray Tubes (CRT) display (Y/N): Yes
   Automated start-up procedures (Y/N): Yes
   Automated normal shutdown procedures (Y/N): Yes
   Automated off normal shutdown procedures (Y/N): Yes
   Use of field buses and smart sensors (Y/N): Yes
   Expert systems or artificial intelligence advisors (Y/N): Yes
   Protection system backup (Y/N): Yes

**D. EMERGENCY POWER SUPPLY SYSTEM**

72. Type (diesel, gas, grid connection): Diesels, Batteries, Auxiliary Grid
73. Number of trains: 2

**E. AC/DC SUPPLY SYSTEM**

74. Type (rectifier, converter, battery): Rectifier, converter, batteries
75. Estimated time reserve (hr): 220V: 2h/24V: 15h

**IV. CONVENTIONAL THERMAL CYCLE**

**A. TURBINE SYSTEM**

76. Type: Steam turbine
77. Overall length (m): ≈ 20m
78. Width (m): ≈ 8m
79. Number of turbines/reactor: 1
80. Number of turbine sections per unit: HP/MP/LP
81. Speed (rpm): 3000 $min^{-1}$

## B. STEAM CHARACTERISTICS

82. H.P. inlet pressure: 180 bar
83. H.P. inlet temperature: 525°C
84. H.P. inlet flowrate: 154 kg/s
85. L.P. inlet pressure
86. L.P. inlet temperature
87. L.P. inlet flowrate (per section)

Dependent from demands of consumer

## C. GENERATOR

88. Type: 3 phase synchronous
89. Apparent power (MVA):
90. Active power (MW):

Nominal active power 62,9 MW, dependent from consumers demand

91. Frequency (Hz): 50 Hz
92. Output voltage (kV): 110 kV
93. Total generator mass (t)
94. Overall length
95. Stator housing outside diameter

Not yet fixed in detail

## D. CONDENSER

96. Number of tubes
97. Heat transfer area
98. Flowrate ($m^3/s$)

Not yet fixed in detail

99. Pressure (bar): ≈ 0.1 bar
100. Temperature (°C): ≈ 45°C

## E. CONDENSATE PUMPS

101. Number: 2
102. Flowrate
103. Developed head
104. Temperature
105. Pump speed

Not yet fixed in detail

### 6.7.5. Project Status

The total plant has been extensively designed. Detailed plant descriptions are given in the safety report dated from November 1988, reviewed in 1988/89 by advisory experts within the conceptual licensing procedure (see below).

Design of the nuclear steam generating system is well progressed up to structural and mechanical engineering. The system behaviour under accidental conditions has been analyzed for design basis accidents and furthermore for events assigned far to the hypothetical realm.

The design of the reactor auxiliary facilities, machines and electric systems has at least reached the status of system and process engineering. As the final design step, the remaining structural and mechanical engineering is to be done.

#### *6.7.5.1. Entities involved*

The development described above of the HTGR in the FRG was managed by HTR G.m.b.H (a subsidiary of ABB and Siemens) on the industrial side carried out jointly by INTERATOM (a subsidiary of Siemens) and HRB (a subsidiary of ABB). Substantial R&D work was contributed by the national Research Center KFA Juelich. It is important to note that the HTR-module concept could draw to a very large extent on the experience gained with AVR and THTR and an extensive long term HTR development program carried out by the system company, various component industries, engineering companies and universities.

The entire program was supported by the Federal Ministry for Research and Technology.

#### *6.7.5.2. Design status*

In 1990 the Basic Design as described above was essentially completed, except for certain R&D tasks and certain proof testing of components. Intensive discussions have been held with potential customers for a first of its kind cogeneration HTR plant. Intensive discussions have also been held with regulatory agencies resulting in a positive statement as to the licensability of the HTR in the FRG (see 6.5.4). According to the definitions selected by IAEA for this report a D2-Status has been reached already.

#### *6.7.5.3. R & D work*

Substantial R&D work has been completed to prove inter alia:

- the fission product retention capability of the coated fuel particles
- the function of the fuel element conveying system
- the reliability of the small absorber sphere shut down system (KLAK)
- the magnitude of mechanical loads of the graphitic core internals e.g. due to thermal elongations within permissible limits
- related qualification of the graphite
- the reliable measuring of the burnup status of recycled fuel elements.

Experience concerning e.g. specific helium processing technologies has been given by the operation of the THTR 300.

Still outstanding R & D tasks are:

- qualification testing of control rod drives
- development of an improved on-line burn-up measurement system
- demonstration tests for an improved fuel discharge facility

*6.7.5.4. Licensing status*

In April 1987 the companies Siemens/KWU and Interatom applied in the West German state of Lower Saxony for a non-site-specific licensing procedure to be initiated for an HTR-Module nuclear power plant for the cogeneration of electricity and process steam/district heat. The application was docketed by the Lower Saxon Ministry for Environment as licensing authority. The licensing board nominated as main expert the "Technischer Überwachungsverein Norddeutschland (TÜV Hanover)" to review and assess the safety concept. In addition the Reactor Safety Commission (RSK), the Reactor Safety Association (GRS) and the Civil Engineering Institute of the University of Brunswick (IBMB) were involved in the review procedure. The review reports of the GRS and IBMB were completed in February 1989 and March 1989, respectively. The final review report of the TÜV Hanover was completed in July 1989. The RSK reported its recommendations in April 1990 in the Bundesanzeiger published by the Federal Minister for Justice.

This can be summarized in the statement that the concept of the HTR-Module fully corresponds to the state of the art of science and technology and fully meets the safety-related requirements to be posed on nuclear power plants in the Federal Republic of Germany. Within the frame of the review of risk minimizing measures the experts especially recognized the high degree of inherent safety margins.

In terms of the terminology selected for this report it can be stated that L2 was successfully completed. A formal licensing procedure for a definite site can only be taken up with a firm commitment for building a NPP with an MHTR at a defined site.

*6.7.5.5. References*

a. Pebble ped reactors: operated or under construction

   - AVR (Germany) 1966 - 1988; 15 MWe. Operated successfully for 22 years at core helium outlet temperatures up to 950°C. Demonstrated successfully the loss of coolant flow accident and the core heatup as well as the naturally governed heat dissipation to the environment.
   - THR 300 (germany) 1985-1988; 300 MWe. Operated successfully at full power, had to be shutdown for political reasons as a consequence of the Chernobyl accident.

b. Other High Temperature Reactors (Research and Power Reactors): Operated or under construction.

- Dragon (England) 1964 - 1976, research reactor operated up to 950°C
- Peach-Bottom (USA) 1967-1974
- Fort St. Vrain (USA) 1979-1989
- HTTR (Japan): test reactor under construction
- MHTR 10 (China): test reactor under construction

### 6.7.6. Economics

The HTR-Module is suitable to supply with high efficiency electrical power, process heat, process steam and district heat.

The small power size of one HTR-Module unit (200 MWth) and the low power density leads principally to high specific investment costs compared to bigger reactors. However, this effect is compensated to a large degree by technical simplifications, (e.g. by omission of a full pressure containment and active safety systems). Additionally the high proof and control effort for nuclear components can be reduced drastically due to the fact that most components of the HTR-Module (e.g. water steam cycle, He-blower) are not safety relevant. The proof effort could be restricted mainly to the fuel elements. A further cost reduction can be reached by manufacturing in larger quantities of modular units.

Investigations showed, that compared to fossil-fired power plants of equal size large HTR-Module power plants (about 8 modular units) can compete in Germany even with the currently low-cost imported hard coal. The necessary surplus costs for a market introduction of the HTR-Module (for a 2 modular lead plant, establishment of a fuel element supply, and FOIK costs until the series production of HTR-plants) are assessed to be regained by the operation of about three to eight HTR-8 Module plants over a period of 20 years.

Smaller HTR-Module plants can become competitive by the economical attractive cogeneration of power and steam or heat e.g. for refineries, chemical industry, district heating grids, tertiary oil recovery and more.

# 7. DESIGN DESCRIPTIONS FOR REACTORS IN THE CONCEPTUAL DESIGN STAGE

## 7.1. REACTOR SYSTEM DESCRIPTION AND DEVELOPMENT STATUS OF BWR-600

### 7.1.1. Basic objectives and features

Siemens, in cooperation with Germany's electric utilities, is developing a design concept for an innovative boiling water reactor plant with a net capacity of approximately 750 MWe.

In this design concept, the primarily "active", highly redundant safety equipment of today's operating plants are replaced by "passive" safety equipment. These function according to basic laws of physics such as gravity, natural convection and evaporation. For this reason they have no need of "active" energy sources such as a permanent power supply.

The main goal of this development work is an enhanced safety concept in which functional capability and reliability are ensured by simple and less sensitive safety equipment. In this way the effects of human error are to be diminished, reactor safety is to be improved even further, and capital cost as well as maintenance reduced.

The development goals specified for designing this innovative boiling water reactor - such as passive or inherent system characteristics, good operating and accident control behavior, and ease of operation both under normal operating conditions and during accidents - have resulted in a concept characterized by the following key features:

- Cooling of the reactor core when the plant is in the shutdown condition or following the occurrence of abnormal events is reliably assured by making use of the natural force of gravity. Provision of a large water inventory inside the reactor pressure vessel as well as of a large source of water inside the containment makes active, fast-response safety equipment, pumps and electric power unnecessary in the event of disturbances in the reactor coolant system.

- The safety systems all operate by passive means. The provision of diverse valve designs as well as the actuation of such valves using system fluid or stored mechanical or hydraulic energy serve to supplement their effectiveness in strategic areas.

- A drop in the water level inside the reactor pressure vessel initiates automatic depressurization, allowing core flooding systems that operate according to the principle of gravity flow to be activated and preventing core melt scenarios from occurring at high reactor pressure levels. Furthermore, facilities are provided for retaining and cooling a molten core.

- Although the occurrence of a core melt accident is not to be realistically expected, the containment is nevertheless protected against such an event by being designed with sufficient capacity to accommodate any hydrogen generated by a zirconium-water reaction as well as by being inerted during plant operation.

- Except for the fact that the mode of reactor cooling has been changed from forced circulation to natural circulation (reactor water recirculation pumps have been eliminated), all other systems and components employed for plant operation are based on the extensive operating experience gained from the boiling water reactor plants currently in service in Germany as well as on the proven system and component designs implemented in these plants.

### 7.1.2. Design description

#### *7.1.2.1. Nuclear steam supply system*

The nuclear steam supply system is located inside the reactor building and is surrounded by a concrete containment with steel liner.

The main dimensions (22.38 m long by 7.55 m I.D.) of the reactor pressure vessel correspond to those of the reactor pressure vessel of Kruemmel Nuclear Power Station (KKK). This means that a comparatively large water inventory is available above the core.

The core also has the same number of fuel assemblies (840) and control rods (205) as Kruemmel. The active height of the core was reduced in accordance with the smaller capacity rating.

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 providet by rupture disks. Once the reactor pressure has been sufficiently reduced, water is able to flow by gravity into the RPV from an elevated pool, the core flooding pool.

For reducing the buildup of pressure occurring inside the drywell in the event of a loss-of-coolant accident (LOCA), the BWR-600 does not have vent pipes of the kind employed in German BWR plants to date, but is provided instead with vertical vent shafts located outside the pressure suppression chamber which terminate at the bottom in horizontal vents discharging into the pool water. The horizontal routing of the relief lines and the location of the vent shafts outside the suppression chamber air space both constitute a further enhancement of safety provided by the BWR-600 design concept since these configurations rule out the possibility of steam being able to leak into the chamber air space.

The reactor is shut down either by driving the control rods into the core using the control rod drives, by rapidly inserting all of the control rods using a hydraulic system or by rapidly injecting boron into the reactor water. The accumulators of this system, which are filled with a boron solution, perform two functions: on the one hand, they serve as a redundant backup to the water tanks of the scram system for rapid insertion of the control rods and, on the other hand, they provide a diverse means for stopping the chain reaction in the core by injecting the boron solution into the reactor water.

The main steam and feedwater lines is equipped with three system-fluid-operated isolation valves, one of every three being of a different design from that of the other two valves. Closure of the main steam valves to isolate the containment is based on the fail-safe principle employing solenoid pilot valves or through passive actuation.

#### *7.1.2.2. Balance of plant systems*

Two main steam lines lead from the RPV to the turbine. Installed parallel to the turbine is a bypass station designed to be able to dump the entire volume of steam being generated inside the reactor directly to the condenser without the steam passing through the turbine. 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 cooled with hydrogen The main steam from the reactor is admitted to the high-pressure (HP) section of the turbine via combined stop and control valves After undergoing partial expansion in the HP turbine section, the steam is passed 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 blading down to the condenser pressure of approximately 0 04 bar, this pressure level being based on the use of a one-through circulating water system supplying water having a mean temperature of about 11°C The condensate collected in the hotwells of the three condensers is discharged into a common header supplying the main condensate pumps These pump the condensate through a demineralizing system before it is heated to the final feedwater temperature in the feedwater heaters The feedwater collected in the feedwater tank downstream of the heaters is pumped to the reactor through two feedwater lines by two variable-speed feedwater pumps Each of the two main steam lines and each of the two feedwater lines are equipped with three isolation valves (one of which is of diverse design) The first valve is located directly at the RPV, while the other two valves are installed immediately outside the containment The first of the two outboard isolation valves is connected to the containment in such a way as to rule out any leakage occurring between the containment penetration and this valve Compared to large-capacity plants of traditional design, this represents a further enhancement in the fulfillment of the safety requirements imposed with regard to containment isolation

#### *7.1.2.3. Instrumentation, control and electrical systems*

The safety concept of the BWR-600 is characterized by an extensive use of passive equipment for accident control This equipment requires neither a power supply nor activation by instrumentation and control (I & C) systems

Certain active components and systems carry out functions which are likewise utilized for accident control Compared to the power reactors presently in operation, the role played by this equipment in assuring plant safety has been greatly reduced

For these reasons it is intended for the BWR-600 to employ components that have already proven their reliability in operation

Electrical Systems

The auxiliary power supply system and the connection to the offsite power system feature state-of-the-art technology that has already proven itself under service conditions The normal power supply system is of two-train configuration and is designed for voltage levels of 10 kV, 690 V and 400 V Emergency power is provided by two trains equipped with diesel generators and designed to supply 690 V and 400 V The direct current supply system also comprises two battery systems with sufficient capacity to supply power for two hours An additional battery system, with sufficient capacity for seven days, is provided for the passive safety features to power accident recording and monitoring equipment

I & C Systems

In view of the change in the significance of I & C systems with regard to plant safety, the three-train approach implemented in the past at large-capacity BWRs has been reduced for the BWR-600

In the case of circuits based on the de-energize to trip principle, a configuration of four sensors designed for two times 1-out-of-2 actuation has been proposed. The resulting output signals act on the pilot valves of system-fluid-actuated main valves. In the case of circuits operating according to the energize to trip principle, a configuration of two sensors per train is planned which would be designed for 2-out-of-2 actuation. The output signals in these cases would act on the dual-redundant 100-%-capacity process systems.

*7.1.2.4. Safety considerations and emergency protection*

The primary objective followed in developing the BWR-600 is to enhance the quality of safety by introducing additional passive systems for performing safety-related functions in the event of transients or accidents. Compared to the reactors of today, the technology employed in these systems is much simpler, operation of the equipment being independent of a power supply and activation by I & C systems.

Passive systems are characterized by the fact that they utilize the laws of nature (e.g. gravity) to perform their safety functions and dispense with active components (e.g. pumps and drives). The supply of coolant to the depressurized reactor by gravity flow from an elevated water pool is a classic example of a passive system.

The most frequent events requiring control of reactor coolant inventory and residual heat removal comprise anomalies in plant operation, so-called transients. In the event of a LOCA, heat from the reactor is discharged as steam to the containment atmosphere via a postulated primary system pipe break. In order to prevent core uncovery, passive systems for supplying reactor water makeup must operate.

The following safety functions must be assured in the case of most transients as well as in the event of accidents:

- Reactor scram
- Containment isolation
- RPV pressure relief and depressurization
- Heat removal from the RPV
- Reactor water makeup and control of core coolant inventory
- Heat removal from the containment.

A short description now follows of the passive systems planned for these tasks.

Reactor Scram, Containment Isolation, Pressure Relief and Depressurization

For the performance of these safety functions, it is planned for the active systems to be additionally activated by passive equipment.

The switching operations required for these safety functions are carried out by passive and diverse means without electric power, actuating fluids or I & C signals as follows. The RPV is connected via a non-isolatable line to a heat exchanger which acts as a passive pressure pulse transmitter. When the water level in the RPV is normal, the tubes inside the heat exchanger are filled with water and do not transfer any heat. If the water level in the RPV starts to drop, the water drains from the heat exchanger tubes and is replaced by steam that is continuously condensed. The heat transferred during this process causes a buildup of pressure on the shell side of the heat exchanger This rise in pressure automatically and passively actuates pilot valves, which in turn initiate the safety functions of reactor scram, containment isolation and RPV depressurization

Reactor and Containment Heat Removal and Makeup of Reactor Coolant Inventory

A passive safety feature of particular importance, especially for controlling transients, are the four emergency condensers (each rated for 63 MW at a pressure of 70 bar) which are located in the core flooding pool and are connected to the RPV by non-isolatable steam discharge and return lines The circuit of each emergency condenser contains an anti-circulation loop so that practically no circulation of condensate takes place through the open lines to the reactor during normal plant operation Only when there has been a drop in the RPV water level does steam enter the condenser, with the resulting condensate being returned to the RPV

The heat removed by the emergency condensers is transferred to the water of the core flooding pool and slowly raises its temperature It takes over 12 hours for the water in the pool to reach a temperature at which it starts to evaporate The resulting steam then causes a buildup of pressure inside the containment

In the event of a LOCA, steam or flashing water is discharged into the containment atmosphere To prevent core uncovery in such a situation, passive systems are activated for supplying reactor water makeup For this, the RPV must first be depressurized The large water inventory inside the RPV enables this to be done without the core becoming uncovered Water from the elevated core flooding pool can then discharge to the RPV by gravity flow via four supply lines and self-actuated check valves The core flooding pool contains a water inventory of approximately 5000 $m^3$ This volume of water is sufficient after the occurrence of a LOCA to fill both the RPV and the drywell of the containment up to a level which is then equal to that in the core flooding pool, this level being situated above the feedwater nozzles on the RPV This not only provides a water cover over potential pipe breaks but also ensures effective cooling of the RPV exterior

At later stages in the accident sequences following both transients and LOCAs, the generation of steam can lead to a rise in temperature and pressure inside the drywell In order to condense this steam and thus limit containment temperature and pressure, containment cooling condensers are provided These discharge their condensate to the core flooding pool while the heat that they remove is rejected to the water in the dryer-separator storage pool situated above the containment A closed, passive cooling circuit for residual heat removal is thus available inside the drywell

The containment cooling condensers provide the plant with a grace period of up to seven days before there is any need for external intervention After this period it will be necessary to make up the water inventory of the dryer-separator storage pool outside the containment, something which can be effected by simple actions

The characteristics of these various safety features enable the reactor to stabilize itself in the event of a transient or LOCA without the need for open- or closed-loop control equipment or external actuation (e g by emergency power or compressed air) Existing active systems required for normal plant operation supplement the passive features provided for accident control

*7.1.2.5. Buildings and structures*

The systems and components are arranged inside the various plant buildings and structures such as to create three clearly delineated structural complexes, something which enables the buildings to be constructed in parallel at the same time.

The structural complex at the center of the plant comprises the reactor building and the turbine building. The second complex contains the switchgear equipment, the systems for radioactive waste treatment and storage, the hot workshop, staff amenities, the main control room and the entrance to the controlled access area. The third structural complex consists of the plant service systems: the circulating water supply systems, the emergency diesel generators, the cold workshops and the demineralized water system.

### 7.1.3. Safety concept

TABLE 7.1.1. MAIN SAFETY RELATED SYSTEMS IN THE BWR-600

| Name | Safety graded | Main characteristics |
|---|---|---|
| Nuclear steam supply system | Y | reactor vessel, natural circulation |
| Scram system | Y | 205 control rods, rapidly inserting hydraulic system |
| Fast-acting boron injection system | Y | 2 accumulators with boron solution |
| Safety-relief valves | Y | 8 valves+8 rupture discs |
| Emergency condenser (passive) | Y | 4x25%, 250 MW, 138 $m^2$ |
| Core flooding pool (passive) | Y | 4 flooding lines to RPV, approx. 5,000 $m^3$ |
| Containment cooling condenser (passive) | Y | 4x25%,12 MW, heat sink: dryer-separator storage pool outside containment |
| Passive pressure pulse transmitter | Y | heat exchanger non-isolatable connected to RPV, actuates pilot valves by pressure rise |

TABLE 7.1.2. MAIN ACCIDENT INITIATORS FOR THE BWR-600

- LOCA: Leak from bottom head of the RPV
- LOCA: Pipe breaks inside containment
- LOCA: Pipe breaks outside containment
- ATWS
- Loss of the main heat sink
- Loss of normal feedwater supply
- Closure of the main steam isolation valves
- Stuck-open safety-relief valve

TABLE 7.1.3. DESIGN FEATURES FOR PREVENTION AND PROTECTION LEVELS THAT REDUCE (R), SUPPRESS (S), THE INITIATORS FREQUENCY OR LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES (L)

| | |
|---|---|
| **PREVENTION LEVEL** | |
| LOCA | - Reduced vessel fluence (R)<br>- RPV connections mainly in steam area (L)<br>- Basic safety of pressurized components (R)<br>- Automatic depressurization (L)<br>- Passive pressure control (R)<br>- Automatic containment isolation (L) |
| ATWS | - Diverse scram system (R) |
| Transients | - slow-acting accident control capabilities by large water inventory inside RPV (R) |
| Loss of electric sources | - All safety systems passive, no need of electrical power (S) |
| Total loss of the cold source (Water) | - Passive ultimate Decay Heat Removal (S) |
| Total loss of feedwater | - Active or passive scram, passive Decay Heat Removal by emergency condenser (L) |
| Station Blackout | - All safety systems passive, no need of electrical power (S)<br>- Passive ultimate Decay Heat Removal (S) |
| **PROTECTION LEVEL** | |
| LOCA | - Increased water inventory above core<br>- Passive coolant makeup by gravity discharge from core flooding pool<br>- Passive pressure pulse transmitter for actuating pilot valves<br>- Deverse containment isolation valves<br>- Inerted containment atmosphere |
| Transients | - Good accident control behavior through slower reaction to off-normal conditions<br>- Control of transients without coolant makeup<br>- Passive activation of safety-related functions |
| Loss of Electric Sources | - Passive accident control over a time period of approx 7 days without electric power, control system intervention or manual operator action |
| Total loss of heat sink | - Passive ultimate Decay Heat Removal |
| Total loss of feedwater | - Passive control by emergency condenser without coolant makeup |

TABLE 7.1.4. DESIGN FEATURES FOR MITIGATION LEVEL OF THE BWR-600

| Safety Functions | Systems | Passive/Active | Design features/ Remarks |
|---|---|---|---|
| **Design Basis** | | | Concrete containment<br>possibly with fibrous concrete |
| Fission Product Containment | Containment with liner | Passive | |
| Coolant inventory | Gravity discharge from core flooding pool<br>Control rod drive cooling system<br>Operational RHR system | Passive<br><br>Active<br>Active | Safety system<br><br>Non-safety system<br>Non-safety system |
| Decay Heat Removal | Emergency condenser<br>Containment cooling condenser<br>Operational RHR system | Passive<br>Passive<br>Active | Safety system<br>Safety system<br>Non-safety system |
| Reactivity control | Control rod<br><br>Fast boron injection | Active<br>(with additional passive actuation)<br>Active | Additional actuation by passive pressure<br>pulse transmitter<br>No passive actuation |
| Pressure control of NSSS | Automatic Depressurization System<br>Rupture discs | Active<br>(with additional passive actuation)<br>Passive | Additional actuation by passive pressure<br>pulse transmitter |
| **Severe Accident** | | | |
| Containment temperature and pressure control | Containment with liner<br>Containment cooling condenser | Passive<br>Passive | Heat transfer to dryer-separator storage pool outside containment |
| Heat removal | Containment cooling condenser<br>Operational RHR system | Passive<br>Active | <br>Non-safety system |
| Tightness control | Containment | Passive | |
| Inflam. gas control | Inerted containment | Passive | design considers pressure loadings of hydrogen production |
| Fission product containment | Containment<br>Flooded RPV leak | Passive<br>Passive | Flooding of drywell out of core flooding pool |
| Corium management | RPV outside cooling<br>Core catcher | Passive<br>Passive | Flooding of drywell out of core<br>Flooding pool |

## 7.1.4. Design Data Questionnaire for BWR600

### I. GENERAL INFORMATION

1 Design name BWR-600
2 Designer/Supplier address Siemens AG
3 Reactor type Integral PWR BWR
Number of modules/per plant 1
4 Gross thermal power (MW-th) per reactor 2200 MW-th
5 Net electrical output (MW-e) per reactor 750 MW
6 Heat supply capacity (MW-th) on demand

### II. BASIC DESIGN DESCRIPTION

### A. CORE AND REACTIVITY CONTROL

7 Fuel material $UO_2$
8 Fuel inventory (tones of heavy metal) 109 t (heavy metal)
9 Average core power density (kW/liter) 42 kW/liter
10 Average fuel power density (kW/kgU) 20 kW/kg U
11 Maximum linear power (W/m) 10,300 W/m
12 Average discharge burnup (MWd/t) 65,000 MWd/t
13 Initial enrichment or enrichment range (Wt%) 4 8% (mean value)
14 Reload enrichment at the equilibrium (Wt%) ---
15 Refueling frequency (months) 24 months
16 Type of refueling (on/off power) off power
17 Fraction of core withdrawn (%) 20%
18 Moderator material and inventory 335,000 kg of light water
19 Active core height (m) 2 8 m
20 Core diameter (m) 5 265 m
21 Number of fuel assemblies 840
22 Number of fuel rods per assembly 91
23 Rod array in assembly 305x305 mm
24 Clad material Zry-4
25 Clad thickness (mm) 0 605 mm
26 Number of control rods or assemblies 205
27 Type cruciform
28 Additional shutdown systems Fast-acting boron injection system
29 Control rod neutron absorber material $B_4C$
30 Soluble neutron absorber boric acid
31 Burnable poison material and form Gadolinium

### B. REACTOR COOLANT SYSTEM

**B1. Coolant**

32 Coolant medium and inventory 335,000 kg of light water (RPV only)
33 Design coolant mass flow through core (kg/s) 8,000 kg/s
34 Cooling mode (forced/natural) natural
35 Operating coolant pressure (bar) 71 bar
36 Core inlet temperature (C) 174 °C (feedwater inlet)
37 Core outlet temperature (C) 287 °C

**B2. Reactor pressure vessel/tube**

38 Overall length of assembled vessel/tube (m) 22 38 m (inside)
39 Inside vessel/diameter (m) 7,550/6,780 mm
40 Average vessel/tube thickness (mm) 182/163 mm
41 Vessel/tube material 20 MnMoNi 5 5
42 Lining material austenitic steel
43 Design pressure (bar) 86 bar
44 Gross weight (tonne) 915 t

**B3. Steam generator** no steam genarator

45 Number of steam generators
46 Type once through
47 Configuration (horizontal/vertical)
48 Tube material
49 Shell material

50 Heat transfer surface per steam generator ($m^2$)
51 Thermal capacity per steam generator (MW)
52 Feed water pressure (bar)
53 Feed water temperature (C)
54 Steam pressure (bar)
55 Steam temperature (C)

**B4. Pressurizer** no pressurizer
56 Pressurizer total volume ($m^3$)
57 Steam volume (full power/zero power, $m^3$)

**B5. Main coolant pumps** no main coolant pumps
58 Number of cooling or recirculation pumps
59 Type
60 Pump mass flow rate (kg/s)
61 Pump design rated head
62 Pump nominal power (kW)
63 Mechanical inertia (kg $m^2$)

## C. CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS) no CVCS

64 Number of extraction lines
65 Number of pumps
66 Number of injection points
67 Feed and bleed connections

## D. CONTAINMENT

68 Type pressure suppression system
69 Overall form (spherical/cyl ) cylindrical
70 Structural material concrete (possibly fibrous concrete)
71 Liner material austenitic steel
72 Simple/double wall simple
73 Dimensions (diameter, height) (m) 30/27 5 m
74 Design pressure (bar) 11 bar
75 Design temperature (C) 165 °C
76 Design leakage rate (% per day) 1% per day

# III. SAFETY RELATED SYSTEMS

## A. DESIGN CONDITIONS

**A1. Fission product retention**
77 Containment spray system (Y/N) N
a Duration (h)
b Flow rate ($m^3$/h)
c Mode of operation (active/passive)
d Safety graded (Y/N)
78 F P sparging (Y/N) N
79 Containment tightness control (Y/N) Y
80 Leakage recovery (Y/N) N
81 Guard vessel (Y/N) N

**A2. Reactivity control**
82 Absorber injection system (Y/N) Y
a Absorber material boric acid
b Mode of operation (active/passive) active
c Redundancy 2x50%
d Safety graded Y
83 Control rods (Y/N) Y
a Maximum control rod worth (pcm) ---
b Mode of operation (active/passive) active
c Redundancy 4x50%
d Safety graded Y

**A3. Decay heat removal**
*A3-1 Passive safety features*
84 Emergency condenser

a Actuation mode (manual/automatic) automatic
b Injection pressure level (bar) respective RPV-pressure

c Flow rate (kg/s) ---
d Mode of operation (active/passive) passive
e Redundancy 4x25%
f Safety graded (Y/N) Y

85 Water recirculation and heat removal
a Intermediate heat sink (or heat exchanger) SG Core flooding pool
b Mode of operation (active/passive) passive
c Redundancy 1x100%
d Self sufficiency (h) 7 days
e Safety graded Y

*43-2 Active non-safety grade features*

86 Operational decay heat removal system
a Actuation mode (manual/automatic) active
b Flow rate (kg/s) <5 bar
c Mode of operation (active/passive) active
d Redundancy 2x100%
e Safety graded N

87 Water recirculation and heat removal
a Ultimate heat sink (cold source)
b Mode of operation (active/passive)
c Redundancy
d Self sufficiency (h)
e Safety graded

*43-3 Primary pressure control*

88 Implemented system (Name)
a Actuation mode (manual/automatic)
b Side location (primary/secondary circuit)
c Maximum depressurization rate (bar/s)
d Safety graded

## B. SEVERE ACCIDENT CONDITIONS*

**B.1 Fission products retention**

89 Containment spray system (Y/N) N
90 F P Sparging (Y/N) N
91 Containment tightness control (Y/N) Y
92 Leakage recovery (Y/N) N
93 Risk of recriticality (Y/N) open

**B.2 Recriticality control** open

94 Encountered design feature
a Mode of operation (A/P)
b Safety graded

**B.3 Debris confining and cooling**

95 Core debris configuration (core catcher) core catcher
96 Debris cooling system (name) RPV outside cooling
a Mode of operation (A/P) passive
b Self sufficiency 7 days
c Safety graded (Y/N) Y

**B.4 Long term containment heat removal**

97 Implemented system Containment cooling condenser
a Mode of operation (A/P) passive
b Self sufficiency (h) 7 days
c Safety graded (Y/N) Y

98 Intermediate heat sink Dryer-separator storage pool
a Self sufficiency (h) 7 days
b Safety graded (Y/N) Y

99 External coolant recirculation Building ventilation system
a Implemented components

* All systems must be qualified to operate under accident conditions

b Mode of operation (A/P)
c Self sufficiency (h)
d Safety graded (Y/N)

100 Ultimate heat sink
a Self sufficiency (h)
b Safety graded (Y/N)

**B.5 Combustible gas control**

101 Covered range of gas mixture concentration open
102 Modes for the combustible gas control
a Containment inertation Y
b Gas burning N
c Gas recombining N

**B.6 Containment pressure control**

103 Filtered vented containment open
a Implemented system
b Mode of operation (A/P)
c Safety graded
104 Pressure suppression system (Y/N) Y
a Implemented system horizontal vents/suppression pool

b Mode of operation active and passive
c Safety graded (Y/N) Y

## C. SAFETY RELATED I&C SYSTEM

Automatic load following (Y/N) Y
* range (% power) 50% power
* maximum rate (%/min) 10% per minute

Load rejection without reactor trip (Y/N) Y
Full Cathode Ray Tubes (CRT) display (Y/N) Y
Automated start-up procedures (Y/N) Y
Automated normal shutdown procedures (Y/N) Y
Automated off normal shutdown procedures (Y/N) Y
Use of field buses and smart sensors (Y/N) open
Expert systems or artificial Y
Protection system backup open

## D. EMERGENCY POWER SUPPLY SYSTEM

105 Type (diesel, gas, grid connection) Diesel units
106 Number of trains 2

## E. AC/DC SUPPLY SYSTEM

107 Type (rectifier, converter, battery) Converter, battery
108 Estimated time reserve (hr) 7 days (for monitoring of plant)

# IV. CONVENTIONAL THERMAL CYCLE

## A. TURBINE SYSTEM

109 Type N30-6x12 5
110 Overall length (m) 23 m
111 Width (m) 10 m
112 Number of turbines/reactor 1
113 Number of turbine sections per unit HP/LP/LP/LP
114 Speed (rpm) 3,000 rpm

## B. STEAM CHARACTERISTICS

115 H P inlet pressure bar 67 bar
116 H P inlet temperature (C) 283 °C
117 H P inlet flowrate (kg/s) 1,097 kg/s
118 L P inlet pressure 11 bar
119 L P inlet temperature ---
120 L P inlet flowrate 880 kg/s

C. **GENERATOR**

121 Type (3-phase synchronous, DC) THDD 115/72
122 Apparent power (MVA) 970 MVA
123 Active power (MW) 776 MW
124 Frequency (hz) 50 hz
125 Output voltage (kV) 27 kV
126 Total generator mass (t) ---
127 Overall length 18 m
128 Stator housing outside diameter 4 m

D. **CONDENSER**

129 Number of tubes ---
130 Heat transfer area ---
131 Flowrate ($m^3/s$) 35 $m^3/s$
132 Pressure (m/bar) 0 0425 bar
133 Temperature (C) 11 °C

E. **CONDENSATE PUMPS**

134 Number 2
135 Flowrate 35 $m^3/s$

### 7.1.5. Project status

#### *7.1.5.1. Entities involved*

In cooperation with Germany's electric utilities, Siemens AG, Power Generation Group (KWU) is developing the design concept. The tests of the emergency condenser are conducted by Siemens in cooperation with Forschungszentrum Jülich GmbH. (KFA) under the sponsorship of the Bundesministerium fuer Forschung und Technologie (Federal Ministry of Research and Technology) and Germany's electric utilities.

#### *7.1.5.2 Design status*

The BWR-600 development program is divided into the four following phases:

**1. Conceptual design phase**

Development of a design concept including predictions regarding feasibility and estimated costs. This phase, which was started at the beginning of 1992, was completed in October 1993.

**2. Consolidation phase**

New technical developments as well as additional technical aspects (such as possibilities for uprating) are to be thoroughly analyzed by the contractual partners (German utilities and Siemens). It is intended for the licensing authorities as well as their consultants and authorized experts to be asked to comment on the safety concept of the BWR-600. This phase is scheduled to be executed between October 1993 and March 1995.

### 3. Basic design phase

This phase is to comprise detailed analysis and design of the individual systems, components and buildings. A non-site-specific safety analysis report is also to be prepared. The basic design phase is planned to be completed by the beginning of 1997.

### 4. Verification phase

During this phase it is planned for the basic design of the BWR-600 to be developed to the stage at which all major plant equipment is ready for construction. Certain safety-related functions are to be experimentally verified and new technical developments tested. This phase should take from the beginning of 1997 to the end of 1999.

*7.1.5.3. Research and development work*

#### A. Executed R&D work

#### B. Ongoing and planned R&D work

Preliminary work for the tests of the emergency condenser started in October 1993. The tests will start in December 1994 and be completed in 1996.

#### C. R&D work needed

The main test programs needed are:

1. Passive pressure pulse transmitter
2. Containment cooling condenser
3. RPV outside cooling

Priliminary work for these tests will start in August 1994. International cooperation is desired.

*7.1.5.4. Licensing process*

The licensing procedure will start in 1995 with the concept certification which is expected 1996. The plan intention for a design certification is by the end of 1997.

### 7.1.6. Project economics

This new reactor design concept featuring natural circulation and passive accident control requires a lower core power density than that of large-capacity plants. This usually also results in a smaller overall plant capacity rating. Compared to a large-capacity plant, the construction costs for a small plant are not proportionate to its smaller capacity but are slightly higher. In the overall cost of such a plant, this is mainly the result of necessary investments which have nothing to do with generating capacity. The cost analyses carried out in the course of the conceptual design phase have, however, shown that the simple design of the passive safety systems accompanied at the same time by a reduction in the complexity of the active systems (designed for normal plant operating functions), as well as a shorter construction period and the higher degree of fuel utilization connected with a 24-month operating cycle, yield power generating costs for a plant rated at approximately 780 MWe which are no more than a few percentage points higher than those of a 1300 MWe plant of traditional design.

The economic analyses have thus demonstrated that the design concept developed in Germany for a medium-capacity BWR is capable of competing with large-capacity plants.

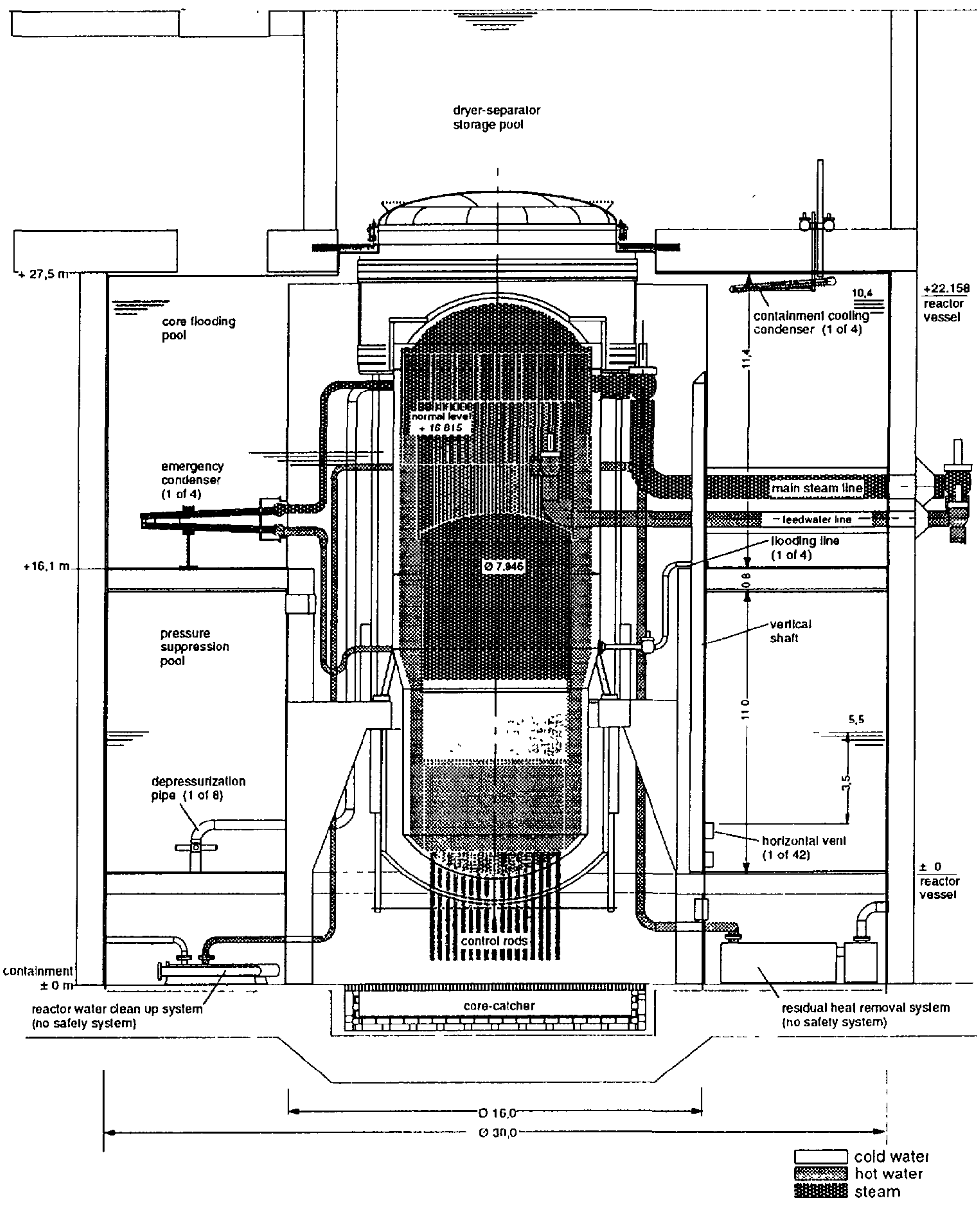
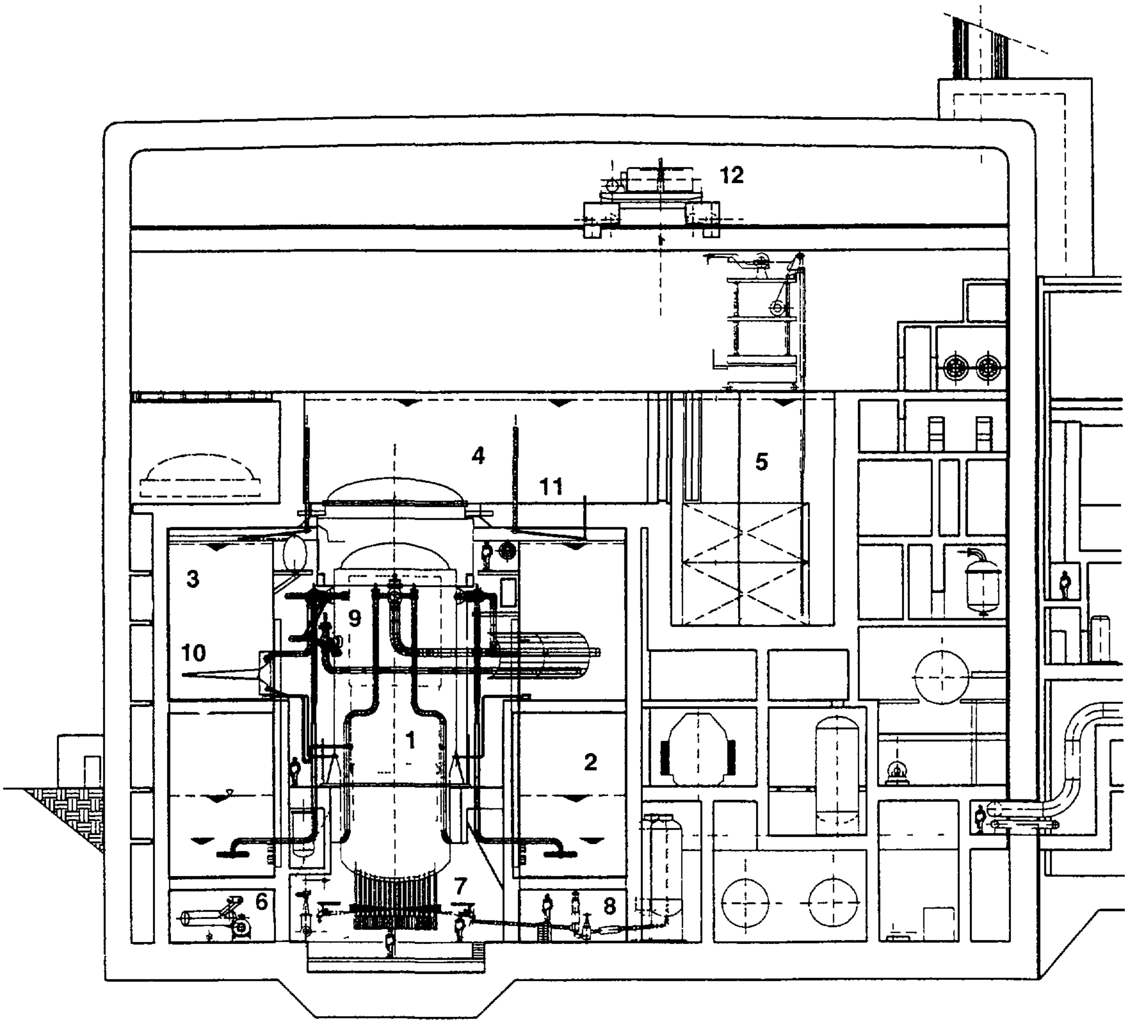



*Fig 7.1.1. BWR-600 - Section through Containment*




Legende

| | | | |
|---|---|---|---|
| 1 | Reactor pressure vessel | 7 | Control rod drives |
| 2 | Pressure suppression pool | 8 | Scram system |
| 3 | Core flooding pool | 9 | Main steam and feedwater valves |
| 4 | Dryer and separator setdown pool | 10 | Emergency condenser |
| 5 | Fuel pool | 11 | Containment cooling condenser |
| 6 | Residual heat removal system | 12 | Reactor building crane |

*Fig 7.1.2. BWR-600 - Section through Reactor Building*

7.2 REACTOR SYSTEM DESCRIPTION AND DEVELOPMENT STATUS OF VPBER-600

### 7.2.1. Basic objectives and features

A nuclear power station with a VPBER-600 reactor plant is intended for production of electricity or co-generation of electricity and heat. The nuclear power station can be used for sea-water desalination.

Original engineering features ensuring the high level of VPBER-600 safety are:

- integral reactor with coolant outflow restrictors;
- leak-tight primary circuit;
- guard vessel housing the reactor and systems operating under primary circuit pressure;
- passive safety systems requiring no external power supply for functioning and actuation.

### 7.2.2. Design description

*7.2.2.1. Nuclear steam supply system*

Reactor

The reactor plant is developed on the basis of an integral reactor (Fig 7.2.1.) the vessel of which encompasses:

Reactor

1. Upper unit
2. CPS drive
3. Level gauge
4. Reactor head
5. Reactor vessel
6. Heat exchanger-condenser
7. Steam generator
8. Guide tubes-connecting devices unit
9. Core barrel
10. Fuel assembly
11. Ionization chamber suspension
12. Electric pump

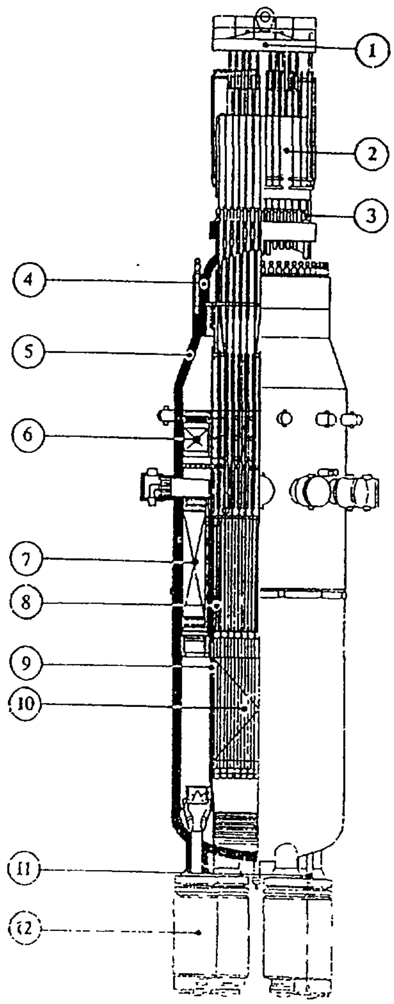



*Fig 7.2.1.*

- core with control and protection devices, heat exchanging surface of steam generator and main circulating pumps (but not their motors) forming a primary coolant main circulation circuit;
- heat exchanger-condensers of the emergency heat removal system;
- steam-gas plenum of the internal pressurizer.

In the reactor cover there are drive mechanisms for the control rods. In the bottom, the main circulating canned-motor pumps are installed. The total volume of the integral reactor together with the cover is 340 $m^3$ of which 260 $m^3$ belongs to coolant and 80 $m^3$ to the steam-gas pressurizer.

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 the 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 of equivalent diameter (ED), 50 mm in the steam-gas region and ED 32 mm in the water volume.
The arrangement of steam generator in the reactor pressure vessel structurally predetermines the enlarged water gap between the core and the vessel which reduces the neutron fluence down to a value of $7 \times 10^{16} n/cm^2$. The reactor vessel service life is thereby extended to 60 years.

Core

The core consists of 151 hexagonal fuel assemblies (FAs) (size across flats is 234 mm) with fuel elements and fuel lattice parameters analogous to those in VVER-1000. Each FA contains boron carbide rods which are combined in a cluster to form a control device. The control devices of 135 FAs are connected to drives of the electromechanical control and protection system (CPS). The core height is 3.53m, its equivalent diameter is 3.05m at average power density of 69.4 kW/1.

The moderate power density of the core allowed the designers to reduce the content of boric acid in the coolant compared to VVER, to dispense with operational control of boric acid concentration at start up and shut down and for power manoeuvres. 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 in the whole range of temperature variation. This allows for 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 core thermal-technical reliability under normal operation and accident conditions.

Steam generator

The heat exchange surface of the steam generator is arranged above the core in the annular gap between the reactor pressure vessel and the in-vessel barrel. The steam generator is of the 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 feed water and removal of steam out of the reactor and guard vessel.

Main circulating pump

In the bottom of the reactor pressure vessel there are six built-in main circulating motor pumps. The pump flow rate is 8250 $m^3$/hr, delivery head is 5.14 MPa. The pump drive is a canned electric motor of 2870 KW power, rotation frequency is 1500 rpm.

The design of the pump and of the nozzle in which the pump is mounted is such that in the event of a guillotine rupture or a breach in any possible part of the nozzle, the cross section for coolant outflow from the reactor is equivalent to a 17.5 mm diameter orifice.

Upper block

The upper block consists of the reactor cover with leak-tight Control Rod Drive Mechanisms (CRDMs), the in-vessel barrel structure and cables.

Primary circuit

The primary circuit (Fig7.2.2.) 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 for transferring the heat to the secondary coolant in six sections of the SG.

Primary Circuit Main Parameters:

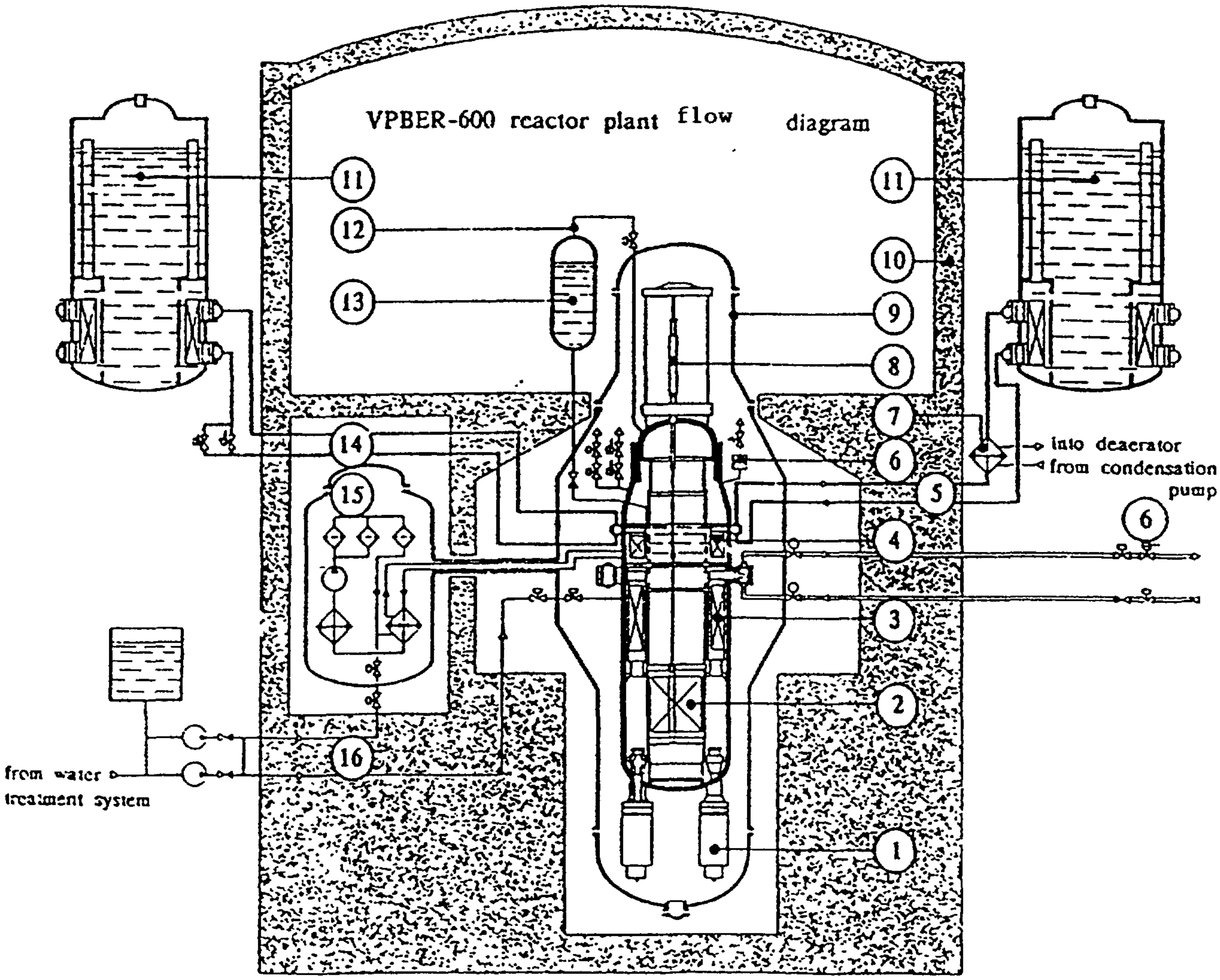
| | |
|---|---|
| - Reactor thermal rating, MW | 1800 |
| - Pressure, MPa | 15.7 |
| - Coolant temperature, °C | |
| - at core inlet | 294.4 |
| - at core outlet | 325 |
| - Coolant flowrate, t/hr | 36,500 |

The pressurizing system is intended for creating and maintaining the primary circuit preset pressure whatever the variation of the coolant volume during 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 done by steam evaporation from the primary coolant surface. The partial pressure of nitrogen above the coolant level is 3.9 MPa.

The purification and boron reactivity control system serves to maintain the required primary coolant quality during operation and for periodic removal of excessive boron (15-20 times for core life) thus compensating the fuel burnup. The system includes a recuperator, cooler, pumps and ion-exchangers.

Guard vessel

The reactor and the purification/boron control system connected to it during plant operation are placed inside the guard vessel (GV) (Fig 7 2.2)




1. Primary coolant pump
2. Reactor
3. Steam generator
4. Heat exchanger - condenser
5. Continuous heat removal system
6. Self-actuating devices
7. Intermediate heat exchanger
8. CPS drives
9. Guard vessel
10. Containment
11. Heat exchangers unit
12. Liquid absorber injection system
13. Tank with boron solution
14. Passive heat removal system
15. Clean up and reactivity boron compensation system
16. Makeup system

*Fig 7.2.2.*

The guard vessel is a passive protective and confinement device ensuring safety at LOCA, and fulfilling the following functions

- keeping the core under coolant level,
- confinement of radioactive products which escape beyond the reactor boundary in the case of coolant leaks

The GV is a factory-built steel vessel consisting of several parts assembled together at the site

*7.2.2.2. Balance of plant system*

In the VPBER-600 nuclear power plant, either a condensing turbine of the K-600-5 9/500 type for a power station or an extraction turbine of the T-600-5 9/500 type for a co-generation plant are used

The turbine with intermediate steam separation and one-stage superheating without controlled bleeds is intended for direct drive of an alternating current generator of the T3B type with water cooling, of 630 MW power and 3000 rpms

In the design with a condensing turbine the maximum electric power is 640 MW and the heat output is up to 215 GCal/hr, (250 MW)

In the design with a steam-extraction turbine the heat output is up to 645 GCal/hr (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

The steam-turbine plant includes also two condensers, three condensate pumps (one back-up), three electric feedwater pumps, condensate and reheaters separated water purification system and a heating facility

*7.2.2.3. Instrumentation, control and electrical systems*

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 allows continuous and periodical diagnosis 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 being presently developed allows solution of problems of plant automatic control, monitoring and diagnosis in normal operation and reliable actuation of safeguards, as well as to diagnose the plant actual state if an accident occurs

To enhance the protection of the reactor plant devices using a diverse principle of operation are used which are actuated by direct action of the working medium Pressure in the reactor and in the GV and coolant level in the reactor are used as safeguards activating parameters. The complex of self-actuated features provide for the reactor shutdown, actuation of the emergency heat removal system and for closing of isolation valves on the steam generator in all anticipated accidents.

*7.2.2.4. Safety considerations and emergency protection*

The reactor plant enhanced safety is based on the realization of engineering decisions which solved 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 reactor shutdown, residual heat removal and confinement of 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 as a result of increased pressure in the reactor 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 (Fig 7.2.2.). 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 reactor shutdown and is capable of keeping the cold unpoisoned reactor in a subcritical state.

Emergency heat removal means

The emergency heat removal system (Fig 7.2.2.) ensures the removal of residual heat from the core should reactor cooling through the steam turbine plant becomes 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 by natural convection of coolant in all the circuits. The pressure in the intermediate circuit of the system is higher than that in the reactor and thereby a barrier against release of radioactive products 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 (signal from pressure or coolant level variations in the reactor).

The heat being removed from the reactor via built-in heat exchanger-condensers is transferred to water storage tanks (heat exchanger units in Fig 7.2.2.) 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 of removing 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 (ED 50 mm in steam-gas plenum, ED 32 mm in water volume and ED 17.5 mm in the pump nozzle);
- cassette-type steam generator with structural limitation of possible inter-circuit leak (ED 40 mm for steam, ED 24 mm for feed water);
- factory-built guard vessel which is designed for the pressure which would develop following loss of the primary circuit integrity;
- steam generator isolation from the steam turbine plant at inter-circuit leakage is provided by three isolation valves, two of them are closed both automatically and remotely by an operator upon a low reactor water level signal, the third one is closed remotely by an operator only;
- steam generator which is designed for primary circuit pressure up to the localizing values;
- 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 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 or personnel intervention for not less than 72 hours following all possible disturbances and accidents (positive reactivity insertion, loss of heat removal, primary circuit depressurization) by means of the reactor inherent safety features and by using the complex of interconnected passive safety systems and devices.

This complex includes:
- guard vessel;
- emergency heat removal system;
- emergency boron injection system.

On positive reactivity addition or loss of reactor heat removal following 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).

Following 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 to enable the removal of 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.

The design solutions adopted for VPBER-600 practically exclude a core melt. The probability of severe core damage is less than $10^{-8}$ per reactor-year.

*7.2.2.5. Building and Structures*

The reactor plant is located within the containment structure, which is arranged in the reactor building. Since the reactor and the primary circuit equipment are located inside the guard vessel a single ferroconcrete containment has been selected without prestressing, but with metallic liner. It is designed for an excess pressure of 0.1 MPa.

The containment serves to protect the reactor plant against external impacts, as well as for confining radioactive products in the event of incidents during reactor refuelling and following accidents with postulated loss of the guard vessel integrity.

The containment is a cylindrical structure of 40 m ID with a spherical dome. The wall thickness is 0.9 m. The leakage from the containment is 1.0% of volume per day.

The tanks of the emergency heat removal system are arranged in annexes to the reactor building on opposite sides of the containment thus excluding their simultaneous failure by air crash or shock wave impact.

The containment, annexes to the reactor building, reactor, guard vessel, primary circuit equipment and the safety systems are designed for a magnitude 8 earthquake (MSK-64 scale).

The reactor plant equipment lay-out provides for its accessibility for repair and maintenance operations.

**Refuelling**

Removal and reloading of in-vessel structures and fuel is carried out is by the universal machine, developed by OKBM.

The reloading machine moves on rails on the reactor hall floor and services the area where the reactor pit, spent fuel storage pool and equipment storage pits are arranged. When manipulating with fuel the reloading machine is used for reshuffling the FAs inside the core, transferring them between the core and the spent fuel storage pool and inside the pool.

All manipulations with FAs beyond the reactor boundary are performed using a special water-filled transfer tube.

## 7.2.3. SAFETY CONCEPTS

TABLE 7.2.1. MAIN SAFETY RELATED SYSTEMS IN THE VPBER-600

| Name | Safety graded | Main characteristics |
|---|---|---|
| Primary Circuit (PC) | X | Integral reactor, twelve steam generator sections, 12 heat exchanger-condensers of emergency heat removal system, leak-tight primary components |
| Reactivity control and protection system (CPS) | X | 139 control rod drives |
| Alternative shutdown system (ASS) | X | Passive and active injection of borated water |
| | | 4 channels are connected to at accidents |
| Passive emergency heat removal system (PEHRS) | X | |
| Guard vessel (GV) | X | Prevention core uncovery at primary circuit leak |
| | | Double valves in each primary circuit pipeline |
| Primary circuit isolation valves (PIV) | X | Triple valves in steam generator pipelines |
| Secondary circuit isolation valves (SIV) | X | |
| Containment | X | 2X 10t/hr capacity |
| Primary circuit make-up system (PMS) | X | 2 channels removing decay heat to cooling water of safety systems |
| Repair cooling system (RCS) | X | |

## TABLE 7.2.2. MAIN ACCIDENT INITIATORS FOR THE VPBER-600

- LOCA (primary) Loss of Primary Coolant Accident,
- LOCA (Secondary) Secondary Pipe Rupture (water or steam),
- LOCA (Interfacing) e g SGTR Steam Generator Tube Rupture,
- ATWS Anticipated Transients Without Scram,
- Primary Transients,
- Secondary Transients (turbine trip),
- Loss of electric sources (all AC sources),
- Total loss of the cold sources,
- Total loss of the steam generator feedwater,
- Station blackout

## TABLE 7.2.3. DESIGN FEATURES FOR PREVENTION AND PROTECTION LEVELS THAT REDUCE (R), SUPPRESS (S), THE INITIATORS FREQUENCY OR LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES (L)

**PREVENTION LEVEL**

LOCA (Primary)
- Integral reactor, low fluence on vessel, reduced thermal shocks onto vessel reduce initiator frequency and limits consequences
- Small diameter of nozzles in reactor vessel limits consequences
- Double valves on the pipelines, connecting the primary circuit systems with the reactor reduce initiator frequency and suppress initiator

LOCA (Secondary)
- Sectioning of steam generator, small diameter of pipelines limits consequences

LOCA (Interfacing)
- Cassette-type steam generator with structural limiting of intercircuit leak, triple valves on pipelines designed for primary circuit pressure suppress initiator

Primary transient
- Reliable canned pumps, high level of natural circulation limits consequences and suppress initiator

Secondary transients
- Reliable structure (heat exchange surface pipelines, isolating valves) reduce initiator frequency

Loss of electric sources Natural circulation in all circuits for removal of residual heat to heat sink (water or atmosphere) suppress initiator

Total loss of the source (Water)
- Emergency heat removal system with permanently connected channels, limits consequences, limitation of consequences of incidents

Total loss of the S G feedwater
- System of emergency heat removal with permanently connected channels limits consequences

Station blackout

**PROTECTION LEVEL**

LOCA (Primary)
- Guard vessel designed for pressure in case of primary circuit leak
- Intermediate circuit for emergency heat removal system
- Intermediate circuit for cooling equipment designed for primary circuit pressure

LOCA (Secondary)
- Reliable isolation of steam generator sections

LOCA (Interfacing)
- Reliable isolation of steam generator

ATWS - Strong negative temperature coefficient

Primary transients

Possibility of operation without one or two pumps, transition to natural circulation at shutting down main circulating pumps

Secondary transient
- Possibility of operation without one section of steam generator, one or two steam generators at unforseen closure of valves

Loss of electric sources non-critical
Total loss of heat sink non-critical
Total loss of S G feedwater non-critical
} Dissipation of decay heat via the residual heat removal system

Station Blackout

TABLE 7.2.4. DESIGN FEATURES FOR MITIGATION LEVEL OF THE VPBER-600

| Safety Functions | Systems (Cf Tab 7 2 1 ) | Passive/Active | Design Features/Remarks |
|---|---|---|---|
| **Design Basis**<br>Fission product containment | Primary circuit<br>GV<br>PIV<br>SIV<br>Containment | Passive<br>Passive<br>Passive<br>Passive<br>Passive | |
| Coolant inventory | PV/GV/PIV/SIV/ASS | Passive | Integral reactor with leak-tight primary components enclosed in GV |
| Decay heat removal | Passive emergency heat removal system<br>RCS | Passive<br>Active | 72 h capacity |
| Reactivity control | CPS<br>ASS | Active/Passive<br>Passive | Cold shut down capability |
| Primary circuit pressure control | CPS/ASS<br>Passive emergency heat removal systems | Passive | Without discharge of primary coolant |
| **Severe Accident**<br>Containment temperature and pressure control | Guard vessel | Passive | |
| Heat removal | Passive emergency heat removal system/guard vessel | Passive | |
| Tightness control | Guard vessel, containment | Passive | |
| Inflam gas control | Igniters | Passive | |
| Fission product containment | PC/Guard vessel/Containment | Passive | |
| Corium management | PC/Guard vessel/Emergency heat removal system | Passive | |
| Others | | | |

### 7.2.4. Design Data Questionnaire (Water Cooled Reactors for VPBER-600)

## I. GENERAL INFORMATION

1 Design name VPBER-600
2 Designer/Supplier address OKB Mechanical Engineering
3 Reactor type PWR Integral Number of modules/per plant 1
4 Gross thermal power (MW-th) per reactor 1800
5 Net electrical output (MW-e) per reactor 630
6 Heat supply capacity (MW-th) 250

## II. BASIC DESIGN DESCRIPTION

### A. CORE AND REACTIVITY CONTROL

7 Fuel material $UO_2$
8 Fuel inventory (tones of heavy metal) 62
9 Average core power density (kW/liter) 69,4
10 Average fuel power density (kW/kgU) 29
11 Maximum linear power (W/m) 26000
12 Average discharge burnup (MWd/t) 50000
13 Initial enrichment or enrichment range (Wt%) 2,1
14 Reload enrichment at the equilibrium (Wt%) 4,15
15 Refuelling frequency (months) 18
16 Type of refueling (on/off power) Off power
17 Fraction of core withdrawn (%) 25
18 Moderator material and inventory Water
19 Active core height (m) 3,53
20 Core diameter (m) 3,05
21 Number of fuel assemblies 151
22 Number of fuel rods per assembly 293
23 Rod array in assembly 18
24 Clad material Zirconium alloy
25 Clad thickness (mm) 0 715
26 Number of control rods or assemblies 139
27 Type Cluster
28 Additional shutdown systems Boric Acid Solution injection
29 Control rod neutron absorber material boron carbide
30 Soluble neutron absorber Boric acid solution
31 Burnable poison material and form Chromium diboride

### B. REACTOR COOLANT SYSTEM

**B1. Coolant**

32 Coolant medium and inventory Water
33 Design coolant mass flow through core (kg/s) 10140
34 Cooling mode (forced/natural) Forced
35 Operating coolant pressure (bar) 157
36 Core inlet temperature (°C) 294,4
37 Core outlet temperature (°C) 325

**B2. Reactor pressure vessel**

38 Overall length of assembled vessel/tube (m) 20,15
39 Inside vessel/diameter (m/mm) 5,97
40 Average vessel/tube thickness (mm) 265
41 Vessel/tube material 15 x 2 MPA steel
42 Lining material 08 x 18 H10T steel
43 Design pressure (bar) 180
44 Gross weight (tone/kg) 830

**B3. Steam generator**

45 Number of steam generators 4 each consisting of 3 sections
46 Type Once-through
47 Configuration (horizontal/vertical) Vertical
48 Tube material titanium alloy
49 Shell material stainless steel
50 Heat transfer surface per steam generator ($m^2$) 13840
51 Thermal capacity per steam generator (MW) 1764

52 Feed water pressure (bar) 78,5
53 Feed water temperature (°C) 230
54 Steam pressure (bar) 6,38
55 Steam temperature (°C) 305

**B4. Pressurizer**

56 Pressurizer total volume ($m^3$) 80, Coated inside reactor
57 Steam volume (full power/zero power, $m^3$)

**B5. Main coolant pumps**

58 Number of cooling or recirculation pumps 6
59 Type Canned motor
60 Pump mass flow rate (kg/s) 1690
61 Pump design rated head 78m
62 Pump nominal power (kW) 2870
63 Mechanical inertia (kg $m^2$)

## C. CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS):

64 Number of extraction lines 1
65 Number of pumps 2
66 Number of injection points 2
67 Feed and bleed connections 2

## D. CONTAINMENT

68 Type Prestressed concrete
69 Overall form (spherical/cyl ) Cylindrical
70 Structural material Concrete
71 Liner material Steel
72 Simple/double wall Simple
73 Dimensions (diameter, height) (m) 40,0, 80,5
74 Design pressure (bar) 2,0
75 Design temperature (°C) 120
76 Design leakage rate (% per day) 1,0%

# III. SAFETY RELATED SYSTEMS

## A. DESIGN CONDITIONS

**A1. Fission product retention**

77 Containment spray system (Y/N) No
- a Duration (h)
- b Flow rate ($m^3$/h)
- c Mode of operation (active/passive)
- d Safety graded (Y/N)

78 F P sparging (Y/N) No
79 Containment tightness control (Y/N) No
80 Leakage recovery (Y/N) Yes
81 Guard vessel (Y/N) Yes

**A2. Reactivity control**

82 Absorber injection system (Y/N)
- a Absorber material Boric acid solution
- b Mode of operation (active/passive) Passive
- c. Redundancy Yes
- d Safety graded Yes

83 Control rods (Y/N)
- a Maximum control rod worth (pcm)
- b Mode of operation (active/passive) passive
- c Redundancy Subcriticality without three control rods
- d Safety graded Yes

**A3. Decay heat removal**

*A3-1 Primary Side*

84 Water injection
- a Actuation mode (manual/automatic) manual
- b Injection pressure level (bar) 180
- c Flow rate (kg/s) 2 77 x 2

d Mode of operation (active/passive) active
e Redundancy Yes
f Safety graded (Y/N) Yes

85 Water recirculation and heat removal
a Intermediate heat sink (or heat exchanger) 4IHXs
b Mode of operation (active/passive) Passive
c Redundancy 4 x 50%
d Self sufficiency (h) 72
e Safety graded Yes

*A3-2 Secondary Side*

86 Feed water
a Actuation mode (manual/automatic) Automatic
b Flow rate (kg/s) 46,3
c Mode of operation (active/passive) Active
d Redundancy 2 x 100%
e Self sufficiency (h)
f Safety graded Yes

87 Water recirculation and heat removal
a Ultimate heat sink (cold source) water
b Mode of operation (active/passive) Passive
c Redundancy 2x100%
d Self sufficiency (h) 72
e Safety graded Yes

*A3-3 Primary pressure control*

88 Implemented system (Name) Emergency heat removal system
a Actuation mode (manual/automatic) Automatic, passive
b Side location (primary/secondary circuit) Primary
c Maximum depressurization rate (bar/s)
d Safety graded Yes

## B. SEVERE ACCIDENT CONDITIONS*

### B.1 Fission products retention

89 Containment spray system (Y/N)
90 F P Sparging (Y/N)
91 Containment tightness control (Y/N)
92 Leakage recovery (Y/N)
93 Risk of recriticality (Y/N)

### B.2 Recriticality control

94 Encountered design feature
a Mode of operation (A/P)
b Safety graded

### B.3 Debris confining and cooling

95 Core debris configuration (core catcher)
96 Debris cooling system (name)
a Mode of operation (A/P)
b Self sufficiency
c Safety graded (Y/N)

### B.4 Long term Guard Vessel heat removal

97 Implemented system
a Mode of operation (A/P) P
b Self sufficiency (h) 72
c Safety graded (Y/N) Y

98 Intermediate heat sink
a Self sufficiency (h) 72
b Safety graded (Y/N) Yes

99 External coolant recirculation No
a Implemented components
b Mode of operation (A/P)
c Self sufficiency (h)
d Safety graded (Y/N)

* All systems must be qualified to operate under the accident conditions

100 Ultimate heat sink
- a Self sufficiency (h) 72
- b Safety graded (Y/N) Yes

**B.5 Combustible gas control**

101 Covered range of gas mixture concentration

102 Modes for the combustible gas control
- a Containment inertation
- b Gas burning
- c Gas recombining
- d Others

**B.6 Containment pressure control**

103 Filtered vented containment (Y/N)
- a Implemented system
- b Mode of operation (A/P)
- c Safety graded

104 Pressure suppression system (Y/N)
- a Implemented system
- b Mode of operation
- c Safety graded (Y/N)

## C. SAFETY RELATED I&C SYSTEM

Automatic load following (Y/N) Yes
- * range (% power) 30-100
- * maximum rate (%/min) 2,5

Load rejection without reactor trip (Y/N) Yes
Full Cathode Ray Tubes (CRT) display (Y/N) Yes
Automated start-up procedures (Y/N) Yes
Automated normal shutdown procedures (Y/N) Yes
Automated off normal shutdown procedures (Y/N) Yes
Use of field buses and smart sensors (Y/N) Yes
Expert systems or artificial intelligence advisors (Y/N) Yes
Protection system backup (Y/N) Yes

## D. EMERGENCY POWER SUPPLY SYSTEM

105 Type (diesel, gas, grid connection)
106 Number of trains

## E. AC/DC SUPPLY SYSTEM

107 Type (rectifier, converter, battery) battery
108 Estimated time reserve (hr) 72

# IV. CONVENTIONAL THERMAL CYCLE

## A. TURBINE SYSTEM

109 Type Steam turbine with uncontrolled steam bleeds
110 Overall length (m) 28 74
111 Width (m) 8
112 Number of turbines/reactor 1
113 Number of turbine sections per unit (e g HP/LP/LP) (1/2/2)
114 Speed (rpm) 3000

## B. STEAM CHARACTERISTICS

115 H P inlet pressure 59 bar
116 H P inlet temperature 300°C
117 H P inlet flowrate 930 Kg/s
118 L P inlet pressure
119 L P inlet temperature
120 L P inlet flowrate (per section)

## C. GENERATOR

121 Type (3-phase synchronous, DC) 3-phase synchronous
122 Apparent power (MVA) 741
123 Active power (MW) 630
124 Frequency (hz) 50
125 Output voltage (kV) 20
126 Total generator mass (t) 420
127 Overall length 12 m
128 Stator housing outside diameter (m) 4 12

## D. CONDENSER

129 Number of tubes 42 000
130 Heat transfer area 44 000 $m^2$
131 Flowrate ($m^3/s$) 27 2
132 Pressure (m/bar) 0,05
133 Temperature (°C) 25

## E. CONDENSATE PUMPS

134 Number
135 Flowrate
136 Developed head
137 Temperature
138 Pump speed

### 7.2.5. Project Status

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 Centre "Kurchatov Institute" (Moscow)- scientific management.

Today the conceptual design of the reactor plant, turbine plant, automated process control system and main principles for the nuclear power station are already completed.

Work continues to develop the technical design of the reactor and the turbine plant, of the control system and the nuclear power station as a whole.

On the basis of the conceptual design of the reactor plant and the main principles of the nuclear power station project the first version of the preliminary safety report has been developed and presented to the Russian regulatory organizations for approval.

The transition to natural coolant circulation in the primary circuit and the reliability and efficiency of the emergency heat removal system have been confirmed by experiment and calculations. A large scale rig (1:200 for power, 1:1 for height) is being set up for the complex verification of the reactor plant safety.

### 7.2.6. Project Economics

A technical-economic assessment made for a nuclear power plant with a VPBER-600 reactor shows that it is competitive with oil-fired power stations: the specific cost of its construction in 1991 prices amounts to approximately 1000 roubles per kW of installed power; the specific cost of electricity is 0.023 roubles per/KW-hour. So the cost of electricity is about equal to a natural gas-fired power plant of a similar size and by 1.5-2.0 times less than a coal-fired power plant.

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## 7 3 HSBWR REACTOR SYSTEM DESCRIPTION AND DEVELOPMENT STATUS

### 7.3.1. Basic objectives and features

The concept of the HSBWR is based on (a) system simplification by adopting natural circulation for coolant recirculation, (b) a reactor building standardization based on 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 adopting steel structured PCVs The features of the HSBWR according to the design bases mentioned above are described below

1 short fuel assemblies of 3 7m (heated length of 3 1m) to avoid seismic resonance between the core and reactor building constructed on soft to firm ground, which make it possible to construct the standard plant underground with a high defence ability against terrorism, if necessary,

2 low volumetric power density of 34 2 kW/l, and a long continuous operation period of 23 months,

3 simple internals without forced recirculation systems (i e natural circulation) and steam separators,

4 no core uncovery in any loss-of-coolant accident (LOCA) 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, automatic depressurization system (ADS) and accumulators at a relatively low pressure of 0 5 ~ 1 0 MPa,

5 decay heat absorption in the suppression pool for one day after the accident initiation and natural heat removal from the primary containment vessel (PCV) by heat conduction through the steel-fabricated PCV to the outer pool for three days, which gives operators enough time recovery actions,

6 RPV flooding by coolant fed from the accumulators through the RPV and direct cooling of the RPV to cover the core with coolant and maintain core cooling even in a RPV bottom break accident,

7 depressurization by the additional ADS and boronated water injection by the accumulator at the pressure of 1 ~ 1 5 MPa to decrease reactivity and shutdown the reactor in an anticipated transient without scram (ATWS),

8 standardized compact PCV, reactor building and turbine building, and the same plant layout 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

### 7.3.2. Design description

*7.3 2 1. Nuclear steam supply system*

The schematic of the reactor pressure vessel (RPV) and the system configuration are shown in Fig 7 3 1 and Fig 7 3 2 respectively The vessel is 23 m long and 6 3 m in internal diameter

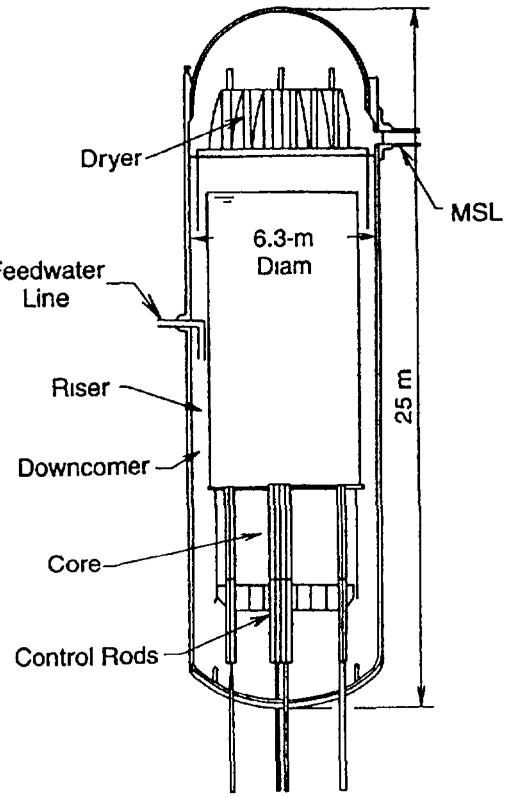


*Fig 7.3.1. Schematic of the Reactor Pressure Vessel*

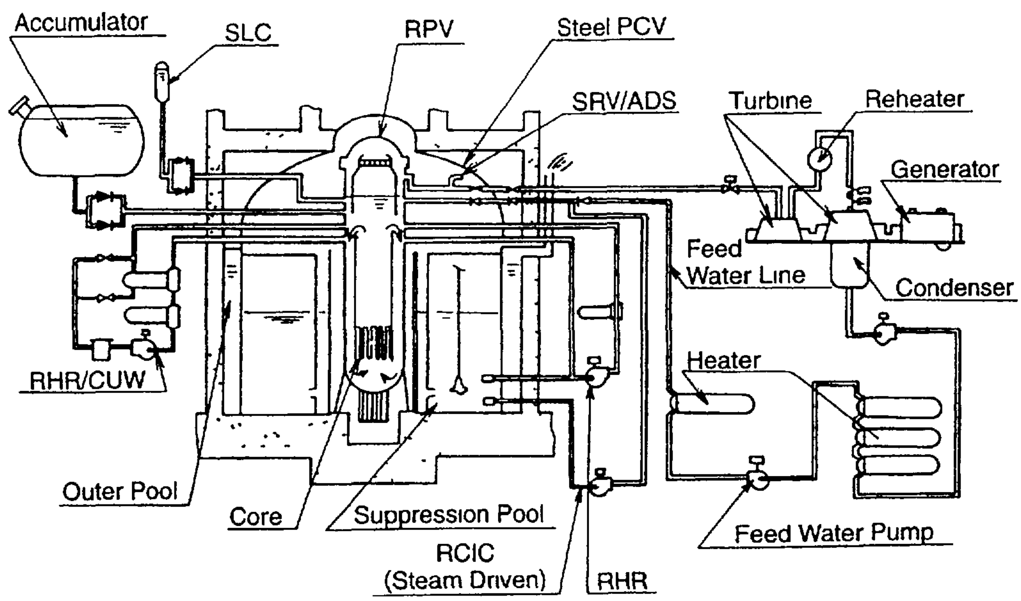


*Fig 7.3.2. System Configeration*

Pumped recirculation systems are eliminated and natural circulation is used for coolant circulation in order to decrease the components driven by external force and 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 bundles (3 1 m active length) is determined to avoid seismic resonance between the fuel bundles and the reactor building, which is constructed on soft to firm ground Therefore standardization of the reactor building and plant layout is possible without any connection to ground firmness The power density is lower in the natural circulation reactor than in a forced circulation reactor, but the lower power density allows a longer continuous operation The short heated length and low power density provide good thermal-hydraulic characteristics With the 8x8 type fuel bundle, the power density is 34 2 kW/l, the number of fuel bundles 708 and the core diameters 4 65 m The uranium enrichment is 3 6% and the average burn-up is 39 GWd/t under the conditions of 23 months continuous operation

By eliminating the separators, the natural circulation flow rate becomes higher due to the absence of flow resistance by them The transit time of steam from the core is 6 sec in the riser of 9 m height and 8 sec in the long steam dome The long transit time decreases the strength of gamma-rays from nitrogen 16 in the turbine building

*7.3.2.2. Balance of plant*

The balance of plant of HSBWR is mainly composed of two main steam lines (pipe diam 700A) with two MSIVs (main steam isolation valves), two feedwater lines (pipe diam 400A) and the turbine systems including one high pressure turbine and three low pressure turbines etc The RCIC (reactor core isolation cooling) system is installed to maintain reactor water level by injection from the condensate storage tank during an isolation event The CUW (reactor water clean-up), system is installed to manage radioactivity in the reactor water or to remove residual heat in case of RHR system failure

A 52" turbine is adopted as model TCDF-52, which had already been developed as TC6F-52 for the 1350 MWe BWR, in order to simplify the turbine island

Furthermore, the condensate transfer pump is also minimized in size by using the re-entry type A high efficiency, of about 33 4%, is estimated under a condenser vacuum of 722 mmHg

*7.3.2.3. I&C and electrical systems*

The concept of the I&C and electrical systems in HSBWR are basically the same as those in ABWR However, the electrical load is much smaller because the recirculation system is eliminated Furthermore, the I&C and electrical systems are very simplified, e g the reduction of emergency diesel generator capacity, because passive safety systems are adopted

*7.3.2.4. Safety considerations*

The HSBWR has a high safety factor for postulated LOCAs by pipe breaks, because there are no large diameter pipes below the top of the core The safety systems have redundancy by the combination of active and passive systems

The steam driven RCIC system is provided for the loss of all AC power (i e station blackout) and is also effective for small break LOCAs The ADS and accumulators with emergency coolant at a low pressure of 0 5 ~ 1 MPa provide short term emergency core cooling instead of the emergency diesel generators and high and low pressure pumped injection systems in the current BWR designs. Elimination of emergency diesel generators and pumped injection systems simplifies the emergency core cooling systems (ECCSs) and provides high system reliability because of fewer elements. 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. The accumulators have the capacity to cover the whole core with coolant during the initial 24 hours after an occurrence of an accident.

Core cooling after normal reactor shutdown and long-term core cooling after reactor scram are performed by residual heat removal (RHR) systems with injection pumps and heat exchangers. The RHR systems have the ability to cool down the rated reactor to 52 C within hours, and it is also enough 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 if the RHR systems are 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-fabricated 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 needed.

*7.3.2.5. Building and construction*

The schematic of the reactor island is shown in Fig 7.3.3. The volume of the reactor building is about 50% of the current BWR building for the same rated capacity, which is realized by simplifying the components and systems as described above and removing the spent fuel pool and control room to other buildings. All water pools use the high level of welding technology and skill available in Japan, steel structures for the PCV are adopted. Simplification of the components and systems and adoption of steel structures in the PCV shorten the construction period. The construction period should be 32 ~ 36 months from the start of construction to commercial operation.

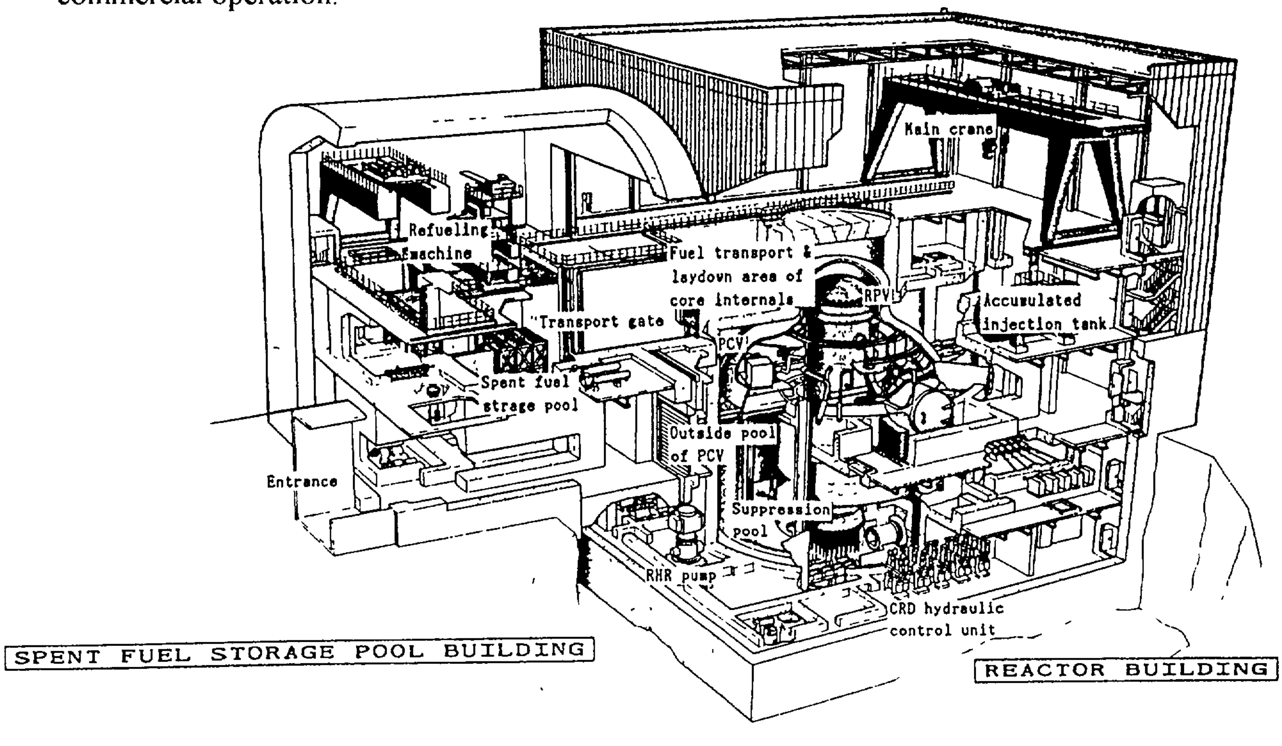


*Fig 7.3.3. The layour of Reactor Building*

*7.3.2.6. Licensing status*

HSBWR is at the status of a conceptual design and has not yet been licensed by any regulator.

### 7.3.3. Safety concept

TABLE 7.3.1. MAIN SAFETY RELATED SYSTEMS IN THE HSBWR

| Name | Safety graded | Main characteristics |
|---|---|---|
| Reactor core isolation cooling (RCIC) | x | - steam driven core injection |
| Automatic depressurization system (ADS) | x | - five valves<br>- total flow area of 0.057 $m^2$ |
| Accumulators | x | - passive emergency core cooling (2 divisions) |
| Residual heat removal system (RHR) | | - active pump and heat exchanger<br>- cooling rated reactor to 52C within 20h |
| Reactor clean-up water (CUW) | | - active pump and heat exchanger<br>- cooling rated reactor to 52C within 20h |
| Primary containment vessel (PCV) | x | - pressure suppression type<br>- one day heat sink after scram<br>- free standing steel containment vessel |
| Water wall | x | - natural circulation heat removal from suppression pool to outer pool through PC wall |
| Flammability control system (FCS) | x | - recombiner in the nitrogen inerted containment |
| Stan-by liquid control (SLC) | x | - boron injection<br>- passive system by accumulator |

TABLE 7.3.2. MAIN ACCIDENT INITIATORS FOR THE HSBWR

| | | |
|---|---|---|
| LOCA (Primary) | - | Main steam line break |
| | - | Feed water line break |
| | - | Accumulator line break |
| | - | Bottom head drain line break |
| Transients | - | Coolant temperature decrease |
| | - | Reactor pressure decrease |
| | - | Loss of off-site power |
| | - | Loss of feed water |
| Station blackout | - | Plant is designed such that this event probability is less than $10^{-7}$/yr |
| ATWS | - | This event probability is less then $10^{-7}$/yr |

TABLE 7.3.3A. DESIGN FEATURES FOR PREVENTION AND PROTECTION LEVELS THAT REDUCE (R), SUPPRESS (S), THE INITIATORS FREQUENCY OR LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES (L)

**PREVENTION LEVEL**

**LOCA:** Design of lines to high quality engineering codes and standards, and to seismic and environmental requirements, Stringent water quality and material specifications to limit corrosion, elimination of longitudinal welds, reduction in number of welds, inspection and maintenance

**Transients:**

- Coolant temperature decrease Caused by (1) steam extraction line to heater is closed, or (2) feedwater is bypassed around heater HSBWR is designed such that no single operator error or equipment failure shall cause a loss of more than 55 6C feedwater heating

- Reactor pressure increase Several events may cause this e g, inadvertent closure of one turbine control value, pressure regulator downscale failure, generator load rejection, turbine trip MSIV closure, loss of condenser vacuum, loss of non-emergency AC power to station auxiliaries, loss of feedwater etc All these have been analyzed, features are included in the instrumentation and control systems or redundancies to maintain reactor pressure through a combination of component automatic responses or operator actions, depending on the identified cause

- Loss of off-site power When this happens, standby AC Power Supply (Two diesel generators) and DC power supply (battery) will supply power

- RO withdrawal error/subcritical/low power the reactor control and instrumentation system (RCIS) and automatic power regulator (APR) systems prevent this event from occurring

- Rod withdrawal at power may be caused by a operator procedural error, causing a malfunction of a control rod(s) withdrawal logic or a automated rod withdrawal logic malfunction The automated thermal limit monitor system (ATLM) performs the rod block monitoring system

**Station blackout:** Capability exists for safe reactor shutdown and maintaining shutdown cooling under a postulated loss of all AC power for more than 72 hours

**ATWS:** The following are provided for prevention, Electrical insertion FMCRDs with sensors and logic independent of RPS

TABLE 7.3.3B. DESIGN FEATURES FOR PREVENTION AND PROTECTION LEVELS THAT REDUCE (R), SUPPRESS (S), THE INITIATORS FREQUENCY OR LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES (L)

| | |
|---|---|
| **PROTECTION LEVEL** | |
| **LOCA:** | Automatic initiation of containment isolation and passive systems actuation (ADS followed by accumulated, and PCCS) |
| **Transients:** | - Coolant temperature decrease caused by (1) steam extraction line to heater is closed, or (2) feedwater is bypassed around heater HSBWR is designed such that no single operator error or equipment failure shall cause a loss of more than 55 6C feedwater heating<br><br>- Reactor pressure increase Several events may cause this e g, inadvertent closure of one turbine control value, pressure regulator downscale failure, generator load rejection, turbine trip MSIV closure, loss of condenser vacuum, loss of non-emergency AC power to station auxiliaries, loss of feedwater etc All these have been analyzed, features are included in the instrumentation and control systems or redundancies to maintain reactor pressure through a combination of component automatic responses or operator actions, depending on the identified cause<br><br>- Loss of off-site power When this happens, standby AC power supply (two diesel generators) and DC power supply (battery) will supply power<br><br>- Rod withdrawal error/subcritical/low power the reactor control and instrumentation system (RCIS) and automatic power regulator (APR) systems prevent this event from occurring<br><br>- Rod withdrawal at power may be caused by a operator procedural error, causing a malfunction of a control rod(s) withdrawal logic or an automated rod withdrawal logic malfunction The automated thermal limit monitor system (ATLM) performs the rod block monitoring system |
| **Station blackout:** | Capability exists for safe reactor shutdown and maintaining shutdown cooling under a postulated loss of all AC power for more than 72 hours |
| **ATWS:** | The following are provided for mitigation Accumulated SLCS under conditions indicative of an ATWS |

TABLE 7.3.4. DESIGN FEATURES FOR MITIGATION LEVEL OF THE HSBWR

| **Safety functions** | **Systems** | **Passive/active** | **Design features/remarks** |
|---|---|---|---|
| **Design basis**<br><br>Fission product containment | Containment | Passive | Pressure suppression type |
| Coolant inventory (high pressure) | - Reactor core isolation cooling (RCIC) | Active | 2 accumulators |
| (low pressure) | - Accumulator<br>- Feedwater system<br>- Depressurization system | Passive<br>Active<br>Passive | - Accumulated boron injection |
| Decay heat removal | - Residual heat removal system (RHR)<br>- Reactor clean-up system<br>- Water wall | Active<br>Active | |
| Reactivity control | - Control rod<br>- Standby liquid control (SLC) | Active<br>Passive | |
| Primary circuit pressure control | - Safety relief value (SRV)<br>- Automatic depressurization system (ADS) | Passive<br>Passive | |
| **Severe accident**<br><br>Containment temperature and pressure control | - Containment/water wall | Passive | |
| Heat removal | - Water wall | Passive | |
| Tightness control | - Containment | Passive | |
| Inflammable gas control | - Interting<br>- Recombiner | Passive<br>Passive | |
| Fission product containment | - Containment | Passive | |
| Corium management | - RPV arrangement | Passive | Lower drywell design includes features to contain and cool debris |
| Others | | | |

### 7.3.4. Design data questionnaire (for water-cooled reactors)

## I. GENERAL INFORMATION

1 Design name HSBWR
2 Designer/Supplier address Hitachi Ltd
3 Reactor type BWR
4 Gross thermal power (MW-th) per reactor 1800
5 Net electrical output (MW-e) per reactor 600
6 Heat supply capacity (MW-th) --

## II. BASIC DESIGN DESCRIPTION

### A. CORE AND REACTIVITY CONTROL

7 Fuel material $UO_2$
8 Fuel inventory (tones of heavy metal) --
9 Average core power density (kW/liter) 34 2
10 Average fuel power density (kW/kgU) --
11 Maximum linear power (W/m) 44 0
12 Average discharge burnup (MWd/t) 39,000
13 Initial enrichment or enrichment range (Wt%) --
14 Reload enrichment at the equilibrium (Wt%) 3 6
15 Refueling frequency (months) 23
16 Type of refueling (on/off power) off-power
17 Fraction of core withdrawn (%) 25
18 Moderator material and inventory Light water
19 Active core height (m) 3 1
20 Core diameter (m) 4 7
21 Number of fuel assemblies 708
22 Number of fuel rods per assembly 60
23 Rod array in assembly 8x8
24 Clad material Zircaloy
25 Clad thickness (mm) 0 86
26 Number of control rods or assemblies 169
27 Type Crossium
28 Additional shutdown systems Boron injection
29 Control rod neutron absorber material B4C/Hf
30 Soluble neutron absorber not used
31 Burnable poison material and form Gd

### B. REACTOR COOLANT SYSTEM

**B1. Coolant**

32 Coolant medium and inventory Light water
33 Design coolant mass flow through core (kg/s) 4,944x103
34 Cooling mode (forced/natural) Natural
35 Operating coolant pressure (bar) 70
36 Core inlet temperature (C)
37 Core outlet temperature (C)

**B2. Reactor pressure vessel/tube**

38 Overall length of assembled vessel/tube (m) 25 0
39 Inside vessel/diameter (m) 6 5
40 Average vessel/tube thickness (mm) --
41 Vessel/tube material Carbon steel
42 Lining material --
43 Design pressure (bar) 87 9
44 Gross weight (tonne) --

**B3. Steam generator** None

45 Number of steam generators --
46 Type once through --
47 Configuration --
48 Tube material --
49 Shell material --
50 Heat transfer surface per steam generator ($m^2$) --
51 Thermal capacity per steam generator (MW) --
52 Feed water pressure (bar) --
53 Feed water temperature (C) --
54 Steam pressure (bar) --
55 Steam temperature (C) --

**B4. Pressurizer** None

56 Pressurizer total volume ($m^3$) --

57 Steam volume (full power/zero power, $m^3$)

**B5. Main coolant pumps** None

58 Number of cooling or recirculation pumps None

59 Type --

60 Pump mass flow rate (kg/s) --

61 Pump design rated head --

62 Pump nominal power (kW) --

63 Mechanical inertia (kg $m^2$) --

## C. CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS) None

64 Number of extraction lines

65 Number of pumps

66 Number of injection points

67 Feed and bleed connections

## D. CONTAINMENT

68 Type pressure suppression

69 Overall form Cylindrical

70 Structural material Steel

71 Liner material --

72 Simple/double wall Simple

73 Dimensions (diameter, height) (m) 23x29

74 Design pressure (bar) 3 16g

75 Design temperature (C) 171

76 Design leakage rate (% per day) 5

## III. SAFETY RELATED SYSTEMS

## A. DESIGN CONDITIONS

**A1. Fission product retention**

77 Containment spray system (Y/N) Y

a Duration (h)

b Flow rate ($m^3$/h)

c Mode of operation (active/passive) Active

d Safety graded (Y/N) No

78 F P sparging (Y/N) Yes

79 Containment tightness control (Y/N) No

80 Leakage recovery (Y/N) No

81 Guard vessel (Y/N) Yes

**A2. Reactivity control**

82 Absorber injection system (Y/N) Yes

a Absorber material Boron

b Mode of operation (active/passive) Passive

c Redundancy No

d Safety graded Yes

83 Control rods (Y/N) Yes

a Maximum control rod worth (pcm) 1 000

b Mode of operation (active/passive) Active

c Redundancy One stuck rod margin

d Safety graded Yes

**A3. Decay heat removal**

*A3-1 Primary side*

84 Water injection

a Actuation mode (manual/automatic) Automatic

b Injection pressure level (bar) 5 0

c Flow rate (kg/s) Accumulator (10 bar)

d Mode of operation (active/passive) Passive

e Redundancy Single failure

f Safety graded (Y/N) Yes

85 Water recirculation and heat removal CUW

a Intermediate heat sink (or heat exchanger) Yes

b Mode of operation (active/passive) Active

c Redundancy None

d Self sufficiency (h) --

e Safety graded Yes

*A3-2 Secondary side*

86 Feed water
- a Actuation mode (manual/automatic) Automatic
- b Flow rate (kg/s) 25
- c Mode of operation (active/passive) Active
- d Redundancy None
- e Self sufficiency (h) 8
- f Safety graded Yes

87 Water recirculation and heat removal RHR
- a Ultimate heat sink (cold source) See water
- b Mode of operation (active/passive) Active
- c Redundancy None
- d Self sufficiency (h) --
- e Safety graded No

*A3-3 Primary pressure control*

88 Implemented system (Name) Safety relief valve
- a Actuation mode (manual/automatic) Automatic
- b Side location (primary/secondary circuit) Primary
- c Maximum depressurization rate flow area 0 057 $M^2$
- d Safety graded Yes

## B. SEVERE ACCIDENT CONDITIONS

### B.1 Fission products retention

89 Containment spray system (Y/N) Yes
90 F P Sparging (Y/N) No
91 Containment tightness control (Y/N) No
92 Leakage recovery (Y/N) No
93 Risk of recriticality (Y/N) No

### B.2 Recriticality control

94 Encountered design feature
- a Mode of operation (A/P) Passive
- b Safety graded Safety

### B.3 Debris confining and cooling

95 Core debris configuration (core catcher) No

96 Debris cooling system (name) accumulator
- a Mode of operation (A/P) Passive
- b Self sufficiency Yes
- c Safety graded (Y/N) Yes

### B.4 Long term containment heat removal

97 Implemented system Water wall
- a Mode of operation (A/P) P
- b Self sufficiency (h) 72
- c Safety graded (Y/N) Yes

98 Intermediate heat sink Suppression pool
- a Self sufficiency (h) 10
- b Safety graded (Y/N) Yes

99 External coolant recirculation
- a Implemented components RHR
- b Mode of operation (A/P) A
- c Self sufficiency (h) --
- d Safety graded (Y/N) N

100 Ultimate heat sink
- a Self sufficiency (h) --
- b Safety graded (Y/N) N

### B.5 Combustible gas control

101 Covered range of gas mixture concentration Whole range

102 Modes for the combustible gas control Yes
- a Containment inertation Yes
- b Gas burning No
- c Gas recombining Yes
- d Others No

### B.6 Containment pressure control

103 Filtered vented containment (Y/N) No
- a Implemented system
- b Mode of operation (Y/N)
- c Safety graded

104 Pressure suppression system (Y/N)
- a Implemented system Suppression pool

b Mode of operation Passive
c Safety graded (Y/N) Yes

### C. SAFETY RELATED I&C SYSTEM

Automatic load following (Y/N) No
* range (% power)
* maximum rate (%/min)

Load rejection without reactor trip (Y/N) No
Full Cathode Ray Tubes (CRT) display (Y/N) --
Automated start-up procedures (Y/N) --
Automated normal shutdown procedures (Y/N) --
Automated off normal shutdown procedures (Y/N) --
Use of field buses and smart sensors (Y/N) --
Expert systems or artificial intelligence advisors (Y/N) --
Protection system backup (Y/N)

### D. EMERGENCY POWER SUPPLY SYSTEM

105 Type (diesel, gas, grid connection) Diesel
106 Number of trains 2

### E. AC/DC SUPPLY SYSTEM

107 Type (rectifier, converter, battery) --
108 Estimated time reserve (hr) --

## IV. CONVENTIONAL THERMAL CYCLE

### A. TURBINE SYSTEM

109 Type TCDF-52
110 Overall length (m) 22
111 Width (m) --
112 Number of turbines/reactor 1
113 Number of turbine sections per unit HP/LP/LP/LP
114 Speed (rpm) 1500

### B. STEAM CHARACTERISTICS

115 H P inlet pressure bar 69 2 bar
116 H P inlet temperature (C) 655 8 C
117 H P inlet flowrate (kg/s) 2,892 t/h
118 L P inlet pressure 6 16 bar
119 L P inlet temperature 710 1 c
120 L P inlet flowrate 2,389 t/h

### C. GENERATOR

121 Type (3-phase synchronous, DC) 3-phase synchronous
122 Apparent power (MVA) 670
123 Active power (MW) --
124 Frequency (hz) 60
125 Output voltage (kV) 20
126 Total generator mass (t) --
127 Overall length --
128 Stator housing outside diameter

### D. CONDENSER

129 Number of tubes --
130 Heat transfer area --
131 Flowrate ($m^3/s$) 41 9
132 Pressure (m/bar) 722 mmHg
133 Temperature (C) 19

### E. CONDENSATE PUMPS

134 Number HPx3, LPx3
135 Flowrate 1850 $M^3/h$
136 Developed head HP 190m, LP 150 m
137 Temperature --
138 Pump speed --

## 7.4. SPWR REACTOR SYSTEM DESCRIPTION AND DEVELOPMENT

### 7.4.1. Basic objectives and features

The SPWR (System-integrated PWR) employs a new concept 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 basic design feature of SPWR is an integrated PWR with 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 counted as further distinctive features.

### 7.4.2. Design description

The SPWR concept could be applied with a wide output range up to 600 MWe. The following descriptions are for a 600 MWe power plant (1,800 MWt).

#### *7.4.2.1. Nuclear Steam Supply System*

Figure 7.4.1. shows the concept of SPWR. The reactor consists of reactor pressure vessel (RPV), reactor core, integrated steam generator (SG), main coolant pump (MCP), and pressurizer. The RPV, covered with a water-tight shell, is installed in a water-filled containment vessel (CV). For cooling the CV water, a heat pipe type passive cooling system is provided.

(1) Reactor Pressure Vessel

As shown in Figure 7.4.1., the RPV is relatively large in size because it is an integrated PWR. This provides a large primary water inventory and increases the distance between the reactor core and the RPV, thus reducing the neutron fluence to the RPV. At present, the fabrication technology of the RPV is well developed and there is no major problem to realize the RPV of a 600 MWe SPWR.

(2) Reactor core

The reactor core consists of 199 hexagonal fuel assemblies. Each fuel assembly consists of 325 fuel rods with burnable poison similar to that of well established PWRs. The dimensions and composition of the fuel rod are the same as those of the existing PWR. The average power density is designed relatively low, 65 $MW/m^3$.

The SPWR has no control rod and reactivity is controlled by the core inherent characteristics and chemical shim. Natural boron content in the primary coolant at the rated operating condition is 1,000 ppm in the BOEC (Beginning of Equilibrium Cycle) and 50 ppm in the EOEC (End of Equilibrium Cycle).

(3) Steam Generator

The once-through helical coil type SG is located in the annulus space above the core and has the 5,384 heat transfer tubes. The heat transfer tubes, made of INCOLOY-800, are 19 mm in outer diameter and 2.0 mm in thickness.

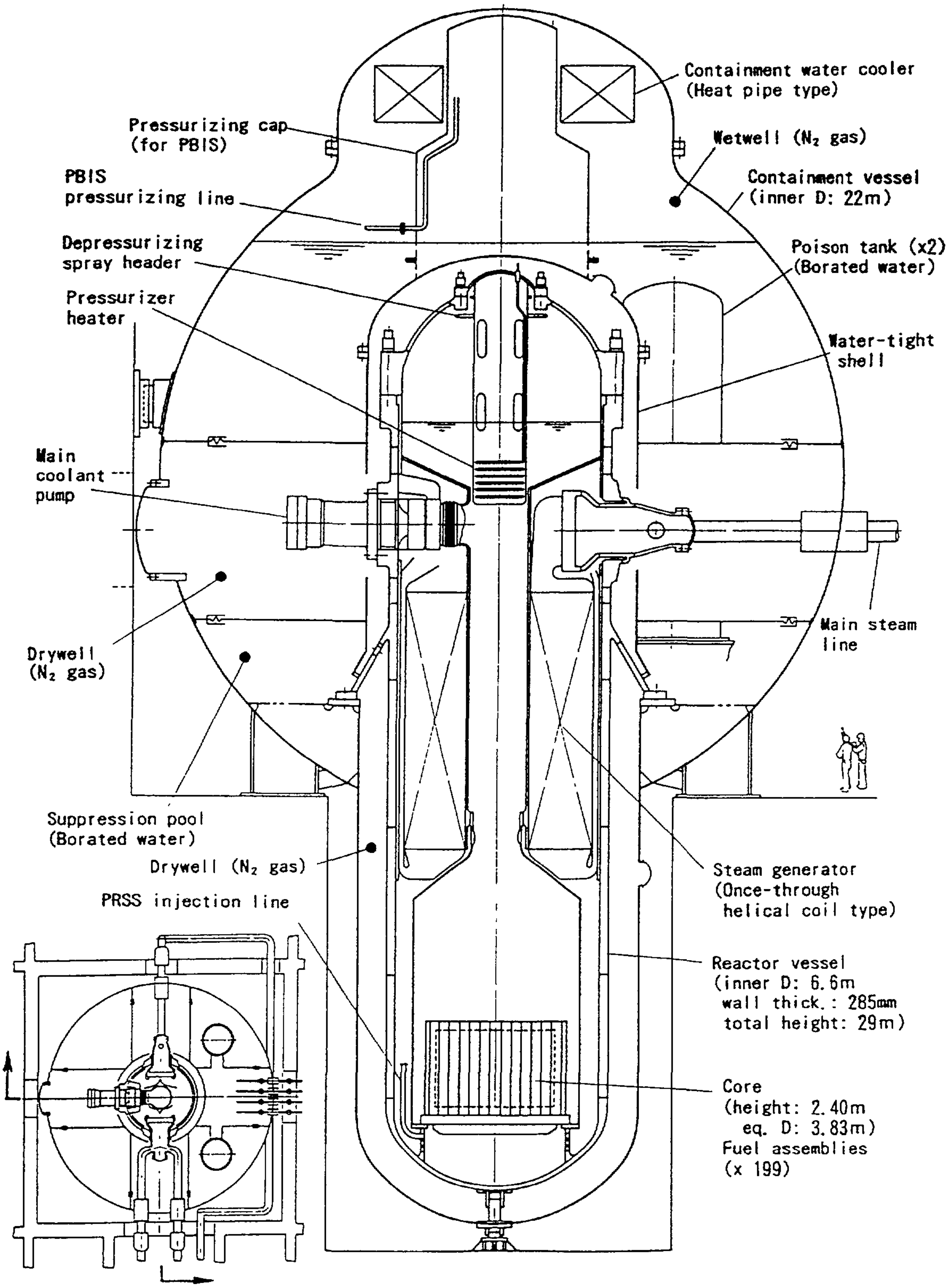


*Fig. 7.4.1. Concept of SPWR (600MWe)*

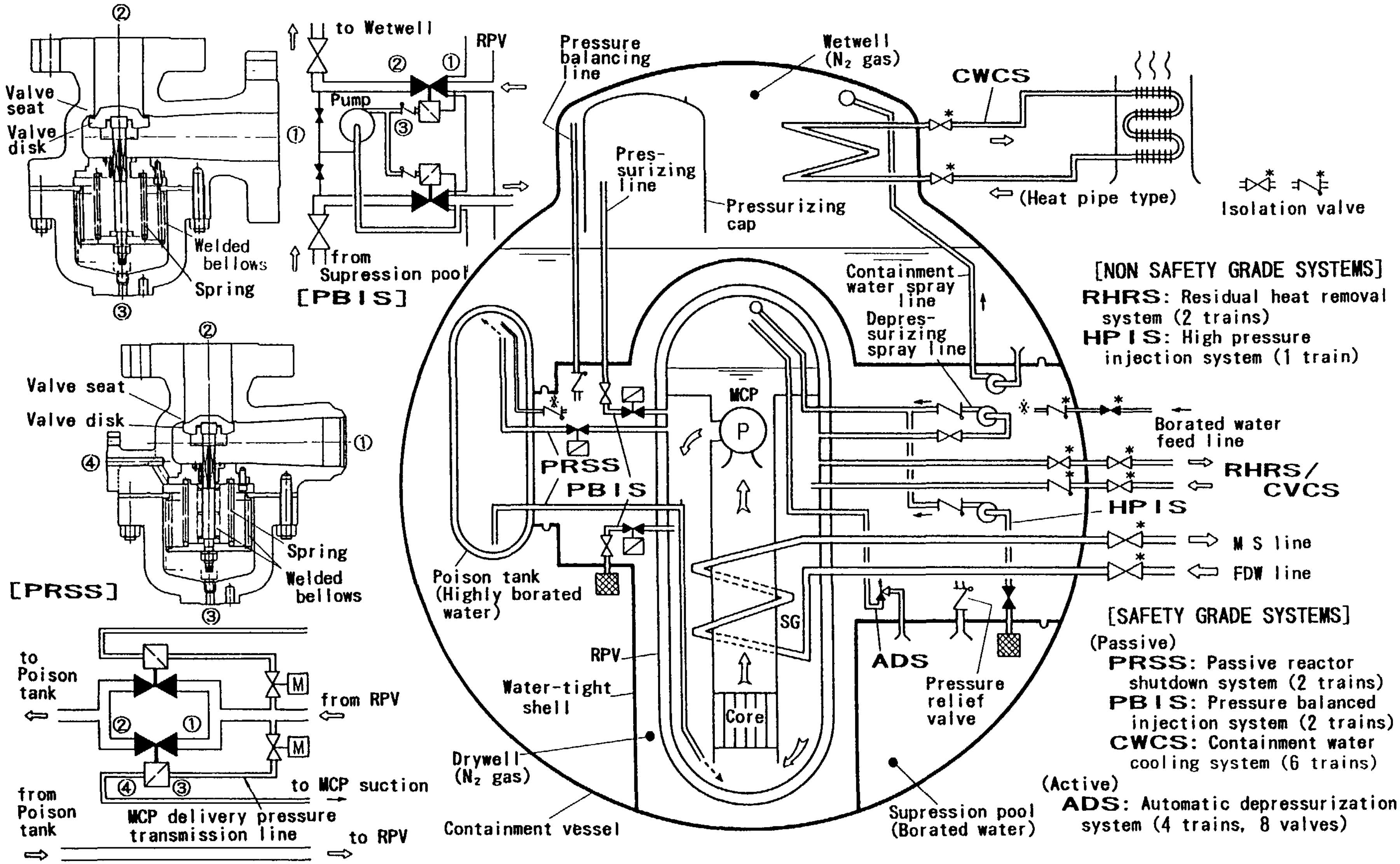


*Fig 7.4.2. Concept of SPWR Safety Systems*

(4) Main coolant pump

The MCP is located at the side of the RPV. Since the SPWR has only one MCP, an MCP seizure accident was considered to be severe in view of the danger of DNBR (Departure from Nuclear Boiling Ratio). A preliminary calculation shows that a burn-out of the fuel never occurs because of the low core power density and a high natural circulation capability due to the low flow resistance and high water head of the primary circuit.

(5) Pressurizer

The pressurizer, which is located at the top of the RPV, consists of an electric heater 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 changes in coolant volume due to temperature changes in the primary system and mitigates any pressure transient. Therefore it contributes to the good controllability of SPWR which has no control rod.

(6) Poison tank for passive reactor shutdown

Two poison tanks which contain highly borated water (12,000 ppm of boron concentration) are installed in the CV for the passive reactor shutdown system (PRSS). Each poison tank (60 $m^3$) is connected to the RPV by two pipes. The upper pipe has two normally closed hydraulic pressure valves. The pressure in the poison tank is maintained to be equal to that of the primary system under normal operation. The borated water in the tank is cooled by the CV water. The inside wall and the nozzle part of the poison tank have liners for mitigating a thermal stress due to temperature change during poison injection.

(7) Containment vessel

The SPWR adopts a water-filled CV. The RPV is covered with a water-tight shell. A mirror insulator of laminated thin stainless steel plates is mounted inside the shell. Heat loss is estimated to be below 1 MW. The space between the water-tight shell and the RPV corresponds to a drywell of the suppression type CV of conventional BWRs. To allow for a pipe rupture in this space, the shell is equipped with pressure relief valves. The advantages of the water-filled CV are the compactness of the reactor plant and ease of application of a passive decay heat removal system.

The CV is made of steel plate of less than 38 mm in thickness. Post-welding annealing is not required with this thickness. Relatively high design pressure and temperature is possible with such thin plate due to the compactness of the CV. Design pressure and temperature of the CV are 11 bar and 184°C, respectively. The large difference between the CV and atmospheric temperatures contributes to the compactness of the CWCS (Containment Water Cooling System).

(8) Fuel handling

Since the upper part of the core is narrow as shown in Figure 7.4.1., 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 has already been made for the former SPWR concept and has clarified its feasibility (See reference 2):

*7.4.2.2. Balance of plant*

(1) Main Steam System

The main steam system of SPWR is similar to that of the existing LWRs, which consists of one high pressure turbine (HPT) and two low pressure turbines (LPTs). The steam re-heater is installed between the HPT and LPTs.

(2) Feedwater System

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 the existing LWRs.

The auxiliary feedwater system is described in section 7.4.2.4.

*7.4.2.3. Instrumentation, control and electrical systems*

(1) It is considered that the next generation reactor will adopt an automatic control system for normal operation and passive mitigation of accidents.

(2) Control scheme

Reactivity control by chemical shim is a well-established technology proved in conventional PWRs. Preliminary analyses show excellent controllability of the SPWR including reactor start-up even though it has no control-rod.

- Reactivity change due to fuel burn-up is slow and it can be easily compensated by control of boron concentration.

- Load change can be followed-up automatically by negative coolant temperature coefficient of reactivity.

- Preliminary analysis shows the power can follow 50%-100% load change. In this analysis, the average temperature of the primary coolant changes only 8°C even at the 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 can occur during slow power changes, such as a daily cycle, is suppressed due to the low core power density. The primary coolant temperature change due to load following is allowed because the steam pressure of the once-through helical coil type SG is capable of being set independently to 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 Xe spatial oscillation. Controllability of power oscillation due to Xe is an important issue because the SPWR has no control rod. It has been confirmed that the SPWR core has satisfactory stability through analysis conducted on the 1,800 MWt SPWR core in the manner commonly applied on PWRs.

*7.4.2.4. Safety consideration*

The safety systems of the SPWR consist of passive and active systems as shown in Figure 7.4. 2. Brief descriptions of the safety consideration are presented below.

(1) Inherent Safety Characteristics

Inherent safe characteristics of SPWR are as follows:

- primary circuit integration in the RPV which eliminates a large break LOCA,
- large primary coolant inventory which allows safety injection systems to have enough time in a LOCA condition,
- negative reactivity coefficient which suppresses power increase under off-normal conditions (similar to the conventional PWRs),
- no possibility of large rapid reactivity insertion, accomplished by the elimination of control rods and fail safe pure water injection system.

(2) Passive Safety Features

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 using the passive safety systems such as Passive Reactor Shutdown System (PRSS), Pressure Balanced Injection System (PBIS) and Containment Water Cooling System (CWCS), and active Automatic Depressurization System (ADS).

- Passive Reactor Shutdown System (PRSS)

The PRSS is a poison injection system driven by natural circulation, which consists of two 100% trains, each of which has one poison tank and two hydraulic pressure valves.

Under normal operation, the valve disk is kept in the closed position by the delivery pressure of the MCP against the spring force. The valves open passively due to loss of the delivery pressure of the MCP in case of MCP failure. If the delivery pressure of the MCP is lost, the disk is pushed back to open position by the spring force. Borated water in the poison tanks has enough ability to maintain a cold shutdown condition.

The PRSS is also used as an active reactor shutdown system. While the MCP continues to run, a scram signal can activate a motor driven valve to close the MCP delivery pressure transmission line and thus the hydraulic pressure valves open. In this case the poison is injected by forced circulation due to the MCP delivery pressure.

The time for the borated water in the poison tank to reach the core would be approximately 24 seconds by natural circulation and approximately 8 seconds by forced circulation. Shutdown time is relatively long in the SPWR. However, this is not regarded as an adverse feature because he SPWR has no possibility of rapid reactivity insertion and system transient response is slow.

- Pressure Balanced Injection System (PBIS)

The PBIS is a passive system that injects CV water (borated water: 2,500 ppm) to the RPV when a LOCA occurs. The PBIS consists of two 100% trains, each of which has an injection line to lead CV water to the RPV and a pressurizing line for equalizing the pressures in the RPV and the CV. Steam from the RPV led by the pressurizing lines is flown into the pressurizing cap in the wetwell and replace $N_2$ gas to rapidly equalize the pressure of the RPV and CV.

Each injection line or pressurizing line has a reliable hydraulic pressure valve actuated by the pressure difference between the RPV and CV to allow gravitational feed of CV water to the RPV. The hydraulic pressure valve which is directly connected to the RPV is a check valve of a spring actuation type and does not open until the design pressure difference (for example, 0.5 MPa) between the RPV and CV is reached.

- Containment Water Cooling System (CWCS)

The CWCS is a heat pipe type passive heat removal system and consists of six 25% trains. The containment water coolers (evaporator of heat pipe) are installed in the upper part of the wetwell of the CV. The heat transferred from the CV water by 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 CV is filled with $N_2$ gas. Therefore, CV water is sprayed onto the heat pipe coolers in order to maintain the normal operating condition of the CWCS.

(3) Active Safety

The objective of the active and non-safety grade systems of the SPWR is to prevent unnecessary challenges threatening safety, to minimize the time to terminate abnormal conditions, and to protect the investment in the facility. The active systems utilize previous experience of similar designs on the well-proven conventional PWR and BWR technology.

Automatic Depressurization System (ADS)

The ADS is an active safety grade system, whose functions are to depressurize the RPV to assist the PBIS to inject CV water to the RPV in a high pressure accident condition and to prevent the over pressure of the RPV. The ADS consists of 4 trains, each of which has two relief valves.

High Pressure Injection System (HPIS)

The HPIS is an active non-safety grade system (1 train) to inject CV water into the RPV under LOCA conditions and to control the primary coolant inventory in transients if necessary.

Residual Heat Removal System (RHRS)

The RHRS is a non-safety grade system (2 trains). The RHRS circulates water in the RPV to a heat exchanger outside the CV to remove decay heat under the 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 (CVCS)

The CVCS is used as a reactor shutdown system to backup the PRSS. The CVCS is an active non-safety grade system (1 train) and has a similar capability to that of conventional PWRs.

Auxiliary Feedwater System (AFWS)

The AFWS is a non-safety grade system to supply feedwater to the secondary side of the SG to remove heat from the core via the SG under a loss of main feedwater and other transients. Two types of pumps, turbine driven and motor driven, are installed to achieve high reliability and diversity.

(4) Safety analysis

Preliminary safety analyses have been performed on the former design (1,100 MWt) with the thermal-hydraulic transient analysis code RETRAN-2 to understand the basic characteristics of 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,
- Station blackout

The results show that the SPWR has an ability to maintain coolability of the fuel rods under accident conditions, even if the functions of the reactor shutdown systems are not considered.

A probabilistic safety assessment has been performed to optimize the design of the SPWR, especially in respect of the combination of passive and active safety systems. Preliminary results indicate the estimated core melt frequency is lower than $10^{-6}$/reactor year.

*7.4.2.5. Building and structures*

(1) Reactor building

The size of the reactor building is much smaller than that of the conventional LWRs because of the adoption of the small CV and simplication of the plant systems (dimensions: 52m x 44m x 53m height).

(2) Accessibility

The RPV, primary piping, valves and pumps can be inspected in the drywell similarly to those of BWRs. A preliminary study shows only 36 days are necessary for full inspection including refuelling (See Reference 8).

(3) Seismic consideration

The seismic design is considered much easier than those of conventional LWRs because the reactor plant is simple and compact

### 7.4.3. Safety Concepts

TABLE 7.4.1. MAIN SAFETY RELATED SYSTEMS IN THE SPWR

| Name | Safety grade | Main characteristics |
|---|---|---|
| Primary Circuit | x | Integrated PWR eliminating large size coolant piping<br>1 main coolant pump<br>1 steam generator |
| Control Element Absorber (PRSS) | x | Highly borated water injection passively from poison tank (2 trains, 1 tank and 2 valves/train) |
| Divers Reactivity Control System | x<br>- | Borated water injection by PBIS<br>Borated water injection by CVCS |
| PBIS | x | 2 trains (4 passive valves) |
| ADS | x | 2 valves X 4 trains |
| Water Filled Containment | x | Water inventory 3,200m$^3$ |
| CWCS | x | Passive cooling CV (6 trains) |
| HPIS | - | 1 train |
| Auxiliary Feedwater System (AFWS) | - | 2 pumps |

CV Containment Vessel
PRSS Passive Reactor Shutdown System (Safety grade)
PBIS Pressure Balanced Injection System (Safety grade)
ADS Automatic Depressurization System (Safety grade)
CWCS Containment Water Cooling System (Safety grade)
CVCS Chemical and Volume Control System (Non-Safety grade)
HPIS High pressure Injection System (Non-Safety grade)

TABLE 7.4.2. MAIN ACCIDENT INITIATORS FOR THE SPWR

- LOCA (primary) small size pipe only ( max size 129mm ID)
- LOCA (secondary) secondary pipe rupture (water or steam)
- LOCA (interfacing) steam generator tube rupture
- ATWS Anticipated transient without scram
- Primary transients
- Secondary transients (turbine trip)
- Loss of electric sources (turbine trip)
- Total loss of heat sink
- Total loss of the steam generator feedwater
- Station blackout

## TABLE 7.4.3. DESIGN FEATURES FOR PREVENTION AND PROTECTION LEVELS THAT REDUCE (R), SUPPRESS (S), THE INITIATORS FREQUENCY OR LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES (L)

| |
|---|
| PREVENTION LEVEL<br>LOCA (Primary)<br>- Reduced vessel fluence reduce initiator frequency<br>- Primary circuit integration avoid large size LOCA, reduces initiator frequency and limits consequences<br>- Adaption of the leak before break limits accident consequences<br>- Automatic depressurization limits accident consequences<br>LOCA (Secondary)<br>LOCA (Interfacing)<br>- Tough structure of SG Tubes, reduce initiator frequency |
| ATWS |
| Primary Transients<br>- Increased design margin reduce initiator frequency (eg fail safe pure water injection system)<br>Secondary transients<br>- Large primary coolant inventory and small secondary coolant inventory reduce the transient phenomena |
| Loss of electric sources<br>- Implementation of passive systems<br>- Battery backup power sources<br>Total loss of the cold sources (water)<br>- Water field containment initiator frequency<br>- Passive CV water cooling system reduce initiator frequency<br>Total loss of the SG feed water<br>Station blackout |
| PROTECTION LEVEL<br>LOCA (Primary)<br>- Primary circuit integration leakage limitation<br>- Large primary inventory<br>- Water injection automatic and passive avoid operators mis-operation<br>- Water filled containment limits the break flow<br>LOCA (Secondary)<br>LOCA (Interfacing)<br>- Easy SG isolation |
| ATWS<br>- Negative temperature coefficient<br>- Passive reactor shutdown system |
| Primary transients<br>- Lower power density<br>Secondary transients<br>- Large natural circulation capability<br>- Large primary inertia |
| LOSS of electric sources not critical<br>Total loss of heat sink not critical<br>Total loss of the SG feed water not critical<br>Station blackout not critical |

TABLE 7.4.4. DESIGN FEATURES FOR MITIGATION LEVEL OF THE SPWR

| Safety functions | Systems (Cf Tab 7 4 1 ) | Passive/Active | Design features/remarks |
|---|---|---|---|
| Design Basis<br><br>Fission product containment | <br>Cladding<br>Primary circuit<br>Containment | <br>P<br>P<br>P | |
| Coolant Inventory | Water filled containment<br>PBIS | P<br>P | |
| Decay heat removal | PBIS<br>CWCS | P<br>P | Natural circulation<br>Heat pipe type cooling system |
| Reactivity control | PRSS<br>CVCS | P<br>P | |
| Primary circuit pressure control | ADS | A | |
| Severe Accident<br>Containment temperature and pressure control | <br>Water filled containment<br>CWCS | <br>P<br>P | |
| Heat removal | PBIS and CWCS | P | |
| Tightness control | Water filled containment | P | |
| Infram gas control | Not necessary | P | Inert gas environment ($N_2$) |
| Fission product containment | Water filled containment | P | |
| Corium management | Core catcher | P | under consideration |

PBIS Pressure Balanced Injection System
ADS Automatic Depressurization System
CWCS Containment Water Cooling System
CVCS Chemical and Volume Control System
PRSS Passive Reactor Shutdown System

### 7.4.4. Design data questionnaire

**I. GENERAL INFORMATION**

1 Design name SPWR (System-integrated PWR)
2 Designer/Supplier address JAERI
3 Reactor type Integrated PWR
Number of modules/per plant 1
4 Gross thermal power (MW-th) per reactor 1 800
5 Net electrical output (MW-e) per reactor 600
6 Heat supply capacity (MW-th) 0

**II. BASIC DESIGN DESCRIPTION**

**A. CORE AND REACTIVITY CONTROL**

7 Fuel material $UO_2$
8 Fuel inventory (tones of heavy metal) 74 8
9 Average core power density (kW/liter) 65 1
10 Average fuel power density (kW/kgU) 24 1
11 Maximum linear power (W/m) 28,000
12 Average discharge burnup (MWd/t) 48,000
13 Initial enrichment or enrichment range (Wt%)
14 Reload enrichment at the equilibrium (Wt%) 4 5 & 4 0
15 Refuelling frequency (months) 24
16 Type of refuelling (on/off power) off
17 Fraction of core withdrawn (%) 34
18 Moderator material and inventory ($m^3$) $H_2O$, 720
19 Active core height (m) 2 4
20 Core diameter (m) 3 83
21 Number of fuel assemblies 199
22 Number of fuel rods per assemblies 325
23 Rod array in assembly Triangle
24 Clad material Zry
25 Clad thickness (mm) 0 57
26 Number of control rods or assemblies Non
27 Type
28 Additional shutdown systems Borated water injection
29 Control rod neutron absorber material
30 Soluble neutron absorber Natural boron
31 Burnable poison material and form Gd (similar to existing PWR)

**B. REACTOR COOLANT SYSTEM**

**B1. Coolant**

32 Coolant medium and inventory ($m^3$) $H_2O$, 720
33 Design coolant mass flow through core (kg/s) 12,300
34 Cooling mode (forced/natural) forced
35 Operating coolant pressure ($kg/cm^2a$) 138
36 Core inlet temperature (°C) 288
37 Core outlet temperature (°C) 314

**B2. Reactor Pressure Vessel**

38 Overall length of assembled vessel (m) 29
39 Inside vessel diameter (m) 6 6
40 Average vessel thickness (mm) 285 (includes liner)
41 Vessel material JIS SFVQ1A
42 Lining material Stainless steel
43 Design pressure ($kg/cm^2a$) 151
44 Gross weight (tone) 1,534

**B3. Steam generator**

45 Number of steam generators 1
46 Type Once-through helical coil
47 Configuration (horizontal/vertical) Vertical
48 Tube material Incoloy 800
49 Shell material
50 Heat transfer surface per steam generator ($m^2$) 17,000
51 Thermal capacity per steam generator (MW) 1,800
52 Feed water pressure ($kg/cm^2a$) 66
53 Feed water temperature (°C) 210

54 Steam pressure ($kg/cm^2a$) 56
55 Steam temperature (°C) 295

**B4. Pressurizer**

56 Pressurizer total volume ($m^3$) 160
57 Steam volume (full power/zero power, $m^3$) 105/77

**B5. Main coolant pumps**

58 Number of cooling or recirculation pumps 1
59 Type Canned motor type
60 Pump mass flow rate (kg/s) 12,300
61 Pump design rated head (m) 24
62 Pump nominal power (kW) 3,600
63 Mechanical inertia ($kg/m^2$)

## C. CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS)

64 Number of extraction lines 1
65 Number of pumps
66 Number of injection points 1
67 Feed and bleed connections

## D. CONTAINMENT

68 Type Water-filled
69 Overall form (spherical/cyl.) Spherical and cyl
70 Structural material JIS SGV49,50
71 Line material Non
72 Simple/double wall Simple
73 Dimensions (diameter, height) (m) 22, 38
74 Design pressure ($kg/cm^2a$) 11
75 Design temperature (°C) 184
76 Design leakage rate (% per day)

# III. SAFETY RELATED SYSTEMS

## A. DESIGN CONDITIONS

**A1. Fission Product Retention**

77 Containment spray system (Y/N) N
a Duration (h)
b Flow rate ($m^3/h$)
c Mode of operation (active/passive)
d Safety graded (Y/N)
78 F P sparging (Y/N) N
79 Containment tightness control (Y/N) Y
80 Leakage recovery (Y/N) Y
81 Guard vessel (Y/N) N

**A2. Reactivity control**

82 Absorber injection system (Y/N) Y
a Absorber material Borated Water
b Mode of operation (active/passive)
c Redundancy Y
d Safety graded Y
83 Control rods (Y/N) N
a Maximum control rod worth (pcm)
b Mode of operation (active/passive)
c Redundancy
d Safety graded

**A3. Decay heat removal**

*A3 1 Primary side*

84 Water injection Y (PBIS)
a Actuation mode (manual/automatic) Automatic
b Injection pressure level ($kg/cm^2a$)
c Flow rate (kg/s)
d Mode of operation (active/passive) Passive
e Redundancy Y
f Safety graded (Y/N) Y

85 Water recirculation and heat removal N
a Intermediate heat sink (or heat exchanger)
b Mode of operation (active/passive)
c Redundancy
d Self sufficiency (h)
e Safety graded

*A3 2. Secondary side*

86 Feed water Y (HPIS)
a Actuation mode (manual/automatic) Automatic
b Flow rate (kg/s)
c Mode of operation (active/passive) Active
d Redundancy N
e Self sufficiency (h)
f Safety graded N

87 Water recirculation and heat removal N
a Ultimate heat sink (cold source)
b Mode of operation (active/passive)
c Redundancy
d Self sufficiency (h)
e Safety graded

*A3 3 Primary pressure control* Y

88 Implemented system (Name) ADS
a Actuation mode (manual/automatic) Automatic
b Side location (primary/secondary circuit) Primary
c Maximum depressurization rate (bar/s)
d Safety graded Y

## B. SEVERE ACCIDENT CONDITIONS*

### B.1. Fission products retention

89 Containment spray system (Y/N) N
90 F P Sparging (Y/N) N
91 Containment tightness control (Y/N) Y

* All systems must be qualified to operate under the accident conditions

92 Leakage recovery (Y/N) Y
93 Risk of recritically (Y/N) N

### B.2. Recriticality control

94 Encountered design feature N
a Mode of operation (A/P)
b Safety graded

### B.3. Debris confining and cooling

95 Core debris configuration (core catcher) Under consideration
96 Debris cooling system (name) CWCS
a Mode of operation Passive
b Self sufficiency
c Safety graded (Y/N)

### B.4. Long term containment heat removal

97 Implemented System Y (CWCS)
a Mode of operation (A/P)
b Self sufficiency (h)
c Safety graded (Y/N)
98 Intermediate heat sink N
a Self sufficiency (h)
b Safety graded (Y/N)
99 External coolant recirculation N
a Implemented components
b Mode of operation (A/P)
c Self sufficiency (h)
d Safety graded (Y/N)
100 Ultimate heat sink Atmosphere
a Self sufficiency (h)
b Safety graded (Y/N)

### B.5. Combustible gas control

101 Covered range of gas mixture concentration
102 Modes for the combustible gas control Y ($N_2$ gas)
a Containment inertiation

b Gas burning
c Gas recombining
d Others

**B.6. Containment pressure control**

103 Filtered vented containment (Y/N) Under consideration
a Implemented system
b Mode of operation (A/P)
c Safety graded

104 Pressure suppression system (Y/N) Y
a Implemented system Suppression pool
b Mode of operation (A/P) Passive
c Safety graded Y

## C. SAFETY RELATED I&C SYSTEM

Automatic load following (Y/N) Y
- range (% power)
- maximum rate (% min)

Load rejection without reactor trip (Y/N) Y
Full Cathode Ray Tubes (CRT) display (Y/N) Y
Automated start-up procedures (Y/N) Y
Automated normal shutdown procedures (Y/N) Y
Automated off normal shutdown procedures (Y/N) Y
Use of field buses and smart sensors (Y/N) Y
Expert systems or artificial intelligence advisors (Y/N) Y
Protection system backup (Y/N) Y

## D. EMERGENCY POWER SUPPLY SYSTEM

105 Type (diesel, gas, grid, connection) Diesel (non-safety grade)
106 Number of trains 2

## E. AC/DC SUPPLY SYSTEM

107 Type (rectifier, converter, batter)
108 Estimated time reserve (hr)

# IV. CONVENTIONAL THERMAL CYCLE

## A. TURBINE SYSTEM

109 Type Steam turbine
110 Overall length (m)
111 Width (m)
112 Number of turbines/reactor 1
113 Number of turbine sections per unit (e g HP/LP/LP) HP/LP/LP
114 Speed (rpm) 1.500 or 1,800

## B. STEAM CHARACTERISTICS

115 H P inlet pressure (kg/cm$^2$a) 54 0
116 H P inlet temperature (°C) 292
117 H P inlet flow rate (t/h) 3,055
118 L P inlet pressure (kg/cm$^2$a) 11 0
119 L P inlet temperature (°C) 240
120 L P inlet flow rate (per section, t/h) 1,219

## C. GENERATOR

121 Type (3-phase synchronous, DC) 3-phase synchronous
122 Apparent power (MVA)
123 Active power (MW) 610
124 Frequency ($H_z$) 50 or 60
125 Output voltage (kV)
126 Total generator mass (t)
127 Overall length
128 Stator housing outside diameter

## D. CONDENSER

129 Number of tubes
130 Heat transfer area
131 Flow rate (m$^3$/S)

132 Pressure (m/bar)
133 Temperature (°C)

**E. CONDENSATE PUMPS**
134 Number
135 Flow rate
136 Developed head
137 Temperature
138 Pump speed

### 7.4.5. Project status

*7.4..5.1. Entities involved*

The design study has been performed by the Japan Atomic Energy Research Institute (JAERI) supported by the Science and Technology Agency (STA) since 1986, cooperated with Ishikawajima-Harima Heavy Industries, Kishikawa Special Valve, Japan Research Institute, Computer Software Development and Mitsubishi Atomic Power Industries.

*7.4..5.2. Design status*

The concept of the SPWR has been well established. At an early stage, the design study was based on a poison tank installed in the RPV and a reactor power of 1,000 MWt (350 MWe). Thereafter, the present design has been established incorporating modifications such as relocation of the poison tank outside the RPV and 1,800 MWt (600 MWe).

A special team consisting of specialists within JAERI independently performed a peer review of the SPWR design in 1992. The design team of the SPWR is now improving the design in order to incorporate the advice made by the special team.

*7.4.5.3. Research and development work*

The function of the hydraulic pressure valve for the PRSS was successfully confirmed by a test with a half scale model. Since there is no operating experience of a reactor without control rods, construction of a demonstration reactor is considered necessary.

JAERI has been developing the marine reactor X (MRX) for marine use (See MRX design). Because basic design characteristics of the MRX are similar to those of the SPWR, JAERI is considering an effective combination of development programmes for these reactors.

### 7.4.6. Project economics

Simplicity and compactness are essential for good economics. As shown in Figure 7.4.1. and 7.4.2., the SPWR realizes a remarkable simplicity and compactness compared with the existing LWRs.

(1) Simplicity

Simplicity of the SPWR is achieved by the following items:

- Simplicity common to all integrated PWRs: elimination of the large primary pipes and reduction of the number of tanks due to primary circuit accommodation in the RPV.

- Simplicity due to absence of control rods: elimination of control rods and drive systems.
- Simplicity due to utilizing the water-filled CV for the passive safety systems: reduction of the number and capacity of systems, tanks and active components.

(2) Compactness

The compact CV and simple systems contribute to reduce the reactor building in size. In other words, the SPWR will have high compatibility to a large scale power plant as well as a small scale one. The reactor building of 1,200 MWe plant consists of two SPWRs requires less space compared with the building of the existing large scale LWRs.

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## 7.5. SIR™ SYSTEM DESCRIPTION AND DEVELOPMENT STATUS

### 7.5.1. Basic objectives and features

The SIR™ reactor was designed to fill a perceived market need for smaller reactors which would nevertheless be economically competitive. It was recognised that to be economically attractive it would have to have design innovations to overcome the economies of scale. The three initial partners, AEA Technology and Rolls Royce and Associates in the UK and ABB Combustion Engineering in the USA had all been independently looking at integral reactors in 1988 and agreed to co-operate in a joint project. Stone and Webster Engineering Company in the USA were brought into the partnership later to deal with balance of plant and architect engineering. It was an objective to produce a design which would be licensable in the USA and in the UK and therefore it would have a good chance of being licensable anywhere.

The objectives were to produce a reactor with the following features:

Enhance the public perception of safety
Reduce the need for operator action
Maximise system reliability and maintainability
Make maximum use of existing technology
Reduce the time needed for construction
Competitive cost

The reactor size was limited by the current diameter of pressure vessel which could be manufactured. This gave an electric power output of 320 MWe allowing for conservatism in the design margins. There is a prospect of increasing the power to 400 MWe.

The reactor vessel contains the core, twelve once through steam generators, six canned rotor pumps at a high level in the vessel, and a control rod assembly in each of the 65 fuel assemblies. The top part of the vessel forms the pressuriser with its electric heaters. There is a passive spray system in the pressuriser which takes water from the riser region and sprays it into the steam space in the event of pressure rise in the core. The containment has a novel form of pressure suppression where the water for pressure control is contained in a tank farm connected to the reactor cavity by large diameter pipes.

Although the module size is limited, a 1200 MWe station could be provided by having four reactor modules. Such an arrangement would take no more land space than a single large reactor.

### 7.5.2. Design description

#### *7.5.2.1. Nuclear steam supply system*

The SIR™ reactor is an integral pressurised water reactor, in which all major components are contained in a single pressure vessel. Figure 7.5.1. shows the general arrangement of the RPV and its internal structures. The principal features are:

Reactor core and internal design based on well proven ABB-CE technology

12 straight tube once-through steam generators located in an annulus around a central flow channel

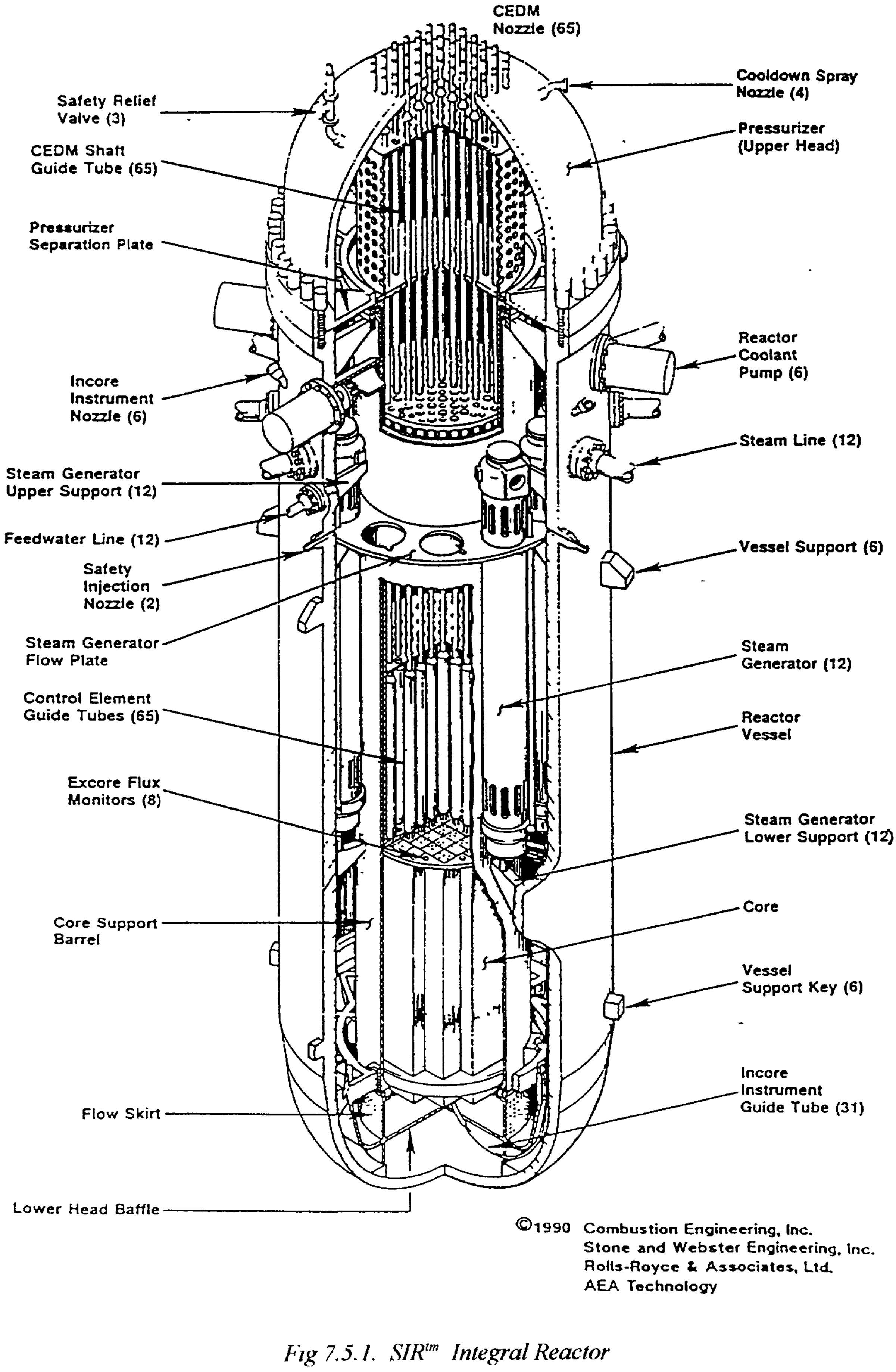


*Fig 7.5.1. SIR$^{tm}$ Integral Reactor*

6 glandless or wet motor pumps located directly onto the RPV above the SGs

The RPV closure head acts also as an integral pressuriser

The reactor core is located low down in the RPV

No bottom head penetrations

All secondary connections are headed up outside the vessel and there are no on site primary welds.

Advantages of the design are as follows:

Single Pressure vessel: This allows the complete elimination of all primary system pipework, the steam generator, pressuriser and accumulator pressure boundaries and therefore the elimination of the large break LOCA. The largest penetration of the RPV is the 70mm control rod guide tubes. The vessel is sized (23.8 m long, 5.8 m diam) so that it can be fabricated using present day methods and is transportable. All vessel internal structures can be completed at the point of fabrication limiting the site work to assembly. The performance of the RPV and its closures can be guaranteed using the methods developed for the Sizewell B PWR vessel. In addition its potential life expectancy is enhanced by the shielding afforded by the 1m of water separating the core from the vessel wall which makes the irradiation damage of the RPV at least $10^4$ times less than for a standard PWR. Also the elimination of the large LOCA reduces the threat of thermal shock to the vessel. All internals can be removed for repair or replacement or for carrying out RPV inspections.

Modular Steam Generators: The once through, straight tube design means that the steam generator tubes are in compressive rather than tensile stress, thereby removing stress corrosion cracking - a principal cause of problems in current designs of a plant. The design allows for in-service inspection and possible tube plugging with access from the secondary side steam lines. All 12 steam generators are individually isolatable, permitting continued operation to a managed outage in the event of a problem in one of them. There is sufficient redundancy to permit full power operation on 11 units. In the SIR™ reactor a steam generator can be replaced during a shut down adding about 15 days to a normal refuelling outage of 14 days.

Canned Rotor Pumps: With no shaft seals the small LOCA associated with seal failure is eliminated. The motors are relatively small and can be replaced by special provisions in the reactor cavity space without removing the vessel closure head. These provisions include diodes in the core barrel to allow natural circulation with the water level reduced to below the level of the pumps.

Reactor Internals Design: The design features standard ABB-CE System 80 type core and fuel. Operation and refuelling is without soluble boron making extensive use of burnable poisons. This eliminates the chemical plant associated with boron concentration changes, reducing intermediate level radioactive liquid waste and costs. The fuel assembly contains a 22 x 22 pin design with provision for 8 large control rods each occupying 4 lattice locations, a similar sized central instrumentation hole and 16 positions for insertion of $Gd_2O_3/Al_2O_3$ burnable poison in the second cycle. There are integral $Gd_2O_3/UO_2$ poisons in the fuel in addition to the insertable ones. Each fuel assembly contains a control rod assembly (CEA), permitting refuelling with all rods in. The length of the vessel provides a substantial capacity for natural circulation with a margin of more than a factor of two for decay heat removal. The core is fully instrumented with no bottom head penetrations. The flow arrangements minimise the horizontal forces on the core barrel.

Control Element Drive Mechanisms (CEDM): There are 65 CEDMs, one per fuel assembly. The design is that of the standard ABB-CE System 80 reactors using two sets of latches engaging onto a ridged drive shaft and energised by electro-magnetic coils, one for raising the latch and one for engaging it onto the ridged shaft. The latches and drive shaft are within the pressure boundary and the coils are outside, dry and air cooled. Tests have shown that these mechanisms are entirely satisfactory for operating in the saturated steam conditions found in the upper part of the integral pressuriser. On de-energising all the coils, the rods drop into the core due to their own weight.

Core Power: Coupled with the operation with burnable poison the core operates at a power density of 54 kW/l. This is approximately half that of a standard PWR. The refuelling interval is extended to two years giving a greater availability.

Integral Pressuriser: The top portion of the top closure head serves the additional role of pressuriser. It contains electric heaters to raise pressure although nuclear heating is necessary for initial pressurisation. There is a passive spray system that takes water from the top of the riser region and sprays it into the steam space in the pressuriser. Outward flow from the pressuriser is through a surge pipe where inward flow is prevented by fluidic diodes. This is less effective than a pumped system using cooler water. However, the pressuriser volume is 5 times larger per MW than many large PWRs and the transient response is more than adequate.

Safety systems: The safety grade residual heat removal system is the Secondary Condensing System (SCS) which uses four of the steam generators in a reconfigured mode with a closed loop to a condenser in the RWST. Water boils in the steam generators, the steam is condensed in the condenser and the condensate returns to the steam generator by gravity flow. There are four completely separate systems which are passive in operation. There is sufficient capacity in the RWST to last for 72 hours without operator attention. For further operation the tanks can be topped up.

There is a passive system for providing make up water in the event of a LOCA, the Emergency Coolant Injection System (ECIS). This uses steam injectors to pump water from the containment tanks into the RPV. The steam is taken from the top of the pressuriser. There is two times redundancy. This system can be combined with a bleed system to provide a second route for decay heat removal. Fig. 7.5.2. gives a schematic diagram of the safety systems and the non-safety grade heat removal systems.

*7.5.2.2. Balance of Plant Systems*

The once through steam generators have been described under the primary system. They are designed to produce steam with 28°C superheat. Four of the steam generators can be automatically reconfiguered to the SCS and four can be reconfigured to an active decay heat removal system ( the Steam Generator Recirculation System), where heat is passed to a process water cooler, which is the system normally used in the shut down condition. There is also the possibility of feed and bleed through the steam generators as an additional method of heat removal under operator control.

The steam generators can all be isolated at inlet and steam outlet to allow operation with a defective unit and to reduce the consequences of a steam tube rupture. All pipework up to and including these valves is designed for full primary pressure.

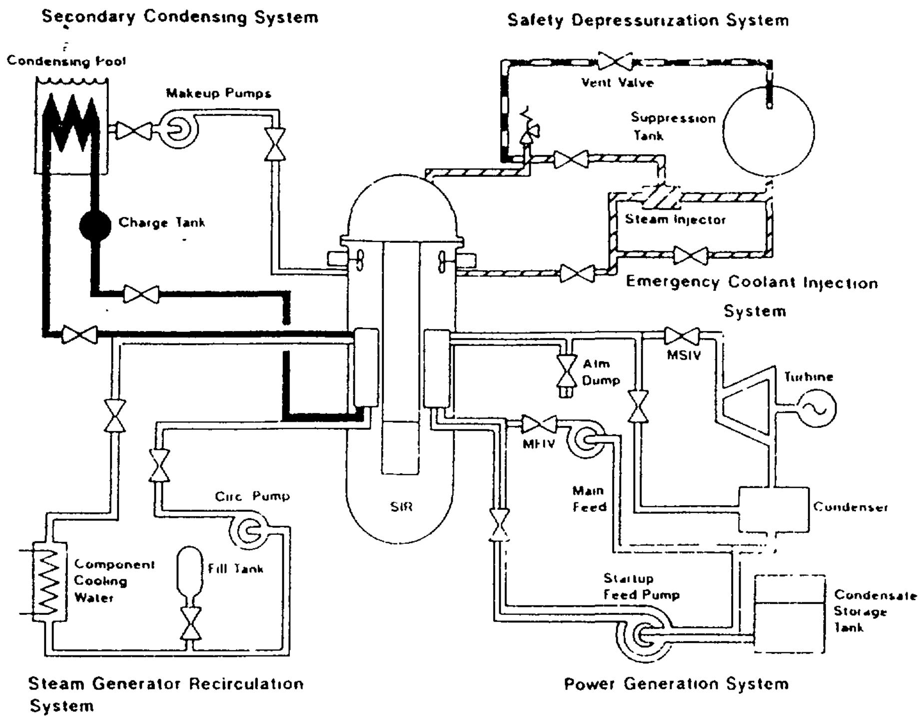


*Fig 7.5.2. Safety and other Heat Removal Systems*

The turbine generator is a tandem compound, dual-flow, condensing, single reheat machine consisting of one double-flow high pressure cylinder, one double-flow low pressure cylinder, one generator with stator and hydrogen intercooled rotor, an exciter, two moisture separator-reheater units, and an electrohydraulic control system. Steam bled off the low pressure turbine is used for feed heating.

The condenser is a single-shell, two tube bundle high integrity design with provision for deaeration and three 50% motor driven condensate pumps.

There is a diverse emergency shut down system which is initiated manually if it is not possible to insert the control rods. It allows for injection of boron into the reactor. Since inadvertent operation of this system must be completely prevented, it requires insertion of a spool piece to connect the boron tank to the reactor. The negative moderator temperature coefficient is sufficient to maintain the reactor in a hot shut down condition for several days until the spool piece can be inserted.

The plant includes all the normal plant facilities necessary for providing supplies of quality controlled air and water to all the areas and plant that need it. It also includes provision for the treatment, handling and disposal of gaseous, liquid and solid wastes.

The primary Chemistry and Volume Control System (CVCS) for the SIR$^{TM}$ reactor is greatly simplified compared to a typical PWR through the elimination of soluble boron. The CVCS comprises a high pressure purification system which provides for water purification, addition of additives as necessary to maintain the specified coolant chemistry conditions, and for removal of noble gases. The secondary condensate treatment is particularly important in once through steam generators to avoid build up of deposits at the boiling boundary. The condensate treatment plant is designed to ensure that there will be no steam generator problems from this source.

### *7.5.2.3. Instrumentation, control and electrical systems*

In the I&C systems the performance requirements to which the plant is designed are:

Reduce costs for design, operation and maintenance by: integrating nuclear supply systems and balance of plant systems into a unified control complex design, using standard components and solid state equipment, simplified field installation by extensive use of multiplexing including fibre optic links where isolation is required.

Improve man-machine interface by: providing plant overview information to the operator to improve comprehension, reducing the operator's information processing while meeting all of his information needs, providing a friendly transition to a new technology reducing the potential for human error that could affect safety or reliability and allow continued operation with failures improving operator comprehension with human engineering principles incorporated through the entire design process.

Improve plant performance by: extensive use of automatic controls, use of load follow systems, control sensor validation, pretrip control actions, and the use of variable protection limits.

Improve reliability by: using a systems approach to the control complex using digital microprocessor based systems for protection, monitoring and control; Designing control systems to be fault tolerant of I&C hardware and software failures, operator and technician errors and failures in components; providing diverse digital processing techniques for the plant protection system, diverse safety and non safety control systems and diverse information systems.

The control complex contains all of the protection, monitoring, and control systems necessary for the safe and efficient operation of the nuclear and balance of plant systems. The control complex is based on ABB-Combustion Engineering's Nuplex 80+ Advanced Control Complex. It is comprised of:

Control Room Panels
Data Processing Systems
Discrete Indication and alarm System
Plant Protection System
Component Control System

These five systems collect data from the plant, efficiently present the information to the operator, perform all automated functions and provide for direct manual control of components. The control room has been designed on anthropometric standards to allow for minimal acceptable manning levels. The panels are based on CRT and touch screen displays. The overall state of the plant is displayed on a large projected screen image.

### *7.5.2.4. Safety considerations and emergency protection*

The SIR™ plant has been designed from the beginning taking into account both deterministic and probabalistic safety considerations. It has taken into consideration both US and UK safety requirements and, in particular, has followed the EPRI design requirement document for innovative advanced reactors in most respects. It has been designed to require no operator intervention for at least 72 hours following any design basis accident and to have a grace period of several hours even following failure of the passive safety systems, which is not conceivable.

The passive safety features have been described in section 7.5.2.1. They comprise:

The Safety Injection System for core inventory control
The Secondary Condensing System for decay heat removal
Negative Moderator Temperature Coefficient which will give subcriticality in the event of core heat up accidents
Inerted Containment to prevent hydrogen combustion or detonation
Naturally cooled containment tanks as an ultimate heat sink.

These passive systems are backed up by active systems which operators will normally use to control the state of the plant and to protect the investment in it.

Analyses have demonstrated that no plant failures result in any core damage. This includes complete failure of control rod insertion or the most severe cooldown transient possible.

Severe accidents leading to core melt and release of radioactivity to the environment in present day PWRs are very unlikely events; the SIR™ design makes them even more unlikely. Since there are no vessel penetrations below 8m above the top of the core, there is no possibility of the sudden ejection of primary water that is associated with a large LOCA and there is adequate, redundant and diverse means for providing make up coupled with a long grace period even in the event of failure of all systems. The calculated core melt frequency is very low indeed. Even so, the consequences of core melt accidents are mitigated by the inerted containment and the pressure suppression containment which also provides a scrubbing action on fission products released into the containment.

*7.5.2.5. Buildings and structures*

The general arrangement of the plant is shown in the isometric diagram in fig 7.5.3. In addition there are the usual workshop and radwaste buildings which abut the main reactor building shown. The Control Room is located behind the wall at the top left of the figure.

The containment geometry is unique to the SIR™ reactor. It uses the pressure suppression principle which is well proven for BWRs is suited to integral PWRs due to the absence of any possibility of fast depressuration through rupture of large pipes. The containment consists of the reactor cavity itself, eight separate factory made pressure suppression tanks located four either side of the reactor building and the ducting which connects the tanks to the cavity. The reactor cavity is a concrete steel lined space below ground level which contains the reactor vessel and minimal surrounding space except at the level of the various pipe and connections and the pumps. The cavity is sealed at the top by a steel dome which covers the control rod drives and is removed for refuelling. It in term is covered by removable shielding. The suppression tanks form a tank farm either side of the building. They are situated behind concrete radiation shields which allow a free movement of air over the finned surfaces of the tanks to provide passive cooling of the water which half fills them. The tanks also contain the nozzles for the primary depressurisation system and the primary safety valves so that primary steam emitted from the vessel for whatever reason finds its way into the suppression tanks. The tanks also provide a source of water for the emergency injection system. When feed and bleed is in operation, the tanks thus provide both the source of feed and the sink for the bleed. Heat is removed by the external air cooling of the tanks by natural convection. To provide protection against seismic damage, both the reactor building and the tank farm are constructed on a single concrete raft.

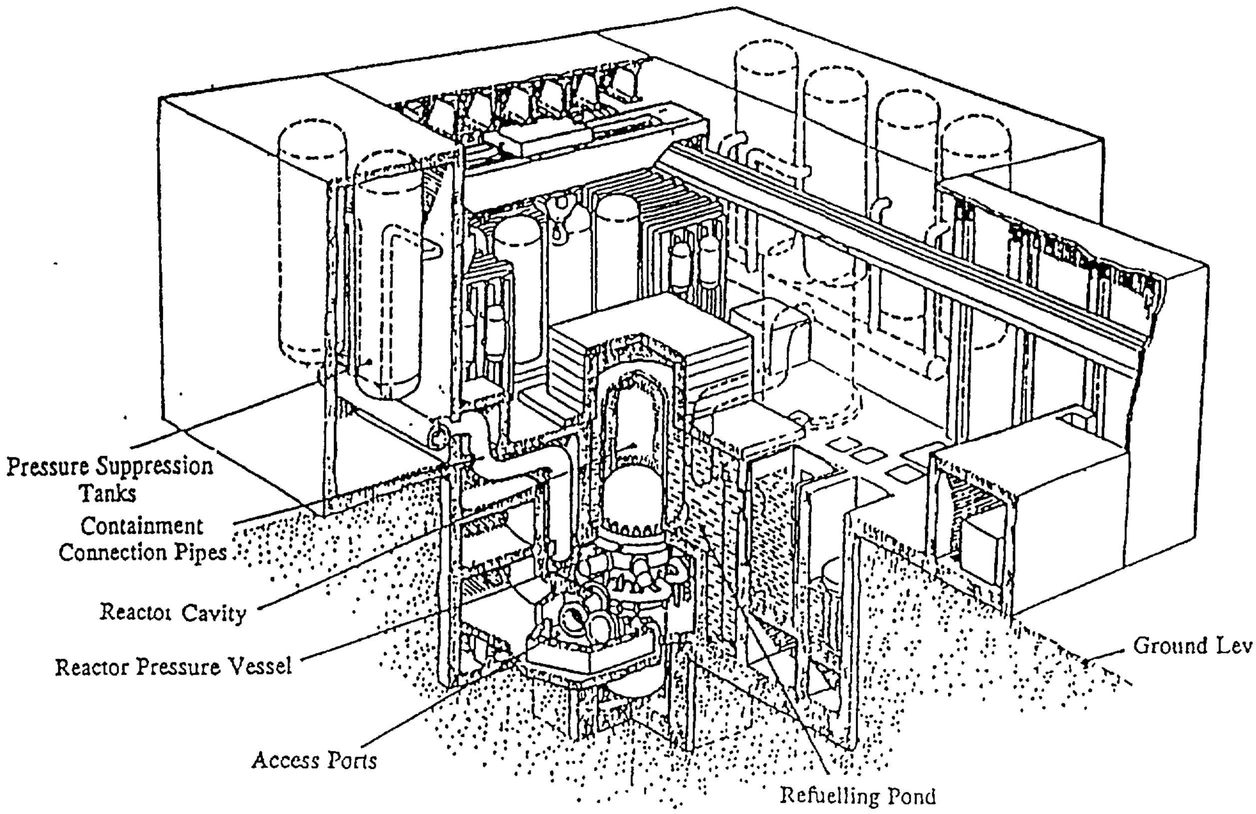


*Fig 7.5.3. 3-D Cut Away of SIR™ Plant Layout*

The RPV penetration level and the pump access level above it have been carefully designed to allow easy removal of a pump for maintenance and for access to the steam and feedline isolation valves. During normal operation the reactor cavity volume is purged with nitrogen to prevent hydrogen conflagration in the unlikely event of a severe accident.

The SIR™ reactor concept lends itself to multimodular stations with shared staff and services

### 7.5.3. Safety concepts

TABLE 7.5.1. MAIN SAFETY RELATED SYSTEMS IN THE SIR™ CONCEPT

| Name | Safety Graded | Main Characteristics |
|---|---|---|
| Primary Circuit | Yes | Reactor vessel<br>4 canned rotor pumps<br>12 steam generators |
| Control Element Absorber | Yes | 65 control rod drives<br>8 large fingers per rod |
| Diverse Reactivity Control System | Yes<br><br>No | negative temperature coefficient<br>Manual emergency boration |
| Emergency Coolant Injection System (ECIS). Passive | Yes | 2 steam injectors take steam from pressuriser and water from containment tanks |
| Secondary Condensing System (SCS). Passive heat removal system | Yes | 4 SGs used to feed steam to condensers in RWST |
| Primary Natural Circulation | Yes | Removes core decay heat in conjunction with SCS |
| Depressurisation System | Yes | Valve area 5800 $mm^2$ |
| Passive Containment Cooling System | Yes | Natural convection of air past finned surface |

The accident initiators have been considered in a PSA approach. The three main safety functions of reactivity control, heat rejection and vessel integrity protection must be preserved under all operational modes and all transient conditions. From a set of six generic transient conditions, under- and overpower, under- and over cooling, and under- and overpressure in all operational states transients corresponding to loss of one of the three safety functions can be defined. This gives a long list which can be reduced by consideration of consequences to a more manageable one. This leads to a short list of transients for detailed deterministic study. ATWS sequences which are caused by a plant failure in coincidence with an initiating transient are also considered. The transients of most significance are listed in Table 7.5.2.

TABLE 7.5.2. MAIN ACCIDENT INITIATORS FOR THE SIR™ REACTOR

- LOCA from the primary circuit
- Overcooling Transients - Secondary pipe rupture
- Inadvertent operation of ECIS
- Steam Generator Tube Rupture (SGTR)
- Loss of Flow (LOF)
- Primary Transients
- Secondary Transients
- Station Blackout

TABLE 7.5.3. DESIGN FEATURES FOR PREVENTION AND PROTECTION STRATEGIES

| **Initiator** | **Prevention** | **Protection** |
|---|---|---|
| LOCA | No large pipe connections<br>Reduced vessel fluence<br>Leak detection in all major penetrations | All pipe connections >8m above core<br>Large primary inventory<br>Passive water injection to maintain inventory<br>Low power density<br>Inertion of the containment atmosphere |
| Overcooling Transients | All secondary penetrations to RPV protected by valves<br>Limited flow rate for pumped injection | Large primary inventory<br>Small secondary inventory |
| SGTR | Secondary piping up to valves designed for primary pressure | Valves to limit release of primary fluid |
| LOF | 6 pumps limit consequences of a single pump failure | Pump inertia coupled with moderator temperature coefficient cause shut down without DNBR |
| Primary Transients | Protection system<br>Negative temperature coefficient<br>High margins due to low power density<br>Boronfree concept | |
| Secondary Transients | | Not critical |
| Station Blackout | | Not critical due to passive systems |
| ATWS | High reliability trip system<br>65 control rods | Negative temperature coefficient<br>Emergency boration system |

TABLE 7.5.4. DESIGN FEATURES FOR MITIGATION OF THE CONSEQUENCES OF ACCIDENTS

| Safety Functions | Sys, Cf Table 7 5 1 | Passive/ Active | Design features/ Remarks |
|---|---|---|---|
| **Design Basis**<br>Fission Product Containment | Primary circuit<br>Containment | Passive<br>Passive | |
| Coolant Inventory | Large coolant volume<br>CVCS<br>Safety Injection System | Passive<br>Active<br>Passive | Integral system<br>Only small pipes connected to RPV |
| Decay Heat Removal | Natural circulation in primary<br>Secondary Condensing Sysem | Passive<br><br>Passive | <br>72 hour capacity |
| Reactivity Control | Negative temperature coefficient<br>Control Rods | Passive<br><br>Active | Full hot shut down capability |
| Primary Circuit Pressure Control | Integral Pressuriser<br><br>Safety Valves<br>Depressurisation System | Passive<br><br>Passive<br>Active | Passive for pressure reduction only<br>Discharge to containment tanks |
| **Severe Accidents**<br>Containment temperature and pressure control | Pressure suppression tanks<br>Air cooling of tanks | Passive<br>Passive | <br>Large finned surface area |
| Heat Removal | Secondary Condensing System<br>Discharge of steam to pressure suppression containment tanks | Passive<br><br>Passive | Ultimate DHR totally Passive |
| Tightness Control | Containment | Passive | |
| Inflammable gas control | Containment inerted with Nitrogen | Passive | |
| Fission Product Containment | Pressure suppression tanks<br>Containment | Passive<br>Passive | Scrub fission products |
| Corium Management | Flooding of reactor cavity | Active | Extremely low probability of core melt |

### 7.5.4. Design data questionnaire (for Water-Cooled Reactors- SIR™)

## I. GENERAL INFORMATION

1. Design name: SIR™ Reactor
2. Designer/Supplier address: Consortium see 6.5.1
3. Reactor type: Integral PWR: Number of modules/per plant 1to4
4. Gross thermal power (MW-th) per reactor: 1000
5. Net electrical output (MW-e) per reactor: 320
6. Heat supply capacity (MW-th): -

## II. BASIC DESIGN DESCRIPTION

## A. CORE AND REACTIVITY CONTROL

7. Fuel material: $UO_2$
8. Fuel inventory (tones of heavy metal): 46.1
9. Average core power density (kW/liter): 54.6
10. Average fuel power density (kW/kgU): 21.7
11. Maximum linear power (W/m):
12. Average discharge burnup (MWd/t): 38,000
13. Initial enrichment or enrichment range (Wt%): 3.3 - 4.0
14. Reload enrichment at the equilibrium (Wt%): 3.3 - 4.0
15. Refueling frequency (months): 24
16. Type of refueling (on/off power): off
17. Fraction of core withdrawn (%): 1/3
18. Moderator material and inventory: $H_2O$
19. Active core height (m): 3.472
20. Core diameter (m): 2.592
21. Number of fuel assemblies: 65
22. Number of fuel rods per assembly: 432
23. Rod array in assembly: 22 x 22
24. Clad material· Zircaloy-4
25. Clad thickness (mm): 0.69
26. Number of control rods or assemblies: 65
27. Type: 8 large fingers
28. Additional shutdown systems: mtc, emergency boration
29. Control rod neutron absorber material: $B_4C$
30. Soluble neutron absorber: -
31. Burnable poison material and form: $Gd_2O_3$ in $UO_2$ fixed / $Gd_2O_3$in$Al_2O_3$inserted

## B. REACTOR COOLANT SYSTEM

**B1. Coolant**

32. Coolant medium and inventory: $H_2O$
33. Design coolant mass flow through core (kg/s): 7500
34. Cooling mode (forced/natural): forced
35. Operating coolant pressure (bar): 155
36. Core inlet temperature (C): 294
37. Core outlet temperature (C): 318

**B2. Reactor pressure vessel/tube**: vessel

38. Overall length of assembled vessel/tube (m): 23.8
39. Inside vessel/diameter (m): 5.8
40. Average vessel/tube thickness (mm): 280
41. Vessel/tube material: Carbon steel
42. Lining material: Stainless steel
43. Design pressure (bar)
44. Gross weight (tonne): 907 ex head

**B3. Steam generator**

45. Number of steam generators: 12
46. Type: once through
47. Configuration (horizontal/vertical): vertical
48. Tube material: Inconel 690
49. Shell material:
50. Heat transfer surface per steam generator ($m^2$): 9283

51 Thermal capacity per steam generator (MW) 91
52 Feed water pressure (bar)
53 Feed water temperature (C) 224
54 Steam pressure (bar) 55
55 Steam temperature (C) 298

**B4. Pressurizer**

56 Pressurizer total volume ($m^3$) 80
57 Steam volume (full power/zero power, $m^3$) 40

**B5. Main coolant pumps**

58 Number of cooling or recirculation pumps 6
59 Type Sealed, wet winding motor
60 Pump mass flow rate (kg/s) 1250 per pump
61 Pump design rated head 37 2
62 Pump nominal power (kW) 700 rated power 1100
63 Mechanical inertia (kg $m^2$) 45

## C. CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS)

64 Number of extraction lines
65 Number of pumps 2
66 Number of injection points
7 Feed and bleed connections

## D. CONTAINMENT

68 Type pressure suppression
69 Overall form (spherical/cyl ) reactor cavity plus tank farm
70 Structural material steel tanks, lined concrete cavity
71 Liner material steel
72 Simple/double wall simple
73 Dimensions (diameter, height) (m) 8 tanks 6m diam, 20 m high
74 Design pressure (bar) 2 41
75 Design temperature (C) 116
76 Design leakage rate (% per day) <0 1%

# III. SAFETY RELATED SYSTEMS

## A. DESIGN CONDITIONS

**A1. Fission product retention**

77 Containment spray system (Y/N) N
- a Duration (h)
- b Flow rate ($m^3$/h)
- c Mode of operation (active/passive)
- d Safety graded (Y/N)

78 F P sparging (Y/N) Y
79 Containment tightness control (Y/N) Y
80 Leakage recovery (Y/N) N
81 Guard vessel (Y/N) N

**A2. Reactivity control**

82 Absorber injection system (Y/N) Y but emergency only
- a Absorber material boron
- b Mode of operation (active/passive) active
- c Redundancy no
- d Safety graded no

83 Control rods (Y/N) Y
- a Maximum control rod worth (pcm) 18
- b Mode of operation (active/passive) active/ passive shutdown
- c Redundancy 65 rods
- d Safety graded Yes

**A3. Decay heat removal**

*A3-1 Primary side*

84. Water injection
 a. Actuation mode (manual/automatic): automatic
 b. Injection pressure level (bar): 172
 c. Flow rate (kg/s): 0.16
 d. Mode of operation (active/passive): passive
 e. Redundancy: x 2
 f. Safety graded (Y/N): Y

85. Water recirculation and heat removal: Primary natural circulation
 a. Intermediate heat sink (or heat exchanger) SG
 b. Mode of operation (active/passive): passive
 c. Redundancy: x 4
 d. Self sufficiency (h)
 e. Safety graded: Yes

*A3-2 Secondary side*

86. Feed water
 a. Actuation mode (manual/automatic): automatic
 b. Flow rate (kg/s)
 c. Mode of operation (active/passive): passive
 d. Redundancy
 e. Self sufficiency (h)
 f. Safety graded

87. Water recirculation and heat removal
 a. Ultimate heat sink (cold source): condenser in IRWST
 b. Mode of operation (active/passive): passive
 c. Redundancy: x 4
 d. Self sufficiency (h):
 e. Safety graded: Yes

*A3-3 Primary pressure control*

88. Implemented system (Name)
 a. Actuation mode (manual/automatic): Manual
 b. Side location (primary/secondary circuit): Primary
 c. Maximum depressurization rate (bar/s)
 d. Safety graded: No

## B. SEVERE ACCIDENT CONDITIONS*

### B.1 Fission products retention

89. Containment spray system (Y/N): N
90. F.P. Sparging (Y/N): Y
91. Containment tightness control (Y/N): Y
92. Leakage recovery (Y/N): N
93. Risk of recriticality (Y/N): N

### B.2 Recriticality control

94. Encountered design feature
 a. Mode of operation (A/P): A
 b. Safety graded: No

### B.3 Debris confining and cooling

95. Core debris configuration (core catcher): N
96. Debris cooling system (name): operator initiated flooding
 a. Mode of operation (A/P)
 b. Self sufficiency
 c. Safety graded (Y/N)

### B.4 Long term containment heat removal

97. Implemented system
 a. Mode of operation (A/P): P
 b. Self sufficiency (h)
 c. Safety graded (Y/N): Y
98. Intermediate heat sink
 a. Self sufficiency (h)
 b. Safety graded (Y/N)
99. External coolant recirculation: natural circulation of air
 a. Implemented components
 b. Mode of operation (A/P): P

---

* All systems must be qualified to operate under the accident conditions.

c Self sufficiency (h)
d Safety graded (Y/N)
100 Ultimate heat sink
a Self sufficiency (h)
b Safety graded (Y/N) Y

**B.5 Combustible gas control**

101 Covered range of gas mixture concentration
102 Modes for the combustible gas control
a Containment inertation Yes
b Gas burning No
c Gas recombining No
d Others No

**B.6 Containment pressure control**

103 Filtered vented containment (Y/N) N
a Implemented system
b Mode of operation (A/P)
c Safety graded
104 Pressure suppression system (Y/N) Y
a Implemented system water tanks
b Mode of operation condensation
c Safety graded (Y/N) Y

## C. SAFETY RELATED I&C SYSTEM

Automatic load following (Y/N) Y
* range (% power)
* maximum rate (%/min)

Load rejection without reactor trip (Y/N) Y
Full Cathode Ray Tubes (CRT) display (Y/N) Y
Automated start-up procedures (Y/N) Y
Automated normal shutdown procedures (Y/N) Y
Automated off normal shutdown procedures (Y/N) Y
Use of field buses and smart sensors (Y/N) Y
Expert systems or artificial
intelligence advisors (Y/N)
Protection system backup (Y/N) Y

## D. EMERGENCY POWER SUPPLY SYSTEM

105 Type (diesel, gas, grid connection) no safety grade diesels
106 Number of trains

## E. AC/DC SUPPLY SYSTEM

107 Type (rectifier, converter, battery) no safety grade emergency supplies
108 Estimated time reserve (hr)

## IV. CONVENTIONAL THERMAL CYCLE

## A. TURBINE SYSTEM

109 Type tandem compound, dual flow, single reheat
110 Overall length (m)
111 Width (m)
112 Number of turbines/reactor 1
113 Number of turbine sections per unit double flow HP and LP
114 Speed (rpm)

## B. STEAM CHARACTERISTICS

115 H P inlet pressure bar 55
116 H P inlet temperature (C) 298
117 H P inlet flowrate (kg/s) 516
118 L P inlet pressure 15
119 L P inlet temperature 252

120. L.P. inlet flowrate: 419

C. **GENERATOR**

121. Type (3-phase synchronous, DC): 3 phase synchronous
122. Apparent power (MVA)
123. Active power (MW): 337
124. Frequency (hz): 50/60
125. Output voltage (kV)
126. Total generator mass (t)
127. Overall length
128. Stator housing outside diameter

D. **CONDENSER**

129. Number of tubes
130. Heat transfer area
131. Flowrate ($m^3/s$)
132. Pressure (m/bar)
133. Temperature (C)

E. **CONDENSATE PUMPS**

134. Number
135. Flowrate
136. Developed head
137. Temperature
138. Pump speed

### 7.5.5. Project status

*7.5.5.1. Entities involved*

The design was prepared by a partnership of:

AEA Technology, UK
Rolls Royce and Associates, UK
ABB Combustion Engineering, USA
Stone and Webster Engineering Corporation, USA

*7.5.5.2. Design Status*

Conceptual design has been completed. There is no current further design work in progress.

*7.5.5.3. R&D status*

The R&D needs have been identified and characterised under the headings of

Work needed to technically prove a feature of the design
Work needed to demonstrate to others the claims made
Work needed to qualify the components and systems

R&D suspended after 1 year of this programme in 1991.

*7.5.5.4. Licensing status*

The plant has been assessed by the developers against the standards of the US and UK licensing authorities and is believed to be licensable in both countries. No formal licensing review applications have been made.

### 7.5.6. Project economics

The SIR™ design has used simplification in its systems and construction to overcome the economies of scale. The on site construction schedule has been estimated at 30 months for the Nth plant with an additional 6 months for start up testing. The result is a generating cost which is competitive with other reactor systems.

## 7.6. REACTOR SYSTEM DESCRIPTION AND DEVELOPMENT STATUS OF ISIS

### 7.6.1. Basic objectives and features

The ISIS (Inherently Safe Immersed System) has been conceived and developed by ANSALDO in recent years beginning from 1987 as an innovative reactor with easily understandable inherent safety characteristics.

It is an integrated PWR, completely immersed in a large pool of cold boronated on water which builds up on 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 the ANSALDO experience in the field of both LWRs and LMFBRs.

The ISIS addresses the following main targets:

Safety targets

- The engineered configuration provides defence, primarily by prevention, against accidents with severe core damage.
- The self-depressurization of the Primary System, the scrubbing effect of the pools and the absence of significant Reactor Containment pressurization under all accident conditions drastically reduce radioactive releases to the environment.

Capital cost target

- A low capital cost and a short construction schedule are expected 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 necessary.

Flexibility to fit demand of power producers

- The whole Primary Circuit contained in the reactor vessel is a feature which leads to a multi-module NPP configuration suitable for large interconnected electrical grids.
- The reduced unit power (200 MWe) of the ISIS reactor permits the ISIS deployment also for local electrical grids.

Polyfunctionality target

- The outstanding features of passive safety of the ISIS reactor should facilitate installation wherever the combined function of electric energy generation and district heating or water desalinisation is required.

## 7.6.2. Design description

### *7.6.2.1. Nuclear steam supply system (NSSS)*

The Primary System of the ISIS reactor is of the integrated type (fig.7.6.2.), with the Steam Generator Unit (SGU) housed in the 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 primary water from the surrounding high boron cold water.

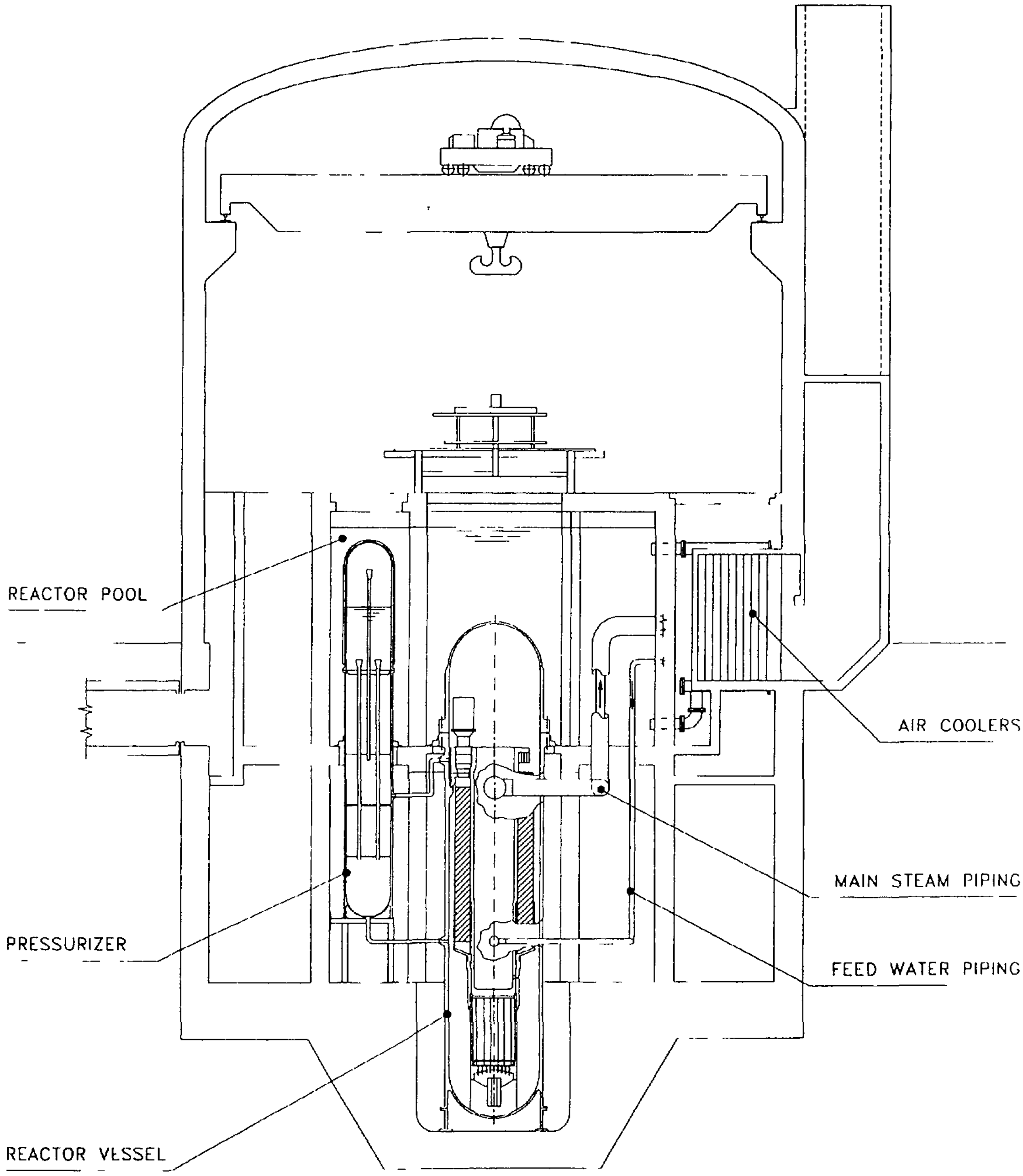


*Fig 7.6.1. ISIS Reactor Building*

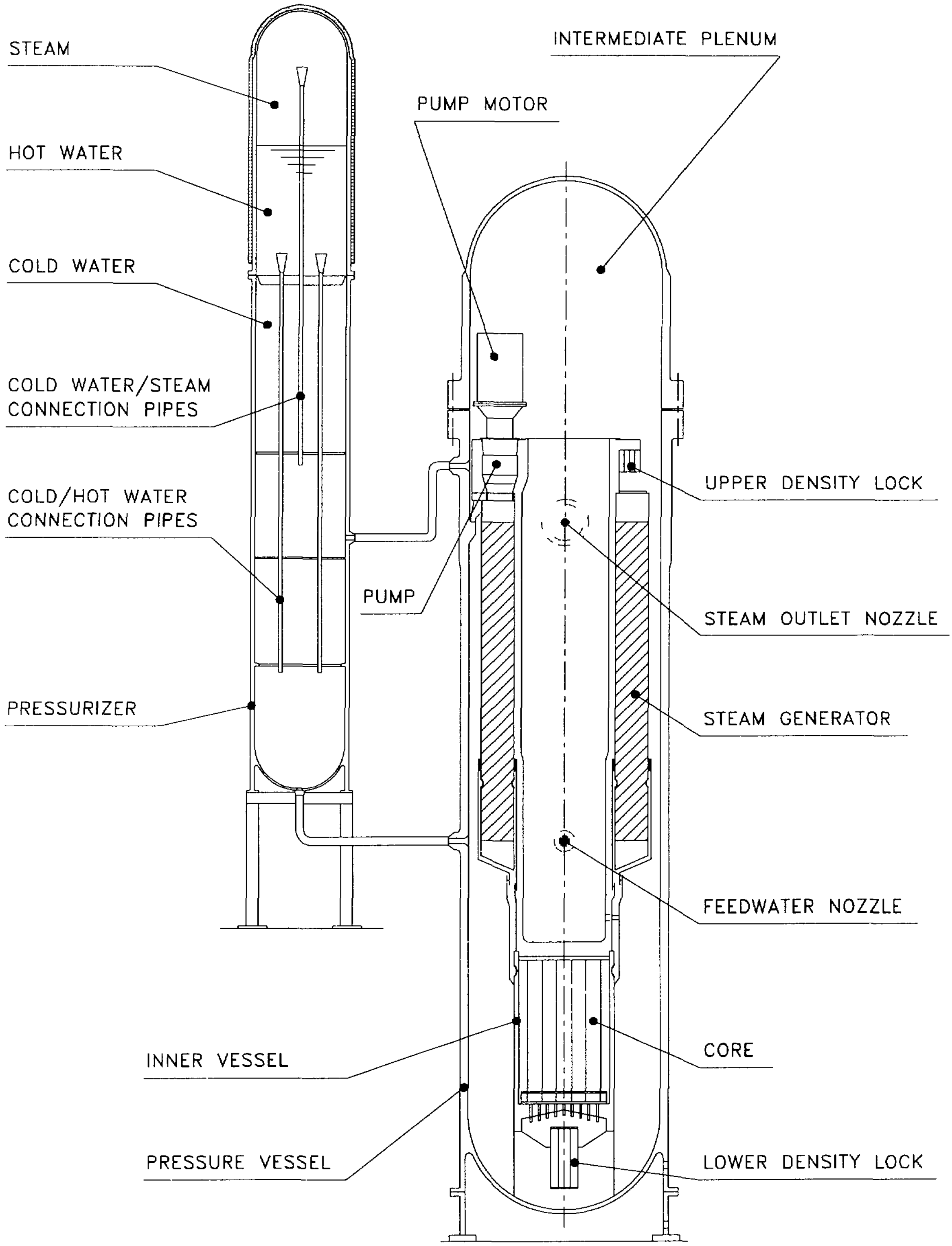


*Fig 7.6.2. ISIS Reactor Module*

Hot and cold plena 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 lower and upper DENSITY LOCKS. The Inner Vessel houses the core, the SGU and the primary pumps.

The outstanding feature is the complete immersion of the Pressure Boundary, made up, for each module, of a Reactor Vessel and of a separated Pressurizer with Interconnecting Piping, in a large pool of boronated cold water.

During normal operation, the heat generated in the core is transferred to the SGU via the water circulated by the Primary Pumps, which are located 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) always enters the Primary Circuit, mixes up with the primary water, shuts down the reactor and cools the core in natural circulation. The same water mixing process heats up the intermediate plenum water and the Pressure Vessel and activates 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 allow this water to remain below the atmospheric boiling point after removal of the decay heat for about a week.

Cooling of the plant pool is guaranteed, for an unlimited time, by virtue of two loops provided with coolers in natural circulation, sized to reject, approximately 2 MW to the external atmosphere and thereby capable of keeping the pool water temperature below 80°C.

Reactor pressure vessel

The Reactor Pressure Vessel (RPV) is of cylindrical shape with hemispherical heads. The main openings of the Reactor Vessel are the water/steam nozzles and the two 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 ID; 26,5 m high) which cumulates several functions: it contains the whole primary system and the Intermediate Water Plenum (IP) and operates as heat exchanger to remove the decay heat towards the reactor pool. The IP limits the total neutron fluence of the RPV to less than $10^{15}$ n/cm$^2$ over its life time.

Core

The reactor core consists of 69 typical (17 X 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 lower soluble boron concentration and assures a non positive moderator temperature coefficient at any operating condition.

Steam generator unit (SGU)

The SG features an annular tube bundle with helical tubing. The steam is generated tube-side. The feed water piping is connected to two feed water headers, located symmetrically inside the reactor vessel within a calm zone, provided each 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 two steam headers connections. 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 higher outer rather than inner tube pressure, a reversed situation with respect to a conventional SGU, reduces the risk of flaw growth in the tubes.

The steam pressure is 46 bar (at the SG outlet) with about 30°C of superheating. The reduced secondary water inventory in comparison to the primary water inventory (3,5 tonnes of secondary water inventory against 20 tonnes of primary water in normal operating condition) inside the tube bundle limits the feedback effects of Steam Line Break Accidents on the core .

Primary circulation pumps

The two Primary Pumps of the glandless, wet winding type (like the pumps manufactured by Hayward Tyler F.D., England) 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 $m^2$).

Above core structure (ACS)

The ACS, 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 ACS 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 ACS is flanged to and suspended from the top of the Inner Vessel for easy removal to allow standard fuel handling.

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.

Inter-connecting piping

Two pipes between Pressurizer and Reactor Vessel connect hydraulically the top and the bottom of the respective cold water plena in order to create a common plenum. The choice of two connection levels makes natural circulation possible in case of temperature difference between cold plena. If the normal decay heat removal route (i.e. the active steam/water system) is lost, the uninsulated wall portion of the Pressurizer would thus help by conducting the decay heat to the Reactor Pool.

Movement of water to and from each vessel, belonging to a common cold water plenum, does not significantly contribute to the thermal loadings on the pressure boundary during transients.

Steam and feedwater system

The normal decay heat removal function of the steam and feedwater systems is not safety related. The closure of both steam and feedwater lines isolation valves allows a complete separation of the NSSS from the non safety grade BOP. No need for pressure relief is anticipated, since the system is designed for the maximum expected pressure.

Relief valves are nonetheless provided in accordance with current codes and regulations.

Chemical and volume control system (CVCS)

The CVCS does not belong to the Safety Grade Systems. It is made of two independent systems; both systems are used for reactor start-up, only one is necessary for reactor normal operation.

The CVCS main functions are:

- primary water chemistry control
- intermediate plenum water chemistry control
- upper density lock level control under normal operation
- pressurizer level control
- primary water boron concentration control. The concentration is controlled by boron transfer between the primary and the intermediate plenum water without buffering vessels. This is to keep the overall boron content inside the reactor module constant and sufficient to bring the reactor to cold shutdown when the primary and intermediate waters mix-up.

Fuel handling and storage system

The ISIS fuel handling and storage systems are conventional. Spent fuel removed from the reactor vessel is temporarily stored underwater. The spent fuel racks are located inside the Spent Fuel Storage Pool (SFP) under sufficient water to provide radiological shielding. It can store up to 3 reactor cores.

*7.6.2.2. Balance of plant*

Steam generated in the steam generators is supplied to the high-pressure turbine via the main steam header. Steam, leaving the high pressure turbine, passes through 2 combined moisture separators/reheaters prior to entering the three low-pressure turbines.

This arrangement implies only one 600 MWe turbogenerator set for a plant with three reactor modules. (The alternative configuration features a 200 MWe turbogenerator set for each module which, in this case, includes a high pressure turbine, a moisture separator/reheater and a low pressure turbine).

The main condenser is a three shell type : each shell is located at each low-pressure stage turbine exhaust. The tube bundle is perpendicular to turbine axis.

The main condenser is a single flow type in case of sea or river circulating water and a 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:

- startup auxiliary system (heat removed from the S.G. secondary side)
- CVCS (heat removed from the primary system)
- intermediate pool cooling system.

The primary chemistry is controlled by the CVCS; additional systems are provided to control the reactor pool chemistry and the secondary 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 modular solution based design a common radwaste building is provided.

*7.6.2.3. Instrumentation, control and electrical systems*

The plant control scheme is basically a "turbine follows reactor", with some degree of coordination to speed up the overall response.

Both the 1E I & C systems (Protection and Safety related monitoring) and the control systems (characterized by the absence of control rods and by the "hot-cold" interface regulation inside the Density Locks) are designed utilizing a proven, distributed microprocessor-based technology. An advanced control room, with a wall panel information station and work stations for operators and supervisor is foreseen, taking into account the most recent achievements and improvements in the man-machine interface systems.

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. The electrical supply system is based on a "two train" approach.

*7.6.2.4. Safety considerations and emergency protection*

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 chain, i.e. with no inputs of "intelligence", nor external power sources or moving mechanical parts, according to Category B Passive Components of the IAEA - TECDOC-626 definition. Active protective measures, aimed at preceding passive system interventions, are included in the design but are not credited in the safety analysis.

The ISIS response to the Design Basis accidents, highlighting the inherently safe behaviour, is illustrated 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 LOSSP the pumps coastdown and natural circulation establishes itself causing reactor shutdown and providing continued core cooling.

Loss of heat sink (LOHS)

Even in the case that all the steam and feedwater lines are instantaneously isolated, so that the SGU heat transfer capability is quickly zeroed, the secondary side of the SGU reaches thermal equilibrium with the primary side with a pressure increase up to about 115 bar 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 integrated Primary System configuration, no significant breaks in the primary circuit are conceivable, but only a loss of pressure boundary of the Intermediate Pool. The cross section of the largest pipe of the Intermediate Pool is limited to about $0.01 m^2$. All pipe connections are furthermore located at a higher elevation than the core.

Scoping parametric analyses carried out assuming conservatively $0.05 m^2$ large breaks located at different elevations, including the vessel bottom, always show the same type of accident evolution. A series of simultaneous water transfers occur. Intermediate water flows out of the break into the Reactor Pool; 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. Soon after, degradation of the pump performance causes entrance of the intermediate water into the primary loop via the lower density lock with consequent reactor scram.

Steam generator tube rupture (SGTA)

The outflow of primary water into the SGU tubing is compensated by pressurizer water. Boronated 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.

Steam line break accident (SLBA)

The sudden depressurization of the secondary water inside the SGU tubing following a pipe break, cools down the primary water flowing through the SGU. A cold water plug enters eventually the core with a consequent transient overpower (TOP).

This primary water cooling is inherently limited, however, by the low tube-side water inventory (about 3,5 tonnes at nominal condition) and low SGU flowrate (high pressure loss of the helical tubing). The TOP is therefore limited to about 30% of the nominal power. Crediting a timely primary pump trip (active, no safety grade system), the consequences of the TOP can be even further reduced.

In conclusion the passive 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 that always guarantees efficient core cooling; even in a LOCA event 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, which provides a scrubbing effect for any radioactivity released from the fuel;
- the "pool reactor like" configuration that determines no containment pressurization, in case of primary water loss, and consequently no significant release of radioactivity outside the containment.

It is pointed out that only one safety grade path for heat rejection to the ultimate heat sink is provided in the ISIS design. It is based on concentric shells (the hot and pressurized water of the primary system, the cold and non-pressurized water of the Intermediate Plenum, the cold and non pressurized water of the Reactor Pool and, ultimately the external atmosphere). The configuration is such that its reliability is extremely high (practically absolute, since the loss of integrity of a pressurized shell does not jeopardize the path effectiveness).

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 common containment building, determines a favourable ratio of hydrogen produced in comparison with the available free volume.

*7.6.2.5. Buildings and structures*

Reactor Building

Each reactor module is located inside a 17 m I.D. reinforced concrete Reactor Pool. The Reactor Pool is located inside the Reactor Building (RB) which houses also auxiliary systems containing primary water, the spent fuel storage pool, the fuel handling equipment and the control room.

The RB is designed for a relative pressure of 0,5 bar which conservatively covers accidental pressure increase due to the evaporation of the water of the Reactor Pool. Two RB sizes are foreseen: a single module size and a three modules size with common fuel element and component handling facilities. Apart from this, each module has its own dedicated auxiliary systems, of which those which process primary fluids are located in segregated areas. A separate access to each potentially contaminated area is provided from the common areas in the upper portion of the reactor building.

Remaining Buildings

Remaining buildings are the turbine building, 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.

Seismic Behaviour

The Nuclear Island offers a generally favourable seismic behaviour especially because of the selected deep embedment into the soil and the low elevation of its centre of gravity.

A free standing configuration for the Reactor Pool, resting on a common foundation mat, but structurally decoupled from the RB, designed for a RG. 1.60 Safety Shutdown Earthquake excitation anchored to 0.3 g, presents reasonable wall thicknesses and reinforcing bar densities.

### 7.6.3. Safety concepts

TABLE 7.6.1. MAIN SAFETY RELATED SYSTEMS IN THE ISIS CONCEPT

| Name | Safety graded | Main Characteristics |
|---|---|---|
| Primary Circuit | X | Two recirculation pumps and a SGU integrated in the Reactor Vessel |
| Density Locks | X | Full passive feature for shutdown and core cooling |
| Reactor pool (decay heat removal and fission product scrubbing) | X | Concrete pool containing 6000 $m^3$ of cold and boronated water |
| Reactor pool air coolers | X | Two passive air coolers of 1 MW each |
| Containment system | X | Reinforced concrete containment<br>Low design pressure |

## TABLE 7.6.2. MAIN ACCIDENT INITIATORS FOR THE ISIS

- LOCA (primary) Loss Of Primary Coolant Accident
- LOCA (Secondary) Secondary Pipe Rupture (water or steam pipes)
- LOCA (Interfacing) e g Steam Generator Tube Rupture
- ATWS Anticipated Transients Without Scram
- Primary Transients
- Secondary Transients (turbine trip)
- Loss of electric grid
- Loss of the normal heat sinks
- Loss of the steam generator feedwater
- LOSSP Loss Of Station Service Power

## TABLE 7.6.3. DESIGN FEATURES FOR PREVENTION AND PROTECTION LEVELS THAT REDUCE (R), SUPPRESS (S), THE INITIATORS FREQUENCY OR LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES (L)

| | |
|---|---|
| **PREVENTION LEVEL** | |
| LOCA (Primary) | Primary circuit integration (pump and steam generator) R, L |
| - | Reduced vessel fluence (< $10^{15}$ n/cm$^2$) R |
| LOCA (Secondary) | Steam and feedwater lines designed at max expected pressure under normal and accident conditions R |
| LOCA (Interfacing) | Small diameter helical tubing of the SGU submitted to external pressure R, L |
| Primary transients | - No control rods R |
| Secondary transients | Secondary system within the reactor vessel designed at max expected pressure R |
| Loss of electric sources | * |
| Total loss of the cold source (water) | * |
| Total loss of the S G feedwater | * |
| Station blackout | * |
| **PROTECTION LEVEL** | |
| LOCA (Primary) | Primary pump shut-down anticipates primary system depressurization I |
| | Self depressurization of the primary system avoids core heat up even in case of assumed worst location of the rupture (including partial ruptures of the RPV) I |
| | Reactor module immersed in a pool of cold water avoids containment pressurization I |
| LOCA (Secondary) | Small secondary water inventory in the SGU I |
| - | No safety valves actuation necessary I |
| LOCA (Interfacing) | Easy SGU isolation L |
| ATWS | Strong negative temperature coefficient I |
| | Passive (IAEA Category B) shutdown I |
| Primary transients | Primary inertia mainly located inside heat transfer component (core and SGU) I |
| Secondary transients | Primary to secondary inertia ratio in the SGU L |
| Loss of electric sources | Not relevant |
| - | DC electrical power supply provided only for protective actions and for monitoring instrumentation |
| Total loss of heat sink | Not relevant * |
| Total loss of S G feedwater | Not relevant * |
| Station blackout | Not relevant * |

SYSTEM NAME ISIS

* N B The cold and borated water entering in natural circulation into the primary system through the density locks shuts down and cools the core The heat transfer to the external pool by natural convection and conduction through the vessel wall and finally the heat release to the external air by means of air cooler in natural circulation make these accidents not significant for safety for indefinite time

## TABLE 7.6.4. DESIGN FEATURES FOR MITIGATION LEVEL OF ISIS

| Safety functions | Systems (Cf Tab 7 6 1 ) | Passive/active | Design features/Remarks |
|---|---|---|---|
| **Design Basis**<br>Fission product containment | Clad<br>Primary Circuit<br>Reactor pool/Containment | Passive (A)<br>Passive (A)<br>Passive (A) | Since the reactor coolant system is normally fully submerged no containment pressurization is possible |
| Coolant inventory | Self-depressurizing primary system<br>Reactor pool | Passive (B)<br>Passive (B) | No core uncovery is possible The loss of few tons of primary coolant inventory determines cold and hot water mixing inside the reactor vessel, this causes depressurization down to the reactor pool pressure Water losses are therefore limited and temporary, being compensated by water inflow from the pool |
| Decay heat removal | Intermediate pool (short term)<br>Reactor pool (medium-long term)<br>Air coolers (indefinite time) | Passive (B)<br>Passive (B)<br>Passive (B) | Mixing of intermediate pool water with primary hot water<br>Heat transfer through the metal vessel wall<br>Reactor pool water temperature kept below 80°C |
| Reactivity control | Boronated water from intermediate pool<br>Primary pump trip | Passive (B)<br>Passive (D) | Boronated water enters the primary system via density locks when unbalance occurs between generated power and extracted power or when forced circulation is lost |
| Primary circuit pressure control | Pressurizer spray<br>Self depressurizing primary system<br>Safety valves | Passive (B)<br>Passive (B)<br>Passive (C) | Spray determined by cold insurge flow |
| **Severe Accident***<br>Containment temperature and pressure control | Reactor pool<br>Air coolers | Passive (B)<br>Passive (B) | No pressurization of the containment is possible being the reactor module immersed in a cold pool<br>Air coolers keep the pool temperature below 80°C for indefinite time |
| Heat removal | Reactor pool<br>Air coolers | Passive (B)<br>Passive (B) | Corium on the vessel bottom is cooled by the water of the reactor pool |
| Tightness control | Containment | Passive (A) | No pressurization of the containment is possible |
| Inflam gas control | Containment | Passive (B) | Large free volume (modular approach) |
| Fission product containment | Intermediate pool and reactor pool<br>Containment | Passive (A)<br>Passive (A) | Scrubbing effect provides significant decontamination factor<br>No containment pressurization is possible |
| Corium management | Vessel | | In Vessel retention |
| Others | | | |

* No conceivable core damage can be caused by any accident initiator of table 7 6 2 In spite of that if a core damage is hypothetically assumed, the corium retention is, in any case, assured at the cold Reactor Vessel bottom

### 7.6.4. Design Data Questionnaire

## I. GENERAL INFORMATION

1. Design name: ISIS
2. Designer/Supplier address: ANSALDO Genova
3. Reactor type: Integrated PWR
   Number of modules/per plant: 3
4. Gross thermal power (MW-th) per module: 650
5. Net electrical output (MW-e) per module: 205
6. Heat supply capacity (MW-th): TBD

## II. BASIC DESIGN DESCRIPTION

## A. CORE AND REACTIVITY CONTROL

7. Fuel material: UO2
8. Fuel inventory (tones of heavy metal): 24.3
9. Average core power density (kW/liter): 70
10. Average fuel power density (kW/kgU): 26.7
11. Maximum linear power (W/m): 30350
12. Average discharge burnup (MWd/t): 38000
13. Initial enrichment or enrichment range (Wt%): 2.0 - 3.0
14. Reload enrichment at the equilibrium (Wt%): 3.5
15. Refuelling frequency (months): 18
16. Type of refuelling (on/off power): Off power
17. Fraction of core withdrawn (%): 33
18. Moderator material and inventory: Light water, 75 tons
19. Active core height (m): 2.92
20. Core diameter (m): 2
21. Number of fuel assemblies: 69
22. Number of fuel rods per assembly: 264
23. Rod array in assembly: 17 x 17
24. Clad material: Zircaloy
25. Clad thickness (mm): 0.57
26. Number of control rods or assemblies: None
27. Type: N.A.
28. Additional shutdown systems: Boronated water injection, pump speed reduction
29. Control rod neutron absorber material: N.A.
30. Soluble neutron absorber: Borid acid
31. Burnable poison material and form: $B_4C$ - WABA

## B. REACTOR COOLANT SYSTEM

### B1. Coolant

32. Coolant medium: Light water
33. Design coolant mass flow through core (kg/s): 2911
34. Cooling mode (forced/natural): Forced
35. Operating coolant pressure (bar): 140
36. Core inlet temperature (°C): 271
37. Core outlet temperature (°C): 310 (with core by-pass mixing)

### B2. Reactor pressure vessel

38. Overall length of assembled vessel (m): 26.5
39. Inside vessel/diameter (m): 4.9
40. Average vessel thickness (mm): 210
41. Vessel material: Carbon steel
42. Lining material: Stainless steel
43. Design pressure (bar): 160

44. Gross weight (ton/kg): 782 tons

**B3. Steam generator**

45. Number of steam generators: 1
46 Type: Helical-tube, once through, integrated
47. Configuration (horizontal/vertical): Vertical
48. Tube material: Inconel 690
49. Shell material: Stainless steel
50 Heat transfer surface per steam generator ($m^2$): 5090
51. Thermal capacity per steam generator (MW): 650
52 Feed water pressure (bar): 52,6
53. Feed water temperature (°C): 120
54. Steam pressure (bar): 46
55. Steam temperature (°C): 290

**B4. Pressurizer**

56. Pressurizer total volume ($m^3$): 22 + 50 of cold water
57. Steam volume (full power/zero power, $m^3$): 11-13

**B5. Main coolant pumps**

58. Number of cooling or recirculation pumps: 2
59. Type: Glandless, wet winding, mixed flow impeller
60. Pump mass flow rate (kg/s): 1565
61. Pump design rated head (bar): 2
62. Pump nominal power (kW): 540
63. Mechanical inertia (kg $m^2$): 40 + 5000 on the common driving system

**C. CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS)**

64. Number of extraction lines: 2
65. Number of pumps: 4
66. Number of injection points: 2
67. Feed and bleed connections

**D. CONTAINMENT**

68. Type: Pool reactor in leaktight containment
69. Overall form (spherical/cyl.): Parallel piped with gabled dome
70. Structural material: Reinforced concrete
71. Liner material: Stainless steel in pools
72. Simple/double wall: Reactor pool + single wall concrete containment
73. Dimensions (diameter, height) (m): 31.5 width, 38 length, 53 height
74. Design pressure (bar): 0.5 gauge
75. Design temperature (°C): 80
76. Design leakage rate (% per day): 1% volume

**III. SAFETY RELATED SYSTEMS**

**A. DESIGN CONDITIONS**

**A1. Fission product retention**

77. Containment spray system (Y/N): N
78. F.P. sparging (Y/N): N
79. Containment tightness control (Y/N): Y
80. Leakage recovery: N
81. Guard Vessel (Y/N): N, pool scrubbing

**A2. Reactivity control**

82. Absorber injection system (Y/N): Y
    a. Absorber material: Boric Acid
    b. Mode of operation (active/passive): Passive (B)
    c. Redundancy
    d. Safety graded: Y

83. Control rods (Y/N): N

**A3. Decay heat removal**

*A3-1 Primary side*

84. Water injection: Y (density locks)
    a. Actuation mode (manual/automatic): N.A.
    b. Injection pressure level (bar): 140
    c. Flow rate (kg/s): determined by transient
    d. Mode of operation (active/passive): Passive (B)
    e. Redundancy: N.A.
    f. Safety graded (Y/N): Y

85. Water recirculation and heat removal
    a. Intermediate heat sinks: Intermediate and reactor pool
    b. Mode of operation (active/passive): Passive (B)
    c. Redundancy: N.A.
    d. Self-sufficiency (h): > 3 months
    e. Safety graded: Y

*A3-2 Secondary side*

86. Feed water: N

87. Water recirculation and heat removal
    a. Ultimate heat sink (cold source): External air
    b. Mode of operation (active/passive): Passive (B)
    c. Redundancy: N
    d. Self sufficiency (h): Indefinitely
    e. Safety graded: Y

*A3-3 Primary pressure control*

88. Implemented system (Name): Self depressurization of primary system
    a. Actuation mode (manual/automatic): Passive (B)
    b. Side location (primary/secondary circuit): Primary circuit
    c. Maximum depressurization rate (bar/s): determined by transient
    d. Safety graded: Y

**B. SEVERE ACCIDENT CONDITIONS**

**B1. Fission products retention**

89. Containment spray system (Y/N): N
90. F.P. Sparging (Y/N): N.A.
91. Containment tightness control (Y/N): Y
92. Leakage recovery (Y/N): N
93. Risk of recritically (Y/N): N

**B2. Recritically control**

94. Encountered design feature
    a. Mode of operation (A/P): N.A.
    b. Safety graded: N.A.

**B3. Debris confining and cooling**

95. Core debris configuration (core catcher): Retention in vessel bottom
96. Debris cooling system (name): Reactor pool water
    a. Mode of operation (A/P): Passive (A)

b. Self sufficiency: Indefinitely
c. Safety graded (Y/N): Y

**B4. Long term containment heat removal**

97. Implemented system: Reactor pool air coolers
    a. Mode of operation (A/P): Passive (B)
    b. Self sufficiency (h): Indefinitely
    c. Safety graded (Y/N): Y
98. Intermediate heat sink: Reactor pool water
    a. Self sufficiency (h): > 3 months
    b. Safety graded (Y/N): Y
99. External coolant recirculation
100. Ultimate heat sink: external air

**B5. Combustible gas control**

101 Covered range of gas mixture concentration: N.A.
102. Modes for the combustible gas control
    a. Containment inertation: N
    b. Gas burning: N
    c. Gas recombining: N
    d. Others: Large free volumes

**B6. Containment pressure control**

103. Filtered vented containment (Y/N): N
104. Pressure suppression system (Y/N): Y
    a. Implemented system: Reactor pool
    b. Mode of operation: Passive (B)
    c. Safety graded (Y/N): Y

**C. SAFETY RELATED I & C SYSTEM**

Automatic load following (Y/N): Y
* range (% power): 100-5-100
* maximum rate (%min): 10%

Load rejection without reactor trip (Y/N): Y
Fuel Cathode Ray Tubes (CRT) display (Y/N): Y
Automated start-up procedures (Y/N): N
Automated normal shutdown procedures (Y/N): Y
Automated off normal shutdown procedures (Y/N): Y
Use of field buses and smart sensors (Y/N): N
Expert systems or artificial intelligence advisors (Y/N): Y
Protection system backup (Y/N): N

**D. EMERGENCY POWER SUPPLY SYSTEM**

105. Type (diesel, gas, grid connection): No safety grade diesels
106. Number of trains: 2

**E. AC/DC SUPPLY SYSTEM**

107. Type (rectifier, converter, battery):
Class 1E uninterruptible power supply
108. Estimated time reserve (hr): Class 1E > 24h

**IV. CONVENTIONAL THERMAL CYCLE**

| | | | |
|---|---|---|---|
| **A.** | **TURBINE SYSTEM** | **200 MWe** | **(3x200 MWe)** |
| 109. | Type: | Reaction type | (Reaction type) |
| 110. | Overall length (m): | 12 | (28) |

| | | | |
|---|---|---|---|
| 111. | Width (m): | 8 | (8) |
| 112. | Number of turbines/reactor: | 1 | (1) |
| 113. | Number of turbine sections per unit: | 1 HP+1 LP | (1 HP+3 LP) |
| 114. | Speed (rpm): | 3000 | (3000) |

**B. STEAM CHARACTERISTICS**

| | | | |
|---|---|---|---|
| 115. | H.P. inlet pressure: | 4.5 MPa | (4,5 MPa) |
| 116. | H.P. inlet temperature: | 288°C | (288°C) |
| 117. | H.P. inlet flowrate: | 247 kg/sec | (740 kg/sec) |
| 118. | L.P. inlet pressure: | 0,55 MPa | (0,55 MPa) |
| 119. | L.P. inlet temperature: | 230°C | (230°C) |
| 120 | L.P. inlet flowrate (per section): | 227 kg/sec | (227 kg/sec) |

**C. GENERATOR**

| | | | |
|---|---|---|---|
| 121. | Type (3-phase synchronous, DC): | 3-phase synchronous | |
| 122. | Apparent power (MVA): | ≈ 238 | (≈715) |
| 123. | Active power (MW): | ≈ 215 | (≈645) |
| 124. | Frequency (hz): | 50 | (50) |
| 125. | Output voltage (kV): | 15 | (22) |
| 126. | Total generator mass (t): | 280 | (500) |
| 127. | Overall length (m): | 10 | (16) |
| 128. | Stator housing outside diameter (m): | 4 | (4) |

**D. CONDENSER** TBD (depends on make and site)

**E. CONDENSATE PUMPS** TBD

### 7.6.5. Project Status

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 ENEA DISP.
- Dynamic Analysis of ISIS Control System.

The further activity progress and schedule will be fundamentally governed by funding. Collaborations with other companies would be of advantage.

This activity should include experimental tests on the Wet Thermal Insulation and the Density Locks for this specific design, even though the Density Lock concept has been extensively tested by ABB for the PIUS reactor.

The remaining ISIS components and systems utilize proven technology in the field of both LWR and LMR reactors. No steps for licensing applications have been undertaken so far.

### 7.6.6. Project Economics

Economic evaluations are in progress; the target of the development of the ISIS design is the competitivity of a three module plant (600 MWe) with fossil fuel and nuclear plants of the same power range.

In spite of the modular plant approach, the integrated reactor configuration and the elimination of the active safety systems allow a compact layout.

Three factors mainly contribute to an expected relatively low cost of the Reactor Building:

- low design containment pressure
- no stringent requirements on containment leak tightness
- short construction time (about 20 months from start of excavation till completion of the civil structure) owing to the building geometry suitable for prefabrication and the full decoupling of the components installation from civil works.

The Reactor Vessel and the integrated components represent, by far, the most complex equipment to be fabricated. Their shipping to the site is foreseen in the final stage of RB construction, with no negative effect on the overall construction schedule. They have a higher cost than those of a classical PWR, but, on the other side, this higher cost should be offset by the elimination of:

- the external primary loops
- the SGU and primary pumps pressurized casing
- the control rods and their driving system.
- the safety grade diesel generator set
- the safety grade active decay heat removal systems
- the safety injection systems

Preliminary scoping studies show that the ISIS modules could be conveniently designed for a reactor power ranging from about 50 MWe up to about 200 MWe and a plant electrical power up to 600 MWe in the configuration with three modules.

Not yet considered in the design activity, 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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## 7.7. ATS-150 NUCLEAR CO-GENERATION PLANT REACTOR SYSTEM DESCRIPTION AND DEVELOPMENT STATE

### 7.7.1. Basic objectives and features

ATS-150 is a nuclear co-generation plant designed for electric power generation and for district heating.

Lay-out and design decisions adopted for the ATS-150 provide for the co-generation of heat and electricity in different ratios, either by reducing or by increasing the heat and electricity output. It is possible to couple the ATS-150 reactor plant with desalination plants.

The following features of the ATS-150 design can be identified as specific:

- integral primary circuit arrangement,
- primary coolant natural convection under all operating modes,
- guard vessel houses the reactor,
- multi-purpose application.

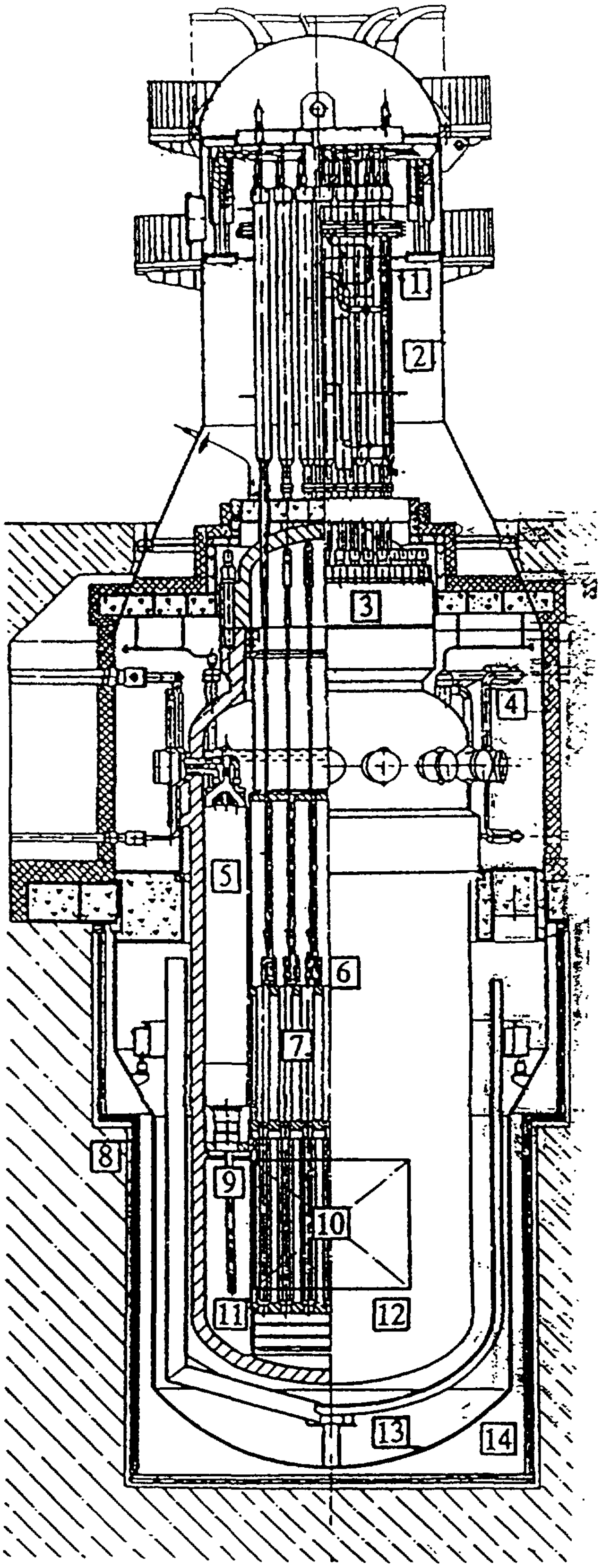
### 7.7.2. Design description

#### *7.7.2.1. Nuclear steam supply system*

Reactor pressure vessel

The reactor plant (RP) is equipped with an integral, vessel-type PWR. The reactor cross-section is shown in Fig 7.7.1. The reactor pressure vessel (RPV) is of 16.75 m height, 5.34 m diameter and 230 m$^3$ volume.

All in-vessel penetrations are of small diameter ($\leq$ 32 mm ID) and are located in the upper part of the RPV.




1. CPS drive
2. GV removable part
3. Upper block
4. Secondary circuit piping
5. SG (steam generator)
6. Drive line-connecting device assembly
7. Chimneys
8. Thermal insulation of reactor pit
9. Ionization chamberf suspension
10. Core
11. Core support barrel
12. Reactor
13. GV (guard vessel)
14. Reactor pit

*Fig 7.7.1. Reactor cross-section*

The RPV provides for a large primary water inventory (up to 150 $m^3$). An increased in-reactor water gap significantly reduces the neutron fluence ($< 5 \cdot 10^{17}$ n/cm$^2$) and provides longer RPV lifetime up to 50 years.

Reactor core

The core is composed of 109 hexagonal fuel assemblies (FAs) of 238 mm width across flats with active fuel length of 2.5 m. The core equivalent diameter is 2.7 m. This results in low core power density (36 MW/m$^3$). The fuel element design is based on the well developed technology of VVER-fuel.

Soluble boron poison is utilized only in an alternate shutdown system. Low worth control rods (36 clusters) are used for power regulation and load follow. The enrichment of the fuel is 3%. The core burnup is 32000 MWd/t.

The reduced power density of the core ensures low thermal loads under normal and emergency operating conditions. The chosen water/uranium ratio provides for favourable fuel cycle economy.

The core has self-regulating and self-stabilizing features due to the negative temperature, power and void reactivity coefficients. Also, the use of burnable poison reduces the excess reactivity margin to be compensated by the mechanical reactivity control system.

Special chimneys above the FAs provide self-controlled flow distribution in the core under natural coolant circulation conditions.

The integrated head package

The integrated head package (IHP) consists of the reactor vessel head, control rod drive mechanism, supports, cables. The condenser of the emergency residual heat removal system is mounted on the RPV head.

Guard vessel

The reactor is placed inside a special guard vessel (GV). The GV fulfils the following safety functions:

- Confinement of the primary radioactive coolant during primary circuit loss-of-coolant accidents;
- Keeping the coolant level in the RPV above the upper core level under all anticipated conditions.

Primary circuit

The primary circuit incorporated inside the RPV includes the main coolant flow path, intended for direct heat removal from the core and its transfer to the secondary circuit in the two internal once-through steam generators (SG), and the pressurization system intended to create and maintain preset pressure in the primary circuit. The 45 $m^3$ steam-gas pressurizer is located under the reactor head. Nitrogen is used for initial pressurization. Additional pressure is created by steam generated from the primary coolant.

Secondary circuit

The secondary circuit serves for heat removal from the primary circuit in the SGs, production of superheated steam, and its further delivery to the turbogenerator set. The secondary circuit is composed of two heat exchange loops each containing an SG made up of 9 sections, piping and valves. Should a primary coolant leak in the SG occur, it can be isolated on water and steam sides by double isolation valves.

To ensure residual heat removal from the reactor at station blackout, at accidents related to the loss of feed water, or in case of an earthquake, an emergency residual heat removal system (ERHRS) is connected to each secondary loop. The reactor main systems and flow diagram are shown in Fig. 7.7.2.

*7.7.2.2. Balance of plant system*

The turbine K-160-4.2 consists of a single - flow high pressure cylinder, and double-flow low pressure cylinder that exhausts to the condenser. The two moisture separator/ reheaters are integrated in the main turbine system. The reactor plant is combined with one turbine generator designed for base load and load-following operation.

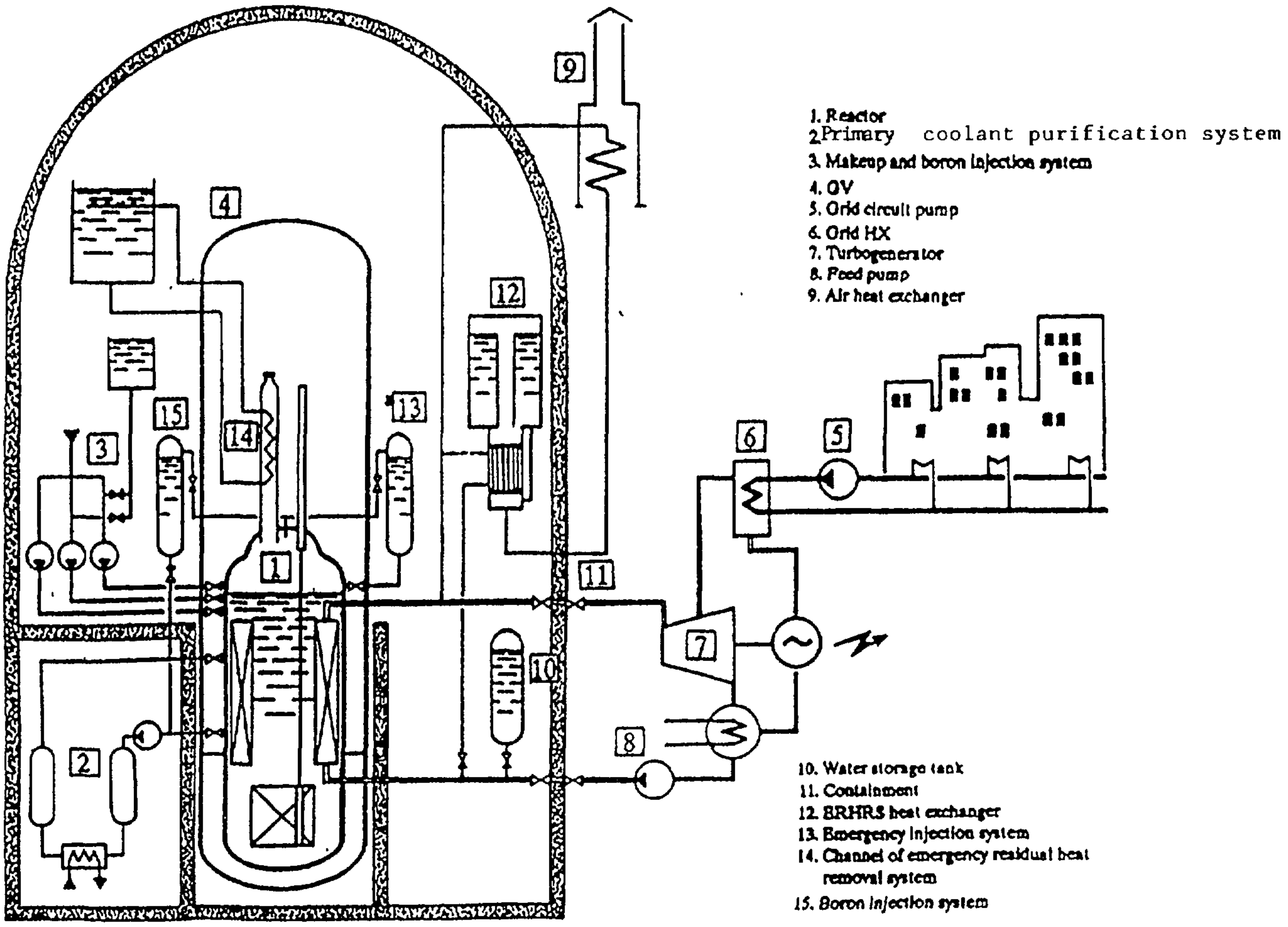



*Fig 7.7.2. Reactor safety systems and flow diagram*

Part of the steam flow from the high pressure cylinder is directed to the district heating grid HX to raise the temperature of grid water up to 150°C.

The preliminary design of the turbine K-160-4.2 was performed by Kharkov Turbine Plant Design Division (Ukraine).

The ATS-150 is equipped with a system for radioactive waste treatment, an auxiliary feed water system, and other supporting systems of standard design ensuring appropriate operation of a nuclear power plant.

*7.7.2.3. Instrumentation, control and electrical systems*

The plant control concept is based on proven technology, using feedback from operating plants, and takes advantage of new approaches aimed at easy plant operation. The goal is to minimize both the possibility and the potential for human error by providing a high degree of automation and well organized displays, controls and operator manuals. Advantage is taken of advancements in electronics and information processing technology.

*7.7.2.4. Safety considerations and emergency protection*

The enhanced safety of the ATS-150 is attained owing to:

- development of the plant self-protection features;
- creation of a multi-barrier system with functional and physical protection (defense in depth).

The following design features ensure the enhanced safety and reliability of the ATS-150:

1. Self-protection, self-regulation and self-limitation of reactor power due to the negative reactivity coefficients (for power, void, fuel and coolant temperature) within the entire range of reactor operation parameters.
2. Natural coolant circulation in the core cooling circuit under all operation modes.
3. Passive principle of the ERHRS operation with natural coolant circulation from the reactor up to the ultimate heat sink in all operation modes, giving a large grace time (not less than 72 hours) during which the RP is in a safe and steady state without operator interference.
4. Redundancy and diversity of safety systems and components.
5. Insertion of control rod assembly (CRA) by gravity into the core from any position in case of the de-energization of the CRA drives. Design of CRA devices preventing absorber rod ejection from the core in the event of rupture of the sleeve or of the CRA drive mechanism casing.
6. Use of self-actuated devices to initiate safety systems should the key parameters of the RP go beyond the preset limits.
7. Low heat density of the core, enhanced margins of departure from nucleate boiling, self regulation of coolant flowrate through the fuel assemblies.
8. High specific volume of primary coolant (0.40 $m^3$/MWth) and as a result high reactor heat accumulation capacity and slow transient dynamics, providing a longer time margin for the operator to analyze the emergencies and, if necessary, to initiate the required accident management procedures.

9. Integral reactor layout, minimization of through-vessel penetrations and their location at the upper part of the reactor vessel, large coolant volume (up to 150 $m^3$) above the core, and low reactor vessel irradiation (fluence $5.10^{17}$ n/$cm^2$).
10. Use of the guard vessel that ensures:

    - prevention of core uncovery in case of the reactor vessel leakage;
    - prevention of any leakage and release of radioactivity from the primary circuit.

11. The containment is an additional external barrier for the retention of the radioactivity in case of beyond-design basis accidents. It also mitigates any impact from external events (airplane crash, explosions, etc.).
12. Ensured protection of the heat consumers taking advantage of the three circuit flow scheme with a pressure barrier, thus eliminating radioactivity ingress into the main heating circuit.

REACTOR SHUTDOWN AND ITS MAINTAINING IN SAFE STATE

1. Reactor shutdown occurs by actuating the electro-mechanical CRA drives which insert the absorber rods into the core.
2. Reactor shutdown due to simultaneous insertion of all absorber rods into the core by gravity if the CRA drives are de-energized. CRA drive de-energization is provided automatically or manually in the case of abnormal conditions (e.g. in the event of pressure rise in the primary circuit). The number of self-actuated devices corresponds to the number of CRA drives. Their design and location in the GV decreases the possibility of spurious and subversive actions causing the devices to fail.

3. Reactor shutdown due to active and/or passive injection of the liquid absorber (boron solution).
4. Power self-limitation of the "hot" reactor at the balance of heat output and its removal by the emergency heat removal systems.
5 Power self-reduction during any emergencies due to the negative reactivity coefficients.

RESIDUAL HEAT REMOVAL

1. Via the SGs if feed water is supplied and the heat is dumped into the condenser of the steam turbine set.

2. Via the SGs if feed water is supplied and the heat is dumped into the main heat exchangers of the district heating grid.
3. Via the SGs and the attached emergency heat removal channels (into the process water and atmosphere) with natural circulation in the secondary coolant.
4. Via the condensers of the emergency heat removal channel on the reactor with natural circulation in all circuits.
5. Via the heat exchangers of the primary coolant purification system.
6. By heat transfer from the core to water in the reactor cavity, with its evaporation and subsequent makeup (accident management system).

1. Fuel with relatively low working temperature.
2. Cladding of fuel elements (under continuous integrity control) at low fuel power density.
3. Integral reactor and leak-tight primary circuit.
4. Limitation of reactor vessel penetrations cross-section (ND-32).
5. Valves in the primary circuit piping in combination with self-actuated isolation devices.
6. Isolation valves in the piping of the systems related to the accident isolating circuit.
7. GV of small volume, ensuring prevention of core dry out during accidents with loss of the reactor vessel integrity.
8. Separated compartments for the primary circuit systems.
9. Normal and emergency steam pressure suppression systems.
10. Containment equipped with filters, emergency pressure suppression systems and isolation valves on penetrations in the containment wall.

These design features of the ATS-150 reduce the probability of a severe accident with a core melt down to $10^{-8}$ per reactor-year and thus eliminate the need for evacuation of the local population.

*7.7.2.5. Buildings and structures*

A nuclear power plant with two ATS-150 power units is under design. The nuclear island and all safety related equipment of each unit are located in a separate containment building.

The cylindrical containment building is made of prestressed concrete with a stainless steel liner of 67.8 m height. Its internal diameter is 37.6 m, and the wall thickness is 1.6 m. The containment is designed for internal pressure of 0.35 MPa, and a temperature of 150°C. It can protect the reactor against external events including aircraft impact.

The design basis earthquake for safety related equipment and structures is based on a load as high as 8 points (MSK-64 scale).

### 7.7.3. Safety concepts

TABLE 7.7.1. MAIN SAFETY RELATED SYSTEMS IN THE ATS-150

| Name | Safety graded | Main characteristics |
|---|---|---|
| Primary Circuit (PC) | X | Integral PWR, build-in steam generators and pressurizer, leak-tight equipment, natural circulation |
| Control and Protection System (CPS) | X | 36 rod cluster assemblies, moved by individual drive mechanisms |
| Diverse Reactivity Control System (DRC) | X | Passive and active borated water injection |
| Emergency Coolant Injection System (ECIS) | X | 2 hydroaccumulators, 4 x 50 $m^3$ capacity |
| Passive Residual Heat Removal System (PRHR) | X | 1 HX permanently connected to the reactor, 2 HX connected to the secondary loops |
| Primary Pipeline Isolation Valves (PIV) | X | Double self-activated isolation valves in each primary pipeline |
| Guard Vessel (GV) | X | Core uncovery prevention at LOCAs |
| Containment (CONT) | X | |
| Containment Pressure Suppression (CPSS) | X | Bubbling through water storage tank |
| Containment Spray System (CSS) | | |
| Reactor Cavity Flooding System (RCF) | X | Core melt is prevented |

TABLE 7.7.2. MAIN ACCIDENT INITIATORS FOR THE ATS-150

- LOCA (Primary): Loss of Primary Coolant Accident
- LOCA (Secondary): Secondary Pipe Rupture (water or steam)
- LOCA (Interfacing): e.g.: SGTR Steam Generator Tube Rupture
- ATWS: Anticipated Transients Without Scram,
- Primary Transients,
- Secondary Transients (Turbine trip),
- Loss of Electric Sources (partial),
- Total loss of the Heat Sink,
- Total loss of the Steam Generator Feedwater,
- Station blackout.

TABLE 7.7.3. DESIGN FEATURES FOR PREVENTION AND PROTECTION LEVELS THAT REDUCE (R), SUPPRESS (S), THE INITIATORS FREQUENCY OR LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES (L)

**PREVENTION LEVEL**

LOCA (Primary)
- Reduced vessel fluence reduce initiator frequency R
- Primary circuit integration reduces initiator frequency and limits consequences R
- Guard vessel limits accident consequences

LOCA (Secondary) Closed double valves in each auxiliary pipeline, R
LOCA (Interfacing) Enhanced design margins and proven technology for SG, R
Primary transients
- Increased design margins R

Secondary transients
- Increased design margins R

Loss of electric sources Two independent off-site power sources S
- Primary coolant natural circulation, R

Total loss of the cold source (Water)
- Passive ultimate decay heat removal (stored water and air) suppress initiator, S

Total loss of the SG feedwater Increased design margins and sec circ make-up system, R
Station Blackout Two independent off-site power sources, DG, batteries, S

**PROTECTION LEVEL**

LOCA (Primary)
- Primary circuit integration leakage limitation
- Increased water inventory above the core,
- Water injection automatic and passive, L

LOCA (Secondary) Double valves in steam and feedwater lines L
LOCA (Interfacing) - Fast SG isolation L
ATWS
- Strong negative reactivity coefficients L

Primary Transients - Great primary inertia, CPS actuation, PRHR actuation L
Secondary Transients - Great primary inertia, CPS and PRHR actuation L
Loss of Electric Sources - Implementation of passive systems (battery power sources)
Total Loss of heat sink not critical*
Total Loss of S G feedwater not critical*
Station Blackout not critical*

NB The possibility for easy natural convection in the primary and secondary circuits is a favourable common factor for all abnormal situations

* The passive DHR and the large PC inertia, ensure that these three situations are not significant for the short term

TABLE 7.7.4. DESIGN FEATURES FOR MITIGATION LEVEL OF ATS-150

| Safety Functions | Systems | Passive/Active | Design features/Remarks |
|---|---|---|---|
| **Design Basis**<br>Fission Product Containment | Primary Circuit<br>GV<br>CPSS/CONT/CSS | Passive<br>Passive<br>Passive | |
| Coolant inventory | ECIS<br>GV<br>PIV | Passive/Active<br>Passive<br>Passive | GV integrates primary systems except clean-up one<br><br>Quick-acting (5s) valves |
| Decay Heat Removal | PRHR<br>CPSS<br>ECIS | Passive<br>Passive<br>Passive/Active | 72 h capacity |
| Reactivity control | CPS<br>DRC | Active/Passive<br>Passive/Active | Cold shut down capability |
| Primary Circuit Pressure Control | CPS/PRHR | Active/Passive | Without discharge of primary coolant |
| **Severe Accident**<br>Containment temperature and pressure control | CONT /CSS | Passive/Active | |
| Heat Removal | ECIS/CPSS/CONT/CSS<br>PRHR<br>RCF/CSS | Passive/Active<br>Passive<br>Passive/Active | |
| Tightness control | GV<br>CONT | Passive<br>Passive | |
| Inflam. gas control | Igniters | Passive | |
| Fission product containment | RV/GV/CONT | Passive | |
| Corium management | RV/GV/RCF | not relevant | |

### 7.7.4. Design Data Questionnaire for ATS 150

## I. GENERAL INFORMATION

1. Design name - ATS 150
2. Designer/Supplier address - Experimental Machine Building Design Bureau
3. Reactor type - Integrated PWR
   Number of modules/per plant - 2
4. Gross thermal power (MW-th) per reactor - 536 MWt
5. Net electrical output (MW-e) per reactor - up to 180 MWe
6. Heat supply capacity (MW-th) - up to 100 MWt

## II. BASIC DESIGN DESCRIPTION

### A. CORE AND REACTIVITY CONTROL

7. Fuel material - $UO_2$
8. Fuel inventory (tones of heavy metal) - 32
9. Average core power density (kW/liter) - 38.5
10. Average fuel power density (kW/kgU)
11. Maximum linear power (W/m) - 260W/m
12. Average discharge burnup (MWd/t) - 24 600
13. Initial enrichment or enrichment range (Wt%) - 3%
14. Reload enrichment at the equilibrium (Wt%)
15. Refuelling frequency (months) - 24
16. Type of refuelling (on/off power) - off power
17. Fraction of core withdrawn (%)
18. Moderator material and inventory - water
19. Active core height (m) - 2.5 m
20. Core diameter (m) - 2.7 m
21. Number of fuel assemblies - 109
22. Number of fuel rods per assembly
23. Rod array in assembly - triangular
24. Clad material - Zirconium alloy
25. Clad thickness (mm)
26. Number of control rods or assemblies - 36 assemblies
27. Type - cluster
28. Additional shutdown systems - Boron solution injection
29. Control rod neutron absorber material - $B_4C$
30. Soluble neutron absorber - Boron solution
31. Burnable poison material and form - Rods on boron basis

### B. REACTOR COOLANT SYSTEM

**B1. Coolant**

32. Coolant medium and inventory - Water, 150 $m^3$
33. Design coolant mass flow through core (kg/s)
34. Cooling mode (forced/natural) - natural
35. Operating coolant pressure (bar) - 158
36. Core inlet temperature (°C) - 265°C
37. Core outlet temperature (°C) - 340°C

**B2. Reactor pressure vessel**

38. Overall length of assembled vessel (m) - 16.75 m
39. Inside vessel/diameter (m) - 5.34 m
40. Average vessel/tube thickness (mm)
41. Vessel/tube material - Carbon steel, 15 x 2 HMPA
42. Lining material - steel
43. Design pressure (bar) - 180
44. Gross weight (ton/kg) - 609 tonnes

**B3. Steam generator**

45 Number of steam generators - 2
46 Type - once-through
47 Configuration (horizontal/vertical) - vertical
48 Tube material - stainless steel
49 Shell material - stainless steel
50 Heat transfer surface per steam generator ($m^2$) - 5940
51 Thermal capacity per steam generator (MW)
52 Feed water pressure (bar)
53 Feed water temperature (°C) - 185
54 Steam pressure (bar) - 45
55 Steam temperature (°C) - 290

**B4. Pressurizer**

56 Pressurizer total volume ($m^3$)
57 Steam volume (full power/zero power, $m^3$) - 45 on full power

**B5. Main coolant pumps**

58 Number of cooling or recirculation pumps - None
59 Type
60 Pump mass flow rate (kg/s)
61 Pump design rated head
62 Pump nominal power (kW)
63 Mechanical inertia (kg $m^2$)

## C. CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS)

64 Number of extraction lines - 2
65 Number of pumps - 2
66 Number of injection points - 2
67 Feed and bleed connections

## D. CONTAINMENT

68 Type - Prestressed concrete with liner
69 Overall form (spherical/cyl ) - cylindrical
70 Structural material - concrete
71 Liner material - stainless steel
72 Simple/double wall - simple
73 Dimensions (diameter, height) (m) - 37 6 m, 67 8 m
74 Design pressure (bar) - 2
75 Design temperature (°C) - 150
76 Design leakage rate (% per day) - 0 2

# III. SAFETY RELATED SYSTEMS

## A. DESIGN CONDITIONS

**A1. Fission product retention**

77 Containment spray system (Y/N) - Yes
- a Duration (h)
- b Flow rate ($m^3$/h)
- c Mode of operation (active/passive) - Active
- d Safety graded (Y/N) - yes

78 F P sparging (Y/N)
79 Containment tightness control (Y/N) - Yes
80 Leakage recovery (Y/N)
81 Guard vessel (Y/N) - Yes

**A2. Reactivity control**

82. Absorber injection system (Y/N) - Yes, but emergency only
    a. Absorber material - boron solution
    b. Mode of operation (active/passive): Active/passive
    c. Redundancy - Yes
    d. Safety graded - Yes
83. Control rods (Y/N)
    a. Maximum control rod worth (pcm) - 36
    b. Mode of operation (active/passive) - Active/Passive
    c. Redundancy - Yes
    d. Safety graded - yes

**A3. Decay heat removal**

*A3-1 Primary side*

84 Water injection
    a. Actuation mode (manual/automatic) - Automatic/manual
    b. Injection pressure level - below 158 bar
    c. Flow rate (kg/s)
    d. Mode of operation (active/passive) - Passive, active
    e. Redundancy - Yes
    f. Safety graded (Y/N) - Yes
85. Water recirculation and heat removal
    a. Intermediate heat sink (or heat exchanger) - SG
    b. Mode of operation (active/passive) - Passive
    c. Redundancy - 3 x 100%
    d. Self sufficiency (h) - 24 h for each channel
    e. Safety graded - yes

*A3-2 Secondary side*

86. Feed water
    a. Actuation mode (manual/automatic) - automatic
    b. Flow rate (kg/s)
    c. Mode of operation (active/passive) - passive
    d. Redundancy - 3.100%
    e. Self sufficiency (h) - 24 h for each channel
    f. Safety graded - Yes
87. Water recirculation and heat removal
    a. Ultimate heat sink (cold source) - Water, air
    b. Mode of operation (active/passive) - Passive
    c. Redundancy - 3.100%
d. Self sufficiency (h)
    e. Safety graded - Yes

*A3-3 Primary pressure control*

88. Implemented system (Name) - without safety valves
    a. Actuation mode (manual/automatic)
    b. Side location (primary/secondary circuit)
    c. Maximum depressurization rate (bar/s)
    d. Safety graded

**B. SEVERE ACCIDENT CONDITIONS***

**B.1 Fission products retention**

89. Containment spray system (Y/N) - Yes
90. F.P. Sparging (Y/N)
91. Containment tightness control (Y/N) - Yes
92. Leakage recovery (Y/N)
93. Risk of recriticality (Y/N)

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* All systems must be qualified to operate under the accident conditions.

**B.2 Recriticality control**

94. Encountered design feature
   a. Mode of operation (A/P)
   b. Safety graded

**B.3 Debris confining and cooling**

95. Core debris configuration (core catcher)
96. Debris cooling system (name)
   a. Mode of operation (A/P)
   b. Self sufficiency
   c. Safety graded (Y/N)

**B.4 Long term containment heat removal**

97. Implemented system
   a. Mode of operation (A/P)
   b. Self sufficiency (h)
   c. Safety graded (Y/N)
98. Intermediate heat sink
   a. Self sufficiency (h)
   b. Safety graded (Y/N)
99. External coolant recirculation
   a. Implemented components
   b. Mode of operation (A/P)
   c. Self sufficiency (h)
   d. Safety graded (Y/N)
100. Ultimate heat sink
   a. Self sufficiency (h)
   b. Safety graded (Y/N)

**B.5 Combustible gas control**

101. Covered range of gas mixture concentration
102. Modes for the combustible gas control
   a. Containment inertation - Yes
   b. Gas burning - No
   c. Gas recombining - No
   d. Others - No

**B.6 Containment pressure control**

103. Filtered vented containment (Y/N) - Yes
   a. Implemented system
   b. Mode of operation (A/P) - Active
   c. Safety graded
104. Pressure suppression system (Y/N) - Yes
   a. Implemented system - SSS, CSS
   b. Mode of operation - Active, passive
   c. Safety graded (Y/N) - Yes

## C. SAFETY RELATED I&C SYSTEM

Automatic load following (Y/N) - Yes
* range (% power) - 10-100%
* maximum rate (%/min) - 10

Load rejection without reactor trip (Y/N) - No
Full Cathode Ray Tubes (CRT) display (Y/N) - Yes
Automated start-up procedures (Y/N) - Yes
Automated normal shutdown procedures (Y/N) - Yes
Automated off normal shutdown procedures (Y/N) - No
Use of field buses and smart sensors (Y/N) - Yes
Expert systems or artificial intelligence advisors (Y/N) - Yes
Protection system backup (Y/N) - Yes

### D. EMERGENCY POWER SUPPLY SYSTEM

105. Type (diesel, gas, grid connection) - diesel
106. Number of trains - 2

### E. AC/DC SUPPLY SYSTEM

107. Type (rectifier, converter, battery) - battery
108. Estimated time reserve (hr) - 24

## IV. CONVENTIONAL THERMAL CYCLE

### A. TURBINE SYSTEM

109. Type - steam turbine
110. Overall length (m)
111. Width (m)
112. Number of turbines/reactor - 1
113. Number of turbine sections per unit (e.g. HP/LP/LP) - HP/LP
114. Speed (rpm)

### B. STEAM CHARACTERISTICS

115. H.P. inlet pressure - 42 bar
116. H.P. inlet temperature - 290°C
117. H.P. inlet flowrate - 905 ton/h
118. L.P. inlet pressure
119. L.P. inlet temperature
120. L.P. inlet flowrate (per section)

### C. GENERATOR

121. Type (3-phase synchronous, DC) - 3 phase synchronous
122. Apparent power (MVA)
123. Active power (MW) - 200
124. Frequency (hz) - 50
125. Output voltage (kV)
126. Total generator mass (t)
127. Overall length
128. Stator housing outside diameter

### D. CONDENSER

129. Number of tubes
130. Heat transfer area
131. Flowrate ($m^3/s$)
132. Pressure (m/bar)
133. Temperature (°C)

### E. CONDENSATE PUMPS

134. Number - 3
135. Flowrate
136. Developed head
137. Temperature
138. Pump speed

### 7.7.5. Project status

The ATS-150 development is currently being carried out by:

- OKB Mechanical Engineering (Nizhny Novgorod): Nuclear island design.
- Design and Engineering Institute "Atomenergoproject" (Nizhny Novgorod): plant-level systems and architect-engineering design,
- Kharkov Turbine Plant Design Division (Ukraine): Turbine generator design,
- Institute of physics and power engineering (Obninsk, Kaluga Region): Scientific support.

At present the conceptual design of the ATS-150 has been completed and some activities for the basic design development are in progress.

The ATS-150 reactor plant (RP) has been developed on the basis of the AST-500 prototype power units and its construction experience, using thoroughly studied and technically proven design decisions and technologies of the VVERs and the Nuclear Steam Supply System being used for icebreakers, whose reliability has been confirmed by the long-term successful operation.

Taking into account the experimental research completed and the available feedback from operating reactors it is possible to conclude that the R&D work has a sufficient degree of completeness for the following components and characteristics of the ATS-150:

- fuel rods and assemblies, core structural elements (there is a possibility to increase fuel lifetime up to 6 years)
- control rod drive design;
- natural convection of primary coolant (thermohydraulics, thermodynamic, safety);
- primary coolant chemistry and corrosion suppression; emergency decay heat removal systems design;
- manufacturing and transportation of the reactor and guard vessels.

In this connection the main part of the plant specific R&D work still required concerns the tests of new equipment items; as well as calculations and experimental research on the safety issues, particularly:

- design basis and beyond-design accident simulation and analysis;
- thermohydraulic experiments;
- validation of reliability and efficiency of the emergency heat removal systems.

Most of the required test facilities are available.

A PSAR for the AST-150 has been prepared. Its review by the licensing body will be done following a decision to be made by the utility. No licensing activity is in progress now.

### 7.7.6. Project economies

The ATS-150 is a small power reactor for multi-purpose application (electricity production, district heating, process heating, desalination). The motivation for the design development was based on the favourable conditions for the use of such reactors in isolated regions with rather low energy demand and lack of local fossil fuel resources. It is possible to consider that the ATS-150 could be a reasonable facility for energy supply, and a starting point for the national nuclear programmes in developing countries.

## 7.8. MARS REACTOR SYSTEM DESCRIPTION AND DEVELOPMENT STATUS

### 7.8.1. Basic objectives and features

The MARS (Multipurpose Advanced Reactor, inherently Safe) plant is a nuclear plant designed for electric power generation and/or for industrial heat generation.

#### *7.8.1.1. Design objectives*

The basic design objectives are: i) reliance only on passive safety; ii) utilisation of physical laws not only to detect the safety action requirements, but also to perform the safety action itself; iii) predicted core damage frequency lower than $1.10^{-10}$ per year; iv) reduced costs and in-shop oriented construction of the plant; v) plant availability goal greater than 94%; vi) acceptance by population (through safety features intelligible by the layman) and by utilities (through drastic simplification of the plant, easy operation and maintenance, lower doses to personnel, certainty in construction times and costs, extension of plant life and drastic simplification of decommissioning, reparability-oriented and testability-oriented plant solutions).

#### *7.8.1.2. Main features*

The main features of the MARS plant are: i) primary-coolant-related components based on the most proven nuclear technology; ii) emergency decay heat removal based on only static components, with the exception of one component only, but of passive type; iii) prevention of any loss of primary coolant, by design; iv) infinite emergency core cooling capability, without requirement of man intervention; v) very simple plant circuitry; vi) advanced pre-assembling of all the plant, including the nuclear part; vii) possibility of easy disassembling, repair and substitution of all components in contact with the primary coolant (including the reactor vessel and the steam generator); viii) possibility of a depreciation period for the plant longer than 80 years; ix) only five primary system moving components in a physical contact with the primary coolant and to be maintained, contributing to limiting the doses to personnel to very low values.

#### *7.8.1.3. Design Criteria*

The major design criteria for the nuclear island are:

- maximum use of proven technology: the nuclear design, the primary coolant overall layout, the steam generator type were selected on the basis of the most experienced nuclear technology in the world;
- adoption of a passive and inherent-safety solution as far as the core shutdown (first cornerstone of nuclear safety) is concerned: a core-temperature-and-gravity-controlled

two-metal scram device is added to traditional PWR electrically controlled and gravity-driven scram system;

- adoption of a passive and inherent-safety solution as far as the residual heat removal (second cornerstone of nuclear safety) is concerned: a completely innovative emergency core cooling system is foreseen, based on: i) natural convection for cooling fluids; ii) very simple circuitry; iii) one moving component (but passive type) only; iv) infinite autonomy of operation; v) a completely testable scheme, during normal plant operation;
- removal of possibility of failure of the primary-coolant boundary, hence removal of loss of coolant accidents (no LOCA, no control rod ejection, relevant limitation in accidental scenarios): the primary-coolant boundary is enclosed in a pressurised containment filled with low-enthalpy water;
- removal of core damage causes, potentially leading to severe accidents and radioactive releases; it is achieved through the combination of: i) the "inherently safe", additional core shutdown system; ii) the innovative, completely passive, residual heal removal system; iii) the enclosure of the whole primary-coolant boundary in the low-enthalpy-water-filled, pressurised containment.
- reduced and certain costs. In the construction phase, it is achieved through plant simplification, thanks to: i) ECCS drastic simplification; ii) drastic reduction of "safety grade" systems and components; iii) primary coolant components pre-assembling, consequence of the primary coolant lower pressure (72 bar) and of the use of "small size", flanged components for the fluid systems; iv) auxiliary systems' pre-assembling, due to the complete accessibility of the reactor building (geometrical optimisation; no constraints imposed by the necessity of withstanding high internal pressures); v) lower unit cost of main components; vi) maximisation of in-shop activities with respect to in-site ones; vii) a test phase drastically simplified; viii) depreciation time for the plant substantially extended. In the operation and maintenance phase, it is achieved through high load factors, minor maintenance requirements, easy tests, easy replacement even of major nuclear components. In the decommissioning phase, it is achieved through a really decommissioning-oriented design of components, fluid systems, structures, buildings.
- lower radiation doses to personnel, achieved through a simple circuitry, very few maintainable components physically interfacing with the reactor coolant, the enclosure of the primary system in a water-filled containment, minimisation of primary-coolant clean-up requirements, complete and easy access of the reactor building, possibility of complete disassembling of all primary-coolant components;
- higher DNBR, achieved through the core power density reduction;
- enterprise risk reduction (limited size of plant units, well proven construction technology for all components, few and "small" components, pre-assembling of all fluid systems);
- extended industrial applications, possible for the very high safety standard.
- protection against external events: the Italian (ENEA-DISP) criteria against external events are adopted (including, among others, plant protection against external missiles);
- highest reliability of the plant against fire accidents, consequence of the drastic reduction in number of electric motors (only one big-size electric motor in the reactor building), of the simplification of instrumentation and of the absence of requirement of big-size diesel generators;
- reduced plant vulnerability;
- walk-away safety: the innovative residual heat removal system, through condensation and natural convection of air, allows indefinite time for natural heat dissipation without need of human intervention.

### 7.8.2. Design description

*7.8.2.1. Nuclear steam supply system*

The reactor is moderated and cooled by pressurised light water (PWR); in the version here described, the nominal core thermal power is 600 MW.

The Primary Cooling System includes one loop only, with one canned pump and one vertical-axis U-tube steam generator. Connected to the reactor vessel is the safety core cooling system. The pressure inside the primary cooling system (72 bar) is controlled by a vapour-bubble pressuriser. On-off valves are installed in the primary cooling loop, in order to isolate, if necessary, the steam generator and the primary pump (i.e., in the event of a SG tube rupture). The primary cooling system and the safety core cooling system are inside a pressurised containment (CPP), filled with water at the same pressure as the primary coolant, but at low temperature (about 70°C).

In the development of the MARS design, well-proven technological features of most experienced PWR's were extensively maintained for all components of the primary circuit (geometry of fuel rods; lay-out of the primary cooling system; type of control rods and of control rod drive mechanisms; most core internals; etc.).

Engineered core safeguards are, on the contrary, innovative. In particular, the safety decay heat removal system, further described, is innovative and its principle of operation (utilisation of natural circulation only) limits the thermal power design value, that cannot exceed about 1000 thermal MW per unit, and in the solution herein described was limited to 600 $MW_{th}$. Another "unusual" parameter is the pressure in the primary system, chosen equal to 72 bar. This choice, leading to a loss in the thermodynamic efficiency of the plant, has nevertheless allowed the adoption of the pressurised Containment for Primary system Protection (CPP, the pressurised boundary that envelopes the primary cooling system and the safety core cooling system, described in paragraph 6.2.4), thus eliminating the possibility of loss-of-primary-coolant accidents.

An insulating system on the external side of the whole primary coolant boundary, realised through matrices of stainless steel wiring, limits thermal losses to about 0.3% of the reactor thermal power.

A Residual Heat Removal system brings the plant to cold conditions during a normal plant shutdown. This system is connected to the primary coolant loop through two low-diameter lines of the Chemical and Volumetric Control System.

Other auxiliary systems of the nuclear supply system are:

- the Pressurised Containment Heat Removal System (CPPHR), with the following main functions: i) to maintain the water temperature inside the CPP below 70° C, in all operation conditions; ii) to control the water pressure in the CPP, maintaining it, in normal operation conditions, roughly equal to that of the primary coolant system; iii) to perform the volumetric control of the water in the CPP; iv) to limit the amount of impurities of the water in the CPP to acceptable levels. The CPPHR system bases its performance on the continuous circulation of the CPP water through a circuitry including a heat-exchanger and pumps.

- the Chemical and Volumetric Control System (CVCS), with the following functions: i) to perform the volumetric control of the water in the Primary Coolant System; ii) to limit the amount of impurities in the primary coolant to acceptable limits; iii) to perform the injection of additives in the primary coolant; iv) to perform the boron concentration control in the primary coolant; v) to remove the residual heat from the primary coolant after normal reactor shutdown; vi) to perform the reactor cavity flooding and draining (for refuelling operations). The CVCS is connected to the primary coolant system only through small-size connections (4"). The outlet connection is equipped with a train of four on/off valves; three valves are passive-type, the fourth valve is active-type. The inlet connection is equipped with a train of four valves; three of these are check valves and the fourth one is an on/off valve. Both the valve trains are placed inside a nitrogen pressurised containment connected with the CPP.

Other auxiliary systems are: the radioactive effluents treatment system, the components intermediate cooling system, the spent-fuel pool cooling and purification system.

*7.8.2.2. Balance of plant systems*

As far as the heat utilisation system is concerned, several alternatives were analysed during the design development, since one of the fundamental design criteria of the MARS reactor is the possibility of using the nuclear power produced, not only for the production of electric power, but also for other industrial purposes such as the production of steam for industrial uses (desalination of sea water; district heating).

The nuclear reactor system, thanks to its reliability and to its inherent safety characteristics, may be installed in proximity of industrial and inhabited centres, while the design criterion of easy operation that characterises its circuits and its auxiliary systems extends its potential use to non-industrialised geographic areas, for which the needs for desalination is well known.

The heat utilisation system will be different, in function of the specific scope of the plant, and will be able to reach - as highest performance limits and with reference to the core characteristics above described - the following performances: a) the production of saturated steam at 17 bar; b) the production of sub-cooled water at about 210°C. Should steam be used for the production of electric power through a turbine-alternator group, a thermal cycle efficiency of 28% could be achieved, with the production of about 170 electric MW.

*7.8.2.3. Instrumentation, control and electrical systems*

Control of core reactivity is assured through a conventional-type control rod cluster system operated electrically (magnetic jack activation) and relying on gravity. This system is associated with the above mentioned innovative, passive-type scram system, operated by gravity and thermal elongation of metal bars (see paragraph 6.2.4). Long-term compensation of reactivity is obtained through boron dilution in the primary coolant (maximum boron concentration at the beginning of cycle, for an irradiation cycle of 18 months and a permanence of fuel in the core for 3 (partially, for 4) cycles, is 700 ppm). Instrumentation systems and electrical systems are characterised by an extreme simplicity.

*7.8.2.4. Safety considerations and emergency protection*

Engineered safeguards are innovative and completely passive type

An innovative core safeguard is the additional, special scram system (ATSS), which was studied and designed to provide the automatic and safe shutdown of the reactor as soon as the fluid temperature in the core rises over a maximum selected value This special scram system eliminates the occurrence of ATWS accidents The reactivity control of the additional scram system is obtained through the insertion, into the core, of control rod clusters with the same geometrical and physical characteristics as the traditional scram system The difference between the two systems is the type of actuator selected In order to assure the required reliability of actuation, each inherent-safety control rod cluster is controlled by a special actuator, based on a simple physical principle the thermal expansion of a rod, due to the variation of temperature of the core coolant, which leads to the disconnection of hooks holding the control rod cluster The geometry and the design of the innovative actuator assure a considerable mechanical force to operate the mechanical hooks The actuator was designed so as to use the mechanical force developed in the sensor itself to open directly the hooks sustaining the ATSS control rod cluster

The Safety Core Cooling System is schematically sketched in the figure During an accidental event (e g , station black-out or steam generator feed-water loss), its actuation is automatic and the operation of the system is completely passive

It comprises two 100% independent trains, each one including

- a Primary Safety Cooling System (PSC)
- an Intermediate Safety Cooling loop (ISC)
- a reservoir of cold water (pool) with piping which brings the steam produced to a special atmospheric-pressure-condenser, cooled by environment air through natural circulation (Pool and Condenser Loop, PCL)

The PSC loop is directly connected to the reactor vessel In case of system intervention, after a first transient phase, the forcing head for the circulation of the coolant is assured by a difference in level of about 7 m between the vessel outlet nozzle and the primary heat exchanger (item "4" in figure)

There is a total mechanical independence between the emergency core cooling trains and the primary loop, the connections of the two trains with the primary coolant boundary are directly on the reactor vessel, through four nozzles The two nozzles of each train are positioned on two different quotes the plane of the hot-leg connection is 1 8 m higher that the cold-leg one, to increase the forcing head which improves the starting of water circulation in the emergency core cooling system In case of an emergency, the insertion of each of the two trains is automatic (the check valves "14" in the figure open, on each train, two independent check valves connected in parallel are present) In such a situation, the coolant, naturally circulating in each train, transfers the core thermal power to the water of the intermediate cooling loop by means of a shell-and-tube heat exchanger (item "4" in the figure) The intermediate loop, still using natural circulation, transfers the power through a tube bundle to the water of a reservoir (in communication with the atmosphere) The intermediate loop provides an additional barrier between the reactor coolant system and the external environment

The difference of level for the natural circulation in the ISC loop is about 10 m Pressure in the ISC loop is 72 bar and it guarantees sub-cooling conditions of water during any accidental situation or transient This value of the pressure is maintained by means of a pressuriser connected to the hot leg of the loop The heat exchanger (item "5" in figure) which transfers heat from the ISC circuit to the water of the emergency reservoir is realised by means of a tube bundle submerged in a pit filled with water, which is the main water reservoir pool.

The emergency water reservoir is built in reinforced concrete, internally lined by stainless steel, and is designed to guarantee the core cooling for a theoretically infinite period.

The reservoir water flows through the pit owing to local natural circulation, absorbing heat from the ISC loop; its temperature rises and then it vaporises. Steam produced is mixed with air initially present in the dome over the pool. The pressure in the dome rises and this causes a flow of the mixture (air+steam) towards a connection path with the atmosphere (item "21" in the figure). Between the pool dome and the connection path is an inclined tubes heat exchanger (item"20"), where steam partially condenses. With the progress of vaporisation in the pool, the air content in the dome decreases, and the blanketing effect of air on condensation is reduced, thus causing an increase in the condensation rate in the heat exchanger, which is cooled by external air in natural circulation. After a short transient from the beginning of vaporisation, the heat transfer capacity of the condenser is such that practically all steam produced in the pool condenses, no further steam losses occur through the communication path, and the condensate produced goes back to the pool.

The performance of the system allows, without presence of pumps, motors, special diesel generators, or any other active component, a core cooling in safe conditions.

The piping and the heat exchangers of the emergency core cooling system in contact with the primary coolant, together with the primary coolant system itself, are placed inside the pressurised Containment for Primary system Protection (CPP), which also houses the pressuriser-safety-and-relief-valve-discharge-tank. This component is designed to avoid pressurisation of the primary system during operational power transients (load rejection). Should an accident of primary-system-pressuriser-safety-valve-struck-open occur, the limited capacity of the discharge tank is such that pressure increases inside it, avoiding an unacceptable primary coolant system depressurisation.

The pressurised Containment for Primary system Protection is filled with water roughly at the same pressure (0.2 bar lower) as the primary coolant and at 70°C. The design selected is aimed at a global easy disassembling of all components. The CPP is manufactured in carbon steel, may be built and tested in shop and transferred and assembled in site.

Special mention should be made to the special coupling among the shell, the tube sheet and the lower head of the steam generator, which allows the realisation of, also through this component, the double barrier enclosing the primary coolant (the shell of the steam generator is designed to withstand the total pressure of the primary coolant (72 bar)).

A slow flow of water is maintained in the CPP, in order to realise a uniformity in its temperature and in order to make it easy, through special check points, the detection of potential small losses of primary flow into the CPP, through the flanged couplings. The detection is made possible by the small under pressure of the CPP boundary with respect to the primary coolant pressure.

*7.8.2.5. Buildings and structures*

The whole nuclear steam supply system may be placed in one "small" building, not necessarily designed to withstand internal over pressure (LOCA accidents are avoided).

Nevertheless, in spite of the capability of this reactor concept to avoid by design LOCA accidents and in spite of its unique capability to face even a core melting accident through re solidification of corium within the reactor vessel (thanks to cooling by external water and to the small reactor vessel thickness), a containment system is foreseen for this plant. A containment building was designed, to house the whole primary system and all the core safeguards, with the following dimensions:

- internal diameter: 24 m
- internal height, cylindrical part: 33 m
- overall internal height: 45 m
- internal gross volume: about 18,500 $m^3$.

### 7.8.3. Safety concepts

TABLE 7.8.1. MAIN SAFETY RELATED SYSTEMS IN THE MARS PLANT

| Name | Safety graded | Main characteristics |
|---|---|---|
| Primary Circuit | X | Reactor vessel:<br>1 circulation pump<br>1 steam generator |
| Control Rods | X | 32 Rod cluster control assemblies |
| Additional Inherently Safe Reactivity Control System | X | 20 gravity driven, coolant temperature actuated control rod clusters |
| Passive Residual Heat Removal System | X | 2 100% trains of natural-circulation water loops permanently connected to the primary circuit and cooled by external air. Only one non-static component, passive type (special check valve), 400% redundant |
| Pressurized Containment for Primary System Protection (CPP | X | low enthalpy-water-filled containment enveloping all the primary coolant pressure boundary |

TABLE 7.8.2. MAIN ACCIDENT INITIATORS FOR THE MARS PLANT

- LOCA (Secondary): Secondary Pipe Rupture
- LOCA (Interfacing): Steam Generator Tube Rupture (SGTR)
- Primary Transients
- Secondary Transients (Turbine trip)
- Loss of the Steam Generator Feed water
- Station blackout

TABLE 7.8.3. DESIGN FEATURES FOR PREVENTION AND PROTECTION LEVELS THAT REDUCE (R), SUPPRESS (S), THE INITIATORS FREQUENCY OR LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES (L)

| |
|---|
| **PREVENTION LEVEL** |
| LOCA (Primary)<br>- Enclosure of the primary coolant boundary in the pressurised, water-filled containment = suppresses LOCA's (LOCA's and control rod ejection)<br>- Reduced vessel fluency = reduces primary vessel failure probability<br>- Canned primary pump = suppresses local break<br>- Monitoring of pressurised containment water = limits accident consequences<br><br>LOCA (Secondary)<br>- SG shell and part of the steam line designed at the primary coolant pressure level = suppresses SG and steam line ruptures<br>- two ON/OFF valves on the main steam line = suppress SLBA consequences<br><br>LOCA (Interfacing)<br>- SG shell and two isolation valves designed at the primary coolant pressure level = limit consequences of SG tube rupture<br>- maximum primary coolant temperature equal to 244°C = reduces tube rupture probability<br>- two ON/OFF valves on the primary coolant circuit = limit consequences of SG tube rupture |
| Primary transient<br>- increased design margins = reduce initiator frequency |
| Secondary transient |
| Loss of electric sources<br>- *not affecting the core integrity and cool ability, thanks to the innovative, passive decay heat removal system* |
| Total loss of the cold source (Water)<br>- not affecting the core integrity and cool ability, thanks to the innovative, passive decay heat removal system |
| Total loss of the SG feed-water<br>- not affecting the core integrity and cool ability, thanks to the innovative, passive decay heat removal system |
| Station Blackout<br>- not affecting the core integrity and cool ability, thanks to the innovative, passive decay heat removal system |
| **PROTECTION LEVEL** |
| LOCA (Primary)<br>- water of the pressurised containment = prevents consequences of possible leakage<br>- increased water inventory around the core<br>- increased ratio primary-coolant-volume/core-thermal-power<br>LOCA (Secondary)<br>LOCA (Interfacing) - easy SG isolation and with several barriers |
| ATWS - the additional "inherent" scram system prevents ATWS's<br>- strong negative temperature coefficient = limits consequences |
| Primary Transient - great primary coolant thermal inertia |
| Secondary Transient - great primary coolant thermal inertia |
| Loss of Electric Sources<br>- implementation of battery-based electric power sources, to feed a few, small-power users |
| Total Loss of heat sink - not critical<br>Total Loss of SG feed water - not critical<br>Station Blackout - not critical |

TABLE 7.8.4. DESIGN FEATURES FOR MITIGATION LEVEL OF THE MARS PLANT

| Safety Function | Systems (Cf. Tab. 7.8.1.) | Passive/Active | Design features/Remarks |
|---|---|---|---|
| **Design Basis**<br>Fission Product Containment | Cladding<br>Primary Coolant System<br>Water-filled, Pressurised Containment<br>Containment | Passive<br>Passive<br>Passive<br>Passive | Traditional type<br>High volume/power ratio<br>Avoids LOCA accidents and solidifies corium<br>Protects internal systems from external events |
| Coolant inventory | Primary Coolant System<br>Water-filled, Pressurised Containment | Passive<br>Passive | High volume/power ratio, lower pressure<br>Additional cooling capacity for severe accidents |
| Decay Heat Removal | Passive Decay Heat Removal System | Passive | Mainly static (only one non-static component, but passive, with 400% redundancy) |
| Reactivity control | Control Rod Clusters<br>Additional passive Control Rod Clusters<br>Boron concentration control | Active<br>Passive<br>Active | Traditional type<br>Additional, independent, passive type (no ATWS)<br>Traditional type, low boron concentration required |
| **Severe Accident**<br>Containment temperature and pressure control | Water-filled, Pressurised Containment<br><br>Containment | Passive<br><br>Passive | Avoids all LOCA category accidents, allows heat removal from corium<br>Pressure, temperature control, radioactivity confinement |
| Heat removal | Containment decay heat removal system | Passive | Static, never ending capacity |
| Tightness control | Containment | Passive | Very low over pressure (lower than 1 bar) even with severe accidents small volume, easy protection |
| Inflam gas control | Igniters | Any type | Practically irrelevant, owing to extremely high defence from cladding/coolant reaction |
| Fission product containment | Water-filled, pressurised containment<br><br>Permanently flooded containment | Passive<br><br>Passive | Relevant, additional obstacle to core degradation (avoids LOCAs additional coolant solidifies corium), additional fission products container<br>Resistant to heavy external events |
| Corium management | Water-filled, pressurised containment<br>Core Catcher | Passive<br>Passive | Absorbs heat from corium and blocks it in vessel<br>Vessel lower head protection |

## 7.8.4. Design data questionnaire

### I. GENERAL INFORMATION

1 Design name MARS
2 Designer/Supplier address University of Rome "La Sapienza", ENEA (Italy)
3 Reactor type PWR / Number of modules/per plant 3
4 Gross thermal power (MW-th) per reactor 600
5 Net electrical output (MW-e) per reactor up to 170
6 Heat supply capacity (MW-th) up to 600

### II. BASIC DESIGN DESCRIPTION

### A.. CORE AND REACTIVITY CONTROL

7 Fuel material $UO_2$
8 Fuel inventory (tones of heavy metal)
9 Average core power density (kW/liter) 63
10 Average fuel power density (kW/kgU) 16 2
11 Maximum lnear power (W/m) 11,783 BOC
12 Average discharge burnup (MWd/t) 30,000
13 Inıtial enrichment or enrichment range (Wt%) (Wt%)
14 Reload enrichment at the equilibrium (Wt%) (Wt%) 3 2 (Zircalloy claddıng)
15 Refuellıng frequency (months) 17
16 Type of refuellıng (on/off power) off
17 Fraction of core withdrawn (%) 33
18 Moderator material and inventory water/36 $m^3$
19 Active core height (m) 2 6
20 Core diameter (m) 2 16
21 Number of fuel assemblies 96
22 Number of fuel rods per assembly 204
23 Rod array in assembly 15 x 15
24 Clad material Zircalloay/AISI 304
25 Clad thickness (mm)
26 Number of control rods or assemblies 32, main clusters
27 Type Cluster-type
28 Additional shutdown systems 20 clusters
29 Control rod neutron absorber material Ag/In/Cd
30 Soluble neutron absorber Boron
31 Burnable poison material and form (BOROSILICATE GLASS)

### B. REACTOR COOLANT SYSTEM

#### B1. Coolant

32 Coolant medium and inventory water/120 $m^3$
33 Design coolant mass flow through core (kg/s) 4800 about
34 Cooling mode (forced/natural) forced
35 Operating coolant pressure (bar) 72
36 Core inlet temperature (°C) 214
37 Core outlet temperature (C) 244

#### B2. Reactor pressure vessel/tube

38 Overall length of assembled vessel/tube (m) 11 091
39 Inside vessel/dıameter (m) 3,000 mm
40 Average vessel/tube thickness (mm) 120 mm
41 Vessel/tube material low alloy carbon steel
42 Lining material stainless steel
43 Design pressure (bar) 80
44 Gross weıght (tonne) 88,000 kg

#### B3. Steam generator

45 Number of steam generators 1

46 Type once through recirculation type
47 Configuration (horizontal/vertical) vertical
48 Tube material inconel 600
49 Shell material low alloy carbon steel
50 Heat transfer surface per steam generator ($m^2$) 7893
51 Thermal capacity per steam generator (MW) 600
52 Feed water pressure (bar)
53 Feed water temperature (C) 150
54 Steam pressure (bar) 17 4
55 Steam temperature (C) 204 3

**B4. Pressurizer**

56 Pressurizer total volume ($m^3$) 25
57 Steam volume (full power/zero power, $m^3$) 10

**B5. Main coolant pumps**

58 Number of cooling or recirculation pumps 1
59 Type Canned
60 Pump mass flow rate (kg/s) 4800
61 Pump design rated head 5 5 bar
62 Pump nominal power (kW) 3000
63 Mechanical inertia (kg $m^2$) 250

## C. CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS)

64 Number of extraction lines 1
65 Number of pumps 2
66 Number of injection points 1
67 Feed and bleed connections (1+1)

## D. CONTAINMENT

68 Type
69 Overall form (spherical/cyl ) spherical + cylindrical
70 Structural material steel
71 Liner material
72 Simple/double wall
73 Dimensions (diameter, height) (m)
74 Design pressure (bar)
75 Design temperature (C)
76 Design leakage rate (% per day)

# III. SAFETY RELATED SYSTEMS

## A. DESIGN CONDITIONS

**A1. Fission product retention**

77 Containment spray system (Y/N) Not applicable
a Duration (h)
b Flow rate ($m^3$/h)
c Mode of operation (active/passive)
d Safety graded (Y/N)
78 F P sparging (Y/N) Not applicable
79 Containment tightness control (Y/N)
80 Leakage recovery (Y/N)
81 Guard vessel (Y/N)

**A2. Reactivity control**

82 Absorber injection system (Y/N) N
a Absorber material
b Mode of operation (active/passive)
c Redundancy
d Safety graded
83 Control rods (Y/N) Y
a Maximum control rod worth (pcm)

b Mode of operation (active/passive) Active and passive
c Redundancy
d Safety graded Y

**A3. Decay heat removal**
*A3-1Primary side*
84 Water injection Not applicable
a Actuation mode (manual/automatic)
b Injection pressure level (bar)
c Flow rate (kg/s)
d Mode of operation (active/passive)
e Redundancy
f Safety graded (Y/N)
85 Water recirculation and heat removal
a Intermediate heat sink (or heat exchanger) SG
b Mode of operation (active/passive) Passive
c. Redundancy 100%
d Self sufficiency (h) infinite
e Safety graded yes
*A3-2 Secondary side*
86 Feed water
a Actuation mode (manual/automatic)
b Flow rate (kg/s)
c Mode of operation (active/passive)
d Redundancy
e Self sufficiency (h)
f Safety graded
87 Water recirculation and heat removal
a Ultimate heat sink (cold source)
b Mode of operation (active/passive)
c Redundancy
d Self sufficiency (h)
e Safety graded

*A3-3 Primary pressure control*
88 Implemented system (Name)
a Actuation mode (manual/automatic)
b Side location (primary/secondary circuit)
c Maximum depressurization rate (bar/s)
d Safety graded

**B. SEVERE ACCIDENT CONDITIONS***

**B.1 Fission products retention**
89 Containment spray system (Y/N) Not necessary
90 F P Sparging (Y/N) Not necessary
91 Containment tightness control (Y/N) Y
92 Leakage recovery (Y/N) Not necessary
93 Risk of recriticality (Y/N) N

**B.2 Recriticality control**
94 Encountered design feature Not applicable
a Mode of operation (A/P)
b Safety graded

**B.3 Debris confining and cooling**
95 Core debris configuration (core catcher) Emispherical, In-vessel
96 Debris cooling system (name) water-filled, pressurized containment (CPP)
a Mode of operation (A/P) A
b Self sufficiency Yes
c Safety graded (Y/N) Y

* All systems must be qualified to operate under the accident conditions

**B.4 Long term containment heat removal**

97. Implemented system
    a. Mode of operation (A/P): A
    b. Self sufficiency (h): Infinite
    c. Safety graded (Y/N): Y
98. Intermediate heat sink
    a. Self sufficiency (h)
    b. Safety graded (Y/N)
99. External coolant recirculation:
    a. Implemented components:
    b. Mode of operation (A/P): A
    c. Self sufficiency (h): Infinite
    d. Safety graded (Y/N): Y
100. Ultimate heat sink
    a. Self sufficiency (h): Infinite
    b. Safety graded (Y/N): Y

**B.5 Combustible gas control**

101. Covered range of gas mixture concentration
102. Modes for the combustible gas control
    a. Containment inertation
    b. Gas burning
    c. Gas recombining
    d. Others

**B.6 Containment pressure control**

103. Filtered vented containment (Y/N)
    a. Implemented system
    b. Mode of operation (A/P)
    c. Safety graded
104. Pressure suppression system (Y/N)
    a. Implemented system
    b. Mode of operation
    c. Safety graded (Y/N)

**C. SAFETY RELATED I&C SYSTEM**

Automatic load following (Y/N)
* range (% power)
* maximum rate (%/min)

Load rejection without reactor trip (Y/N)
Full Cathode Ray Tubes (CRT) display (Y/N)
Automated start-up procedures (Y/N)
Automated normal shutdown procedures (Y/N)
Automated off normal shutdown procedures (Y/N)
Use of field buses and smart sensors (Y/N)
Expert systems or artificial
intelligence advisors (Y/N)
Protection system backup (Y/N)

**D. EMERGENCY POWER SUPPLY SYSTEM**

105. Type (diesel, gas, grid connection): Not necessary
106. Number of trains

**E. AC/DC SUPPLY SYSTEM**

107. Type (rectifier, converter, battery): Battery
108. Estimated time reserve (hr): 120

**IV. CONVENTIONAL THERMAL CYCLE**

**A. TURBINE SYSTEM**

109. Type
110. Overall length (m)

111. Width (m)
112. Number of turbines/reactor
113. Number of turbine sections per unit (e.g. HP/LP/LP)
114. Speed (rpm)

**B. STEAM CHARACTERISTICS**

115. H.P. inlet pressure bar
116. H.P. inlet temperature (C)
117. H.P. inlet flowrate (kg/s)
118. L.P. inlet pressure
119. L.P. inlet temperature
120. L.P. inlet flowrate

**C. GENERATOR**

121. Type (3-phase synchronous, DC)
122. Apparent power (MVA)
123. Active power (MW)
124. Frequency (hz)
125. Output voltage (kV)
126. Total generator mass (t)
127. Overall length
128. Stator housing outside diameter

**D. CONDENSER**

129. Number of tubes
130. Heat transfer area
131. Flowrate ($m^3/s$)
132. Pressure (m/bar)
133. Temperature (C)

**E.CONDENSATE PUMPS**

134. Number
135. Flowrate
136. Developed head
137. Temperature
138. Pump speed

### 7.8.5. Project status

The development of the project of the MARS plant was started in 1984 by the University of Rome "La Sapienza", Italy.

The preliminary design of the plant was completed in 1989, and concerned:

- the primary coolant system, including the nuclear core, the reactor vessel, the steam generator, the pressure control system;
- the emergency core cooling system, with all innovative components as the no-hydraulic-resistance special check valve;
- the low-enthalpy-water-filled containment (CPP), with its auxiliaries;
- the chemical and volumetric control system;
- the main reactivity control system and the additional, passive reactivity control system (ATSS);
- the main reactor building.

At the same date also a preliminary cost evaluation was performed, and a preliminary probabilistic risk assessment and a safety analysis limited to main accidental scenarios were carried out.

Later on, research and design activities concerned the development of special components as the improved check valves and the atmospheric-pressure condenser of the passive decay heat removal system. Experimental tests were also performed on prototypes.

In 1993, a thorough review of the project started in the framework of a co-operation agreement between ENEA (Italy) and the University of Rome "La Sapienza", with the aim at developing a basic design and at performing a Preliminary Safety Report of MARS nuclear plant, to be submitted to the Italian Nuclear Safety Authority. The Preliminary Safety Report was completed in May 1994.

### 7.8.6. Project economics

The MARS plant design is, at the moment, at a stage which does not allow the provision of cost figures based on a detailed cost analysis. A preliminary and rough cost evaluation performed in 1989 showed that the cost of the electric energy produced through the MARS plant was comparable with the cost of electric energy produced through traditional nuclear plants. The present design development is aimed at introducing all possible plant simplifications which may be compatible with the special level of safety which is the characteristic of the MARS concept. The aim is the achievement of even higher safety standards, with even lower costs.

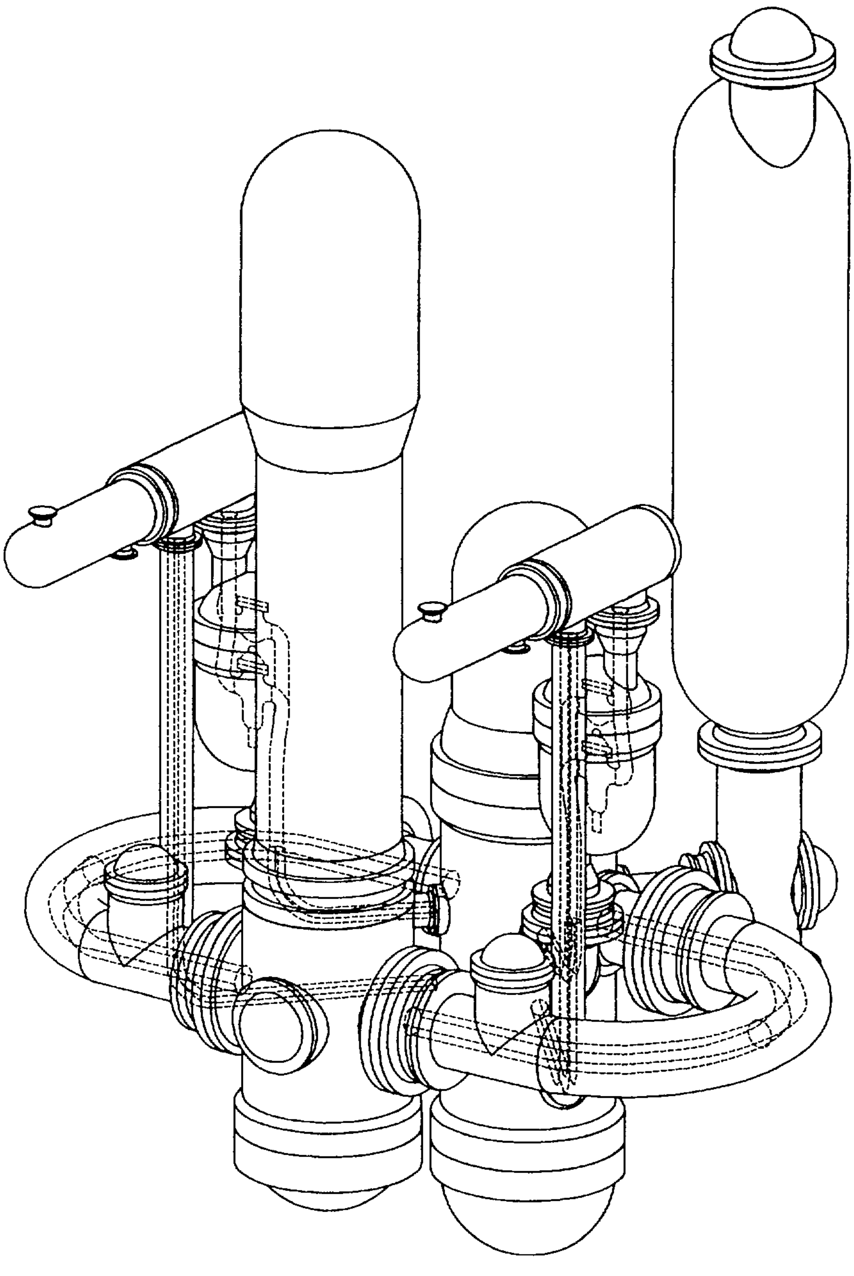

*Fig 7.8.1. MARS Primary system and pressurised containment*

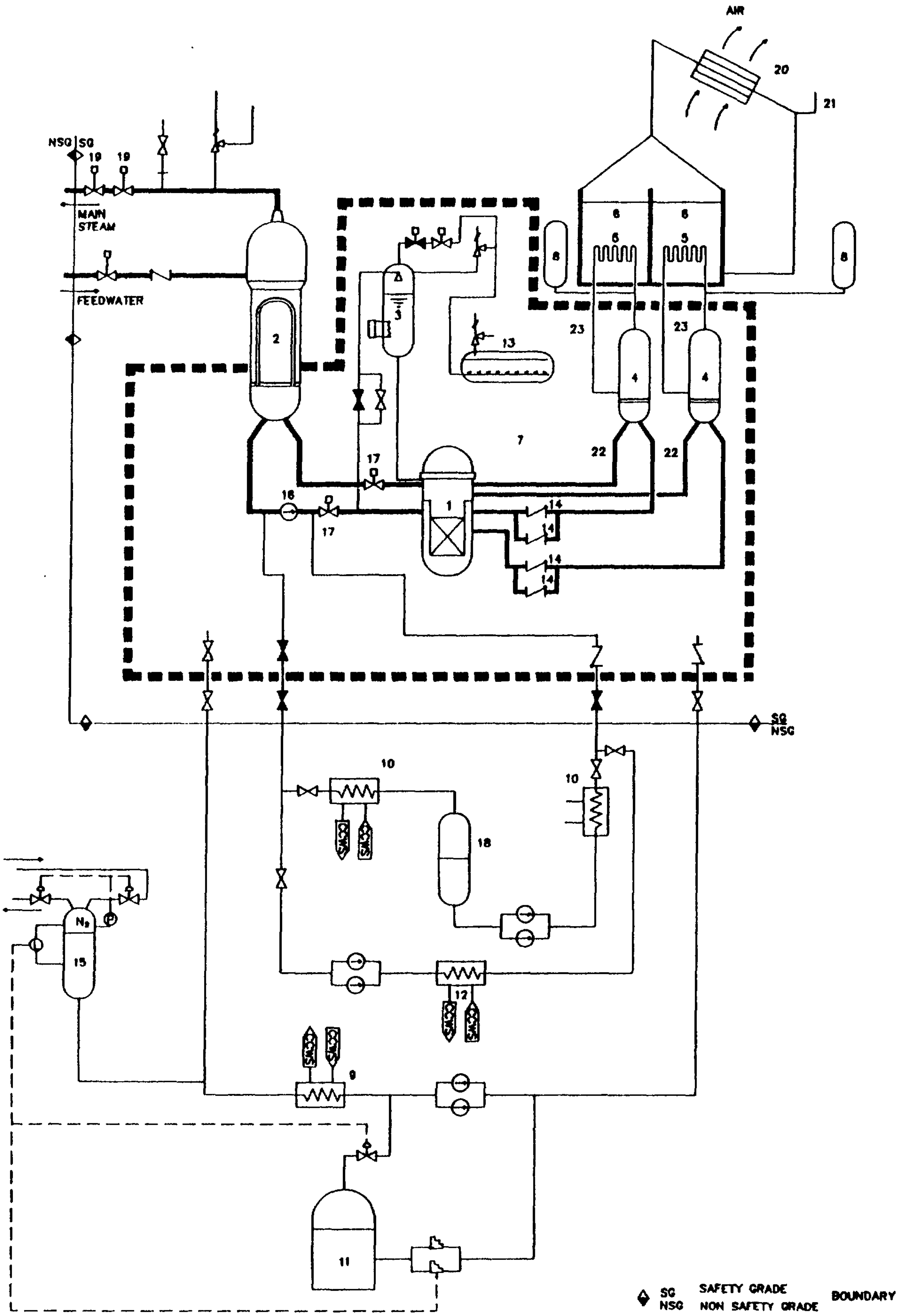


*Fig 7.8.2. MARS Primary coolant system flow diagram*

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## 7.9. RUTA-20 REACTOR SYSTEM DESCRIPTION AND DEVELOPMENT STATUS

### 7.9.1. Basic objectives and features

A. Purpose

The pool-type reactor RUTA-20 was developed as a heat source for district heating in small settlements and towns. Apart from heating purposes, the reactor can be used as a power source for summer air conditioning in residential and industrial complexes. It is also possible to use the reactor as a power source for desalination of sea water. Thermal power of a single unit is 20 MW.

B. Prototype

The RUTA-20 reactor concept is based on long-term successful experience in operating pool-type research reactors. In addition, RUTA-20 reactor safety has been particularly enhanced due to technological progress in manufacturing of different reactor units, as well as modern approaches to the reactor design process with emphasis on inherent safety features based on fundamental laws of nature.

C. Consumer features

- fossil fuel saving;
- reduction of fossil fuel transportation burden;
- reduction of environmental impact in the heating district because of the reduction of waste products released from fossil fuel burning;
- direct economical benefit due to cheap nuclear heat;
- improvement of heat production industrial culture.

D. Basic safety features

The high safety level of the NHP RUTA-20 arises mainly from two dominant features of the pool-type reactor operating at atmospheric pressure:

- possibility for reactor power self-control in the case of reactor control errors or under accident conditions, independent of control and protection system rod positioning; this is due to a high negative water density coefficient of reactivity, and the selection of reactor normal operation parameters at the boundary of the start of intensive steam generation in the core;

- impossibility of a severe loss of core coolant accident, even if the reactor tank is leaking, due to the large amount of water stored in the reactor tank proper (more than 240 $m^3$), and the natural losses of heat from the tank walls and the metal surfaces of the secondary circuit. All this ensures the emergency core cooling and decay heat removal from the reactor for an unlimited period of time, and makes it possible to eliminate the need for additional engineered systems.

The use of double clad fully tested fuel pins under conditions where fuel pin overheating is technologically excluded, ensures long-term and reliable localization of fission products inside the fuel rods.

The activity released during normal operating conditions is within the limits of natural background radiation, and in the case of severe accidents, the additional equivalent dose does not exceed the radiation dose from the natural background radiation.

### 7.9.2. Design description

#### *7.9.2.1. Reactor plant*

A. Primary circuit

- Reactor tank:

The reactor tank is a cylindrical vessel of 15 m height and 4.8 m inner diameter, manufactured from stainless steel. The reactor tank wall thickness is 20 mm. Inside the tank a reactor coolant flow guiding structure of 1.6 m diameter is installed, with a lower inlet chamber at the bottom below the core and connections to hot legs at the top. In the upper part of the reactor tank, primary heat-exchangers are installed and fixed on a support frame on which there are also drums for spent fuel storage. Water circulation in the reactor tank is natural without boiling. The reactor tank is installed in a concrete cavity lined with carbon steel, 20 mm in thickness. The inner diameter of the reactor cavity is 5.2 m, and its height is 16 m. The reactor pool is covered by a lid sealed with a hydroseal.

- Core:

The reactor core consists of 61 hexagonal FAs whose casing is made out of zirconium with 1 mm wall thickness. The width across flats of the FA is 145.5 mm. Each FA has 61 cells for fuel rods. The outer diameter of the cladding is 13.6 mm and its wall thickness is 0.965 mm. Inside the first cladding there is a second zirconium cladding with an outer diameter of 10.3 mm and a 0.75 mm wall thickness. The gap between the claddings is filled with silumin. The fuel in the form of $U0_2$ pellets, with 4 % enrichment is located in the internal cladding. Some fuel rods in the FA are replaced by fixed absorber rods with burnable poison (gadolinium). In some FAs, there are mobile elements (clusters) containing boron carbide. The core is designed for an operating period of 1850 effective days. The maximum reactivity margin is 2%, in a fresh core, and the average fuel burn-up is 23000 MWd/t to uranium. At full power, the power coefficient amounts to minus $1.1*10^{-4}$ per % of power change, and the reactivity void coefficient to minus $2.5*10^{-3}$%.

- Control and Protection System (CPS);

The CPS is designed to start up the reactor, sustain the specified power level, compensate reactivity changes, and shut down the reactor under emergency conditions.

The CPS systems include:

- neutron flux sensing ionization chambers;
- reactivity control units (cluster drives) with indications of control- rod position;
- electronic and logic devices.

The CPS also ensures:

- continuous control and recording of neutron flux/power (starting with a subcritical condition);
- monitoring of the power changes;
- emergency and alarm signalling.

The reactor shut down system (RSS) is a constituent part of the CPS and is designed in such a way that a CPS failure does not affect operability of the RSS. An important feature of the reliability of the RSS actuation is the fact that apart from 6 RSS clusters all 19 clusters of the CPS fall to their ultimate lowest position under gravity.

- Primary heat exchangers;

Primary plate-type heat exchangers are composed of standard stainless steel plates with 1 mm wall thickness forming a welded 7 m high structure including two sequential stacks. Each stack contains 503 plates. The thermal power of the heat exchanger under the given reactor parameters is 7 MW. Three heat exchangers, identical in design, are installed in the reactor tank.

- Spent fuel handling and storage method;

On the supporting frame of the primary heat exchangers inside the reactor tank there are three local spent fuel stores (LSFS)with positioning holes for used FAs. The capacity of the three LSFS is 61 FAs. During the refuelling procedure, all FAs from the core are placed into temporary drum-type stores in the reactor, the LSFS. After storage for 2-3 years, the spent FAs are removed from the temporary store and placed into a transport container which brings the spent fuel to a regional accumulator-store (temporary repository), and afterwards to the reprocessing facility.

B. Secondary circuit

- Secondary heat exchangers and circulation system;

Coolant circulation in the secondary circuit is natural. For one reactor it is envisaged to have three loops, each of which includes a primary heat exchanger, circulation pipes and a secondary heat exchanger. The level difference between the places of location of the primary and secondary heat exchangers is 25 m.

The secondary heat exchanger is contained in a stainless steel cylinder vessel. The air pressure above the water level is 0.39 MPa. Below the eater level, the plates which are similar in design to the primary heat exchanger, are placed below the water level. The secondary heat exchanger, apart from the function of heat supply to the heating grid, performs the function of a compensator for thermal expansion of the secondary circuit. The diameter of the connecting pipes in the second circuit is 200 mm.

C. Reactor safety system

- Diverse Reactivity Control System (DRCS);

The design foresees an independent direct action DRCS in which absorber balls, stored in an ampule at the top of the fuel assembly, can be dropped down a control tube in each FA. If the FA outlet temperature increases to 120°C, a special local isolating device in the ampule opens and the balls drop into the central tube (i.e. in the core). However, the real necessity for including the DRCS in the RUTA reactor is presently being studied, since the reactor safety can be ensured by the inherent and passive features without the DRCS.

- Passive Decay Heat Removal System (PDHRS);

The reactor has natural coolant circulation in the primary circuit under all modes of operation. Before returning to the reactor, the secondary coolant passes through a system of parallel finned tubes which form the heat exchanger to atmosphere of the PDHRS. To ensure adequate heat transfer from these tubes, they are contained in a vertical duct designed to enhance the air flow. However such heat transfer is undesirable in normal operation so the entry to and exit from the tube region of the duct are closed by a system of louvres. When the temperature in the chamber between the louvres rises to 80 - 90C, direct actioning devices (thermostatic or based on a 'shape-memory material') will actuate to open the louvres and initiate the PDHRS action to cool the reactor.

The secondary circuit pressure is higher than that in the primary circuit and the secondary water is not radioactive. This allows positioning of most of the secondary circuit, including its hear exchanger and the PDHRS convector in normal unsealed compartments.

- Passive Containment Cooling and Steam Condensation System (PCCSCS);

In the event of failure of all the heat removal loops of the secondary circuit, the primary reactor water heats up and starts to evaporate. The heat-up time depends on conditions in the reactor and in the environment and ranges from several hours up to several days. Evaporated primary water passes to the air space of the reactor hall and condenses on the inner surfaces of the containment which conducts heat to the environment. There is also a quite substantial heat loss into the ground surrounding the reactor aided by the large surface area of the reactor tank. Devices to assist condensation in the leak tight reactor hall are envisaged, as is return of the condensate to the reactor.

D. Reactor Auxiliary Systems

- Reactor gas ventilation and purge system;

The reactor gas ventilation and purage system is designed to restrict radioactive release into the reactor plant compartments and to exclude the possibility of generation of dangerous concentrations of hydrogen in the space above the water surface in the reactor pool. The components of the gas system includes:

- hydrogen recombiners;
- blowers;
- condenser for air dryout;
- aerosol and iodine filters.

The system is active. It ensures that the hydrogen concentration does not exceed 0.2% per volume. Some small amounts of air in the gas system are released through the vent stack equipped with a filter and sensor system; this arrangement allows continuous monitoring of reactor fuel pin integrity.

- Reactor pool water purification system;

To ensure the water quality of the primary and secondary circuit water, a purification system is deployed. In the primary circuit the system consists of pumps and ion-exchange filters. The maximum temperature of the water intended for purification is up to 80°C. The secondary circuit does not have a permanently acting purification system. Once every half year, partial replacement of the secondary circuit water is performed. The water is replaced by makeup water from the fresh water storage tanks. Contaminated water from the secondary circuit is later forwarded through the evaporator to tanks as pure condensate.

- Radioactive waste storage and their quantities;

It is envisaged that all radioactive wastes (RW) of the NHP will be transported to a Central Regional Repository for subsequent processing.

Amounts of Liquid RW: 50 $m^3$/year, activity - 1 x $10^{-6}$ Ci/liter.
Amounts of Solid RW: 4.6 $m^3$/year
of which:

| | |
|---|---|
| 3.4 | $m^3$/year, low radioactive |
| 1.0 | $m^3$/year, medium radioactive |
| 0.2 | $m^3$/year, high radioactive |

### *7.9.2.2. Balance of plant systems - Usage of the reactor for heat supply systems and for other purposes*

- To ensure requirements of heating grid, heating station, network pumps;

An NHP with two RUTA reactors (2 x 20 MW) is designed to replace the base heating load of a group of boilers which can then be used as sources of heat for peak load conditions. The NHP supplies hot water at a temperature of 88/60°C under maximum load. To boost the hot water supply during peak winter months, the thermal power from the boiler operating on fossil fuel is used. The rated power of the boiler amounts to 30 MW. The water flow rate in the heat supply system of the NHP is 167.4 kg/s. Pressure in the network pumps is within the range 0.6 to 2.5MPa depending upon local conditions. The pumps may be installed either at the NHP or in a district heating station so as to be compatible with any particular district heating network. The NHP and the boiler ensure heat and hot-water supply for a district with a population as high as 25.000 people.

- Possibility of air-conditioning in a heating district;

The project foresees installation on an NHP of an absorptive refrigerating machine for cooling water down to 7°C which can be used for air-conditioning needs in a heating district during summer time. The capacity of the refrigerating machine is up to 10 MW.

- Desalination of seawater;

The reactor plant can also be used as the power source for sea water desalination. The capacity of a desalination facility is up to 315 t/h of fresh water per reactor. A standard desalination facility is used.

*7.9.2.3. Instrumentation, Control and Electrical Systems - Computer-aided I&C system*

The computer-aided I&C system is designed to control the NHP heat production process, and to provide automation of operation and limitation of possible mistakes by the operating personnel.

The computer-aided I&C system performs the following functions:

- NHP equipment control;
- automatic startup of the protection systems;
- radiation safety monitoring;
- assurance of reliable electric power supply;
- information interaction assurance;
- recording of engineering and production data;
- monitoring and control of fire safety;
- information support of configuration issues and engineering processes;
- physical security of the NHP.

*7.9.2.4. Safety consideration and emergency protection*

Enhanced safety and reliability of the NHP with RUTA reactors is ensured by the following:

- development and utilization of the inherent safety characteristics;
- implementation of passive safety features;
- realization of the defence in depth principle with multi-barrier protection against radioactivity release into the environment;
- safety oriented selection of the reactor design features, performance and layout.

The inherent and passive safety features of the reactor are based on:

- Negative temperature (of coolant and fuel), power and water density reactivity coefficients over the whole range of parameter changes.
- Large specific (per unit of power) heat capacity.
- Natural primary and secondary coolant convection under all modes of operation.

Safety functions fulfilled by passive systems:

- Reactor scram is ensured by gravity driven insertion of absorber rods into the core.
- Decay heat removal is ensured by passive systems under natural primary and secondary coolant convection. Atmospheric air and/or ambient ground is used as an ultimate heat sink.

No emergency coolant injection system is needed in case of primary circuit rupture. Under conditions of rupture in any place or of any size, cooling of the reactor core and the primary heat exchangers is ensured by placing the reactor tank in a leak-tight cavity (thus performing the function of redundant casing).

In the event of a rupture of heat exchange surface there is no leakage from the reactor tank, because the pressure in the primary circuit is lower than in the secondary one.

Barriers to the release of radioactivity:

- Fuel matrix ($UO_2$) whose temperature under the normal conditions does not exceed 630°C.
- Double zirconium fuel rod cladding and silumin material filling the clearance space between them.
- Leak tight reactor tank with a closed system of ventilation of the gaseous space above the water level in the reactor tank.
- Leak tight reactor cavity and containment where all primary circuit equipment is located.
- Exclusion of radioactivity release to the network water is achieved by the use of an intermediate circuit with the following circuit pressure distribution:

$$(P_1 = 0.1 \text{ MPa}) < (P_2 = 0.4 \text{ MPa}) < (P_3 > 0.6 \text{ MPa})$$

where $P_1$ is the primary pressure, $P_2$ is the secondary (intermediate) pressure and $P_3$ is the pressure in the district heating network.

The following features contribute to enhancing the RUTA safety:

- integral layout of the reactor without excessive pressure in the primary circuit;
- utilization of burnable absorbers to reduce excess reactivity.
- presence of additional reactivity compensation system;
- low power density and parameters (temperature, pressure) of the coolant in the circuits:
- three-loop arrangement of heat removal from the reactor;
- presence of containment or underground location of NHP.

Safety analysis

A. Design basis accidents

The design basis accident considered for the RUTA system, are the following:

- unauthorized withdrawal of absorber rods;
- depressurization of the primary circuit by a circuit break in any position;
- blackout, loss of power supply.

In these accidents the reactor conditions do not exceed permissible limits.

B. Beyond design basis accidents

- Unauthorized withdrawal of absorber rods with a worth of 0.63% accompanied by failure of all reactivity reduction systems. The reactor increases its power, and then spontaneously reduces it to the power level of current heat removal at the expense of coolant temperature and steam reactivity effects. Maximum parameter values: power - 1.9 $N_{nom}$, fuel temperature 900°C.

- Blackout accompanied by the failure of all loops of the heat removal systems. The slow heat up of the plant does not result in boiling in the reactor. Heat is accumulated in the coolant (where temperature rises) and is then released to the environment (ground air). After 13 days the heat dissipation capability is equal to the decay heat release (43 kW), the coolant attains its maximum temperature (< 100°C) and then it drops..

### *7.9.2.5. Building and engineering structures on the NHP site*

A. General planning and basic engineering structures at the NHP site

The project foresees an underground location of the reactor plant and a ground level location of supplementary engineering services.

The underground part includes all the process systems of the reactor.

- reactors with the Central Hall;
- gas system and primary circuit purification system;
- compartments of the secondary circuit with heat exchangers
- filters and ventilation system of the primary compartments.

The ground level part includes:

- administrative building, inlet-outlet of ventilation, blowers, control panel, diesels and batteries, network pumps;
- refuelling platform with crane bridge;
- ventilation stack.

The volumes of the underground and the ground level structures are 35000 $m^3$ and 16000 $m^3$, respectively.

B. Seismic stability

All the equipment and engineering structures of the NHP are designed for a design basis earthquake with an intensity 8 on the MSK-64 scale.

### 7.9.3. Safety Concept

TABLE 7.9.1. MAIN SAFETY RELATED SYSTEMS IN THE RUTA NHP

| Name | Safety | Main characteristics |
|---|---|---|
| Primary Circuit (PC) | | Reactor pool: 3 built-in heat exchangers |
| Control and protection system (CPS) | X | 13 control rod cluster assemblies and 6 shut-down rod cluster assemblies |
| Diverse reactivity control system (DRCS) | X | Independent, ampule-type in each FA central tube |
| Passive decay heat removal system (PDHRS) | X | 3 reactor heat exhangers; 3 secondary loops cooled by air |
| Guard vessel (GV) | X | Reactor cavity with steel liner excludes core dry-out |
| Containment (CONT) | X | External impact protection |
| Passive Containment Cooling and Steam Condensation system (PCCSCS) | | Steel plates for steam condensation and heat transfer to the air (or ground) in severe accident |

TABLE 7.9.2. MAIN ACCIDENT INITIATORS FOR THE RUTA NHP

- LOCA (primary): Loss of Primary Coolant Accident
- LOCA (secondary): Secondary Pipe Rupture (Water)
- ATWS: Anticipated Transients Without Scram
- Primary Transients (control rod assembly withdrawal)
- Secondary Transients (one or two loops failure)
- Loss of electric sources (partial)
- Total loss of heat sink
- Station blackout

TABLE 7.9.3. DESIGN FEATURES FOR PREVENTION AND PROTECTION LEVELS THAT REDUCE (R), SUPPRESS (S), THE INITIATORS FREQUENCY OR LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES (L)

**PREVENTION LEVEL**

**LOCA** (Primary):
- Reduced reactor tank fluence: R
- Primary circuit integration: R and L
- Pump absence: S
- Low coolant temperature (100°C) and pressure (0.1 MPa): R&S

**LOCA** (Secondary):
- Pump absence: S
- Low coolant temperature (95°C) and pressure (0.4 MPa): R&S
- Absence of temperature control valves: S

Primary transients:
- Low speed control rod drivers: L
- Low reactivity margin of control rod assembly (< 0.6%): L
- High reliability of CPS and computer reactor control: R
- Low initial heat flux level and fuel temperature: R
- Pump absence: S

Secondary transients:
- Three loops of secondary circuit: L
- Pump absence: S

Loss of electric sources:
- Natural primary and secondary coolant circulation: S
- Reserved diesels: L

**PROTECTION LEVEL**

LOCA (Primary):
- Safeguard tank: L
- Increased water volume in reactor: L
- Tight reactor hall: L

LOCA (Secondary): Not critical*
ATWS: Strong void fraction negative coefficient: - L & S
Primary transients: Grate primary inertial: - L
Natural convection: - L
Secondary transients: Natural convention: - L
Loss of electric sources: Battery power: - R
Total loss of heat sink: Passive DHR to the air and ground: - S
Station blackout: Not critical*

NB: The natural convection in the primary and secondary circuits is a favourable common factor for all the abnormal situations.

* The passive DHR and large PC intertia contribute to make these situations not significant for short and long term.

TABLE 7.9.4. DESIGN FEATURES FOR MITIGATION LEVEL OF THE RUTA NHP

| Safety functions | Systems (Tab. 1) | Passive (P) /Active (A) | Design Features/remarks |
|---|---|---|---|
| **Design Basis**<br>Fission product containment | Double clad/PC/GV+Cont | P/P/P | Double clad fuel pins with low power density ensure good safety barrier |
| Coolant inventory | PC/GV/PCCSCS | P/P/P | External water injection no needed at all |
| Decay heat removal | PC/PDHRS<br>PC/PCCSCS | P/P<br>P/P | Totally passive |
| Reactivity control | CPS<br>DRCS | A<br>P | Low speed drivers |
| Primary circuit pressure control | -- | -- | No need |
| **Severe accident**[*]<br>Containment temperature and pressure control | PCCSCS | P | No need |
| Heat removal | PC/PDHRS | P/P | Loss of heat to the ground from tank |
| Tightness control | Cont. | P | |
| Inflame gas control | Cont. | P | |
| Fission product containment | | | |
| Corium management | | | |
| Others | | | |

[*] - no core degradation considered to be probable for RUTA. In this connection Severe Accidents mean Beyond Design Basis Accidents with manifold Safety and Reactor System Failure

## 7.9.4. Design data questionnaire (for Water-Cooled Reactors)

### I. GENERAL INFORMATION

1. Design name: RUTA NHP
2. Designer/Supplier address: RDIPE, Moscow, Russia
3. Reactor type: Pool type Number of modules/per plant: 2
4. Gross thermal power (MW-th) per reactor: 20
5. Net electrical output (MW-e) per reactor: --
6. Heat supply capacity (MW-th): 20

### II. BASIC DESIGN DESCRIPTION

### A. CORE AND REACTIVITY CONTROL

7. Fuel material: $UO_2$
8. Fuel inventory (tones of heavy metal): 1.9
9. Average core power density (kW/liter): 16.8
10. Average fuel power density (kW/kgU): 10.5
11. Maximum linear power (W/m): 11200
12. Average discharge burnup (MWd/t): 23000
13. Initial enrichment or enrichment range (Wt%): 4
14. Reload enrichment at the equilibrium (Wt%): 4
15. Refuelling frequency (months): 60 eff. months
16. Type of refuelling (on/off power): Off power
17. Fraction of core withdrawn (%): 100
18. Moderator material and inventory: $H_20$
19. Active core height (m): 1
20. Core diameter (m): 1,23
21. Number of fuel assemblies: 61
22. Number of fuel rods per assembly: 61,57,54
23. Rod array in assembly: 61
24. Clad material: Double zirconium alloy cladding
25. Clad thickness (mm): External: 0,965, Internal: 0,75
26. Number of control rods or assemblies: 19
27. Type: Rod cluster control
28. Additional shutdown systems:
    Special, shape memory material basis
29. Control rod neutron absorber material: $B_4C$
30. Soluble neutron absorber: --
31. Burnable poison material and form: Gd; rod

### B. REACTOR COOLANT SYSTEM

**B1. Coolant**

32. Coolant medium and inventory: Water
33. Design coolant mass flow through core (kg/s): 136,4
34. Cooling mode (forced/natural): Natural
35. Operating coolant pressure (bar): Atmospheric
36. Core inlet temperature (°C): 60
37. Core outlet temperature (°C): 95

**B2. Reactor pressure vessel**

38. Overall length of assembled vessel/tube (m): 15
39. Inside vessel/diameter (m/mm): 4.8 m
40. Average vessel/tube thickness (mm): 20
41. Vessel/tube material: Stainless steel
42. Lining material: --
43. Design pressure (bar): Atmospheric
44. Gross weight (tone/kg): 40 tone

**B3. Steam generator: Absent**

45. Number of steam generators
46. Type
47. Configuration (horizontal/vertical)
48. Tube material
49. Shell material
50. Heat transfer surface per steam generator ($m^2$)
51. Thermal capacity per steam generator (MW)
52. Feed water pressure (bar)

53. Feed water temperature (°C)
54. Steam pressure (bar)
55. Steam temperature (°C)

**B4. Pressurizer: Absent**

56. Pressurizer total volume ($m^3$)
57. Steam volume (full power/zero power, $m^3$)

**B5. Main coolant pumps: Absent**

58. Number of cooling or recirculation pumps
59. Type
60. Pump mass flow rate (kg/s)
61. Pump design rated head
62. Pump nominal power (kW)
63. Mechanical inertia (kg $m^2$)

## C. CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS)

64. Number of extraction lines: 1
65. Number of pumps: 2
66. Number of injection points: 1
67. Feed and bleed connections: 1/1 inside system

## D. CONTAINMENT

68. Type: Underground cavity
69. Overall form (spherical/cyl.): Square
70. Structural material: Basalt + concrete
71. Liner material: Steel
72. Simple/double wall: Simple
73. Dimensions (diameter, height) (m): 12 X 12 X 23
74. Design pressure (bar): Atmospheric
75. Design temperature (°C): 100
76. Design leakage rate (% per day): 1% per day

# III. SAFETY RELATED SYSTEMS

## A. DESIGN CONDITIONS

**A1. Fission product retention**

77. Containment spray system (Y/N): No
    a. Duration (h): --
    b. Flow rate ($m^3$/h): --
    c. Mode of operation (active/passive): --
    d. Safety graded (Y/N): --
78. F.P. sparging (Y/N): No
79. Containment tightness control (Y/N): Yes
80. Leakage recovery (Y/N): No
81. Guard vessel (Y/N): Yes

**A2. Reactivity control**

82. Absorber injection system (Y/N): No
    a. Absorber material: --
    b. Mode of operation (active/passive): --
    c. Redundancy: --
    d. Safety graded: --
83. Control rods (Y/N): Yes
    a. Maximum control rod worth (pcm): 9.49%
    b. Mode of operation (active/passive): active
    c. Redundancy: 4.2%
    d. Safety graded: Yes

**A3. Decay heat removal**

*A3-1 Primary side*

84. Water injection: **Absent**
    a. Actuation mode (manual/automatic)
    b. Injection pressure level (bar)
    c. Flow rate (kg/s)
    d. Mode of operation (active/passive)
    e. Redundancy

f. Safety graded (Y/N)

85. Water recirculation and heat removal
   a. Intermediate heat sink (or heat exchanger): heat exchanger + heat sink to the ground
   b. Mode of operation (active/passive): Passive
   c. Redundancy: --
   d. Self sufficiency (h): Not restricted
   e. Safety graded: No

*A3-2 Secondary Side*

86. Feed water: **Absent**
   a. Actuation mode (manual/automatic)
   b. Flow rate (kg/s)
   c. Mode of operation (active/passive)
   d. Redundancy
   e. Self sufficiency (h)
   f. Safety graded

87. Water recirculation and heat removal
   a. Ultimate heat sink (cold source): Air
   b. Mode of operation (active/passive): Passive
   c. Redundancy: --
   d. Self sufficiency (h): not restricted
   e. Safety graded: Yes

*A3-3 Primary pressure control*

88. Implemented system (Name): Reactor cover gas system
   a. Actuation mode (manual/automatic): Automatic
   b. Side location (primary/secondary circuit): Primary
   c. Maximum depressurization rate (bar/s): --
   d. Safety graded: No

**B. SEVERE ACCIDENT CONDITIONS ***

**B.1 Fission products retention**

89. Containment spray system (Y/N): No
90. F.P. Sparging (Y/N): No
91. Containment tightness control (Y/N): Yes
92. Leakage recovery (Y/N): No
93. Risk of recriticality (Y/N): No

**B.2 Recriticality control: Absent**

94. Encountered design feature
   a. Mode of operation (A/P): --
   b. Safety graded: --

**B.3 Debris confining and cooling**

95. Core debris configuration (core catcher): Absent
96. Debris cooling system (name): absent
   a. Mode of operation (A/P): --
   b. Self sufficiency: --
   c. Safety graded (Y/N): --

**B.4 Long term containment heat removal**

97. Implemented system: PCCSCS (tab.1)
   a. Mode of operation (A/P): P
   b. Self sufficiency (h): not restricted
   c. Safety graded (Y/N): No
98. Intermediate heat sink: --
   a. Self sufficiency (h): --
   b. Safety graded (Y/N): --
99. External coolant recirculation: --
   a. Implemented components: --
   b. Mode of operation (A/P): --
   c. Self sufficiency (h): --
   d. Safety graded (Y/N): --
100. Ultimate heat sink: --
   a. Self sufficiency (h): --
   b. Safety graded (Y/N)

*) All systems must be qualified to operate under the accident conditions.

**B.5 Combustible gas control**

101. Covered range of gas mixture concentration: 0,5 - 4%
102. Modes for the combustible gas control
   a. Containment inertation: No
   b. Gas burning: Yes
   c. Gas recombining: Yes
   d. Others: No

**B.6 Containment pressure control**

103. Filtered vented containment (Y/N): Yes
   a. Implemented system: Reactor hall ventilation system
   b. Mode of operation (A/P): A
   c. Safety graded: No
104. Pressure suppression system (Y/N): Absent
   a. Implemented system: --
   b. Mode of operation: --
   c. Safety graded (Y/N): --

## C. SAFETY RELATED I&C SYSTEM

Automatic load following (Y/N): Yes
* range (% power): 0 - 100%
* maximum rate (%/min): 0.15

Load rejection without reactor trip (Y/N): Yes
Full Cathode Ray Tubes (CRT) display (Y/N): Yes
Automated start-up procedures (Y/N): No
Automated normal shutdown procedures (Y/N): Yes
Automated off normal shutdown procedures (Y/N): Yes
Use of field buses and smart sensors (Y/N): Yes
Expert systems or artificial intelligence advisors (Y/N): Yes
Protection system backup (Y/N): Yes

## D. EMERGENCY POWER SUPPLY SYSTEM

105. Type (diesel, gas, grid connection): Diesel/batteries
106. Number of trains: 3

## E. AC/DC SUPPLY SYSTEM

107. Type (rectifier, converter, battery): Battery
108. Estimated time reserve (hr): 72

# IV. CONVENTIONAL THERMAL CYCLE

## A. TURBINE SYSTEM: Absent

109. Type
110. Overall length (m)
111. Width (m)
112. Number of turbines/reactor
113. Number of turbine sections per unit (e.g. HP/LP/LP)
114. Speed (rpm)

## B. STEAM CHARACTERISTICS

115. H.P. inlet pressure
116. H.P. inlet temperature
117. H.P. inlet flowrate
118. L.P. inlet pressure
119. L.P. inlet temperature
120. L.P. inlet flowrate (per section)

## C. GENERATOR

121. Type (3-phase synchronous, DC)
122. Apparent power (MVA)
123. Active power (MW)
124. Frequency (hz)
125. Output voltage (kV)
126. Total generator mass (t)

127. Overall length
128. Stator housing outside diameter

**D. CONDENSER**

129. Number of tubes
130. Heat transfer area
131. Flowrate ($m^3/s$)
132. Pressure (m/bar)
133. Temperature (°C)

**E. CONDENSATE PUMPS**

134. Number
135. Flowrate
136. Developed head
137. Temperature
138. Pump speed

## 7.9.5. Project status

### *7.9.5.1. Entities involved*

The RUTA-20 development is being carried out by:

- RDIPE, Research and Development Institute of Power Engineering (Moscow, Russia): Reactor plant and main reactor systems design;
- VNIPIET, Design and Engineering Institute (St-Petersburg, Russia): auxiliary systems and architectural design;
- PPI, Institute of Physics and Power Engineering (Obninsk, Kaluga region, Russia): scientific support.

### *7.9.5.2. Design status*

The conceptual design of RUTA NHP was completed in 1992 (D = 1).

### *7.9.5.3. R&D works*

The purpose of the remaining R&D work is to improve the thermohydraulic performance of the reactor, to substantiate the steady-state heat removal modes in the core, and to optimize the heat-exchanger equipment design. A separate purpose of the R&D work is to demonstrate the inherent and passive safety features of the reactor under different emergency conditions.

The simplicity of the reactor design, operation experience with pool-type research reactors and utilization in its construction of proven components (fuel rods, control rods and their driving mechanisms, heat exchangers, tanks, control systems) has allowed the designers to reduce the scope of the Research and Development Work.

Some information is needed on the performance of the "double-walled" fuel pins.

a) Completed research and experimental work.

- Investigation of the true volumetric steam content and heat exchange rate in the reactor core;
- Investigation of inter channel and whole-circuit hydroneutronic stability;
- Investigation of hydrodynamics and heat transfer in the reactor by means of transparent reactor modelling;
- Investigation and long-term in-pile testing of 150 double clad fuel rods of the RUTA dimension and materials with enrichment 4.4, 5, 10 and 17%. A maximum fuel burnup of 50,000 MWd/t of uranium was reached with a maximum linear power of 87,000kW/m. No damage occurred.

b) Required and planned research work

- Investigation of the true volumetric steam content and heat exchange rate in the core under emergency conditions and non-design basis modes of operation;
- Investigation of inter channel and whole-circuit instability and ways of achieving stability under emergency conditions;
- Upgrading of heat-exchangers (reduction of header effect);
- Substantiation of in-reactor compensating and optimizing facilities;
- Verification and licensing of computer codes.

#### *7.9.5.4. Licensing process*

The design licensing process includes 5 major stages, and all of them pass through the process of agreement and licensing in the State Nuclear Inspection of Russia (SNIR, Gosatomnadzor)

- Requirement specifications for the design of the reactor plant, heating plant and major NHP systems and equipments;
- Conceptual design of the reactor plant and NHP;
- Preliminary safety analysis report;
- Basic design of the reactor plant and NHP;
- Detailed design of the NHP.

The first two stages were completed in 1992.

### 7.9.6. Project Economics

An NHP with two RUTA-20 reactor units offers heat at low cost due to the simplicity of design, the modular method of construction and low manning requirements. The main input data used in the cost-benefit calculations are as follows:

- US 1991 dollars;
- NHP construction period 3 years;
- capacity factor - 0.65;
- NHP lifetime - 40 years;
- real interest on capital investment is about 5 %.

The capital cost of an NHP is $ 60 millions. This cost includes installation and commissioning. Underground siting takes about 30% of the total capital costs. The total unit energy cost is 2.334 cent per kWh.

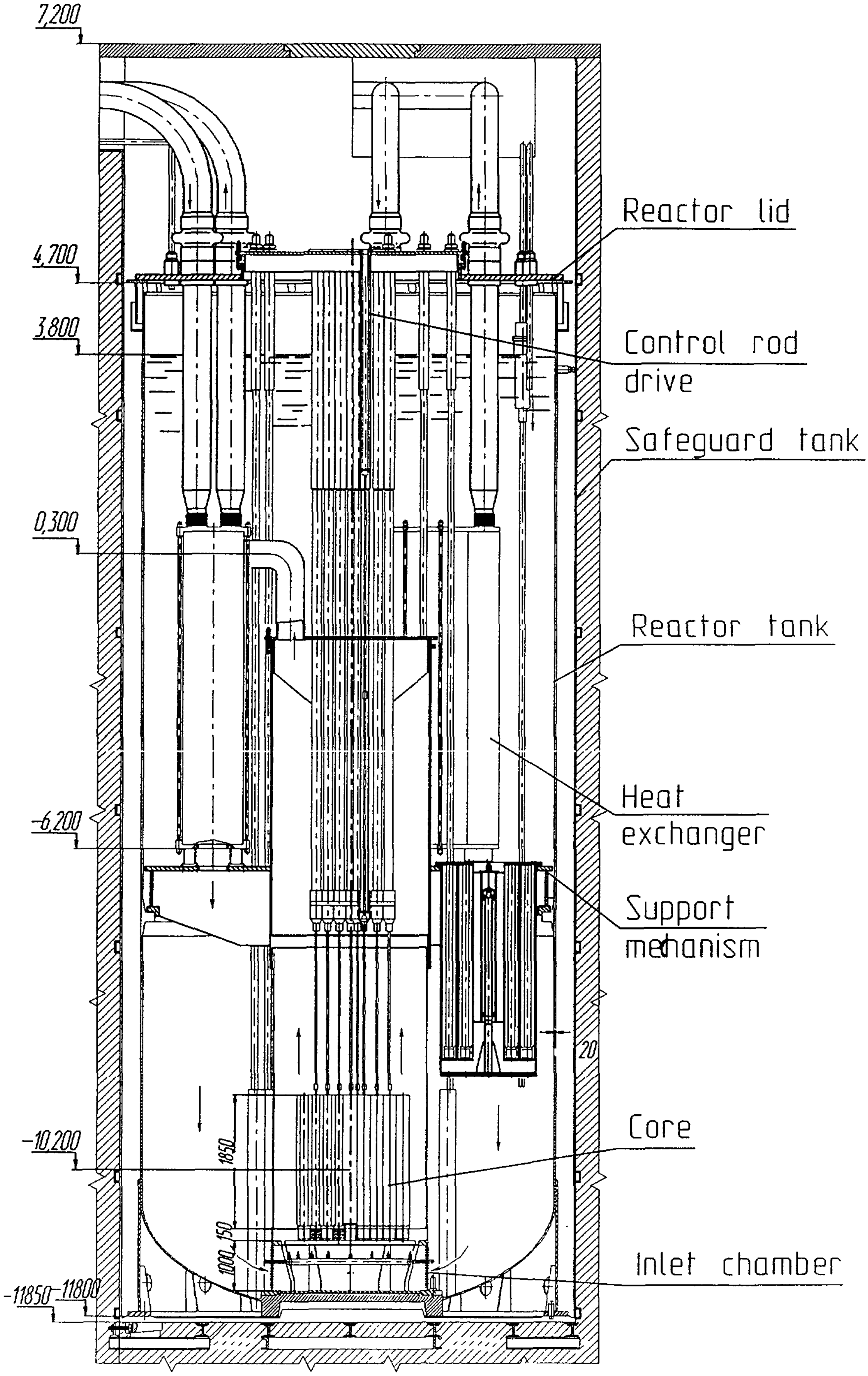



*Fig 7.9.1. Cross-section of RUTA-20 reactor*

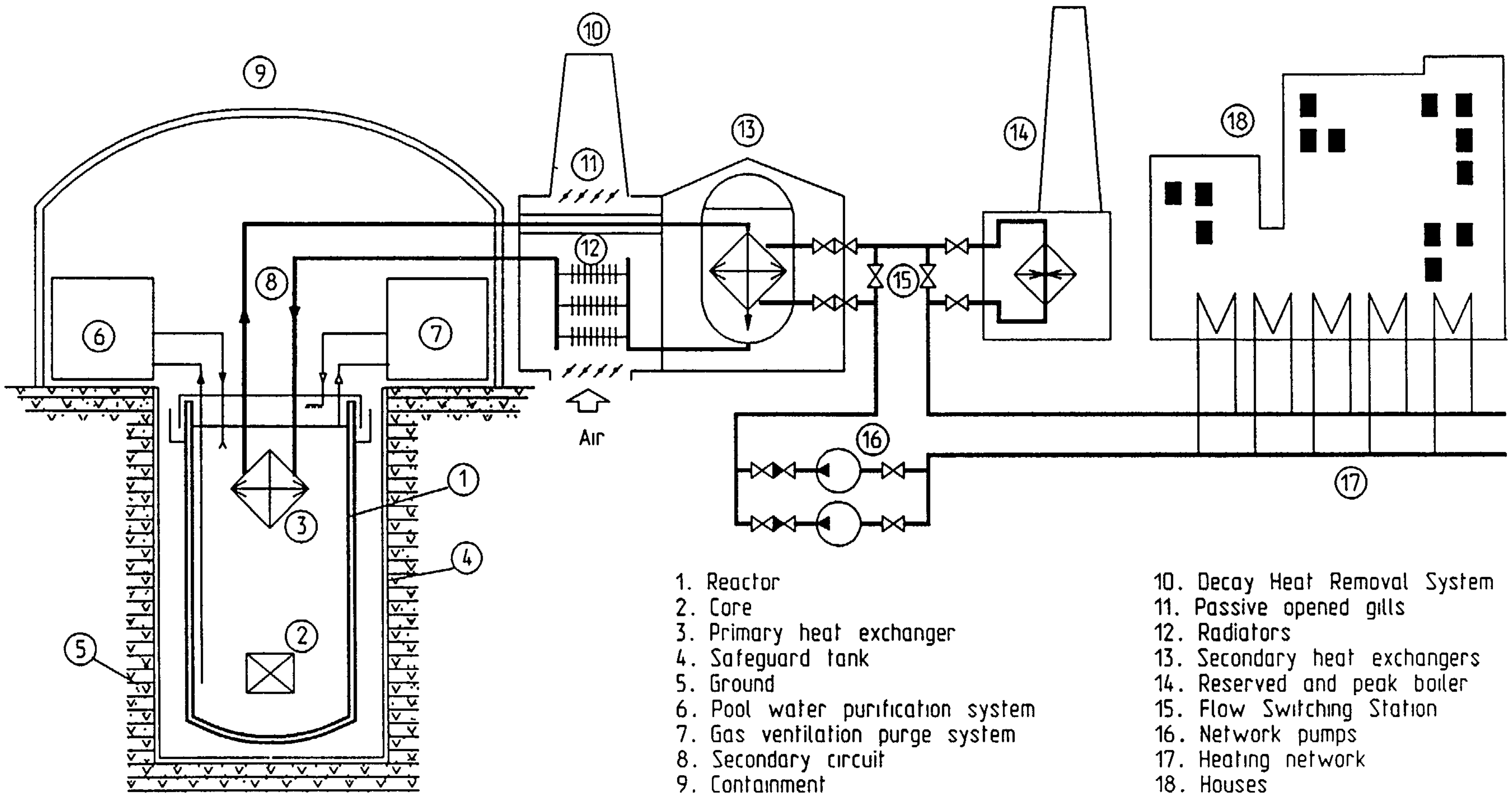



*Fig 7.9.2. RUTA NHP flow diagram*

## 7.10. REACTOR SYSTEM DESCRIPTION AND DEVELOPMENT STATUS OF UNATTENDED LOW POWER NP SAKHA-92

### 7.10.1. Basic objectives and features

The low power SAKHA-92 nuclear power plant is intended for generation of electric energy or combined generation of electricity and heat in independent regions with poor access isolated from power networks. The NPP may be used for sea water desalination as well.

The major design decisions which provide for the high safety level of SAKHA-92 are the following:

- integral reactor in a guard vessel (GV);
- natural coolant circulation in the primary circuit (main circulators are absent);
- leak-tight secondary circuit, designed for the primary circuit pressure;
- self-regulation of reactor power with load following without the need for control rod movement and automatic self shut-down of the reactor in case of accidents. Absence of positive reactivity coefficient during NPP operation;
- passive systems for residual heat removal, no need for electric energy either for operation or for start up;
- asynchronous electric generator, having soft loading characteristic, provides fire safety;
- leak-tight turbogenerator which simplifies the secondary circuit;
- only one fuel load in the core which provides for reactor operation during the entire service life, thus excluding the complexity of refuelling equipment.

### 7.10.2. Design Description

The NPP structure is designed for prefabrication and delivery to the site in the form of three main modules weighing 90 tons each.

*7.10.2.1. Nuclear steam supply system*

The reactor plant (Fig.7.10.1.) uses a self-pressurized integral reactor (Fig.7.10.2.) in the vessel in which the core with the protection system, the heat exchanger and steam-gas pressurizer are located.

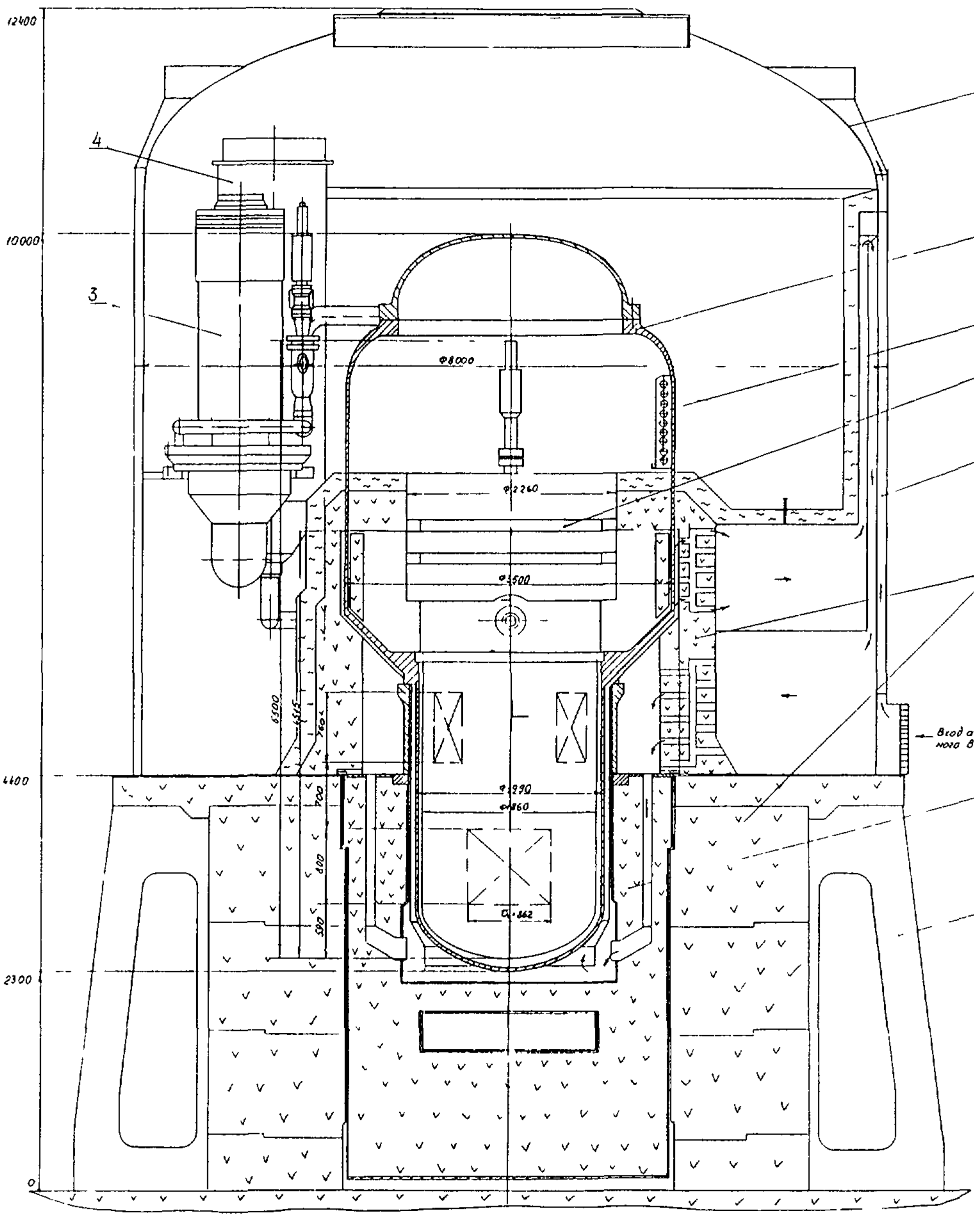
Reactor vessel

The reactor vessel is a prefabricated forged-welded vessel with elliptical bottom of 4230 mm height and 1 860 mm diameter. There are 2 nozzles (ND 80) for steam removal and 2 (ND 40) for feed water supply in the upper part of the vessel.

Reactor core

The core consists of 55 shroudless fuel assemblies (width across flats 101.5 mm). The core height, its diameter and average power density are 800 mm, 860 mm and 17.1 kW/l respectively.

The core is designed for continuous operation during 20-25 years without refuelling the reactor.




1. Reactor (NSSS)
2. Guard vessel
3. Turbogenerator
4. Condenser
5. Containment
6. Heat exchanger
7. Central shielding block
8. Biological shield
9. Guard vessel air coolers
10. Support structures

*Fig 7.10.1.Reactor Plant*

1. CPS drive
2. Shielding unit
3. Reactor head
4. Pressurizer
5. Steam outlet
6. Steam generator
7. Removable unit
8. Reactor vessel
9. Core
10. Feed water inlet

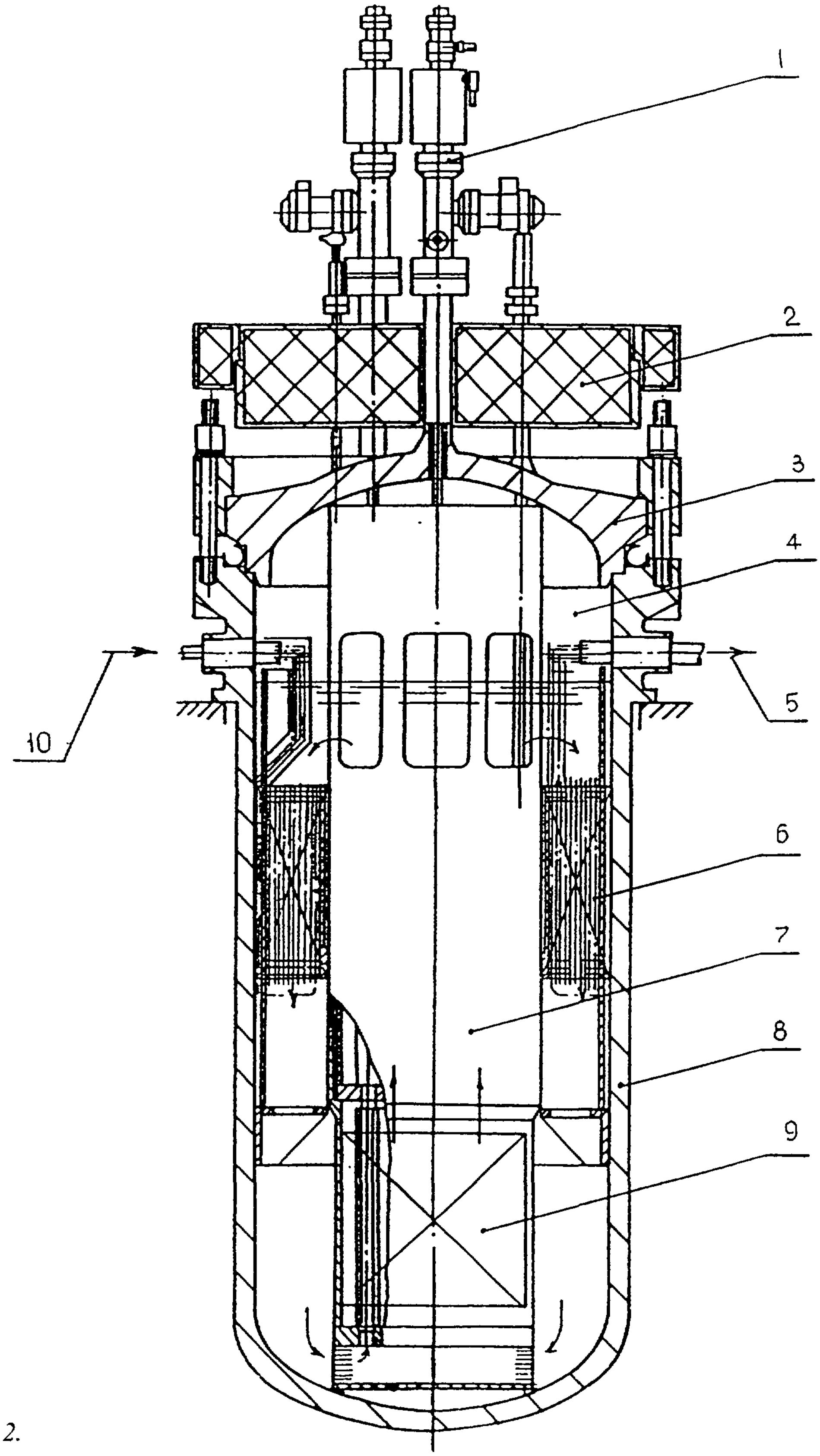



*Fig 7.10.2.*

Steam generator

The SG tube system, which forms a multistart multi-row coil with secondary water inside the tube side, is located in the annular gap between removable internals and the vessel above the core. The tube system consists of 48 tubes. The SG is designed for operation during the entire service life of the NPP without scheduled repair; but its structure is traditionally maintainable with the possibility of plugging each tube without the need to open the primary circuit disclosure.

Secondary circuit

The secondary circuit is intended for primary circuit heat removal, superheated steam generation and feeding the steam to turbine generator.

The main structural feature of the secondary circuit is its leak-tightness. The prototypes of leak-tight condensate and feed pumps were fabricated and tested in OKBM.

*7.10.2.2. Balance of plant system*

Turbogenerator

A special asynchronous vertical turbogenerator is used in the SAKHA-92 NPP. The prototype of the turbogenerator was fabricated and tested in OKBM. Electric output of the turbogenerator is 1 MW.

In the mode of co-generation of electricity and heat the turbogenerator output values are: $N_{el}$=825 KW, $N_{th}$=3000 KW.

*7.10.2.3. Instrumentation, control and electrical systems*

The integrated I&C system is designed for NPP self-control (operator-free) operation under supervision of a remote control centre covering several NPPs of this type in a given region.

The main purpose of the plant control and protection system is to provide for plant start up, data acquisition, accumulation and transmission of telemetric information about the state of the plant and its support systems (diagnostic messages) to the control centre.

*7.10.2.4. Safety considerations and emergency protection*

Enhanced safety is provided primarily by intrinsic self-protective features of the reactor plant which rely upon increased safety margins and passive safety systems.

Residual heat removal in all heat removal accidents is provided by passive reactor vessel cooling system by a leak-tight air circuit. Reactor heat is removed through the guard vessel to a leak-tight containment and then to the environment.

There is a second system of residual heat removal providing heat removal from the core by injection of water from the water storage tank of the district heating circuit. Both systems for emergency heat removal operate by natural circulation.

The design concept of SAKHA-92 practically excludes core melt and significant radiological impact on the environment.

Reactor Shutdown Features

The reactor is normally shutdown by the Control and Protection System (CPS). In emergency situations the reactor is shutdown by control rod drives de-energization.

Means for Confinement of Radioactive Products

The reactor plant represents a single steam generating mono block enclosed in a strong leak-tight guard vessel (GV), providing the confinement of any accident consequences, associated with primary circuit loss-of-integrity accidents. The GV is a steel leak-tight vertically oriented cylindrical vessel of 8 m diameter (Fig.7.10.1).

*7.10.2.5. Buildings and structures*

All equipment of the reactor plant, condensate-feedwater system and electric power system are located inside the containment. The steel leak-tight containment prevents radioactivity release to the atmosphere in case of emergencies. The GV is designed for an internal pressure up to 10 MPa.

The NPP equipment is protected against natural atmospheric phenomena by the reactor building, constructed of prefabricated ferroconcrete.

One of the versions of the NPP design provides for the plant to be sited in permafrost regions. In this case a pile foundation will be used to prevent the permafrost from thawing out.

## 7.10.3. Safety Concepts

TABLE 7.10.1. MAIN SAFETY RELATED SYSTEMS IN THE SAKHA-92

| Name | Safety graded | Main characteristics |
|---|---|---|
| Primary Circuit (PC) | X | All primary components and systems are integrated in reactor pressure vessel |
| Reactor Control and Protection System (CPS) | X | 5 control members for reactivity compensation and emergency protection |
| Alternative Shutdown System | | Injection of liquid absorber (boron solution) |
| Passive alternative heat removal system (AHRS) | X | Heat removal by air, through GV and containment to atmosphere. |
| Passive residual heat removal system (PRHRS) | X | Heat removal by sec.water through SG and main condenser to air tower |
| Guard vessel (GV) | X | Steel leak-tight pressure vessel |
| Containment | X | Steel leak-tight shell |

## TABLE 7.10.2. MAIN ACCIDENT INITIATORS FOR THE SAKHA-92

- LOCA (primary): Loss of Primary Coolant Accident
- LOCA (Secondary) Secondary Pipe Rupture (water or steam)
- LOCA (Interfacing) e g : SGTR Steam Generator Tube Rupture
- ATWS. Anticipated Transients Without Scram,
- Primary Transients,
- Secondary Transients (turbine trip),
- Loss of electric sources (all AC sources),
- Total loss of the cold sources,
- Total loss of the steam generator feedwater,
- Station blackout.

## TABLE 7.10.3. DESIGN FEATURES FOR PREVENTION AND PROTECTION LEVELS THAT REDUCE (R), SUPPRESS (S), THE INITIATORS FREQUENCY OR LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES (L)

**PREVENTION LEVEL**

LOCA (Primary):
- Reduced fluence of the reactor vessel, absence of primary circuit pipelines, absence of links with ventilation systems in the other rooms R,S

LOCA (Secondary):
- The circuit is designed for the primary circuit pressure; cut-off valves in steam lines (for repair) R

LOCA (Interfacing). Reliable design of SG tube system based on well proven technology R

Primary transient:
- Increased design margins, simple reliable structure of NSSS, natural coolant circulation, enhanced power self-control properties of the reactor R

Secondary transients.
- Reliable design of steam turbine R

Loss of electric sources
- Off-site power is not needed for the reactor plant operation S

Total loss of cold source (Water).
- Passive system for residual heat removal by air to the atmosphere

Total loss of the S G feedwater
- Leak-tight sec circuit and related equipment

Station blackout non-critical

**PROTECTION LEVEL**

LOCA (Primary)
- Guard vessel, flow restrictors in small nozzles, two passive systems for residual heat removal L

LOCA (Secondary) Reliable isolation of leaky portion (1/48) of SG by twin isolation valves, L

LOCA (Interfacing) The circuit is designed for the primary circuit pressure L

ATWS
Primary transients Increased design margins and negative reactivity feedbacks, L
Secondary transient Increased design margins, dumping steam into condenser through pressure reduction/cooling device, L

Loss of electric sources Plant's transition to safe state without power consumption is provided, L

Total loss of heat sink non-critical due to heat dissipation into atmosphere
Total loss of S G feedwater non-critical due to passive removal of heat into atmosphere
Station Blackout. non-critical, due to passive safety systems

TABLE 7.10.4. DESIGN FEATURES FOR MITIGATION LEVEL OF SAKHA-92

| Safety functions | Systems (Cf.Tab.7.10.1.) | Passive/ Active | Design features/Remarks |
|---|---|---|---|
| **Design Basis**<br>Fission product containment | PC/GV/Containment | Passive | |
| Coolant inventory | GV | Passive | |
| Decay heat removal | PRHRS/AHRS | Passive | Natural circulation of water/air AHRS is permanently connected |
| Reactivity control | CPS | Active/ Passive | 5 groups of control rods gravity inserted into core |
| Primary circuit pressure control | CPS/PRHRS | Active/ Passive | No safety valves and prim. coolant discharge for overpressure protection |
| **Severe Accidents**<br>Containment temperature and pressure control | CPS/PRHRS | Passive/ Active | |
| Heat Removal | PRHRS/AHRS | Passive | |
| Tightness control | Guard vessel | Passive | |
| Inflam. gas control | | | |
| Fission product containment | GV, Containment | Passive | |
| Corium management | GV | Passive | |
| Others | | | |

## 7.10.4. Design Data Questionnaire
## (Water Cooled Reactors for SAKHA-92)

### I. GENERAL INFORMATION

1 Design name SAKHA-92
2 Designer/Supplier address OKBM
3 Reactor type PWR Number of modules/per plant 1
4 Gross thermal power (MW-th) per reactor 7
5 Net electrical output (MW-e) per reactor Up to 1
6 Heat supply capacity (MW-th) Up to 3

### II. BASIC DESIGN DESCRIPTION

#### A. CORE AND REACTIVITY CONTROL

7 Fuel material Uranium - aluminium alloy
8 Fuel inventory (tones of heavy metal) 77 3 kG U235
9 Average core power density (kW/liter) 17,1
10 Average fuel power density (kW/kgU) 90,5
11 Maximum linear power (W/m) 5966
12 Average discharge burnup (MWd/t)
13 Initial enrichment or enrichment range (Wt%) 30
14 Reload enrichment at the equilibrium (Wt%) not required
15 Refuelling frequency (months) 20-25 years
16 Type of refuelling (on/off power) Without refuelling
17 Fraction of core withdrawn (%)
18 Moderator material and inventory Water
19 Active core height (m) 800 mm
20 Core diameter (m) 860 mm
21 Number of fuel assemblies 55
22 Number of fuel rods per assembly 48
23 Rod array in assembly triangular
24 Clad material Zirconium alloy
25 Clad thickness (mm) 0 65
26 Number of control rods or assemblies 5
27 Type rod cluster
28 Additional shutdown systems Liquid absorber system
29 Control rod neutron absorber material boron and rare earth metals
30 Soluble neutron absorber boron solution
31 Burnable poison material and form rods on gadolinium basis

#### B. REACTOR COOLANT SYSTEM

**B1. Coolant**

32 Coolant medium and inventory Water
33 Design coolant mass flow through core (kg/s)
34 Cooling mode (forced/natural) Natural
35 Operating coolant pressure (bar) 14 MPa (not more)
36 Core inlet temperature (°C) 303,7 (not more)
37 Core outlet temperature (°C) 336

**B2. Reactor pressure vessel**

38 Overall length of assembled vessel/tube (m) 4230 mm
39 Inside vessel/diameter (m/mm) 1860 mm
40 Average vessel/tube thickness (mm) 90
41 Vessel/tube material Thermal-resistant steel
42 Lining material vessel steel with corrosion-resistant facing
43 Design pressure (bar) 155
44 Gross weight (tone/kg) 49000 kG

**B3. Steam generator**
45 Number of steam generators 1 (two sections)
46 Type Once-through, coil
47 Configuration (horizontal/vertical) Vertical
48 Tube materıal Corrosion-resistant metal
49 Shell material
50 Heat transfer surface per steam generator ($m^2$) 100
51 Thermal capacity per steam generator (MW) 7
52 Feed water pressure (bar) 3,83 MPa
53 Feed water temperature (°C) 30/170
54 Steam pressure (bar) 3,2 MPa
55 Steam temperature (°C) 300-335

**B4. Pressurizer**
56 Pressurızer total volume ($m^3$)
57 Steam volume (full power/zero power, $m^3$) 0 5

**B5. Main coolant pumps: Not Relevant**
58 Number of cooling or recirculation pumps
59 Type
60 Pump mass flow rate (kg/s)
61 Pump design rated head
62 Pump nominal power (kW)
63 Mechanical inertia (kg $m^2$)

**C. CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS)**
64 Number of extraction lines No
65 Number of pumps No
66 Number of injection points 1
67 Feed and bleed connections Welded

**D. CONTAINMENT**
68 Type Independent, Steel
69 Overall form (spherical/cyl ) Cylindrical
70 Structural material Steel
71 Liner material
72 Simple/double wall Sımple
73 Dimensions (diameter, height) (m) 8 0, 8 0
74 Design pressure (bar) 2 0
75 Design temperature (°C) 180
76 Design leakage rate (% per day)

**III. SAFETY RELATED SYSTEMS**

**A. DESIGN CONDITIONS**
**A1. Fission product retention**
77 Containment spray system (Y/N) No
 a Duration (h)
 b Flow rate ($m^3$/h)
 c Mode of operation (active/passive)
 d Safety graded (Y/N)
78 F P sparging (Y/N)
79 Containment tightness control (Y/N) Yes
80 Leakage recovery (Y/N)
81 Guard vessel (Y/N) Yes
**A2. Reactivity control**
82 Absorber injection system (Y/N) No
 a Absorber material Boron and rare earth metals
 b Mode of operation (active/passive) Passive

c. Redundancy: Yes
d. Safety graded: No

83. Control rods (Y/N): Yes
   a. Maximum control rod worth (pcm): 20-25% ΔK/k in cold state
   b. Mode of operation (active/passive): Active/Passive
   c. Redundancy: Yes
   d. Safety graded: Yes

**A3. Decay heat removal**

*A3-1 Primary Side*

84. Water injection: No
   a. Actuation mode (manual/automatic)
   b. Injection pressure level (bar)
   c. Flow rate (kg/s)
   d. Mode of operation (active/passive)
   e. Redundancy
   f. Safety graded (Y/N)
85. Water recirculation and heat removal: Heat removal through II circuit
   a. Intermediate heat sink (or heat exchanger): heating grid HX
   b. Mode of operation (active/passive): Passive/active
   c. Redundancy: Yes
   d. Self sufficiency (h): Unlimited
   e. Safety graded: Yes

*A3-2 Secondary side*

86. Feed water
   a. Actuation mode (manual/automatic): Automatic self-insertion
   b. Flow rate (kg/s): 3.08
   c. Mode of operation (active/passive): Passive/Active
   d. Redundancy: Yes
   e. Self sufficiency (h): to final cooling down
   f. Safety graded: Yes

87. Water recirculation and heat removal
   a. Ultimate heat sink (cold source): Atmosphere
   b. Mode of operation (active/passive): Passive
   c. Redundancy: Yes
   d. Self sufficiency (h): Unlimited
   e. Safety graded: Yes

*A3-3 Primary pressure control*

88. Implemented system (Name): CPS/PRHRS
   a. Actuation mode (manual/automatic): automatic/manual
   b. Side location (primary/secondary circuit)
   c. Maximum depressurization rate (bar/s)
   d. Safety graded: Yes

## B. SEVERE ACCIDENT CONDITIONS*

**B.1 Fission products retention**

89. Containment spray system (Y/N): No
90. F.P. Sparging (Y/N): No
91. Containment tightness control (Y/N): Yes
92. Leakage recovery (Y/N): Yes
93. Risk of recriticality (Y/N): No

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* All systems must be qualified to operate under the accident conditions

**B.2 Recriticality control**

94 Encountered design feature boron injection system
- a Mode of operation (A/P) Active
- b Safety graded No

**B.3 Debris confining and cooling**

95 Core debris configuration (core catcher) Guard vessel
96 Debris cooling system (name) PRHRS, AHRS
- a Mode of operation (A/P) Passive
- b Self sufficiency Unlimited
- c Safety graded (Y/N) Yes

**B.4 Long term Guard Vessel heat removal**

97 Implemented system AHRS
- a Mode of operation (A/P) Passive
- b Self sufficiency (h) to final cooling down
- c Safety graded (Y/N) Yes

98 Intermediate heat sink secondary coolant
- a Self sufficiency (h) to final cooling down
- b Safety graded (Y/N) Yes

99 External coolant recirculation
- a Implemented components No
- b Mode of operation (A/P) Passive
- c Self sufficiency (h) to final cooling down
- d Safety graded (Y/N) Yes

100 Ultimate heat sink Atmosphere
- a Self sufficiency (h) Unlimited
- b Safety graded (Y/N) No

**B.5 Combustible gas control**

101 Covered range of gas mixture concentration
102 Modes for the combustible gas control
- a Containment inertation Nitrogen
- b Gas burning
- c Gas recombining
- d Others

**B.6 Containment pressure control**

103 Filtered vented containment (Y/N) Yes
- a Implemented system Water-cooled vent system
- b Mode of operation (A/P) Active
- c Safety graded No

104 Pressure suppression system (Y/N) No
- a Implemented system
- b Mode of operation
- c Safety graded (Y/N)

## C. SAFETY RELATED I&C SYSTEM

Automatic load following (Y/N) Yes
- * range (% power) 20-100
- * maximum rate (%/min)

Load rejection without reactor trip (Y/N) Yes
Full Cathode Ray Tubes (CRT) display (Y/N) Yes
Automated start-up procedures (Y/N) Yes
Automated normal shutdown procedures (Y/N) Yes
Automated off normal shutdown procedures (Y/N) Yes
Use of field buses and smart sensors (Y/N)

Expert systems or artificial intelligence advisors (Y/N): Yes (for central control post)
Protection system backup (Y/N): Yes

**D. EMERGENCY POWER SUPPLY SYSTEM (not required)**

105. Type (diesel, gas, grid connection)
106. Number of trains

**E. AC/DC SUPPLY SYSTEM**

107. Type (rectifier, converter, battery)
108. Estimated time reserve (hr): unlimited

## IV. CONVENTIONAL THERMAL CYCLE

**A. TURBINE SYSTEM**

109. Type: Active
110. Overall length (m): 520 mm
111. Width (m): 840 mm
112. Number of turbines/reactor: 1
113. Number of turbine sections per unit (e.g. HP/LP/LP): 1
114. Speed (rpm): 3000

**B. STEAM CHARACTERISTICS**

115. H.P. inlet pressure: 3,2 MPa
116. H.P. inlet temperature: 300-330°C
117. H.P. inlet flowrate: 11.1 t/h
118. L.P. inlet pressure: No
119. L.P. inlet temperature: No
120. L.P. inlet flowrate (per section): No

**C. GENERATOR**

121. Type (3-phase synchronous, DC): 3-phase asynchronous
122. Apparent power (MVA): 1000 kWA
123. Active power (MW): 1000 kW
124. Frequency (hz): 50
125. Output voltage (kV): 400 V
126. Total generator mass (t): 16,9
127. Overall length: 2,88 m
128. Stator housing outside diameter: 900 mm

**D. CONDENSER**

129. Number of tubes: 133
130. Heat transfer area
131. Flowrate ($m^3/s$)
132. Pressure (m/bar)
133. Temperature (°C)

**E. CONDENSATE PUMPS**

134. Number: 1
135. Flowrate: 11,1 t/h
136. Developed head
137. Temperature: max 170°C
138. Pump speed: 3000 rpm

### 7.10.5. Project Status

The design of SAKHA-92 NPP is being developed by:

- OKB Mechanical Engineering (Nizhny Novgorod) - overall NPP design
- NPO "Avrora" (St.Petersberg) - project of control and protection system using satellite telemetry technology,
- Institute of Physical and Power Engineering (Obninsk, Kaluga region) - scientific supervision.

SAKHA NPP is being developed on the basis of solid experience of construction and operation of nuclear ice-breakers' power plants, on proven prototypes. Further R& D is needed. Licensing status: Preliminary design study.

### 7.10.6. Project economics

Preliminary evaluation of SAKHA-92 NPP economics showed the plant competitiveness with power plants based on diesel-generators, when using them in regions difficult to access (for example in the regions of Far North or deserts of Middle Asia).

## 7.11. MODULAR DOUBLE POOL REACTOR SYSTEM DESCRIPTION AND DEVELOPMENT STATUS (MDPR)

### 7.11.1. Basic objective and features

The objective is to establish a modular reactor concept which can compete with a LWR on construction cost and schedule and can also support advanced safety features to contribute to the early practical application of FBRs.

To achieve competitive construction costs with a LWR, rationalization of the intermediate heat transport system is required. Considering the direct effect of a sodium-water reaction on the core, even if it is BDBA (Beyond Design Base Accident), it is difficult to secure general acceptance of eliminating the secondary heat transport system at an early stage of FBR practical application.

The double pool design concept[(1), (2)] reduces the size of the intermediate heat transport system by installing steam generators and secondary pumps into the sodium filled annular space formed between the primary vessel and the secondary vessel. This concept allows reduction of, i) size and volume of a nuclear island building, ii) electrical equipment and utilities such as heating, ventilating and air conditioning systems, iii) piping support structures, sodium leak monitoring systems, and iv) countermeasure devices for sodium fire. On the other hand, due to this compact arrangement of components, that is, steam generators are located adjacent to the primary reactor vessel, the structural integrity of the primary reactor vessel against any sodium-water reaction event should be assured.

In addition to the rationalization of the reactor module by the double pool system, standardization of design by seismic isolation, reduction of plant site work and shorter construction schedule are considered to compensate the scale-disadvantage of the small reactor.

For safety, an enhancement of inherent safety is expected by the neutronic and thermal characteristics of metallic fuel. Redundancy and diversity of safe shutdown systems are employed, and a long grace period is to be expected because of increased thermal capacity due to the large amount of sodium in both primary and secondary vessels.

## 7.11.2. Design description

### *7.11.2.1. Nuclear steam supply system*

Figures 7.11.1. and 7.11.3. show the schematic diagram of the reactor module. Electric power output of 325 MWe has been estimated, considering the size of the primary vessel, the secondary vessel and cavity wall combination that can be manufactured at a factory and transported to a site. To allow start up of each reactor built in series, each module has its own turbine generator.

1. Heat transport system

The principal components of the reactor system consist of i) the primary vessel which includes the core and its support structures, four IHXs (Intermediate Heat Exchanger), four primary circulating EM (Electro-Magnetic) pumps integrated with cold traps, ii) the secondary vessel which includes four SGs (Steam Generator), four secondary circulating EM pumps, two cold traps, iii) upper internal structure, iv) fuel handling mechanism, v) roof-slab, and vi) short secondary sodium piping.

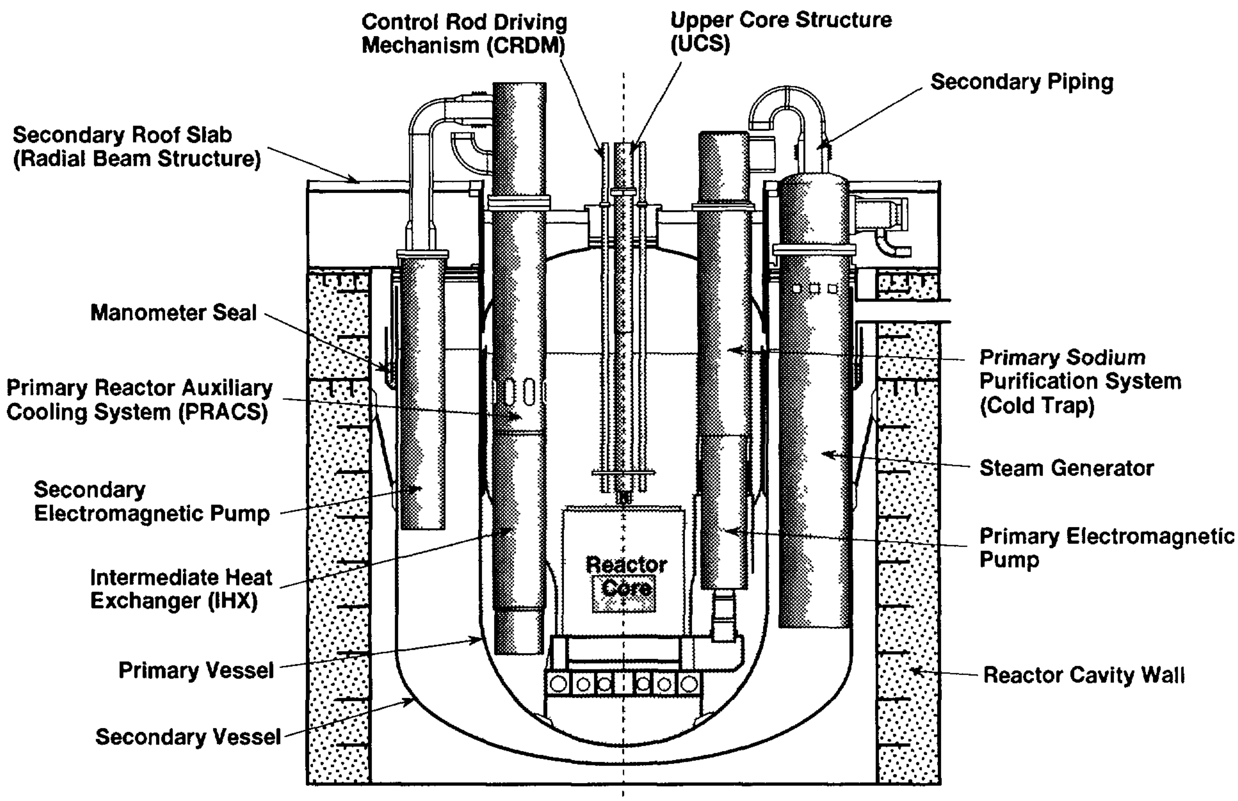



*Fig 7.11.1. Vertical View of Reactor Structure (Modular Double Pool, 325MWe)*

The sodium in the primary vessel flows upward from the core and flows downward in the tube side of the IHX to reach the cold plenum. The sodium in the cold plenum is driven into the core by the primary EM pump. On the other hand, the secondary sodium pressurized by the secondary EM pump enters the IHX through the secondary cold leg piping. The sodium flows upward in the IHX shell side and enters the SG through the secondary hot leg piping. After heat transfer to water/steam, the sodium flows to the secondary EM pump. To keep the secondary vessel in an isothermal condition, a small amount of coolant leaks into the secondary vessel from the upper part of the EM pump.

## 2. Reactor structure

The primary vessel of 316SS is supported from the roof-slab. The bottom of the primary vessel is hemispherical to withstand the external pressure of a sodium-water reaction accident. The secondary vessel of 2 1/4Cr-1Mo steel retains the secondary coolant and provides the cooling flow path for heat exchange between the secondary coolant and the water/steam. This secondary vessel also serves as a safety vessel, if a leakage occurs in the primary vessel. It functions to prevent the sodium level from lowering in the primary vessel. The secondary vessel is supported from the cavity wall. To cope with the thermal expansion of the secondary vessel and to form the cover gas boundary between the secondary cover gas space and cavity space, a manometer seal of low melting point alloy is used. This manometer also works to release the pressure in the case of a beyond design base sodium-water reaction accident. As for the pressure rise after a design base sodium-water reaction accident, four piping systems with rupture disks are used in common with four SGs to release the pressure.

For the roof-slab, a radial beam structure of eight box-beams 2.5m high and 0.5m wide of 38mm thick carbon steel are used based on the results of a 3 dimensional eigenvalue analysis. This number of beams conforms with the arrangement of the SGs and the secondary pumps.

## 3. Core

The core design aims at passive shutdown capability based on the features of metallic fuel and the small-size core. A homogeneous core is used to achieve the compact radial core size which has a marked influence on the vessel size. The core consists of driver assemblies, blanket assemblies, shielding assemblies, control rods and in-vessel storage (for spent fuel). A quarter of the core (drivers and blankets) is changed every two years.

## 4. Reactivity control

Reactivity control for normal operations (burnup compensation and load following) is accomplished by six control rods which constitute two individual shutdown systems (3+3). Each system has its own trip parameters and can independently shutdown the reactor to the fuel changing temperature.

To prevent an ATWS (Anticipated Transient Without Scram), this separated shutdown system, low excess reactivity core and the primary flow coast down control system are adopted. Furthermore, features of the metallic fuel, thermal expansion of control rod drive line, radial thermal expansion of core and large heat capacity are expected to mitigate an ATWS.

5 Main components

A shell and tube type IHX integrated with heat transfer tubes for the decay heat removal system is used Considering their contribution to the simplification of the head access area, sodium immersed EM pumps are employed as both primary and secondary circulating pumps Cold traps are integrated with the primary EM pumps

Once through helical coil type SGs are adopted and they are suspended from the roof slab Mod 9Cr-1Mo steel is used as the material of the major element These SGs have double wall outer shells Between these walls, argon gas is enclosed It works as a thermal insulator between hot sodium in the SGs and cold sodium in the secondary vessel The technical feasibility of a rectangular helical coil type SG is being examined, because it has a high potential to reduce the diameter of the secondary vessel

6 Decay heat removal system

Four primary reactor auxiliary cooling systems (PRACS) are used A cooling coil is installed in the inlet plenum of each IHX and a heat transfer coil is installed in the air cooler of ultimate heat sink Coolant is circulated by EM pumps supported by emergency AC power The air cooler consists of a blower, a stack, vanes and dampers The blower is supported by the emergency AC power The vanes and the dampers are operated by the emergency DC power Decay heat removal by natural circulation is possible to mitigate a total blackout event (loss of all AC power)

*7.11.2.2. Balance of plant systems*

1 Plant layout

Figure 7 11 2 shows the plant layout for a four module constellation producing 1300 MWe As described before, each reactor has its own turbine generator, because considerations are given to the start-up of each module in order and to assure of the independence of each module The control building, the fuel transport equipment, etc , are common

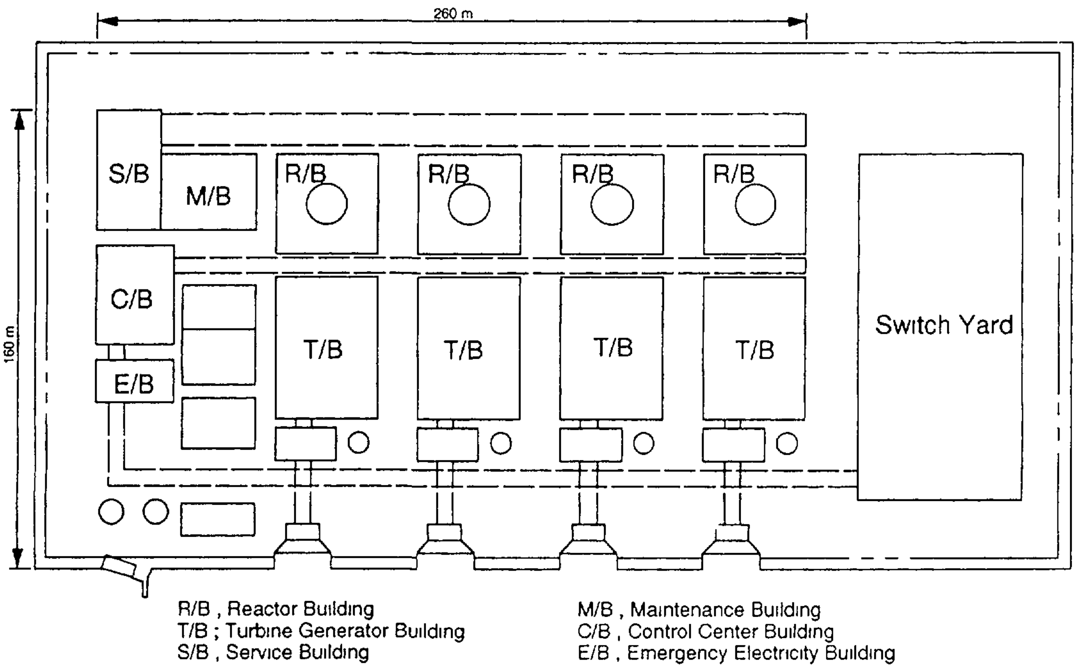



*Fig 7.11.2. Plant Layout*

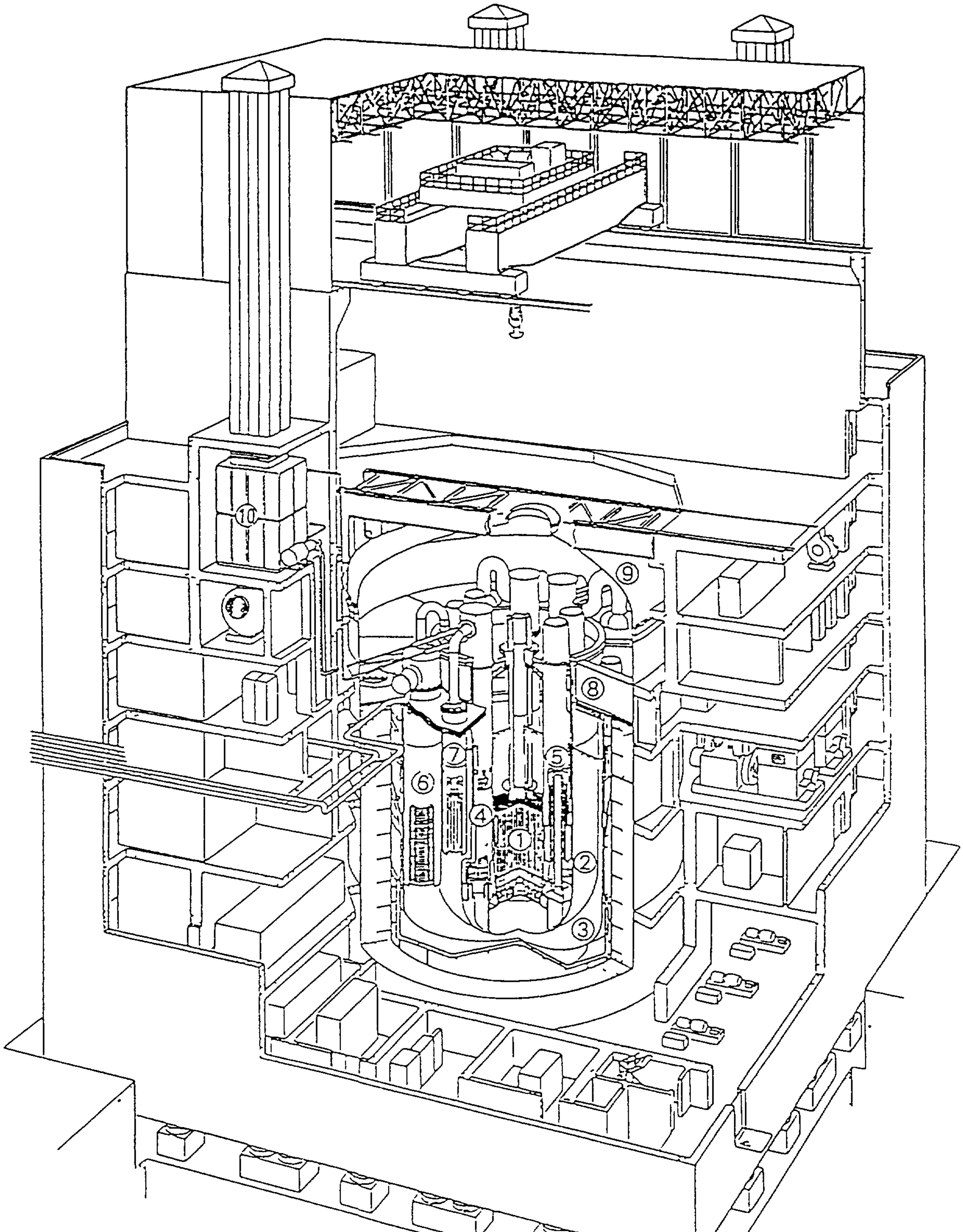



*Fig 7.11.3 Bird's-eye view of Reactor Structure ( Modular Double Pool, 325 MWe)*

2. Turbine building

A turbine building contains a turbine system, an electric generating system, a condenser, a feed water system and connecting piping. All systems are non-nuclear grade. The feed water system has three low pressure feed water heaters and one high pressure feed water heater, and has two main feed water pumps.

*7.11.2.3. Instrumentation, control and electrical systems*

The instrumentation and control system consists of the reactor protection system, engineered safety features operation system, plant control system and reactor monitoring system. The reactor protection system has two diverse shutdown systems. The engineered safety features are the decay heat removal system (PRACS) and the containment system.

The plant control system is hierarchical. The top level control task is to receive grid dispatcher demands and to coordinate load demands on each module. Each module has its module controller, which coordinates the turbine generator controller and local controllers (set points, diagnostics and operator actions). Each module has two emergency AC power units (diesel generator). The plant control system has its own emergency power units (both AC and DC) for power plant control. These power units can provide its power to each module.

*7.11.2.4. Safety considerations and emergency protection*

In the double pool reactor structure, SGs are located adjacent to the primary vessel, which is the primary coolant boundary. If a water leak should occur from the SG tubes, the primary vessel is exposed to the external buckling load. As a result the mitigation system against sodium-water reaction events is different from other FBRs. In a beyond design basis leak event, quasi-steady state pressure of the secondary cover gas is released through the manometer seal structure passively.

*7.11.2.5. Buildings and structures*

A three dimensional seismic isolation system for the reactor building is adopted to achieve the standardization of design. The base size of the building is 40mx40m, and the height is 55m. Shortening the construction schedule on-site is very important to reduce the plant construction costs, thus an optimum combination of steel structure and concrete structure is being considered.

The containment structure consists of a top dome and a reactor cavity. Limit pressure and temperature are summarized in the design data (see 7.11.4).

Core debris is planned to be retained inside the primary vessel (on the core support structure).

The head access area (above the roof slab) and operation floor for fuel handling operations are prohibited for entry in normal operation because of the high radiation dose rate. The other areas can be entered by the use of proper shielding.

### 7.11.3. Safety concepts

*7.11.3.1. Transient analysis*

The primary vessel of the double pool is in the secondary coolant, so some heat from the primary coolant (1% of rated power) can be transferred to the secondary coolant through the primary vessel A combination of vertical and linear structure is being considered for the thermal protection of the primary vessel in normal operation It works as a heat transfer structure under transient conditions because the sodium level difference between the redan and the primary vessel is low in normal operation and rises if the primary pump trips

To evaluate the effect of the secondary sodium surrounding the primary vessel, a preliminary analysis was made for a transient event of total blackout The temperature transient analysis of both the primary hot plenum and the secondary plenum indicate that the hot plenum temperature is lowered by the effect of heat transfer compared to that of the insulated case Therefore it should be noted that the large heat capacity of the cold secondary coolant works as a heat sink

*7.11.3.2. Sodium-water reaction evaluation*

SGs are located adjacent to the primary vessel and are immersed in the secondary sodium If a water leak should occur in SG tubes, the primary vessel is exposed to the external buckling load and all the secondary sodium is contaminated by the sodium-water reaction products Therefore the following prevention and mitigation defense in depth strategy against a sodium-water reaction has to be established

1 Quality control (manufacture and water purification)
2 Fast leak detection by hydrogen meter, acoustic detector, cover gas pressure gauge, rupture sensor and manometer seal level meter
3 Protective action due to isolation and rapid blow-down of water/steam system
4 Fast pressure release by a sodium-water reaction pressure release system for a DBL (Design Basis Leak) event and BDBL (Beyond Design Basis Leak)
5 Purification of the secondary sodium after a sodium-water reaction As the safety design criteria for a DBL, 4DEG (Double Ended Guillotine) breaks are specified based on the result of leak propagation analysis using conservative conditions

The following counter measures are provided to cover the whole range of leak rates

Instruments

1 Hydrogen meters are provided at SG outlet for small leakage, less than 10g/s
2 Acoustic detectors and cover gas pressure gauges are provided for medium leakage, less than 1kg/s
3 Rupture sensors are provided for large leakage, less than DBL
4 Manometer seal level meters are provided as the back-up leak detection system for a BDBL event

Pressure release systems

5 Sodium-water reaction product/pressure release system which consists of 30Bx4 piping and rupture disks provided at the secondary vessel for a DBL event

6 A Manometer seal in addition to the pressure release system is provided at the circumference of the secondary vessel for BDBL event

From the preliminary analysis, the increase of the quasi-steady-state pressure of the cover gas in the secondary vessel is about 0 13 MPa and 0 2MPa for a DBL and a BDBL, respectively These values are much less than the external buckling pressure of the primary vessel (0 56MPa), thus the structural integrity of the primary vessel can be secured

The safety concept is shown in Table 7 11 1 through Table 7 11 4

TABLE 7.11.1. MAIN SAFETY RELATED SYSTEMS IN THE MDP CONCEPT

| Name | Safety graded | Main characteristics |
|---|---|---|
| Primary circuit | x | Primary vessel (reactor vessel) |
| | x* | Four recirculation EMP (25% Flow each) |
| | x* | Four intermediate heat exchange |
| Diverse reactor shutdown system | x | Neutron absorber insertion Group 1 |
| | x | Neutron absorber insertion Group 2 |
| Reactivity control system | -- | Six neutron absorber rods |
| Counter measurement of extra-reactivity insertion | | None |
| Passive and diverse decay heat removal system | x | PRACS |
| | -- | Steam/water system |
| Containment system | x | - Reactor cavity outside the secondary vessel |
| | x | - Top dome at the top of the reactor vessel |
| Plant protection system | x | Individual signal processing system for two reactor shutdown system |
| Water-sodium reaction | -- | Pressure release system |

* Boundary function

TABLE 7.11.2. MAIN ACCIDENT INITIATORS FOR THE MDP

- LOF Loss of flow
- TOP Transient over power
- ATWS Anticipated Transients Without Scram,
- Primary transients,
- Secondary transients (turbine trip),
- Loss of electric sources (partial),
- Total loss of the heat sink,
- Total loss of the steam generator feedwater,
- Station blackout
- Water sodium reaction

TABLE 7.11.3. DESIGN FEATURES FOR PREVENTION AND PROTECTION LEVELS THAT REDUCE (R), SUPPRESS (S), THE INITIATORS FREQUENCY OR LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES (L)

| |
|---|
| **PREVENTION LEVEL** |
| LOF: Reduced vessel fluence : reduce initiator frequency<br>Primary circuit integration : reduce initiator frequency<br>Primary cover gas release : reduce initiator frequency<br>Four primary EMP : reduce failure frequency according to failure mode<br>External electronic power : depends on the grid reliability<br>Plant control system : reduce initiator frequency<br><br>TOP: Core support structure : reduce initiator frequency<br>Six control rods system : reduce initiator frequency<br>Feed water flow control system : reduce initiator frequency<br>Gas entrainment into the core : reduce initiator frequency |
| Primary transients: Increased design margins : reduces frequency |
| Secondary transients: Increase design margins : reduces frequency |
| Loss of electric sources : reduces frequency |
| Total loss of cold source (water): reduces frequency |
| Total loss of the SG feedwater : reduces frequency |
| Station blackout : reduces initiator frequency |
| Water-sodium reaction : Increased design margins: reduces initiator frequency |
| **PROTECTION LEVEL** |
| LOF: Reactor protection system, shutdown system, PRACS<br>Flow coastdown control system, low flow operation system<br>TOP: Reactor protection system, shutdown system, PRACS<br>Mechanical rod stopper (design option) |
| ATWS: Negative temperature coefficient : limit accident consequence<br>Primary flow coastdown control system |
| Primary transients: Limit consequence<br>Flow coastdown control system |
| Secondary transients: Limit consequence |
| Loss of electric sources: Emergency power generator (diesel generator) |
| Total loss of heat sink: PRACS: Large thermal inertia (with scram)<br><br>Total loss of SG feedwater: Negative reactivity feedback by thermal expansion of core structure (without scram)<br><br>Station blackout: Two diesel generators<br>N/C PRACS operation (total blackout) |

TABLE 7.11.4. DESIGN FEATURES FOR MITIGATION LEVEL OF MDP

| Safety functions | Systems (Cf. Table 7.11.1.) | Passive/active | Design features/remarks |
|---|---|---|---|
| **Design Basis**<br>Fission product containment | Cladding<br>Primary vessel (reactor vessel)/IHX tube-roof slab<br>Reactor cavity/Top dome | Passive<br>Passive<br>Passive | |
| Coolant inventory | Secondary vessel | Passive | |
| Decay heat removal | Primary reactor auxiliary cooling system (PRACS)<br>Steam/water system | Active<br>Active | N/C operation available<br>Non-safety grade |
| Reactivity control | Control rod (Group 1)<br>Control rod (Group 2) | Active<br>Active | 3 rods<br>3 rods |
| Primary circuit pressure control | Cover gas pressure control | Active | |
| **Severe accident**<br>Containment temperature and pressure control | Reactor cavity : 270°C, 0.3 MPa<br>Top dome : 150°C, 0.2 MPa | Passive<br>Passive | No pressure control<br>No pressure control |
| Heat removal | PRACS | Active | |
| Tightness control | Isolation valve (penetrating piping at top dome) | Active | |
| Inflam. gas control | Igniter (in sodium-water reaction accident only) | Active | for $H_2$ |
| Fission product containment | Reactor cavity and Top dome<br>Reactor building (confinement) | Passive<br>Passive | |
| Corium management | None | | |
| Others | | | |

## 7.11.4. Design data questionnaire (for Liquid Metal-cooled Reactors)

### I. GENERAL INFORMATION

1 Design name Modular Double Pool
2 Designer/Supplier address Central Research Institute of Electric Power Industry
3 Reactor type LMR number of modules/per plant 4
4 Gross thermal power (MW-th) per reactor 840
5 Net electrical output (MW-e) per reactor 325
6 Heat supply capacity (MW-th)

### II. BASIC DESIGN DESCRIPTION

### A. CORE AND REACTIVITY CONTROL

7 Fuel material U-Pu-Zr
8 Fuel inventory (tones of heavy metal) 25 4
9 Average core power density (kW/liter) 241
10 Average fuel power density (kW/kg) 47 7
11 Maximum linear power (W/m) 39000
12 Average discharge burnup (MWd/t) 140000
13 Initial enrichment or enrichment range (Wt%)
14 Reload enrichment at the equilibrium (Wt%) 13 7
15 Refuelling frequency (months) 24
16 Type of refuelling (on/off power) off power
17 Fraction of core withdrawn (%) 25
18 Moderator material and inventory Not applied
19 Active core height (m) 1 15
20 Core diameter (m) 1 90
21 Number of fuel assemblies 115
22 Number of fuel rods per assembly 271
23 Rod array in assembly Triangular
24 Clad material Oxiside-dispersion-strengthened Ferritic Steel
25 Clad thickness (mm) 0 42
26 Number of control rods or assemblies 6 segments
27 Type Neutron Absorber Rod
28 Additional shutdown systems Not applied
29 Control rod neutron absorber material $B_4C$
30 Soluble neutron absorber Not applied
31 Burnable poison material and form Not applied

### B. REACTOR COOLANT SYSTEM

**B1. Coolant**

32 Coolant medium and inventory Sodium
33 Design coolant mass flow through core (kg/s) 4311
34 Cooling mode (forced/natural) Forced
35 Operating coolant pressure (bar) very low
36 Core inlet temperature (°C) 375
37 Core outlet temperature (°C) 530

**B2. Reactor pressure vessel/tube**

38 Overall length of assembled vessel/tube (m) 15 6
39 Inside vessel/diameter (m/mm) 9 0
40 Average vessel/tube thickness (mm) 45
41 Vessel/tube material Austenitic stainless steel
42 Lining material Not applied
43 Design pressure (bar) 4 4
44 Gross weight (tonne) 156 (ton)

**B3. Steam generator**

45 Number of steam generators 4
46 Type Helical coil once-through type
47 Configuration (horizontal/vertical) vertical
48 Tube material Mod 9 Cr-1Mo
49 Shell material Stainless steel
50 Heat transfer surface per steam generator ($m^2$) 965
51 Thermal capacity per steam generator (MW) 210

52 Feed water pressure (bar) 135
53 Feed water temperature (°C) 200
54 Steam pressure (bar) 129
55 Steam temperature (°C) 480

**B4. Pressurizer** (Not applied)
56 Pressurizer total volume ($m^3$)
57 Steam volume (full power/zero power, $m^3$)

**B5. Main coolant pumps**
58 Number of cooling or recirculation pumps 4
59 Type EMP
60 Pump mass flow rate (kg/s) 1078
61 Pump design rated head 70mNa
62 Pump nominal power (kW) 2768 (kVA)
63 Mechanical inertia (kg $m^2$) 36400 (external MGset)

## C. CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS)

64 Number of extraction lines
65 Number of pumps
66 Number of injection points
67 Feed and bleed connections

## D. CONTAINMENT

68 Type Guard vessel / Top dome
69 Overall form (spherical/cyl.)
Reactor cavity, Cylinder Top dome, Spherical
70 Structural material steel
71 Liner material Covered with adiabatic material
72 Simple/double wall Simple
73 Dimensions (diameter, height) (m) 20 x 28
74 Design pressure (bar) 3 (reactor cavity), 2 (top dome)
75 Design temperature (°C) 270/150
76 Design leakage rate (% per day) below 1%/d

# III. SAFETY RELATED SYSTEMS

## A. DESIGN CONDITIONS

**A1. Fission product retention**
77 Containment spray system (Y/N) No
a Duration (h)
b Flow rate ($m^3$/h)
c Mode of operation (active/passive)
d Safety graded (Y/N)
78 F P sparging (Y/N) No
79 Containment tightness control (Y/N) Yes
80 Leakage recovery (Y/N) No
81 Guard vessel (Y/N) Yes (Secondary vessel)

**A2. Reactivity control**
82 Absorber injection system (Y/N) Yes
a Absorber material $B_4C$
b Mode of operation (active/passive) Active
c Redundancy Primary and Secondary
d Safety graded SC-1
83 Control rods (Y/N) Yes
a Maximum control rod worth (cent) 50
b Mode of operation (active/passive) Active
c Redundancy Fine and Coarse
d Safety graded SC-3

**A3. Decay heat removal**
*A3-1 Primary side*
84 Water injection
a Actuation mode (manual/automatic)
b Injection pressure level (bar)
c Flow rate (kg/s)

d Mode of operation (active/passive)
e Redundancy
f Safety graded (Y/N)

85 Water recirculation and heat removal
a Intermediate heat sink (or heat exchanger) SG
b Mode of operation (active/passive)
c Redundancy
d Self sufficiency (h)
e Safety graded

*A3-2 Secondary side*

86 Feed water
a Actuation mode (manual/automatic)
b Flow rate (kg/s)
c Mode of operation (active/passive)
d Redundancy
e Self sufficiency (h)
f Safety graded

87 Water recirculation and heat removal
a Ultimate heat sink (cold source)
b Mode of operation (active/passive)
c Redundancy
d Self sufficiency (h)
e Safety graded

*A3-3 Primary pressure control*

88 Implemented system (Name)
a Actuation mode (manual/automatic)
b Side location (primary/secondary circuit)
c Maximum depressurization rate (bar/s)
d Safety graded

## B. SEVERE ACCIDENT CONDITIONS*

### B.1 Fission products retention

89 Containment spray system (Y/N) No
90 F P Sparging (Y/N) No
91 Containment tightness control (Y/N) Yes
92 Leakage recovery (Y/N) No
93 Risk of recriticality (Y/N) Yes

### B.2 Recriticality control

94 Encountered design feature Accident management
a Mode of operation (A/P)
b Safety graded

### B.3 Debris confining and cooling

95 Core debris configuration (core catcher) Core support structure
96 Debris cooling system (name) PRACS (N/C operation)
a Mode of operation (A/P) Passive
b Self sufficiency Depend on PRACS failure mode
c Safety graded (Y/N) N (cf F/C Yes)

### B.4 Long term containment heat removal

97 Implemented system
a Mode of operation (A/P) Active
b Self sufficiency (h) Very long
c Safety graded (Y/N) Yes

98 Intermediate heat sink Water cooling system
a Self sufficiency (h) Very long
b Safety graded (Y/N) Yes

99 External coolant recirculation
a Implemented components Seawater cooling system
b Mode of operation (A/P) Active
c Self sufficiency (h) Very long
d Safety graded (Y/N) Yes

100 Ultimate heat sink Atmosphere
a Self sufficiency (h)
b Safety graded (Y/N)

---

* All systems must be qualified to operate under the accident conditions

**B.5 Combustible gas control**

101 Covered range of gas mixture concentration

102 Modes for the combustible gas control

- a Containment inertation $N_2$
- b Gas burning None
- c Gas recombining None
- d Others

**B.6 Containment pressure control**

103 Filtered vented containment (Y/N) No

- a Implemented system
- b Mode of operation (A/P)
- c Safety graded

104 Pressure suppression system (Y/N) No

- a Implemented system
- b Mode of operation
- c Safety graded (Y/N)

## C. SAFETY RELATED I&C SYSTEM

Automatic load following (Y/N) Yes

- * range (% power) 40-100
- * maximum rate (%/min) 0 4

Load rejection without reactor trip (Y/N) No

Full Cathode Ray Tubes (CRT) display (Y/N)

Automated start-up procedures (Y/N) No

Automated normal shutdown procedures (Y/N) No

Automated off normal shutdown procedures (Y/N) Yes

Use of field buses and smart sensors (Y/N) No

Expert systems or artificial intelligence advisors (Y/N) No

Protection system backup (Y/N) No

## D. EMERGENCY POWER SUPPLY SYSTEM

105 Type (diesel, gas, grid connection) Diesel

106 Number of trains 2

## E. AC/DC SUPPLY SYSTEM

107 Type (rectifier, converter, battery) Battery

108 Estimated time reserve (hr) 10

# IV. CONVENTIONAL THERMAL CYCLE

## A. TURBINE SYSTEM

109 Type TC6F-23"

110 Overall length (m) 27 85

111 Width (m) 8 0

112 Number of turbines/reactor 1/1

113 Number of turbine sections per unit (e g HP/LP/LP) HP/LP/LP/LP

114 Speed (rpm) 3000

## B. STEAM CHARACTERISTICS

115 H P inlet pressure bar 127 $kg/cm^2g$

116 H P inlet temperature (°C) 450

117 H P inlet flowrate (kg/s) 400/h

118 L P inlet pressure

119 L P inlet temperature

120 L P inlet flowrate (per section)

## C. GENERATOR

121 Type (3-phase synchronous, DC) TAKS-LCH

122 Apparent power (MVA) 365

123 Active power (MW) 325

124 Frequency (hz) 50

125 Output voltage (kV) 17

126 Total generator mass (t)

127 Overall length 13 34

128. Stator housing outside diameter: 4.91

D. **CONDENSER**

129. Number of tubes
130. Heat transfer area: $3.64 \times 10^3$ (kcal/h)
131. Flowrate ($m^3/s$)
132. Pressure (m/bar): 722 (mmHgV)
133. Temperature (°C): 17 (coolant)

E. **CONDENSATE PUMPS**

134. Number: 3
135. Flowrate: 660 ($m^3/h$)
136. Developed head: 190 (m)
137. Temperature
138. Pump speed

## 7.11.5. Project Status

### *7.11.5.1. Entities involved*

CRIEPI (Central Research Institute of Electric Power Industry) has designed the Modular Double Pool Reactor in collaboration with Toshiba Corporation.

### *7.11.5.2. Design status*

Conceptual design has been completed.

### *7.11.5.3. R&D and licensing status*

R&D items have been discussed. The items are classified from two viewpoints. The first is on importance; A) items essential for safety design, B) items effective on economy and C) items of desirable demonstration tests. The second is dependency on structure; 1) items peculiar to the double pool structure, 2) items peculiar to small reactors and/or modular reactors and 3) items common to all LMRs. Typical items are a thermal-hydraulic test with a scale model (A-1), and the fast pressure release system and demonstration experiment for beyond design basis leak events (B-1). Licensing will be started after completion of detailed design.

## 7.11.6. Project economics

The modular double pool reactor has two economic characteristics. One is design simplification and reduction of on-site work by the double pool type configuration. The other is modularization by design standardization and common use of equipment.

Construction cost of the first of a kind (FOAK) module and the first 4-module plant (see Fig.7.11.2.) is evaluated as 2.0 and 1.2 times the cost of an existing LWR respectively. The second 4-module plant cost is expected to be equal to an existing LWR as a result of standardization and the learning effect.

## REFERENCES


[1] Hattori, S., An Innovative LMFBR Concept "Double Pool Type LMFBR", Proceedings of the IAE International Symposium of LMFBR Development, November 1984, p.233.

[2] Kinoshita, I. et al., "Development of Small Modular Double Pool Reactor for Early Realization of FBR Practical Application", Proceedings of the International Conference on Fast Reactors and related Furl Cycles (FR '91), Oct. 1991, 14.3.


## 7.12. 4S REACTOR SYSTEM DESCRIPTION AND DEVELOPMENT STATUS

### 7.12.1. Basic objective and features

4S (Super Safe, Small and Simple) is a small sodium-cooled fast reactor, in which intensive efforts are concentrated with the aim of meeting the energy power source requirements of the global market.

To correspond to the global market, 4S is designed on the principles of simple operation, simplified maintenance including refueling, higher safety, improved economic features and non proliferation of nuclear materials. More specific design policy for 4S is as follows:

- The void reactivity and all reactivity temperature coefficients are negative,
- No refueling for 10 years,
- Simple core burnup control without a control rod and its driving mechanism,
- Safety system independent of emergency power and active decay heat removal systems,
- Elimination of control and adjustment components from the reactor system,
- Load following without operation of reactor control system,
- Minimum maintenance and inspection of reactor components,
- Quality assurance and short construction period based on shop fabrication,
- No core damage in any conceivable transient events without scram,
- Complete containment of nuclear materials for a long period.

### 7.12.2. Design description

#### *7.12.2.1. Nuclear steam supply system (NSSS)*

(1) Core and burnup control system

The 4S reactor employs a burnup control system with an annular reflector in place of the control rod and its driving mechanism. Burnup control by vertical movement of the annular reflector eliminates the necessity for complicated control rod operations.

The core geometry with reflector control system should have a core equivalent diameter less than 90cm and length of the reflector at least 1.5m in order to meet requirements for negative void reactivity and no refueling for 10 years. When the length of the sub-assembly is restricted to 7m maximum from the view point of the manufacturing process, thermal power of the core could be less than about 125 MWth. The active length of the core as designed is 4m and equivalent diameter is 83cm.

(2) Annular reflector

Graphite is employed for the reflector to prevent deformation by irradiation. The upper part of the reflector has to be composed of a substance with lower reflecting effect than the coolant in order to raise the reactivity value of the reflector. The reflector consists of six segments with a thickness of 15cm.

(3) Reflector drive mechanism

The reflector drive mechanism consists of two hydraulic systems that are used for startup and shutdown, and six motor systems that are used during normal operation. Each motor system corresponds to a reflector segment. The reflector is moved upward by the hydraulic pump for startup. For normal shutdown of the reactor, the reflector is moved down by means of the hydraulic system. For reactor scram, the scram valves of the hydraulic systems open, and cause the reflector to move downward rapidly. As the reflector goes down 1m, the reactor enters the subcritical cold shutdown state.

The length of the downward movement of the reflector is determined by the capacity of the hydraulic cylinder. It cannot move further.

Upon completion of the startup by hydraulics, the reflector is moved up for burnup control at a constant speed of around 1mm/day by a motor using electricity which is supplied from the 4S's own generator so that the reflector is positioned by integration of the generated power. Uncontrolled withdrawal cannot occur in principle since the motor system does not have any feedback control circuit. A reduction mechanism composed of planetary gears is installed within the motor system to attain the fine speed of 1mm/day.

(4) Reactor system

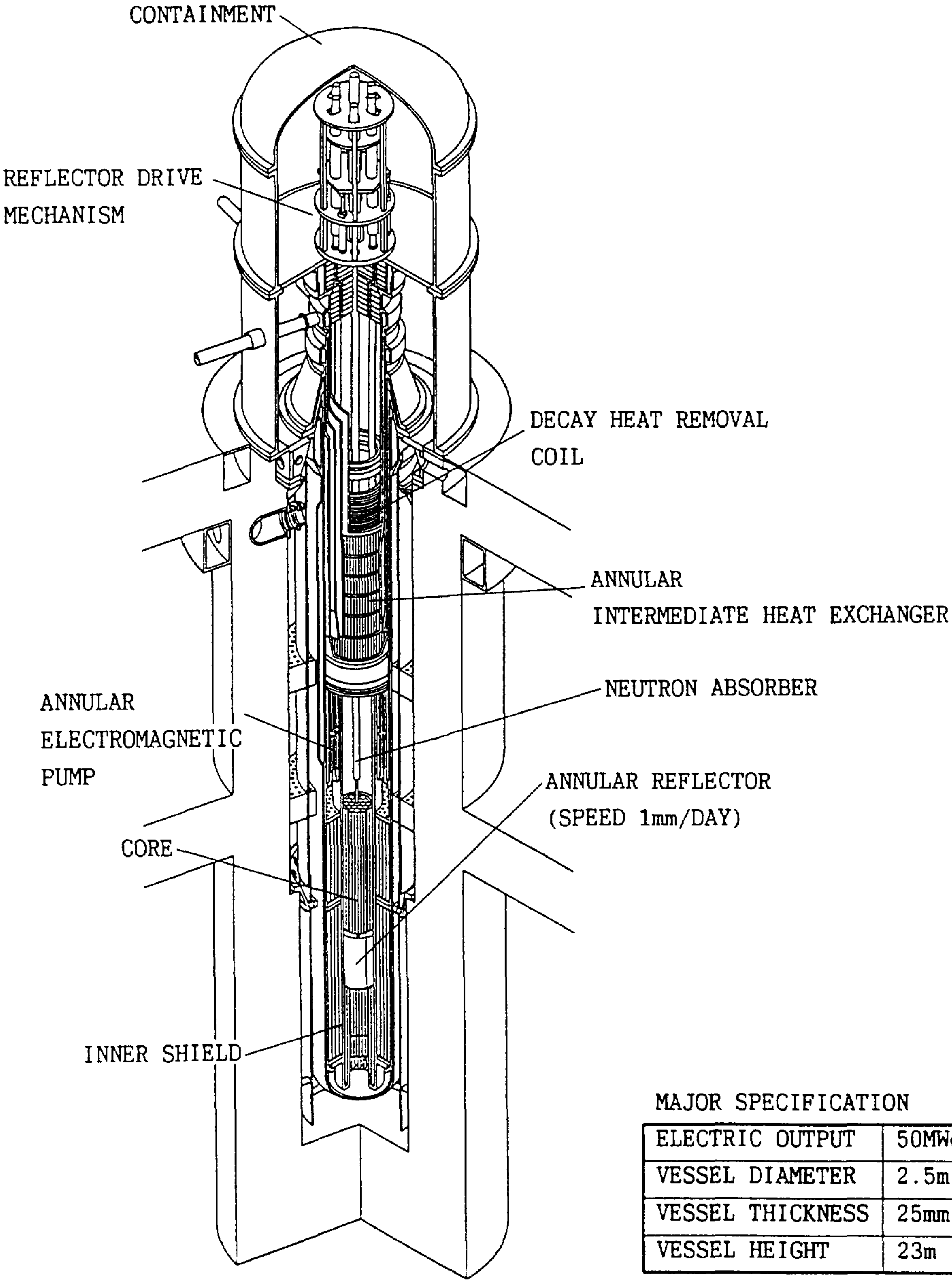
Major specifications and a bird's eye view of the reactor are shown in Fig.1.

The sodium coolant flows out of the core, ascends the hot pool and descends in the intermediate heat exchanger (IHX) through which the heat is transferred to the secondary coolant. It is pressurized by the primary electromagnetic pump (EMP) at the bottom of the IHX and flows down along the inner hole of in-vessel reflector. Then, it turns at the bottom of the reactor vessel and enters the core. These IHX and EMP in the primary loop have annular shapes. An annular vertical redan is installed to form the annular path of flow. A space is provided outside the core barrel, in which the reflector moves vertically. The sodium flows through this space to cool the reflector.

The primary pump is a sodium immersed self-cooled EMP which consists of two annular single stator EMPs joined in series.

For the radial shield, a 60cm Modified 9Cr-1Mo shield is installed in the reactor vessel to prevent Ar gas activation in the reactor cavity.

For the reactor internals, since the neutron fluence to the core barrel, reflector and reflector guide is high, Mod.9 Cr-1Mo, which has a good irradiation characteristics, is




MAJOR SPECIFICATION

| | |
|---|---|
| ELECTRIC OUTPUT | 50MWe |
| VESSEL DIAMETER | 2.5m |
| VESSEL THICKNESS | 25mm |
| VESSEL HEIGHT | 23m |

*Fig 7.12.1. Reactor Assembly*

employed. Two stages of shield are installed in the reactor cavity, in order to prevent neutron streaming from the reactor cavity. A 60cm $B_4C$ shielding block is installed on the subassembly in order to protect the EMP and reduce activation of the secondary sodium in the HX.

The top part of reactor is composed of small and large steel plate decks with thermal shielding. Concrete for neutron shielding on the upper deck is not necessary since there is a sodium layer 8m deep over the top of core.

(5) Fuel exchange system

The refueling interval is 10 years. All of the 18 subassemblies are replaced every 10th year. First, the neutron absorber is withdrawn, then the smalldeck is removed. They are stored in the maintenance pit of the reactor building. Fuel assemblies are withdrawn from the core using the fuel exchange machine.

The fuel assemblies are withdrawn and inserted with ease. Assemblies removed from the core are inserted into a cask in the pit and the cask is filled with helium gas.

(6) Reactor cavity

A natural air cooling system is employed for cooling the reactor cavity. While its main purpose is removal of heat from the reactor in normal operation, it has also a function of decay heat removal. Points to be noted in the natural air cooling system include: uniform temperature distribution circumferentially; low pressure loss, and prevention of air activation. Particular attention is placed on the uniformity of circumferential temperature distribution.

The natural air cooling system in the reactor vessel is designed so that it is capable of removing decay heat without operation of the Primary Reactor Auxiliary Cooling System (PRACS).

(7) Secondary system

An EMP is also employed for the secondary loop, and joined to the cold leg of the steam generator (helical tube type) in order to reduce the piping length between the secondary pump and the steam generator and reduce the size of the secondary system. The cover gas of the steam generator is tightly enclosed during operation. In the sodium-water reaction product release system, two lines of 24inch piping (diameter of 60cm) are connected to the bottom of the steam generator and led to a dump tank so that the reactor system is not affected in the event of total break of heat transfer tubes.

*7.12.2.2. Balance of plant systems*

(1) Turbine generator system

The Nuclear Steam Supply System (NSSS) of 4S is attached to one turbine generator system, which is intended to be compact insize and to be a simple system.

(2) Radioactive waste management system

Since the refueling interval is 10 years, the frequency of radioactive waste management operation through plant life is very low.

A radioactive waste management system for individual 4S plants is not necessary and a common system for several 4S plants is provided.

(3) Heat and ventilation system

The heat released from the reactor vessel, support structure of the vessel and heat from sources in the reactor containment are all removed by natural ventilation. An HVAC system for the primary loop is thus not required.

Cooling of the secondary loop and the electric equipment room is performed by forced open cycle ventilation. The switch board room and the room where the coastdown system is situated, are cooled by a heat storage system in the event of loss of power, so that the temperature in each of these rooms is not elevated during loss of power for a long period. As a result, an emergency HVAC system and water cooling system as the final heat sink of the HVAC system are not required.

*7.12.2.3. Instrumentation, control and electric systems*

The reactor system does not have any feedback control system at all. The plant is started up by heat from the primary pump which raises the system temperature to 350°C. Then, the reflector is lifted by the hydraulic system to reach the critical state at 350°C. The system temperature is raised up to 430°C by nuclear fission heating and maintenance thre by moving the reflector upwards.

Feed water flow is then gradually increased. The core inlet temperature decreases and the power is increased by reactivity insertion. When the reactor power is increased to equal the house load, the main steam control valve is opened to increase the speed of the turbine to a specified frequency, and then the generator is connected. The reactor power output is increased to 100% by raising the water flow. In the present design, it takes 16hrs to reach 100% power after completion of raising system temperature up to 350°C.

Burn up compensation is attained by moving the reflector upward at a constant speed of 1 mm/day to compensate for the reactivity decrease due to the burn-up of the core. Since no feedback system or control system is used, the reflector speed remains constant and the electric output is adjusted by varying the feed water flow rate to control the core inlet temperature. The controllable range of the power level by water flow variation is ±10% at the rated power, which is limited by the steam generator heat balance. Beyond this range, a back-up control mechanism to adjust the reflector position is installed in the driving mechanism.

To load follow, the core inlet temperature is changed by controlling the water flow so that the generator output coincides with the load-following control, thus causing the reactor output to follow.

*7.12.2.4. Safety consideration and emergency protection*

(1) Inherent safety

(a) Negative void reactivity coefficient

One of the attractive features of the fast reactor is its hard neutron spectrum. To expand this feature, a metallic fuel core(U-Pu-Zr or U-Zr) is employed in the 4S. However, it is more difficult to reduce void reactivity for a core with a harder spectrum. It is very important to design the void reactivity to be negative in order to prevent a severe nuclear accident in the event of sudden loss of coolant, sudden loss of coolant flow or a large gas bubble entrainment in the core.

There are two generic approaches to reduce the void reactivity. Reducing the core height is one popular approach, and a core with a small diameter is another effective method. In the 4S, making the core diameter small is the preferred approach because this reduces the vessel diameter and enhances the value of the reflector reactivity. By reducing the core diameter, neutron leakage is enhanced in the radial direction so that a negative void reactivity coefficient is maintained during the entire core life time.

For the selected core, the void reactivity of the total core is -1$ at the end of life based on a transport calculation. Other temperature feedback coefficients are also negative.

(b) Preventing potential super prompt criticality

It is essential for the safety of the reactor to exclude the possibility of a super prompt critical state at all times. This requires that the inserted reactivity from potential events should be below 1$ under conservative conditions, neglecting reactivity feedback coefficients.

The largest reactivity change occurs during plant start up. The reactivity decrease from criticality at zero power under cold temperature conditions to full power is generally above 1$. The worst case is reactivity insertion under cold temperature conditions.

During plant start up in the 4S, the system temperature is raised to 350°C by heat input from the EMP before raising the reflector. This procedure greatly reduces the reactivity temperature swing. The reactivity to be inserted to increase the power is about 86c, which counters the following reactivity effects; thermal expansion of the fuel, structure, coolant, core support grid and doppler reactivity. Because metallic fuel is employed in the 4S, the reactivity requirement is small compared with the 150c for MOX fuel, mainly due to its small doppler effect.

The power transient would reflect the super prompt critical condition if a large reactivity insertion were to occur. By contrast, the power transient is small for the 4S during any potential reactivity insertion during the plant start up phase.

(c) Neutron leakage control by reflector

In the 4S core, all reactivity change is controlled by the reflector. This neutron leakage control system has a decisive advantage compared with a control rod system from the safety point of view.

The active length of the core is 4m, which is surrounded by a 1.5m long reflector. The reflector is separated into six azimuthal parts, each of which can move from the bottom to the top of the core during the core burn-up. If an uncontrolled lift of any part of the reflector occurs, the core criticality cannot be sustained. The new geometry of the reflected region causes negative reactivity insertion because of the enhanced neutron leakage. In this design, lifting up parts, in contrast to raising all parts together, gives strong negative reactivity of the reflector.

(2) Safety system

The safety system of 4S consists of the reactor shutdown system, decay heat removal system and containment. The reactor is shutdown by insertion of a neutron absorber into the core center and lowering of the reflector which surrounds the core. The reflector is divided into six parts and managed by two hydraulic systems corresponding to 3 reflector segments. The reflector moves down by opening two scram valves of a hydraulic system which has the required redundancy. If the valve of a hydraulic system is opened during upward motion of the reflector, scram is initiated and the reflector drops. Should the neutron absorber and reflector not work, the power drops due to the negative feedback coefficient to lead the reactor to its inherent shutdown.

The decay heat is removed by a system consisting of the decay heat removal coil installed in the reactor and the natural ventilation of air from outside the guard vessel. Heat removal by natural circulation takes place inside the reactor after shutdown, eliminating necessity for the operation of active components.

The containment is composed of the guard vessel installed outside the reactor vessel and the top dome over the reactor vessel. The top dome has only few penetration to maintain high leaktightness.

Two types of plant protection system are provided, the neutron detector installed on the outside of the reactor vessel and the core outlet temperature detector in the reactor vessel.

*7.12.2.5. Building and structures*

The reactor building is an embedded structure of high seismic class. It contains the reactor, secondary system, steam generator, coast down control system, power switch board and refueling pits as shown in Fig. 2. The dimensions of the building are 26mx16m, requiring only a small ground space.

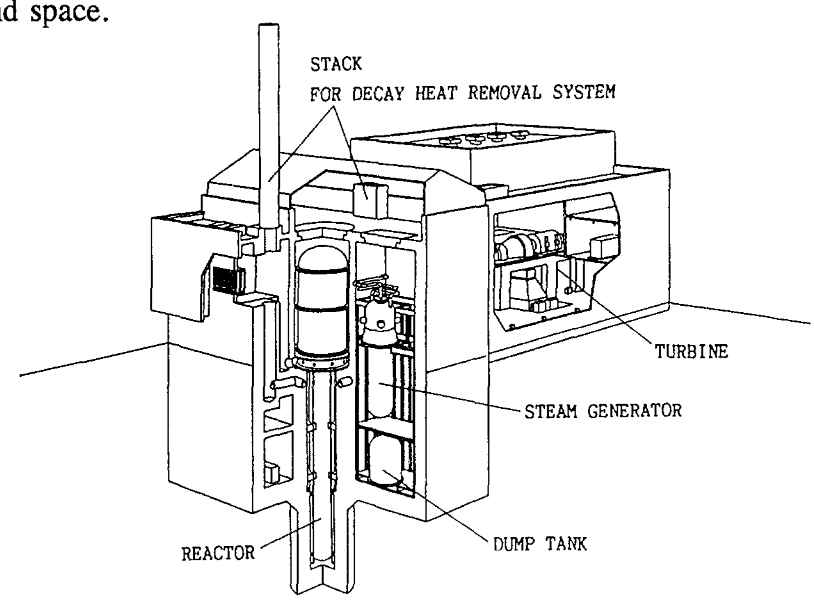



*Fig 7.12.2. Plant Concept*

### 7.12.3. Safety concepts

ABLE 7.12.1. MAIN SAFETY RELATED SYSTEMS IN THE 4S CONCEPT

| Name | Safety graded | Main characteristics |
|---|---|---|
| Primary circuit | x<br>-<br><br>- | Reactor vessel<br>Two recirculation EMP (50% flow each)<br>One intermediate heat exchange |
| Diverse reactor shutdown system | - | Neutron absorber insertion<br>Reflector withdraw |
| Reactivity control system | - | Six reflector drive system<br>Feed water flow control system |
| Counter measurement of extra-reactivity insertion | -<br>- | Hydraulic drive system of reflector<br>Self connected mechanism (to hold the neutron absorber only above 350°C) |
| Passive and diverse decay heat removal system | - | PRACS<br>RVACS |
| Containment system | x<br><br>x | Guard vessel at outside of the reactor vessel<br>Top dome at the top of the reactor vessel |
| Plant protectin system | x<br>x | Neutron detector<br>Core outlet temperature detector |
| Water-sodium reaction | - | Pressure release system |

TABLE 7.12.2. MAIN ACCIDENT INITIATORS FOR THE 4S

- LOF : Loss of Flow
- TOP : Transient Over Power
- ATWS : Anticipated Transients Without Scram
- Primary Transients
- Secondary Transients (turbine trip)
- Loss of electric sources (partial)
- Total loss of the heat sink
- Total loss of the steam generator feedwater
- Station blackout
- Water-sodium reaction

TABLE 7.12.3. DESIGN FEATURES FOR PREVENTION AND PROTECTION LEVELS THAT REDUCE (R), SUPPRESS (S), THE INITIATORS FREQUENCY OR LIMIT THE POTENTIAL ACCIDENT CONSEQUENCES (L)

| **PREVENTION LEVEL** |
|---|
| LOF: Reduced vessel fluence : reduce initiator frequency<br>Primary circuit integration : reduce initiator frequency<br>Low pressure: reduce initiator frequencey<br>Adaptation of leak before break : limits accident consequence<br>Two EMP : Limits accident consequence<br>TOP: Six reflector driving systems : limits accident consequence<br>Feedwater flow control system : limits accident consequence<br>Hydraulic drive system of reflector : reduce initiator frequency<br>Self connected mechanism of the neutron absorber : reduces initiator frequencey<br>Constant speed of relfector : reduce initiator frequency |
| Primary transients: Increased design margins : reduces frequency |
| Secondary transients: Increased design margins : reduces frequency |
| Loss of electric sources: reduces frequency |
| Total loss of the cold source (water): reduces frequency |
| Total loss of the SG feedwater: reduces frequency |
| Station blackout: reduces initiator frequency |
| Water-sodium reaction : increased design margins : reduces initiator frequency |
| **PROTECTION LEVEL** |
| LOF: Primary Reactor Auxiliary Cooling System (PRACS)<br>Reactor Vessel Auxiliary Cooling System (RVACS)<br>TOP: Mechanical stop of hydraulic drive system of reflector |
| ATWS: Strong negative temperature coefficient : limit accident consequence |
| Primary transients : limit consequence |
| Secondary transients : limit consequence |
| Loss of electric sources : Emergency power generator |
| Total loss of heat sink : PRACS and RVACS : limit consequence<br>Total loss of SG feedwater : negative reactivity feedback by thermal expansion of core support place<br>Station blackout : passive heat strage system for switch board room : limit HVAC accident consequence |

TABLE 7.12.4. DESIGN FEATURES FOR MITIGATION LEVEL OF 4S

| Safety functions | Systems (Cf. Table 7.12.1.) | Passive/active | Design features/Remarks |
|---|---|---|---|
| **Design Basis**<br>Fission product containment | Clad<br>Reactor Vessel<br>Guard vessel / Top dome | Passive<br>Passive<br>Passive | |
| Coolant inventory | Guard Vessel | Passive | |
| Decay heat removal | Primary Reactor Auxiliary Cooling System (PRACS)<br>Reactor Vessel Auxiliary Cooling System (RVACS) | Passive<br>Passive | 1 system<br>1 system |
| Reactivity control | Reflector withdraw<br>Neutron Absorber Insertion | Active<br>Active | 2 systems (3 segment/system)<br>One neutron absorber |
| Primary circuit pressure control | | | |
| **Severe Accident**<br>Containment temperature and pressure control | Guard vessel / Top dome / RVACS | Passive | |
| Heat removal | RVACS | Passive | 1 system |
| Tightness control | Guard Vessel / Top dome | | |
| Inflam. gas control | | | |
| Fission product containment | | | |
| Corium management | | | |
| Others | | | |

### 7.12..4. Design data questionnaire

## I. GENERAL INFORMATION

1. Design name: 4S
2. Designer/Supplier address: Central Research Institute of Electric Power Industry
3. Reactor type: LMR / Number of modules/per plant: 1
4. Gross thermal power (MW-th) per reactor: 125
5. Net electrical output (MW-e) per reactor: 50
6. Heat supply capacity (MW-th)

## II. BASIC DESIGN DESCRIPTION

## A. CORE AND REACTIVITY CONTROL

7. Fuel material: U-Pu-Zr or U-Zr
8. Fuel inventory (tones of heavy metal): 9.9
9. Average core power density (kW/liter): 61
10. Average fuel power density (kW/kgU): 12.6
11. Maximum linear power (W/m): 25300
12. Average discharge burnup (MWd/t): 45500
13. Initial enrichment or enrichment range (Wt%)
14. Reload enrichment at the equilibrium (Wt%)
15. Refueling frequency (months): 120
16. Type of refueling (on/off power): off power
17. Fraction of core withdrawn (%)
18. Moderator material and inventory
19. Active core height (m): 4
20. Core diameter (m): 0.83
21. Number of fuel assemblies: 18
22. Number of fuel rods per assembly: 217
23. Rod array in assembly
24. Clad material: Improved Austenitic stainless steel
25. Clad thickness (mm): 0.5
26. Number of control rods or assemblies: 6 segments
27. Type: refelctor
28. Additional shutdown systems: Neutron absorber
29. Control rod neutron absorber material: $B_4C$
30. Soluble neutron absorber: None
31. Burnable poison material and form: None

## B. REACTOR COOLANT SYSTEM

**B1. Coolant**

32. Coolant medium and inventory: Sodium
33. Design coolant mass flow through core (kg/s): 633
34. Cooling mode (forced/natural): Forced
35. Operating coolant pressure (bar)
36. Core inlet temperature (C): 355
37. Core outlet temperature (C): 510

**B2. Reactor pressure vessel/tube**

38. Overall length of assembled vessel/tube (m): 27
39. Inside vessel/diameter (m): 2.5
40. Average vessel/tube thickness (mm)
41. Vessel/tube material: stainless steel
42. Lining material
43. Design pressure (bar)
44. Gross weight (tonne)

**B3. Steam generator**

45. Number of steam generators: 1
46. Type: once through: Integrated type of EMP and steam generator
47. Configuration (horizontal/vertical): vertical
48. Tube material: Mod. 9Cr-1Mo
49. Shell material: Stainless steel
50. Heat transfer surface per steam generator ($m^2$): 554
51. Thermal capacity per steam generator (MW): 125
52. Feed water pressure (bar): 127
53. Feed water temperature (C): 210
54. Steam pressure (bar): 107
55. Steam temperature (C): 453

**B4. Pressurizer**

56. Pressurizer total volume ($m^3$)
57. Steam volume (full power/zero power, $m^3$)

**B5. Main coolant pumps**

58. Number of cooling or recirculation pumps: 2 (50% each)
59. Type: EMP
60. Pump mass flow rate (kg/s): 633
61. Pump design rated head
62. Pump nominal power (kW)
63. Mechanical inertia (kg $m^2$)

## C. CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS)

64. Number of extraction lines
65. Number of pumps
66. Number of injection points
67. Feed and bleed connections

## D. CONTAINMENT

68. Type: Guard vessel / Top dome
69. Overall form (spherical/cyl.): Guard vessel;
    Sylinder; Top Dome; Spherical
70. Structural material: Steel
71. Liner material:
72. Simple/double wall: Simple
73. Dimensions (diameter, height) (m): Guard vessel 3x22
    Top Dome 6x11
74. Design pressure (bar)
75. Design temperature (C)
76. Design leakage rate (% per day)

## III. <u>SAFETY RELATED SYSTEMS</u>

## A. DESIGN CONDITIONS

**A1. Fission product retention**

77. Containment spray system (Y/N)
    a. Duration (h)
    b. Flow rate ($m^3$/h)
    c. Mode of operation (active/passive)
    d. Safety graded (Y/N)
78. F.P. sparging (Y/N)
79. Containment tightness control (Y/N)
80. Leakage recovery (Y/N)
81. Guard vessel (Y/N): Yes

**A2. Reactivity control**

82. Absorber injection system (Y/N): Yes
    a. Absorber material: $B_4C$
    b. Mode of operation (active/passive): Passive

c. Redundancy
d. Safety graded

83. Control rods (Y/N)
a. Maximum control rod worth (pcm)
b. Mode of operation (active/passive)
c. Redundancy
d. Safety graded

**A3. Decay heat removal**

*A3-1 Primary side*

84. Water injection
a. Actuation mode (manual/automatic)
b. Injection pressure level (bar)
c. Flow rate (kg/s)
d. Mode of operation (active/passive)
e. Redundancy
f. Safety graded (Y/N)

85. Water recirculation and heat removal
a. Intermediate heat sink (or heat exchanger) SG
b. Mode of operation (active/passive)
c. Redundancy
d. Self sufficiency (h)
e. Safety graded

*A3-2 Secondary side*

86. Feed water
a. Actuation mode (manual/automatic)
b. Flow rate (kg/s)
c. Mode of operation (active/passive)
d. Redundancy
e. Self sufficiency (h)
f. Safety graded

87. Water recirculation and heat removal
a. Ultimate heat sink (cold source):
b. Mode of operation (active/passive):
c. Redundancy:
d. Self sufficiency (h)
e. Safety graded:

*A3-3 Primary pressure control*

88. Implemented system (Name)
a. Actuation mode (manual/automatic)
b. Side location (primary/secondary circuit)
c. Maximum depressurization rate (bar/s)
d. Safety graded:

**B. SEVERE ACCIDENT CONDITIONS***

**B.1 Fission products retention**

89. Containment spray system (Y/N)
90. F.P. Sparging (Y/N)
91. Containment tightness control (Y/N)
92. Leakage recovery (Y/N)
93. Risk of recriticality (Y/N)

**B.2 Recriticality control**

94. Encountered design feature
a. Mode of operation (A/P)
b. Safety graded

**B.3 Debris confining and cooling**

95. Core debris configuration (core catcher)
96. Debris cooling system (name)

* All systems must be qualified to operate under the accident conditions.

a. Mode of operation (A/P)
- b. Self sufficiency
- c. Safety graded (Y/N)

**B.4 Long term containment heat removal**

97. Implemented system
- a. Mode of operation (A/P)
- b. Self sufficiency (h)
- c. Safety graded (Y/N)

98. Intermediate heat sink
- a. Self sufficiency (h)
- b. Safety graded (Y/N)

99. External coolant recirculation
- a. Implemented components
- b. Mode of operation (A/P)
- c. Self sufficiency (h)
- d. Safety graded (Y/N)

100. Ultimate heat sink
- a. Self sufficiency (h)
- b. Safety graded (Y/N)

**B.5 Combustible gas control**

101. Covered range of gas mixture concentration
102. Modes for the combustible gas control
- a. Containment inertation
- b. Gas burning
- c. Gas recombining
- d. Others

**B.6 Containment pressure control**

103. Filtered vented containment (Y/N)
- a. Implemented system
- b. Mode of operation (A/P)
- c. Safety graded

104. Pressure suppression system (Y/N)
- a. Implemented system
- b. Mode of operation
- c. Safety graded (Y/N)

## C. SAFETY RELATED I&C SYSTEM

Automatic load following (Y/N): Yes
- * range (% power)
- * maximum rate (%/min)

Load rejection without reactor trip (Y/N)
Full Cathode Ray Tubes (CRT) display (Y/N)
Automated start-up procedures (Y/N): Yes
Automated normal shutdown procedures (Y/N): Yes
Automated off normal shutdown procedures (Y/N): Yes
Use of field buses and smart sensors (Y/N)
Expert systems or artificial intelligence advisors (Y/N): Yes
Protection system backup (Y/N)

## D. EMERGENCY POWER SUPPLY SYSTEM

105. Type (diesel, gas, grid connection)
106. Number of trains

## E. AC/DC SUPPLY SYSTEM

107. Type (rectifier, converter, battery)
108. Estimated time reserve (hr)

## IV. CONVENTIONAL THERMAL CYCLE

### A. TURBINE SYSTEM

109. Type
110. Overall length (m)
111. Width (m)
112. Number of turbines/reactor
113. Number of turbine sections per unit (e.g. HP/LP/LP)
114. Speed (rpm)

### B. STEAM CHARACTERISTICS

115. H.P. inlet pressure bar
116. H.P. inlet temperature
117. H.P. inlet flowrate (kg/s)
118. L.P. inlet pressure
119. L.P. inlet temperature
120. L.P. inlet flowrate

### C. GENERATOR

121. Type (3-phase synchronous, DC)
122. Apparent power (MVA): 62.5
123. Active power (MW): 50
124. Frequency (hz): 60
125. Output voltage (kV): 13.8
126. Total generator mass (t): 106
127. Overall length· 7.6m
128. Stator housing outside diameter

### D. CONDENSER

129. Number of tubes
130. Heat transfer area
131. Flowrate ($m^3/s$)
132. Pressure (m/bar)
133. Temperature (°C)

### E. CONDENSATE PUMPS

134. Number
135. Flowrate
136. Developed head
137. Temperature
138. Pump speed

## 6.5. Project status

### 6.5.1. Entities involved

Under the sponsorship of the Central Research Institute of Electric Power Industry (CRIEPI), Toshiba Corporation has designed the 4S-50 Liquid Metal Fast Reactor, which supplies electricity to a desalination plant, supported by the Technical Advisory Committee of 4S.

### 6.5.2. Design status

Conceptual design has been completed.

### 6.5.3. R&D and licensing status

R&D items have been discussed. Licensing will be started after completion of the detailed design.

## 6.6. Project economics

The case study shows that the construction cost of 4S will be a little higher than a large LWR in the case of continuous shop fabrication with 10 units per year.

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# APPENDIX I

## ECONOMICS OF NUCLEAR POWER

There may be reasons for nuclear power introduction for a particular purpose or location which override economic considerations. However this situation is unusual and economics often dominate any decision to deploy a nuclear station. Many of the issues are common to all sizes of nuclear reactor but there are aspects where SMRs have a potential advantage over larger plants. The following discussion deals with the economic issues involved and brings out the areas where SMRs are advantageous.

Most of the existing nuclear power plants whether large or in the SMR range generate base load electricity safely, reliably and at lower cost than fossil-fired plants in the same country. On the other hand, the margin of competitiveness of future nuclear power plants compared to fossil fired baseload generation has become smaller than previously. An important reason for this is that fossil fuel prices have decreased in real terms. While this also applies to nuclear electricity generation costs, the impact on fossil electricity generation costs is greater.

A recent study by the Nuclear Energy Agency (NEA) and the International Energy Agency (IEA) of the OECD, Reference [1], in which the IAEA co-operated, shows that the projected nuclear and fossil based levelized generation costs are rather close in a number of countries. As can be seen from Fig. A1.1, there are only a few countries where a certain energy technology is economically preferable across the range of parameters, e.g. nuclear power in France and Japan, coal in western USA or natural gas in the UK (compared to a single PWR). In most countries, which option is economically preferable depends on the interest/discount rate, operating regime and other factors.

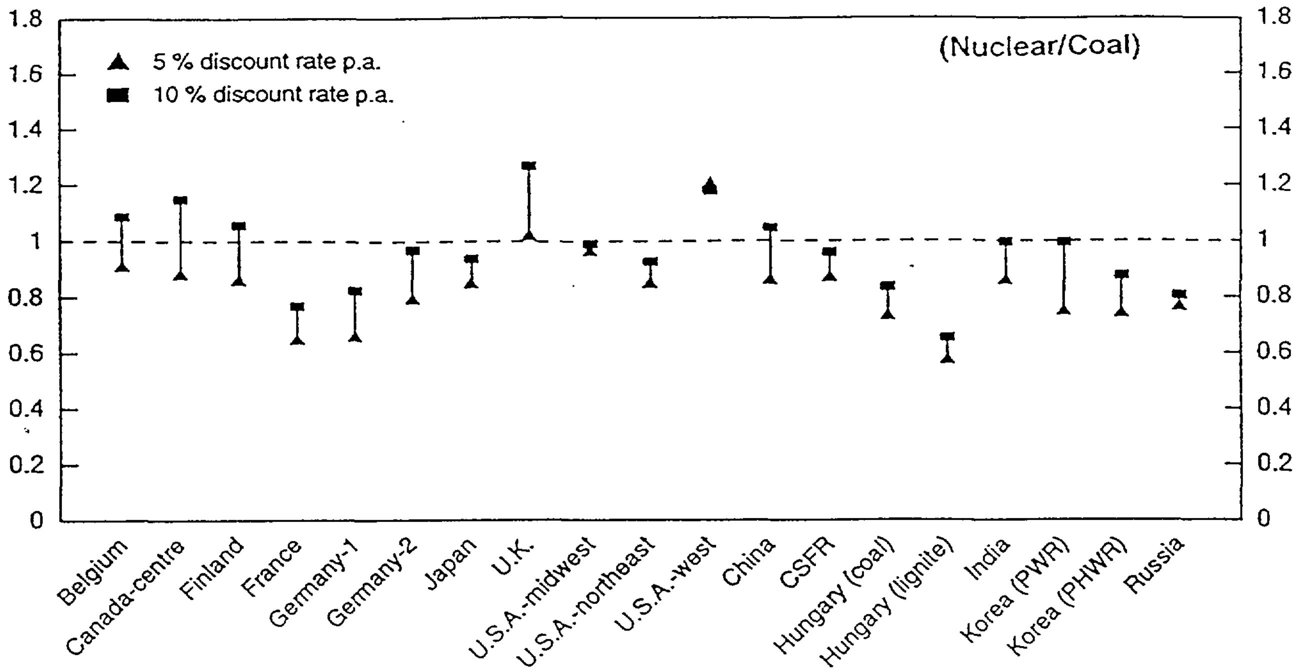
Some cost issues, which currently tend to detract from the competitiveness and market prospects of nuclear power, are discussed below:

- Low fossil fuel prices: The real prices (net of general inflation) of coal and gas have fallen in the last 10 years in most countries. They are expected to escalate somewhat in the future, but not at high rates. The reasons for this include large resources and sufficient mining capacities for coal, and improved drilling techniques for gas exploration and production.

- Improved combined cycle efficiency has improved the competitiveness of this technology.

- The high capital intensity of NPPs, in particular of SMRs, results in electricity generation costs in the first years of service that will usually be higher than for fossil-fuel based electricity. Utilities may favour less capital-intensive fossil fuel technologies, even if the levellized costs are higher.

- There is a mixed cost experience; some recently constructed NPPs were completed on budget and schedule, whereas others experienced long delays and large cost overrun.




1. Germany - Domestic coal.
2. Germany - Imported coal.

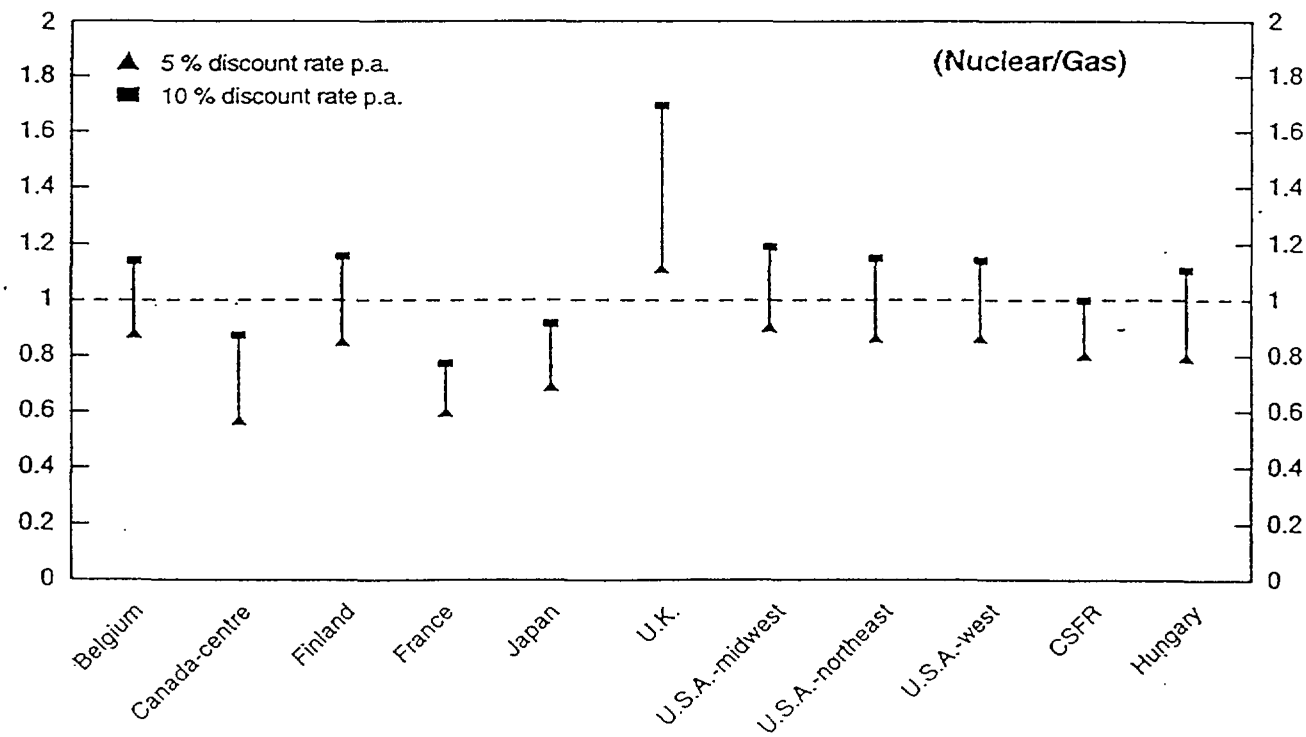



Figure A1.1 Generation Cost Ratio

- Financing difficulties: financing requirements exceeding one billion US$ with a long repayment period, and with the risks of start-up delay and cost overrun, are seen as a high risk by the financial community. Commercial banks will demand higher interest rates than for smaller loans with faster repayment. In the past, many loans to utilities were guaranteed by the government, but some governments may be reluctant to continue this practice.

- Trend to increase competition in the energy sector. Measures include third party access to formerly monopolized grids and privatization of state-owned utilities; large utilities may be split into smaller ones (e.g. as was done in the UK). This more competitive environment will let utilities strive for higher capital return rates than in the past, and may tend to favour less capital-intensive technologies and smaller unit sizes than in the past.

- Trend to even higher nuclear safety standards which may entail cost increases.

- In North America and Europe, the electricity demand is almost stagnant or only slowly increasing, and new base load capacity is currently not needed in most of the countries. In Contrast many countries in South and East Asia are experiencing high economic growth and high electricity demand growth and need additional base load capacity.

On the other hand, fossil-fired power plants must meet higher environmental protection standards than in the past, which requires additional costs for flue gas cleaning equipment and for solid waste handling and disposal. Increasing fossil fuel demand may also require high investment in mines, railways, pipelines, harbours and other transport facilities, and may raise the average fuel cost. Many countries aim at conserving fossil fuel and have committed themselves to reducing $CO_2$ emission levels.

It is very important to keep the construction costs of nuclear power plants as low as possible, while maintaining high quality standards and further improving plant safety. Reactor manufacturers are improving current plants and developing designs for new plants, aiming to simplify the plants where possible without compromising plant safety and reliability. Relevant cost savings can be achieved in large nuclear programmes through standardization, modularization, replication and multi-unit siting. Such measures, and the application of advanced construction techniques, which can also help to minimize the construction period, are discussed in more detail in References [2] and [3].

In particular, it is vitally important to aim at a predictable construction cost and schedule, and to have:

- efficient project management with full responsibility and authority for the project manager;

- clear division of responsibilities and appropriate allocation of financial risk among the project partners;

- avoidance of major changes in organization, project management and shift of responsibilities during the implementation of the project;

- adequate secured funding in all currencies involved;

- co-operative regulatory, utility and supplier interaction with strong feedback of experience and appropriate risk allocation;

- political and regulatory issues resolved before construction;

- straightforward and efficient licensing procedure;

- detailed design largely completed before start of construction;

- continuous control of quality, costs and schedules in all areas.

Accurate costs and a predictable schedule are vitally important to secure the financing and to make the project a success. In order to reach low generating costs, it is also important to achieve:

- low capital costs as referred to above;

- effective plant life management with the objectives of:

  - maintaining plant safety and reliability,
  - improving plant operating performance through exchange of operating experience among operators of similar plants,
  - performing plant improvements aimed at raising availability and net power output and at extending plant life,
  - improving fuel utilization (design, burnup and in-core and out-of-core management).

To be able to achieve good operating performance, it is essential that the plant is built according to high quality standards and with sufficient operating and safety margins. It is also essential to have highly qualified and well trained operation and maintenance staff and well planned operation and maintenance procedures.

These issues are discussed in more detail in Reference [4].

## REFERENCES


[1] OECD, Projected Costs of Generating Electricity (1993).

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[3] IAEA, Methods and technologies for cost reduction in the design of water cooled Reactor Power Plants, Proceedings of a Specialists Meeting, Helsinki, 3-6 Sep. 1990.

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## APPENDIX II

## DESIGN AND LICENSING STATUS

This appendix proposes a simple classification to provide some coherence to the analysis and intercomparison between the different design projects, for the purpose of the present IAEA activities in this field. The basis of this system is a set of two coordinates aimed at classifying quantitatively the status of each project. These coordinates are the design and the licensing levels, the range of each variable being 0-3. The milestones corresponding to the coordinate values are defined below. Only the integers and the intermediate number 0.5, 1, 1.5 and 2.5 are used. The integers mean that the assigned milestone is achieved (or rather essentially achieved) and the intermediate numbers would represent any situation between two of the defined milestones.

*Design Status*

Three levels or milestones for the D coordinate are defined with the following (or similar) items completed, achieved and/or established, whatever is applicable:

D=1 Conceptual design completed

* Design QA programme established
* Initial concept and plant layout
* Single line flow diagrams for reactor coolant and other main processes (power production) safety-related systems
* Essential core configuration and composition
* Specific safety features, including accident management
* Overall sizes for main components, long delivery items and buildings
* Main quantified parameters: power, flow rates, temperatures, pressures sizes etc.
* Computer code development and adaption
* Fuel cycle characteristics, if not conventional
* Identification of required R&D: materials, components, systems, tests, etc.
* Economic evaluation

D=2 *Basic design completed*

* QA program for detailed design and procurement established
* Essential R&D completed (except non critical items)
* Plant technical description
* Engineering tools
* Component conformity and principle feasibility tests
* Design criteria manual "Engineering manual"
* System descriptions for the main reactor and auxiliary systems, with piping and instrumentation diagrams developed
* Functional specifications for main components
* Plant general layout: plans, evaluations, building sizes, floor loading, and embodiments

* Basic safety studies and accident evaluation, Part of Preliminary Safety Analysis Report
* First cost estimates
* Marketing file
* Preliminary schedule for construction

D=3 *Detailed design completed*

* Construction and commissioning QA programmes established
* Complete design of the plant, except very minor items. It can be site-independent (for example, for an envelope of site conditions) or site-specific if construction has started or is considered for a specific site
* Large scale integral system tests
* The complete detailed design includes:
    - completion of equipment qualification testing
    - design/engineering for systems and components
    - detailed design, or detailed specifications, for procurement of all materials, components, systems, package units, construction/erection services etc.
    - Preliminary safety analysis report completion
    - Completion of all detailed design reviews
* Detailed construction planning
* Final cost estimate
* Final tender document

*Licensing status*

The different approaches for licensing in the individual Member States make it difficult to establish milestones with the same precise meaning. The following phases may represent a common approach to the licensing process, with values assigned for the parameter L at completion of each phase as follows:

| | Parameter L |
|---|---|
| - Preliminary licensability assessment | 1 |
| - Formal licensing application submitted | 2 |
| - Review process by regulator | 3 |
| - Permit(s) | 4 |

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