IAEA-TECDOC-1485

Status of innovative small and medium sized reactor designs 2005

Reactors with conventional refuelling schemes

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FOREWORD

There is a renewed interest in Member States in the development and application of small and medium sized reactors (SMRs). In the near term, most new NPPs are likely to be evolutionary designs building on proven systems while incorporating technological advances and often the economics of scale, resulting from the reactor outputs of up to 1600 MW(e). For the longer term, the focus is on innovative designs aiming to provide increased benefits in the areas of safety and security, non-proliferation, waste management, resource utilization and economy, as well as to offer a variety of energy products and flexibility in design, siting and fuel cycle options. Many innovative designs are reactors within the small-to-medium size range, having an equivalent electric power less than 700 MW(e) or even less than 300 MW(e). The projected timelines of readiness for deployment are generally between 2010 and 2030.

The objective of this report is to provide Member States, including those just considering the initiation of nuclear power programmes, and those already having practical experience in nuclear power, with a balanced and objective information on important development trends and objectives of innovative SMRs for a variety of uses, on the achieved state-of-the-art in design and technology development for such reactors and on their design and regulatory status.

The report is intended for many categories of stakeholders, including regulators, electricity producers, designers, non-electrical producers and policy makers.

The main chapters of this report, addressed to all abovementioned groups of stakeholders, provide a summary of major specifications, applications and user-related special features of innovative SMRs, outline the achieved design and regulatory status and its progress since previous IAEA publications, review targeted deployment dates, fuel cycle options, design approaches used to meet design objectives in specific subject areas, enabling technologies and current research and development (R&D) programmes in Member States, and non-technical factors and arrangements that could facilitate successful development and deployment of innovative reactors within the small-to-medium power range.

The annexes, intended mostly for designers and technical managers, provide detailed design descriptions of innovative SMRs under development worldwide and are patterned along a newly developed common format, which makes it possible to identify the design philosophy, objectives and approaches, as well as technical features and non-technical factors and arrangements with a potential to provide solutions in the specific areas of concern associated with future nuclear energy systems.

Detailed design descriptions in this report were prepared firsthand by the designers and reviewed, updated and approved for publication by the respective vendors, research and design organizations and academic institutions in Member States.

The scope of this report is limited to reactors with conventional, proven in operation refuelling schemes and does not include small reactors that could operate without reloading and shuffling of fuel for a long period, from 10 years and more, with no fresh or spent fuel being stored at the site during reactor operation. Because of a large number of inputs, such reactors would require a separate dedicated publication.

The IAEA officer responsible for this publication was V. Kuznetsov of the Division of Nuclear Power.

CONTENTS

1. INTRODUCTION

1.1. Background

1.1.1. Developments in Member States

In the early decades, civil nuclear power essentially borrowed from the experience of nuclear submarine reactors, which came first and were essentially small sized reactors. Since 1970s, the major focus for nuclear power was on the design and construction of nuclear plants of increasing size, with average size levelling out at about 1000 MW(e) with a tendency for further increase. This was and is generally appropriate for many industrialized countries, which could add generation capability to their electrical grids in larger increments and benefit from the construction costs reduced due to scale factor. However, it might be not appropriate for many developing countries that have small electricity grids, limited capacity for investment and less developed infrastructure.

At the time when this report was prepared (in 2005), 146 small and medium sized reactors[1] (SMRs) were operated worldwide, accounting for 61 GW(e) of electricity generation, and 12 more were under construction[2]. These were mostly earlier generation reactors still in operation and a few prototype or tests reactors, intended to support development and deployment of new larger-capacity commercial plants. Their share in worldwide nuclear electricity production was around 16.5%.

All analyses and forecasts of global energy needs project large increases in the century ahead. For example, a projection from the International Panel on Climate Change [2] indicates that primary energy demand in the world may double by 2050, see Fig. 1.

Many analyses reach the conclusion that nuclear energy has a strong role to play, particularly if the goal is truly sustainable development[3], not just temporary economic growth. For example, the IPCC projection [2] indicates a 2.5 times median growth of nuclear power by 2030, see Fig. 2. The IAEA's Nuclear Technology Review (2005 update) [5] projects a 34% to 86% increase of nuclear generation by the year 2030.

The trends in the world at large contributing to increased expectations of the future role of nuclear power, as identified in the Medium Term Strategy of the IAEA [6], are the following:

- The use of nuclear technologies in developing countries is growing as local infrastructures improve and technology transfer increases;
- As the demand for electricity continues to increase and the drive for sustainable development gains momentum, the need to exploit energy sources with limited environmental impacts (in particular to meet commitments made in connection with the Kyoto Protocol) could revitalize the nuclear power option.

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[1] According to the classification adopted by the IAEA, small reactors are reactors with the equivalent electric power less than 300 MW, medium sized reactors are reactors with the equivalent electric power between 300 and 700 MW.

[2] These data as of June 2005 are derived from the IAEA's Power Reactor Information System [1].

[3] According to the definition adopted by the Generation-IV International Forum [4], "sustainability is the ability to meet the needs of the present generation while enhancing the ability of future generations to meet society's needs indefinitely into the future". According to the definition adopted by the IAEA's International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) [3], sustainable development is the "development that meets the needs of the present without compromising the ability of future generations to meet their own needs".

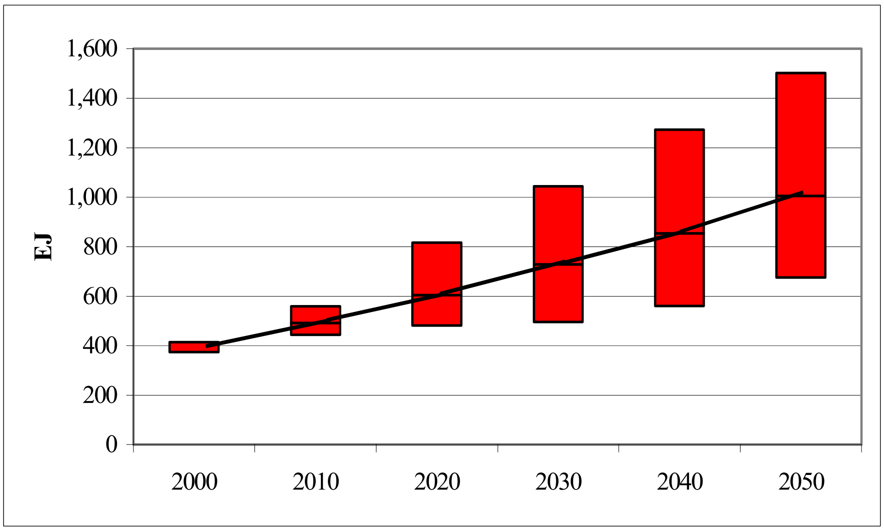


FIG. 1. Range of future primary energy demand in SRES scenarios, 2000–2050. Solid line represents median [2, 3].

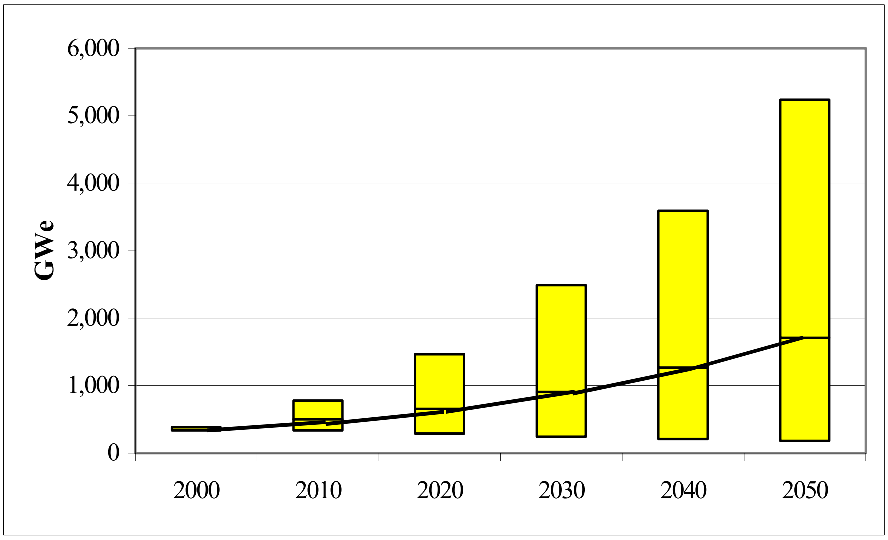


FIG. 2. Range of nuclear power in SRES scenarios, 2000–-2050. Solid line represents median [2, 3].

The principal drivers behind projected large increase in global energy needs are population growth and economic development in today's developing countries [2]; therefore, in defining pathways for nuclear power it is important to address the specific needs of such countries. These needs may vary between different groups of developing countries, but are often defined by weak electricity grids and insufficient infrastructure [7]. Certain areas in some developing countries suffer from the deficiency of potable water [8]. Legal, institutional and human resource provisions for nuclear power are in many cases insufficient [9]. Many developing countries suffer from corruption and poverty, which fosters political instability and makes them an attractive domain for international terrorism. A transfer of traditional nuclear power and, especially, nuclear fuel cycle technologies to such countries would apparently pose a proliferation risk.

At present, there is no general consensus on the future role SMRs. A balanced view is that SMRs are an option, not a universally best option that will suit in all cases. In a longer term, similar view generically applies also to large-capacity reactors.

Large utilities with a big grid size will still favour large units for reasons of the economy of scale. In order to cope with economics of scale, SMRs have to incorporate specific design features that result into reduced complexity of the overall plant design, modularization and mass production.

On the other hand, many developing countries have small electricity grids and limited turnover of capital in the energy market, which means that under liberalized energy markets SMRs may become the only affordable nuclear power option for such countries. It's the absolute and not specific overnight capital cost of the plant that matters in this case.

In industrialized countries, the market deregulation and resulting competition drives the utilities toward shorter time of capital recovery and lower financial risks, which could perhaps be achieved by enabling the incremental capacity addition to a network that would match the incremental increase of demand. On the other hand, an essential simplification of plant operating and maintenance requirements may be requested to justify for, say, the reduced number of control rooms at a site with several modular reactors.

Some SMRs offer the possibility of very long core lifetimes with reduced core power density, burnable absorbers or high conversion ratio in the core. An infrequent refuelling interval may provide certain guarantees of sovereignty for those countries that have a less developed infrastructure and would prefer to lease fuel rather than master an autonomous fuel cycle [7]. SMRs are also the preferred option for near-term (desalination of seawater or district heating) and advanced (e.g. hydrogen production) process heat applications [7].

Some countries such as Canada, the USA and the Russian Federation have low populated Far North areas with severely cold climatic conditions and permanent frost, complicating the transport of fossil fuels and resulting in the cost of energy being several times higher than in the rest of the country. Small autonomous nuclear reactors for electricity generation and district heating are viewed as a secure and perhaps competitive option of energy supply in such regions [7].

Last but not least, reflecting on the experiences of other industries, such as an aircraft or a motorcar industry as well as electricity production from fossil fuels, an observation could be made that 'one size fits for all' approach never works.

More than 50 concepts and designs of innovative[4] SMRs are under development in more than 15 IAEA Member States representing both industrialized and developing countries [7].

Innovative SMR designs are under development for water cooled, gas cooled, liquid metal cooled and molten salt cooled reactor lines, as well as some non-conventional combinations thereof. The targeted timelines of readiness for deployment vary between 2010 and 2030; the major concerns addressed by the innovation cover a broader spectrum of subject areas as compared to the operating and near term evolutionary NPPs, see Table 1. Such extended consideration is apparently due to the anticipated growth and geographical expansion of nuclear power.

[4] The IAEA-TECDOC-936 [13] defines an innovative design as the design "that incorporates radical conceptual changes in design approaches or system configuration in comparison with existing practice" and would, therefore, "require substantial R&D, feasibility tests and a prototype or demonstration plant to be implemented".

TABLE 1. SUBJECT AREAS FOR INNOVATIVE SMRs [7]

SUBJECT AREAS CONSIDERED BY DESIGNERS OF INNOVATIVE SMRS	SOLUTIONS PROPOSED (EXAMPLES)
Economics and maintainability	Incremental capacity increase through modular approach; design standardization and mass production; reduced design complexity and simplified operational requirements achieved through strong reliance on passive safety design options.
Safety and reliability	Strong reliance on passive safety design options: inherent and passive safety features, reliable passive systems; finding an effective combination between passive and active systems, etc.
Proliferation resistance and physical protection	Broader reliance on intrinsic proliferation-resistance features, specifically, to facilitate implementation of the extrinsic measures such as safeguards.
Resources, waste management and environmental impacts	Improved fuel utilization, e.g. achieved through higher fuel burn-up or recycling; clearly defined strategies of spent nuclear fuel and waste management; design features to reduce off-site emergency planning.
Applications	Generation of electricity, district heating, production of potable water, hydrogen etc; various cogeneration options, including purposeful use of the reject heat.
User-related special features	Modular approach to reactor design and/or fabrication and construction; transportability (e.g. floating NPP options); infrequent refuelling; flexible design, siting and applications.
Fuel cycle options	Flexible fuel cycle options, e.g. once-through use or recycling of U, Pu and Th fuel; centralized (e.g. regional) fuel cycle services.
Enabling technologies	An enabling technology is the technology that needs to be developed and demonstrated to make a certain reactor concept viable. Upon a diversity of SMR designs, it may be useful to identify enabling technologies that are common to several SMR designs and, therefore, could benefit from being developed on a common or shared basis.
Marketing strategy and deployment scenarios	Tailoring designs to specific market needs; considering SMRs as part of innovative nuclear energy systems; figuring out deployment strategies on the basis of dynamic system simulations.
Non-technical factors and arrangements to facilitate deployment	International cooperation; infrastructure developments to support centralized fuel cycle services or NPP leasing; guarantees of sovereignty to countries that would prefer to lease fuel; reciprocity of licensing/ design certification regimes between countries; simplified licensing procedures, e.g. license-by-test and reduced or eliminated off-site emergency planning.

Reflecting on the developments in Member States, the IAEA's Medium Term Strategy [6] defines one of the IAEA's objectives as the following:

"Providing a forum for, and encouraging, the review of developments associated with new nuclear power and fuel cycle technologies, including:

- Small and medium size reactors for electricity generation and heat production.
- Co-generation and heat applications, including seawater desalination.
- New technological developments relevant to competitiveness, safety and efficiency.
- Improving proliferation resistance in reactors and associated fuel cycles.
- Reduction of radioactive waste arisings."

1.1.2. Previous IAEA publications

Upon the advice and with the support of IAEA Member States, the IAEA provides a forum for the exchange of information by experts and policy makers from industrialized and developing countries on the technical, economic, environmental, and social aspects of SMRs development and implementation in the 21st century, and makes this information available to all interested Member States by producing status reports and other publications dedicated to advances in SMR technology [10, 11].

In the field of SMRs, the last status report published was IAEA-TECDOC-881 [10], issued in 1995. Since that time many developments took place; for some designs the development activities have resulted in a significant progress towards detailed design and licensing, while for the others development activities for whatever reasons have been stopped. Many new developments for innovative SMRs have originated and progressed since that time.

A more recent review of the progress of evolutionary and innovative SMRs, issued in 2001, can be found in [11]. It is noted that this publication belongs to the category of proceedings and, therefore, provides the summaries and descriptions of SMRs in an unevenly structured form, generally as found appropriate by the authors.

In 2001–2002, the International Energy Agency (IEA), the OECD Nuclear Energy Agency (OECD/NEA) and the IAEA have been conducting a joint project to examine R&D needs on the innovative nuclear fission reactor technologies and to explore the potential for enhanced international collaboration in developing these technologies. This project, called "Three Agency Study" [12], also intended to highlight how new reactor designs are addressing the issues currently rated as critical for further deployment of nuclear power and, to this end, it has defined an approach partially similar to that used in the present report. At the same time, the "Three Agency Study" was neither a dedicated publication on SMRs nor presented their technical descriptions as detailed as provided for by a status report.

With these developments in mind, the IAEA recommended preparation of a new status report on innovative SMRs, with a focus on their potential to provide solutions in the specific areas of concern associated with future nuclear energy systems. To support the preparation of this report, an IAEA technical meeting on Innovative Small and Medium Sized Reactors: Design Features, Safety Approaches and R&D Trends was held on 7–11 June 2004 in Vienna, and its final report was published as IAEA-TECDOC-1451 in May 2005 [7]. This TECDOC presents a variety of innovative water cooled, gas cooled, liquid metal cooled and non-conventional SMR designs developed worldwide and examines the technology and infrastructure development needs that may be common to several concepts or lines of such reactors. Both, the technical meeting and the IAEA-TECDOC-1451 provided recommendations on the objectives, structure, scope and content of this report.

1.2. Objectives

The general objective of this report is to provide Member States, including those just considering the initiation of nuclear power programmes, and those already having practical experience in nuclear power, with a balanced and objective information on important development trends and objectives of innovative SMRs for a variety of uses, on the achieved state-of-the-art in design and technology development for such reactors and on their design and regulatory status.

The specific objectives of this report are the following:

(1) Through direct cooperation with the designers in Member States, to define, collate and present the state-of-the art in the design objectives, design approaches and technical features of innovative SMRs making a focus on their potential to provide solutions in the following subject areas, important for future nuclear energy systems:

- Economics and maintainability;
- Safety and reliability;
- Proliferation resistance and physical protection;
- Resource utilization, waste management and environmental impacts;
- Fuel cycle options;
- Applications;
- User-related special features;
- Enabling technologies; and, when possible
- Marketing strategy and deployment scenarios.

(2) To identify non-technical factors and arrangements that could facilitate development and deployment of innovative SMRs;

(3) To provide a technical and information background to assist the designers of innovative SMRs in defining consistent design strategies regarding the selected subject areas;

(4) To provide various categories of stakeholders in Member States, including regulators, electricity producers, designers, non-electrical producers and policy makers, with a balanced and objective summary information on the application potential, development status and prospects of innovative SMRs;

(5) To provide an information support to high-level technical managers and policy makers in Member States who are planning to assess innovative SMRs projects with a potential of deployment between 2010 and 2030.

1.3. Scope

The structure and scope of this report were defined through a series of consultants and technical meetings [7], with the support from the IAEA Technical Working Groups (TWGs) on advanced water-cooled, gas-cooled, and fast reactors and accelerator driven systems, SMR designers in Member States and the International Coordinating Group (ICG) of the IAEA's International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO).

No limits were set regarding innovative SMR types, so that the report includes design descriptions and summaries of water-cooled, gas-cooled, liquid metal cooled, and liquid salt cooled designs. The upper limit for targeted deployment dates was set at roughly 2050, and it was generally accepted that some innovative SMRs might be just at the conceptual or even pre-conceptual design stage. Regarding the objectives of this report, which addresses

innovative SMRs targeted for deployment between 2010 and 2050, bringing out as many design approaches as possible was rated useful to foster their further adjustment, modification, merging and transformation and, perhaps, the origination of new concepts and designs of innovative SMRs that might better fit the requirements to future nuclear energy systems, e.g. [3, 4].

It was agreed that, in order to be present in the report, an SMR concept should be clearly defined in main parameters and supported by at least a preliminary set of analyses in major areas. Another requirement was that an innovative SMR concept should be 'alive'. With an understanding that many efforts for innovative reactors are nowadays financed unevenly, the word 'alive' was used to identify that at the time when this report was prepared, there was at least a research team with particular R&D experience willing to continue design and technology development for a given innovative SMR concept.

To collect information on innovative SMRs, a new common outline for design descriptions was developed, which provides for a structured description of the features and anticipated performance of an innovative SMR in all considered subject areas, see Appendix I. Reflecting on the fact that some innovative SMRs may be at a design stage too early to provide all data requested, a shorter version of the outline was developed also, see Appendix II.

The designers in Member States were then contacted with an offer of participation in this report and informed about its objectives and the approach to be used in its preparation. Specifically, the designers were informed about the adopted definition of innovative designs (see Chapter 2 and [13]) and the design description outlines to be applied.

In response to the abovementioned activity, 54 descriptions of innovative SMR designs were collected from Member States by mid 2005. Of them, 26 innovative SMR concepts presented reactors with the so-called conventional, i.e. proved-in-operation refuelling schemes, such as refuelling in batches, or on-line refuelling, or pebble transport. Twenty-eight descriptions presented small reactors without on-site refuelling, i.e. the reactors that can operate without reloading and shuffling of fuel for a reasonably long period, from 5 to 30 years and more, with no fresh or spent fuel being stored at the site during reactor operation. For such reactors, whole core refuelling is performed once-at-a-time, either at a factory to which the reactor is transported after its fuel lifetime expires, or at the site with the use of special remote refuelling equipment that is moved away immediately after the refuelling is accomplished.

Small reactors without on-site refuelling, defined and addressed in more detail in [7], have a common incentive, which is to provide certain guarantees of sovereignty to those countries that would prefer to lease fuel from a foreign vendor or, perhaps, an international fuel cycle centre. Small reactors without on-site refuelling have common technology and infrastructure development issues related to the provision and qualification of reliable long-lived core operation, in-service reactor inspection and maintenance, and whole core remote refuelling and transportation. The timeline of readiness for deployment is generally more remote for such reactors. Because of a large overall number of inputs, it was decided to limit the scope of this report by innovative SMRs with conventional refuelling schemes. Small reactors without on-site refuelling could then be addressed in future IAEA publications.

In the abovementioned way, twenty-four full and 2 short design descriptions of innovative SMRs with conventional refuelling schemes were selected for this report. All design descriptions, included as Annexes I through XXVI, were reviewed and approved for publication by their respective designers (representatives of vendors, research and design organizations and academic institutions in Member States).

Since the design description outline provides for a justification of why a particular SMR could be rated as innovative (see the definition of an innovative design in Chapter 2), the decision

on whether to include a certain SMR in this report was generally left up to the designers, who also provided the corresponding justifications.

1.4. Structure

The report includes an introduction, 6 chapters, 3 Appendices and 26 Annexes.

The introduction (Chapter 1) describes the background and identifies the objectives, the scope and the structure of this report.

Chapter 2 collates the definitions and terms used in this report. The outlines (formats) used in the preparation of design descriptions for this report as enclosed as Appendices II and I.

Chapter 3 gives a summary table of SMR concepts and designs addressed in this report, and also identifies the designs that were not included. For SMR designs addressed in this report, this chapter provides a review of:

- Major specifications and applications.
- User-related special features.
- Targeted deployment dates.
- Achieved design and regulatory status and its progress since previous IAEA publications.
- Fuel cycle options.
- The sources of additional information.

Chapter 4 summarizes the design approaches and technical features that define innovative SMRs' performance in the subject areas of economics, sustainability (resource utilization, waste management, environmental impacts), safety, proliferation resistance and physical protection. Short reviews in this section, based on the information provided in the Annexes, are structured according to the reactor types.

Chapter 5 reviews the enabling technologies for innovative SMRs with a focus on further necessary R&D. The crosscut tables of ongoing and planned R&D and a crosscut table of the available and planned test facilities are given in Appendix III.

Chapter 6 summarizes the non-technical factors and arrangements that, in view of designers, could facilitate successful development and deployment of innovative SMRs, and reviews the deployment strategies as outlined in the design descriptions of Annexes I–XXVI.

As a conclusion, Chapter 7 provides a review of the on-going programmes for innovative SMR development in Member States.

Annexes I–XXVI present the contributions from Member States — structured design descriptions of the innovative water cooled, gas cooled, liquid metal cooled and non-conventional SMRs.

2. DEFINITIONS, TERMS AND FORMATS FOR SMR DESIGN DESCRIPTION

2.1. Definitions and terms

2.1.1. Small and medium sized reactors (SMRs)

According to the classification currently in use in the IAEA, small reactors are reactors with the equivalent electric power less than 300 MW, medium sized reactors are reactors with the equivalent electric power between 300 and 700 MW [7, 14].

2.1.2. Innovative design

IAEA-TECDOC-936 "Terms for describing new, advanced nuclear power plants" [13] defines an *advanced design* as a "design of current interest for which improvement over its predecessors and / or existing designs is expected. Advanced designs consist of evolutionary designs and designs requiring substantial development efforts".

Evolutionary design is an advanced design that "achieves improvements over existing designs through small to moderate modifications, with a strong emphasis on maintaining design proveness to minimize technological risks".

Innovative design is a design "that incorporates radical conceptual changes in design approaches or system configuration in comparison with existing practice" and would, therefore, "require substantial R&D, feasibility tests and a prototype or demonstration plant to be implemented".

2.1.3. Safety related terms

Definitions from IAEA safety standards

The formats (outlines) for SMR design description, provided in Section 2.2, were developed keeping in mind the following consensus definitions taken from the IAEA safety standard NS-R-1 [15]:

ACTIVE COMPONENT. A component whose functioning depends on an external input such as actuation, mechanical movement or supply of power.

PASSIVE COMPONENT. A component whose functioning does not depend on an external input such as actuation, mechanical movement or supply of power.

Plant equipment (see Fig. 3).

SAFETY SYSTEM. A system important to safety, provided to ensure the safe shutdown of the reactor or the residual heat removal from the core, or to limit the consequences of anticipated operational occurrences and design basis accidents.

PROTECTION SYSTEM. System which monitors the operation of a reactor and which, on sensing an abnormal condition, automatically initiates actions to prevent an unsafe or potentially unsafe condition.

Plant states (see Fig. 4).

NORMAL OPERATION. Operation within specified operational limits and conditions.

POSTULATED INITIATING EVENT. An event identified during design as capable of leading to anticipated operational occurrences or accident conditions.

ANTICIPATED OPERATIONAL OCCURRENCE. An operational process deviating from normal operation which is expected to occur at least once during the operating lifetime of a facility but which, in view of appropriate design provisions, does not cause any significant damage to items important to safety or lead to accident conditions.

ACCIDENT CONDITIONS. Deviations from normal operation more severe than anticipated operational occurrences, including design basis accidents and severe accidents.

DESIGN BASIS ACCIDENT. Accident conditions against which a nuclear power plant is designed according to established design criteria, and for which the damage to the fuel and the release of radioactive material are kept within authorized limits.

SEVERE ACCIDENTS. Accident conditions more severe than a design basis accident and involving significant core degradation.

ULTIMATE HEAT SINK. A medium to which the residual heat can always be transferred, even if all other means of removing the heat have been lost or are insufficient.

SINGLE FAILURE. A failure which results in the loss of capability of a component to perform its intended safety function(s), and any consequential failure(s) which result from it.

COMMON CAUSE FAILURE. Failure of two or more structures, systems or components due to a single specific event or cause.

SAFETY FUNCTION. A specific purpose that must be accomplished for safety.

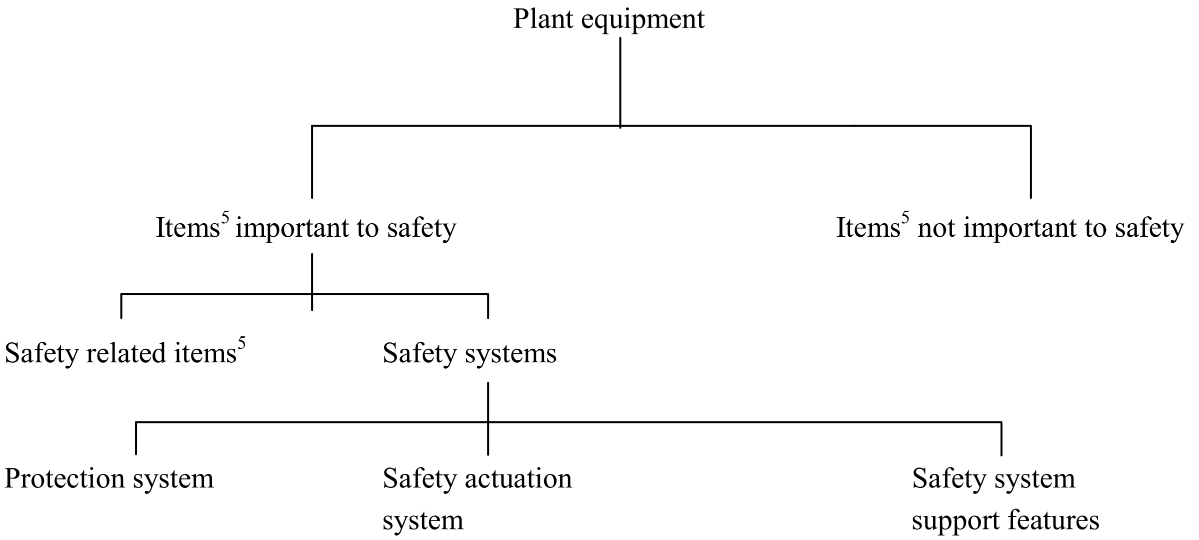
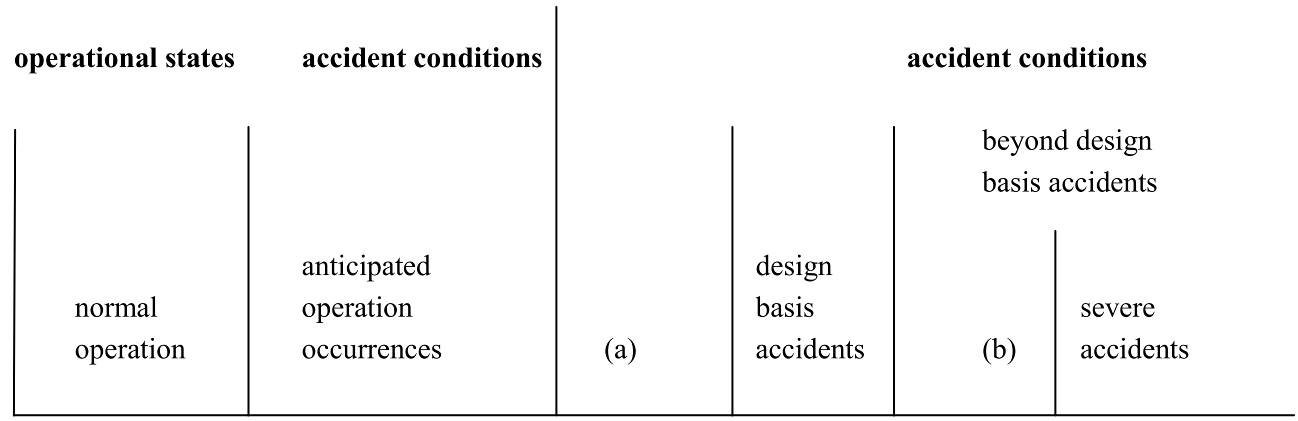


FIG. 3. Plant equipment [15].



(a) Accident conditions which are not explicitly considered design basis accidents but which they encompass;
(b) Beyond design basis accidents without significant core degradation.

FIG. 4. Plant states [15].

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[5] In this context, an 'item' is a structure, system or component [15].

Non-consensus definitions from IAEA TECDOCs

At the moment, the IAEA safety standards do not provide a complete set of definitions necessary for the description of safety features of NPPs with innovative reactors. In view of this, some missing definitions related to passive safety features could be taken from IAEA-TECDOC-626 [17]:

INHERENT SAFETY CHARACTERISTIC. Safety achieved by the elimination of a specified hazard by means of the choice of material and design concept.

PASSIVE COMPONENT. A component, which does not need any external input to operate.

PASSIVE SYSTEM. Either a system which is composed entirely of passive components and structures or a system which uses active components in a very limited way to initiate subsequent passive operation.

GRACE PERIOD. The grace period is the period of time during which a safety function is ensured without the necessity of personnel action in the event of an incident/accident.

Recommendations from International Nuclear Safety Advisory Group (INSAG)

Although the IAEA safety standard NS-R-1 [15] provides a consensus definition of the defence in depth levels, the definitions suggested in INSAG-10 [18] may better suit for NPPs with innovative reactors. For the future reactors, reference [18] envisages the following trends of the different levels of defence in depth:

"— Level 1, for the prevention of abnormal operation and failures is to be extended by considering in the basic design a larger set of operating conditions based on general operating experience and the results of safety studies. The aims would be to reduce the expected frequencies of initiating failures and to deal with all operating conditions, including full power, low power and all relevant shutdown conditions.

— Level 2, for the control of abnormal operation and the detection of failures, is to be reinforced (for example by more systematic use of limitation systems, independent from control systems), with feedback of operating experience, an improved human-machine interface and extended diagnostic systems. This covers instrumentation and control capabilities over the necessary ranges and the use of digital technology of proven reliability.

— Level 3, for the control of accidents within the design basis, is to consider a larger set of incident and accident conditions including, as appropriate, some conditions initiated by multiple failures, for which best estimate assumptions and data are used. Probabilistic studies and other analytical means will contribute to the definition of the incidents and accidents to be dealt with; special care needs to be given to reducing the likelihood of containment bypass sequences.

— Level 4, for the prevention of accident progression, is to consider systematically the wide range of preventive strategies for accident management and to include means to control accidents resulting in severe core damage. This will include suitable devices to protect the containment function such as the capability of the containment building to withstand hydrogen deflagration, or improved protection of the basemat for the prevention of melt-through.

— Level 5, for the mitigation of the radiological consequences of significant releases, could be reduced, owing to improvements at previous levels, and especially owing to reductions in source terms. Although less called upon, Level 5 is nonetheless to be maintained."

Terms to be avoided

The designers were not requested to adjust safety related terminology of their projects accordingly when preparing the design descriptions for this report; they had rather followed the definitions accepted in their respective Member States. However, in line with the recommendations of [13] and upon the approval from designers, terms such as "revolutionary design", "passive, simplified and forgiving design", "inherently safe design", "deterministically safe design", "catastrophe free design" etc. were edited out from design descriptions, except for the cases when they appear in the names of certain innovative SMRs.

2.1.4. Proliferation resistance related terms

The terms and definitions used in the design description outline correspond to reference [16]:

- *Proliferation resistance* is that characteristic of a nuclear energy system that impedes the diversion or undeclared production of nuclear material, or misuse of technology, by States in order to acquire nuclear weapons or other nuclear explosive devices.
- The degree of proliferation resistance results from a combination of, inter alia, technical design features, operational modalities, institutional arrangements and safeguards measures.
- *Intrinsic proliferation resistance features* are those features that result from the technical design of nuclear energy systems, including those that facilitate the implementation of extrinsic measures.
- *Extrinsic proliferation resistance measures* are those measures that result from States' decisions and undertakings related to nuclear energy systems."

2.1.5. Design and regulatory status

The following simple classification, defined in [10], illustrates the design stages of an NPP project. Bullets indicate the items to be defined, determined, completed, achieved or established (whatever is applicable) within each stage.

(1) Conceptual design stage:

- Initial concept and plant layout.
- Single line flow diagrams for reactor coolant and other main processes (power production), and 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, verification and validation.
- Fuel cycle characteristics, if not conventional.
- Identification of required R&D: materials, components, systems, tests, etc.
- Economic evaluation.
- Design QA programme.

(2) Basic (in some Member States – preliminary) design stage:

- 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 (PSAR).
- First cost estimates.
- Marketing file.
- Preliminary schedule for construction.
- QA program for detailed design and procurement.

(3) Detailed design stage:

- Complete design of the plant, except very minor items. It can be unified (for example, for an envelope of site conditions) or site-specific.
- Large scale integral system tests.
- Equipment qualification testing.
- Design/engineering for systems and components
- Detailed specifications for procurement of all materials, components, systems, package units, construction/erection services, etc.
- Preliminary safety analysis report.
- Detailed design reviews.
- Detailed construction planning.
- Final cost estimate.
- Final tender document.
- Construction and commissioning QA programmes.

The abovementioned classification is given as a reasonable example. The designers were not requested to adjust the design stages of their projects accordingly when preparing design descriptions for this report; they had rather followed the patterns established in their respective Member States.

Regarding the *regulatory status*, different approaches for licensing in the individual Member States make it difficult to establish milestones with precise meaning [10]. The following phases may represent a common approach to the licensing process:

- Licensing pre-application submitted.
- Preliminary licensability assessment by regulator.
- Formal licensing application submitted.
- Review process by regulator.
- Permit(s) or licenses issued.

Regarding innovative reactors, an important observation is that R&D on key enabling technologies is often started before completion of the conceptual design stage, to obtain the proofs of technological feasibility at as early stage as possible. Licensing pre-application at early design stages is rated useful by several designers of innovative SMRs [7] to establish good working relations with the regulator and secure more time for the resolution of the regulatory problems associated with a certain innovative design.

2.1.6. Enabling technologies

The enabling technology is a technology that needs to be developed and demonstrated to make a certain reactor concept viable [7]. Within this report, the term 'enabling technology' is used in a broad sense; for example, it could denote the technology of a particular system, structure or component as well as a combination of design approaches used to ensure inherent or passive safety features or high economic competitiveness of a certain innovative design. Calculation technologies and data sets necessary for validation of SMR performance also fall under this definition.

2.2. Formats for SMR design description

The formats (outlines) used in the preparation of full and short design descriptions of innovative SMRs for this report are enclosed as Appendices I and II, respectively.

3. SUMMARY OF INNOVATIVE SMRs

3.1. Innovative SMRs included in this report

Of the twenty-six concepts and designs addressed, 13 (50%) are water cooled SMRs, 6 (23%) are gas cooled SMRs–high temperature gas cooled reactors (HTGRs), 6 are sodium or lead-bismuth cooled fast reactors, and 1 is a non-conventional very high temperature reactor concept, a liquid salt cooled reactor with HTGR type prismatic fuel.

For all innovative SMRs addressed in this report, Table 2 provides a summary of the user-related features, the targeted deployment dates, the achieved design and regulatory status and its progress since previous IAEA publications, identifies fuel cycle options and specifies the recommended sources of additional information.

A short review of these data, structured according to the reactor types, is provided in Sections 3.1.1–3.1.4 below.

3.1.1. Water cooled SMRs

Reactor types

Of the thirteen water cooled SMR designs, 6 are pressurized water reactors — PWRs (SMART, IRIS, CAREM, MARS, SCOR, and VBER-300); 3 are boiling water reactors — BWRs (VK-300, CCR, and RMWR); one is an innovative light water reactor, which could be classified as an indirect cycle BWR or, alternatively, a PWR with coolant boiling in the upper part of the core (IMR); one is a light water cooled heavy water moderated pressure tube reactor (AHWR); one is pool type light water reactor with non-pressurized primary circuit (RUTA-70); and one is a direct flow pool type pressure tube light water reactor with the separated fuel elements and pressure tubes embedded in graphite blocks to provide a separation from the pool water (KAMADO).

(Text continues on page 29)

TABLE 2. SUMMARY TABLE OF INNOVATIVE SMR DESIGNS WITH CONVENTIONAL REFUELLING SCHEME

SMR-name	**SMART**	**IRIS**
Member State (MS)	ROK	USA
Company / Institution	KAERI, MOST	International consortium (10 countries) lead by Westinghouse (USA)
Reactor type	PWR	PWR
Reactor capacity, thermal / electric (MW)	330 / 90	1000 / 335
Reactor style	Integral type	Integral type
Reactor core outlet temperature (^oC)	310	330
Primary circuit pressure (MPa)	15	15.5
Thermodynamic cycle	Indirect	Indirect
Applications	- Electricity (90 MW(e)) plus - Potable water (40 000 t/day)	- Electricity plus potable water; or - Electricity plus district heating; or - Electricity plus process heat generation
Special features	- Increased operation cycle (4 years) - Modular approach to construction - Pre-fabrication of components	- Floating, e.g., barge-mounted NPP option - Shop fabrication of reactor components - Incremental capacity addition (modules) - Simplified operational licensing requirements (possible) - Extended operation cycle (30 – 48 months in near term; up to 96 months in mid term) - Licensing with reduced or eliminated off-site emergency planning zone (possible) - A reduced power (~50 MW(e)) version with full natural circulation, no on-site refuelling, for deployment in remote locations as an autonomous energy source, with more than 15 years core lifetime
Fuel cycle option, basic	Once-through / low enrichment U dioxide fuel	Once-through / low enrichment U dioxide fuel
Fuel cycle options, alternative	-	Closed fuel cycle with regional or centralized reprocessing of U or MOX fuel
Design and regulatory status 1995*	-	-
Design and regulatory status 2005	- Conceptual design completed in 1999 - Basic design with desalination completed in 2002	- Preliminary design (2002-2005) - Licensing pre-application completed (spring 2006) - Final design approval 2010
Targeted deployment date 1995*	-	-
Targeted deployment date 2005	- 1/5th prototype – 2008 - Commercialization of SMART desalination plant – beginning in 2009	First-of-a-kind deployment: 2012-2015 (near term)
Design description in this TECDOC	Annex I	Annex II
Design description in other TECDOCs	IAEA-TECDOC-1391 (May 2004), p. 680	IAEA-TECDOC-1391 (May 2004), p. 581
Similar or relevant SMRs	CAREM, IRIS, MRX	- IRIS shares some design features within the Integral Primary System Reactors (IPSR) group of the International Near-Term Deployment (INTD)/ GIF - IRIS shares with INPRO [3] the goal of revising the emergency planning requirements

* Here and after, with reference to IAEA TECDO-C-881

2. SUMMARY TABLE (Continued-1)

SMR-name	**CAREM-25 [CAREM-300]**	**MARS (Italy)**
Member State (MS)	Argentina	Italy
Company / Institution	CNEA, INVAP	University of Rome “La Sapienza”, ENEA
Reactor type	PWR	PWR
Reactor capacity, thermal / electric (MW)	100 / 27 [900 / 300]	600 / 150
Reactor style	Integral type	Modular, loop type
Reactor core outlet temperature (°C)	326	254
Primary circuit pressure (MPa)	12.25	7.5
Thermodynamic cycle	Indirect	Indirect
Applications	- Electricity - Seawater desalination (CAREM could supply heat or electric energy for a reverse osmosis desalination plant)	- Electricity plus potable water; or - Electricity plus district heating; or - 80 - 100 MW(e) of electricity and water or steam at ~ 100°C for food industry (conservation industry)
Special features	- Incremental capacity addition (modules) - Floating, e.g., barge-mounted NPPs could be considered - Potential for design standardization, series production and shop fabrication of equipment	- Shop fabrication of reactor components - Incremental capacity addition (modules)
Fuel cycle option, basic	Once-through / low enrichment U dioxide fuel	Once-through/ low enrichment uranium dioxide fuel
Fuel cycle options, alternative	Closed U or MOX fuel cycle with reprocessing of fuel by advanced PUREX process	Innovative fuel forms could be accommodated
Design and regulatory status 1995	- Basic design completed - Preliminary SAR available and sent to Regulatory Authority. No formal licensing application submitted yet	- Development of the project started in 1984 - Preliminary Safety Report completed in 1994
Design and regulatory status 2005	- Conceptual engineering for a 27 MW(e) prototype reactor is completed - Detailed design stage - Licensing pre-application made	- Basic design to be finished by the end of 2006 - Final design for a prototype could be finished by the end of 2008
Targeted deployment date 1995	Ready for construction in 1995	-
Targeted deployment date 2005	Next step is the construction of a 27 MW(e) engineering prototype; exact date not provided	Construction of a prototype plant possible in 2012
Design description in this TECDOC	Annex III	Annex IV
Design descriptions in other TECDOCs	IAEA-TECDOC-1391 (May 2004), p. 660	IAEA-TECDOC-881 (1996), p. 470
Similar or relevant SMRs	IMR, IRIS, SMART CAREM has been recognized as an International Near Term Deployment (INTD) reactor by the Generation IV International Forum (GIF)	-

2. SUMMARY TABLE (Continued-2)

SMR-name	**SCOR**	**IMR**
Member State (MS)	France	Japan
Company / Institution	CEA	Mitsubishi Heavy Industries, Kyoto University, CRIEPI
Reactor type	PWR	Light water reactor (LWR)
Reactor capacity, thermal / electric (MW)	2000 / 630	1000 / 350
Reactor style	Integral type	Integral type, modular
Reactor core outlet temperature (^oC)	285.4	345
Primary circuit pressure (MPa)	8.8	15.5
Thermodynamic cycle	Indirect	Indirect
Applications	Mainly electricity; but may be used for seawater desalination	- Electricity production - General capability for co-generation, but design work for these applications has not yet been accomplished
Special features	Simplified operational licensing requirements	- Shop fabrication of reactor building - Shop fabrication of reactor components - Incremental capacity addition (modules) - Extended fuel cycle 790 EFPD**
Fuel cycle option, basic	Once-through/ low enrichment uranium dioxide fuel	Once-through/ low enrichment uranium dioxide fuel
Fuel cycle options, alternative	Closed fuel cycle (U, MOX)	Closed fuel cycle (U, MOX); fuel reprocessing by aqueous methods; pyro-metallurgical reprocessing with low decontamination factors could be examined at later stages
Design and regulatory status 1995	-	-
Design and regulatory status 2005	Conceptual design started in 2000	-Conceptual design to be completed in 2005 - Target to start licensing is 2011 at the earliest
Targeted deployment date 1995	-	-
Targeted deployment date 2005	In the next 15 years	After 2011
Design description in this TECDOC	Annex V	Annex VI
Design descriptions in other TECDOCs	-	IAEA-TECDOC-1391 (May 2004), p. 715
Similar or relevant SMRs	IRIS, CAREM, MARS (Italy) The concept has been developed in line with the European utility requirements for enhanced reliability and safety and improved economics for next generation reactors	-

** EFPD - effective full power days

2. SUMMARY TABLE (Continued-3)

SMR-name	**VBER-300**	**VK-300**
Member State (MS)	Russian Federation	Russian Federation
Company / Institution	OKBM, Atomenergoproekt, RRC-KI, Lazurit	RDIPE, RRC-KI, IPPE
Reactor type	PWR	BWR
Reactor capacity, thermal / electric (MW)	850 / 295	750 / 250
Reactor style	Modular, loop type	Integral type, monolithic
Reactor core outlet temperature (^{o}C)	332	284.5
Primary circuit pressure (MPa)	15.7	6.86
Thermodynamic cycle	Indirect	Direct
Applications	- Potable water; or - Electricity plus potable water; or - Electricity plus district heating	- Electricity production; or - 150 MW(e) electricity plus up to 400 GCal/h for district heating - Desalination option could be considered
Special features	- Floating, e.g., barge-mounted NPPs; or - Land based cogeneration plants - Increase of capacity by increasing the number of primary system loops (two- or three-loop NPP layout)	Nuclear desalination complex using the afloat construction technology is considered
Fuel cycle option, basic	Once-through/ low enrichment uranium dioxide fuel	The same as in standard VVER-1000
Fuel cycle options, alternative	Radkowsky Thorium Fuel (RTF): once-through fuel cycle with heterogeneous fuel layout; bimetallic enriched uranium (<20%) seed subassemblies; thorium dioxide blanket subassemblies	-
Design and regulatory status 1995	-	-
Design and regulatory status 2005	- Preliminary design completed in 2002 - Detailed design development, including licensing, could be completed within 3 years	- Detailed design of VK-300 reactor for a cogeneration plant completed - Basic regulations for a cogeneration plant using VK-300 prepared - Substantiation of investments into construction of a nuclear cogeneration plant with the VK-300 in the Arkhangelsk region of the Russian Federation prepared
Targeted deployment date 1995	-	-
Targeted deployment date 2005	Plant construction (including licensing and major demonstrations): - Nuclear cogeneration plant - within 8 years; - Floating NPP – within 7 years	First power unit planned for 2012
Design description in this TECDOC	Annex VII	Annex VIII
Design descriptions in other TECDOCs	IAEA-TECDOC-1391 (May 2004), p. 612	IAEA-TECDOC-1391 (May 2004), p. 568
Similar or relevant SMRs	-	-

** EFPD is for effective full power days

2. SUMMARY TABLE (Continued-4)

SMR-name	**CCR**	**RMWR**
Member State (MS)	Japan	Japan
Company / Institution	Toshiba, JAPC	JAERI, JAPC, Hitachi, TITech
Reactor type	BWR	BWR
Reactor capacity, thermal / electric (MW)	900 / 300	955 / 330
Reactor style	Modular type	Integral tank type, modular
Reactor core outlet temperature (^oC)	560 K	288
Primary circuit pressure (MPa)	7	7.2
Thermodynamic cycle	Direct	Direct
Applications	- 182 MW(e) of electricity plus 102 000 m^3/day of potable water by reverse osmosis - Applicable to district heating	Electricity production only
Special features	- Incremental capacity addition (modules) - Option of an increased refuelling interval (15 years)	Breeding ratio > 1.0 with MOX fuel
Fuel cycle option, basic	Once-through/ low enrichment uranium dioxide fuel	Closed fuel cycle (U, MOX)
Fuel cycle options, alternative	Closed fuel cycle (U, MOX); aqueous and dry reprocessing; MOX recycle in BWRs with fast neutron spectrum	-
Design and regulatory status 1995	-	-
Design and regulatory status 2005	- Conceptual design completed - Detailed design by mid-2010	- Conceptual design - Detailed design within 3 years
Targeted deployment date 1995	-	-
Targeted deployment date 2005	Start of construction related actions by mid-2010	-
Design description in this TECDOC	Annex IX	Annex X
Design descriptions in other TECDOCs	-	For a 1300 MW(e) class RMWR see IAEA-TECDOC-1391 (May 2004), p. 418
Similar or relevant SMRs	No other similar SMRs are under design elsewhere	-

2. SUMMARY TABLE (Continued-5)

SMR-name	**AHWR**	**RUTA-70**
Member State (MS)	India	Russian Federation
Company / Institution	BARC	IPPE, Rosatom
Reactor type	Light water cooled heavy water moderated	Light water reactor (LWR)
Reactor capacity, thermal / electric (MW)	920 / 300	70 / no conversion to electricity
Reactor style	Vertical pressure tube type	Integral, pool type
Reactor core outlet temperature (^{o}C)	558 K	101
Primary circuit pressure (MPa)	7 (steam at steam drum outlet)	Non-pressurized
Thermodynamic cycle	Indirect	Indirect
Applications	300 MW(e) (gross) of electricity plus 500 m^3/day of desalinated water	- 60 GCal/hr for district heating; or - Desalination option with 30 000 m^3/day of potable water
Special features	-	Low temperature in the primary circuit (< 101 ^{o}C at core outlet)
Fuel cycle option, basic	- Once-through, with Pu – Th and ^{233}U - Th dioxide fuel (only in the near term) - Closed fuel cycle with Pu – Th and ^{233}U - Th dioxide fuel; three-stream U, Pu and Th reprocessing method	Once-through/ low enrichment uranium dioxide fuel
Fuel cycle options, alternative	- Plutonium burner mode using plutonium-thorium MOX and a fully thorium-^{233}U fuelled mode in a closed fuel cycle - Synergy with fast breeder reactors and accelerator driven systems	Once-through/ low enrichment U in advanced fuel forms, e.g., micro-particles in metallic matrix
Design and regulatory status 1995	-	- Conceptual design of RUTA-20 was completed in 1992 - Licensing for the conceptual design of RUTA-20 was completed in 1992
Design and regulatory status 2005	- Basic design completed - Pre-licensing safety appraisal started	- Preliminary design stage - Licensing not yet started
Targeted deployment date 1995	-	-
Targeted deployment date 2005	-	The time required for deployment of the RUTA-70 is estimated as ~ 3 years in the Russian Federation and ~ 4 - 5 years outside the Russian Federation
Design description in this TECDOC	Annex XI	Annex XII
Design descriptions in other TECDOCs	IAEA-TECDOC-1451 (May 2005), p. 143	IAEA-TECDOC-881 (1996), p.487
Similar or relevant SMRs	No other similar SMRs are under design elsewhere	- SLOWPOKE - Design and licensing experience of RUTA-20 (a predecessor)

2. SUMMARY TABLE (Continued-6)

SMR-name	**KAMADO**
Member State (MS)	Japan
Company / Institution	CRIEPI
Reactor type	Direct flow pressure tube light water reactor
Reactor capacity, thermal / electric (MW)	1000 / 300
Reactor style	Vertical pressure tubes; fuel rods and pressure tubes separately embedded in graphite blocks forming fuel elements; fuel elements submerged in non-pressurized water pool (temperature < 60 ^{o}C)
Reactor core outlet temperature (^{o}C)	About 300
Primary circuit pressure (MPa)	~ 7
Thermodynamic cycle	Direct
Applications	Electricity generation and production of several thousand m^3/hr hydrogen; an option to produce process steam (800 ^{o}C)
Special features	- Incremental capacity addition (increasing the number of fuel elements in the pool) - Floating NPP not excluded - Simplified construction and transportation (no reactor pressure vessel)
Fuel cycle option, basic	Once-through/ low enrichment uranium dioxide fuel
Fuel cycle options, alternative	Closed fuel cycle (U, MOX); graphite combustion or mechanical destruction before applying an aqueous reprocessing method
Design and regulatory status 1995	-
Design and regulatory status 2005	Preliminary conceptual design in progress
Targeted deployment date 1995	-
Targeted deployment date 2005	-
Design description in this TECDOC	Annex XIII
Design descriptions in other TECDOCs	-
Similar or relevant SMRs	No other similar SMRs are under design elsewhere

2. SUMMARY TABLE (Continued-7)

SMR-name	PBMR	GT-MHR
Member State (MS)	South Africa	USA, Russian Federation
Company / Institution	ESKOM, Industrial Development Corporation of South Africa, BNFL	US National Nuclear Security Administration, General Atomics, Rosatom, OKBM
Reactor type	High temperature gas cooled reactor (HTGR)	HTGR
Reactor capacity, thermal / electric (MW)	400 / 165	600 / 287
Reactor style	Pebble bed fuel	Prismatic core; pin-in-block fuel
Reactor core outlet temperature (^oC)	900	850
Primary circuit pressure (MPa)	9	7.0
Thermodynamic cycle	Direct gas turbine cycle	Direct gas turbine cycle
Applications	- Cogeneration applications, including electricity generation and process heat production; -Potable water and hydrogen production are being considered	- Pu utilization (GT-MHR design for the Russian Federation); - 200 t of H_2 per day at 600 MW(th); or - Electricity plus 42 000 m^3/day of potable water at 600 MW(th) - Low temperature heat applications
Special features	- Multi-module plant as basic option - Shop fabrication of major reactor and internal components	- 4-module plant as basic option - Simplified operational licensing requirements (possible) - Flexible siting
Fuel cycle option, basic	Once-through fuel cycle; U dioxide fuel in TRISO coated particles within graphite spheres; on-line refuelling	Once-through fuel cycle; U dioxide fuel in TRISO coated particles within graphite fuel compacts
Fuel cycle options, alternative	-	- Closed fuel cycle (U, MOX); or - Hybrid fuel cycle (U, Th)
Design and regulatory status 1995 **Design and regulatory status in 2001 (IAEA-TECDOC-1198)**	- Conceptual design finalized; licensing pre-application made	Basic design for a steam turbine 4-module plant Conceptual design completed
Design and regulatory status 2005	- Detailed design stage on-going	- Basic design - Pre-application licensing interactions with the US NRC began in 2001
Targeted deployment date 1995 **Targeted deployment date in 2001 (IAEA-TECDOC-1198)**	- 2005	- 2009 (in the Russian Federation)
Targeted deployment date 2005	- The demonstration plant site preparation is scheduled to begin at the Koeberg NPP site in the first quarter of 2007 with fuel loading anticipated for mid-2010. - The commercial acceptance by ESKOM is scheduled for early 2011	Around 2015
Design description in this TECDOC	Annex XIV	Annex XV
Design descriptions in other TECDOCs	IAEA-TECDOC-1198 (February 2001), p. 19	IAEA-TECDOC-1198 (February 2001), p. 73
Similar or relevant SMRs	GT-MHR, GTHTR-300, ACACIA	GTHTR-300, PBMR, HTR-PM,

2. SUMMARY TABLE (Continued-8)

SMR-name	**GTHTR-300**	**HTR-PM**
Member State (MS)	Japan	China
Company / Institution	JAERI	INET, Tsinghua University
Reactor type	HTGR	HTGR
Reactor capacity, thermal / electric (MW)	600 / 274	380 / 160
Reactor style	Prismatic core; pin-in-block fuel	Pebble bed fuel
Reactor core outlet temperature (^{o}C)	850	750
Primary circuit pressure (MPa)	7.0	7.0
Thermodynamic cycle	Direct gas turbine cycle	Indirect, steam turbine cycle
Applications	- Electricity generation - Cogeneration, including high temperature process heat for hydrogen production and process steam and low temperature heat for seawater desalination and district heating	Electricity production only
Special features	- Multi-module plants	- Incremental capacity addition (modules) - Design is aimed at standardization and modularization
Fuel cycle option, basic	Once-through fuel cycle; U dioxide fuel in TRISO coated particles within graphite fuel compacts	Once-through fuel cycle; U dioxide fuel in TRISO coated particles within graphite spheres; on-line refuelling
Fuel cycle options, alternative	Closed fuel cycle; feasibility of fuel reprocessing was investigated	Reprocessing option could be considered
Design and regulatory status 1995	-	-
Design and regulatory status 2005	- Basic design completed - The detailed design and development of the GTHTR300 including R&D for the gas turbine system will be completed by the end of March 2008	Conceptual design stage
Targeted deployment date 1995	-	-
Targeted deployment date 2005	- Prototype plant demonstration in 2008 – 2018 - The system would be upgraded to deliver 950^{o}C helium, which will contribute to the deployment of other attractive systems such as the GTHTR300C cogeneration system for electricity and hydrogen around 2020	Construction of a demonstration plant around 2010
Design description in this TECDOC	Annex XVI	Annex XVII
Design descriptions in other TECDOCs	-	-
Similar or relevant SMRs	PBMR, GT-MHR, other HTGRs	HTR-10; other HTGRs developed around the world

2. SUMMARY TABLE (Continued-9)

SMR-name	**FAPIG-HTGR**	**ACACIA**
Member State (MS)	Japan	The Netherlands
Company / Institution	FAPIG: Fuji Electric Systems, Kawasaki Plant Systems, Ltd., Shimizu Corporation	NRG
Reactor type	HTGR	HTGR
Reactor capacity, thermal / electric (MW)	220 / 100	60 / (18.1 - 23.2)
Reactor style	Pebble bed fuel	Pebble bed fuel
Reactor core outlet temperature (°C)	900	900
Primary circuit pressure (MPa)	6	4.1
Thermodynamic cycle	Direct gas turbine cycle (recuperated Brayton cycle)	Indirect
Applications	- Electricity generation; - Applicable to other high temperature heat applications with some system modifications	- 23.2MW(e) electricity production (combined cycle of a gas turbine and a steam turbine) - 18.1 MW(e) electricity plus 27.8 t/h process steam at 425 °C and 4.14 MPa - Potable water production was considered within cogeneration option
Special features	- Multi-module plants	- Concept is well suited as an autonomous energy source (nuclear cell) - Incremental capacity addition possible (modules) - Simplified on-site refuelling once in 3 years
Fuel cycle option, basic	Once-through fuel cycle; U dioxide fuel in TRISO coated particles within graphite spheres; on-line refuelling	Once-through fuel cycle; U dioxide fuel in TRISO coated particles within graphite spheres; once-at-a-time whole core refuelling
Fuel cycle options, alternative	-	-
Design and regulatory status 1995 **Design and regulatory status in 2001 (IAEA-TECDOC-1198)**	-	- Pre-conceptual design stage
Design and regulatory status 2005	Pre-conceptual design stage	Design team has the intention to go on with further development; however, no further R&D is planned at the moment
Targeted deployment date 1995 **Targeted deployment date in 2001 (IAEA-TECDOC-1198)**	-	-
Targeted deployment date 2005	-	-
Design description in this TECDOC	Annex XVIII	Annex XIX
Design descriptions in other TECDOCs	-	IAEA-TECDOC-1198 (February 2001), p.169
Similar or relevant SMRs	-	PBMR, HTR-PM, ENHS

2. SUMMARY TABLE (Continued-10)

SMR-name	**KALIMER**	**BMN-170**
Member State (MS)	The Republic of Korea	Russian Federation
Company / Institution	KAERI	OKBM, SPb AEP, IPPE
Reactor type	Sodium cooled fast reactor	Sodium cooled fast reactor
Reactor capacity, thermal / electric (MW)	392.2 / 150	400 / 170
Reactor style	Pool type; intermediate heat transport system	Modular type; intermediate heat transport system
Reactor core outlet temperature (^{o}C)	530.0	550
Primary circuit pressure (MPa)	Non-pressurized	Non-pressurized
Thermodynamic cycle	Indirect; steam turbine Rankine cycle	Indirect; steam turbine cycle
Applications	Electricity production only	- 170 MW(e); or - 75 MW(e) and 630 t/hr of steam; or - 100 MW(e) and 260 GCal/h of heat for district heating
Special features	- Multiple unit construction - Breeding ratio of about 1.05	- Flexible plant capacity achieved through modular design - High degree of pre-fabrication - Operation within a multi-component nuclear energy system with optimized nuclide flows (possible); breeding ratio >1
Fuel cycle option, basic	- Closed fuel cycle (U, TRUs, and some fission products will be recovered from the spent fuel and recycled); an option of on-line refuelling is being considered; U–Pu–Zr ternary fuel - For a start-up core, the fresh fuel is composed of recovered LWR transuranics and depleted uranium; in subsequent cycles, where the discharged TRU become available for manufacturing new fuel feeds, the fissile make-up of a core load will be based on recycled Pu - Pyro-processing of metallic fuel will be applied; the fuel cycle facilities (fabrication and reprocessing) will be co-located with the reactor at the same site	- Closed fuel cycle (U, Pu) - Flexibility in fuel: oxide, nitride or metallic fuel options - Non-aqueous methods of fuel reprocessing
Fuel cycle options, alternative	-	Option of ^{233}U production in blankets (for future thermal reactors)
Design and regulatory status 1995	-	-
Design and regulatory status 2005	- Conceptual design of the KALIMER–150 has been completed in 2001 - The key design technologies are being developed currently	Conceptual investigations
Targeted deployment date 1995	-	-
Targeted deployment date 2005	Sodium-cooled fast reactor was chosen as one of the two future reactor options deployable by 2030	-
Design description in this TECDOC	Annex XX	Annex XXI
Design descriptions in other TECDOCs	-	-
Similar or relevant SMRs	JSFR, S-PRISM	BMN-170 concept was shaped up based on experience in the development and operation of NPPs with BOR-60, BN-350, BN-600, and BN-800 reactors

2. SUMMARY TABLE (Continued-11)

SMR-name	**MDP**	**RBEC-M**
Member State (MS)	Japan	Russian Federation
Company / Institution	CRIEPI	RRC-KI
Reactor type	Sodium cooled fast reactor	Lead-bismuth cooled fast reactor
Reactor capacity, thermal / electric (MW)	840 / 325 (per module); 1300 MW(e) with a 4-module plant	900 / 340
Reactor style	Modular double pool type; intermediate heat transport system	Integral pool type; modular; gas lift system; no intermediate heat transport system
Reactor core outlet temperature (^{o}C)	530	519
Primary circuit pressure (MPa)	Non-pressurized	Non-pressurized
Thermodynamic cycle	Indirect; steam turbine cycle	Indirect; steam turbine cycle
Applications	Electricity production only	- Electricity production in base load mode
Special features	- Multi-module plants - Breeding ratio 1.16	- Shop fabrication of reactor components including the mono-block - RBEC-M is designed as an element of a multi-component nuclear power system with optimized nuclide flows. The main functions of the RBEC-M within such a system are to provide effective closure of the nuclear fuel cycle with respect to U and Pu, and extended breeding of nuclear fuel (BR>1)
Fuel cycle option, basic	- Closed fuel cycle (U, Pu) - U-Pu-Zr ternary alloy - Pyro-processing and injection casting for fuel fabrication	- Closed fuel cycle (U, Pu) - U-Pu nitride fuel - On-site reprocessing could be considered
Fuel cycle options, alternative	-	Thorium blankets for ^{233}U production to feed future thermal reactors
Design and regulatory status 1995	Conceptual design has been completed	-
Design and regulatory status 2005	Preliminary conceptual design has been completed; at the moment, there is no financial support for further R&D	Conceptual design stage
Targeted deployment date 1995	-	-
Targeted deployment date 2005	-	~2025
Design description in this TECDOC	Annex XXII	Annex XXIII
Design descriptions in other TECDOCs	IAEA-TECDOC-881 (1996), p. 518	-
Similar or relevant SMRs	-	-

2. SUMMARY TABLE (Continued-12)

SMR-name	**PEACER-300 (PEACER-550)**	**Medium Scale Lead-bismuth Cooled Reactor**
Member State (MS)	The Republic of Korea	Japan
Company / Institution	Nuclear Transmutation Reactor Engineering Center Korea (NuTRECK), Seoul National University	JNC
Reactor type	Lead-bismuth cooled fast reactor	Lead-bismuth cooled fast reactor
Reactor capacity, thermal / electric (MW)	850 / 300 (1560 / 550)	1875 / 710 (per module)
Reactor style	Loop type; no intermediate heat transport system	Tank-type, modular; no intermediate heat transport system
Reactor core outlet temperature (^{o}C)	400	445
Primary circuit pressure (MPa)	Non-pressurized	Non-pressurized
Thermodynamic cycle	Indirect; steam turbine cycle	Indirect; steam turbine cycle
Applications	Electricity production and incineration of TRU and fission products	Electricity production
Special features	- High conversion reactor, not a breeder - A concept of the energy park with several PEACERs and closed fuel cycle facilities is considered to utilize spent fuel from LWRs	4-module NPP of 2840 MW(e) as basic
Fuel cycle option, basic	Closed fuel cycle; U-TRU-Zr metallic fuel; on-site pyro-metallurgical reprocessing; starting with U fuel and on to the recycling of all self-produced TRUs	Nitride fuel; 100% ^{15}N enriched; closed fuel cycle
Fuel cycle options, alternative	Incineration of spent fuel components from other reactors, e.g., LWRs	-
Design and regulatory status 1995	-	-
Design and regulatory status 2005	Conceptual design of the reactor and basic design of its major components have been completed	Conceptual design stage
Targeted deployment date 1995	-	-
Targeted deployment date 2005	Initiation of prototype-construction related actions by the end of 2010	-
Design description in this TECDOC	Annex XXIV	Annex XXV
Design descriptions in other TECDOCs	-	-
Similar or relevant SMRs	BREST-300 and 1200, SVBR-75/100; and other Pb and Pb-Bi cooled reactor designs developed worldwide	-

2. SUMMARY TABLE (Continued-13)

SMR-name	**AHTR**
Member State (MS)	USA
Company / Institution	ORNL, SNL and the University of California at Berkeley
Reactor type	Non-conventional: liquid (molten) salt cooled very high temperature reactor with HTGR type block fuel
Reactor capacity, thermal / electric (MW)	600 to 2400 / 300 to 1200
Reactor style	Integral, pool type; liquid salt intermediate heat transport system (heat transport to process applications or to the power circuit)
Reactor core outlet temperature (^{o}C)	705 or 800 or 1000
Primary circuit pressure (MPa)	Non-pressurized
Thermodynamic cycle	Indirect, multi-reheat helium or nitrogen Brayton power cycle
Applications	The AHTR is designed to produce electricity and/or high temperature heat. The heat may be used for hydrogen production or other applications.
Special features	Factory assembled modular units with ease of transport
Fuel cycle option, basic	The same as for HTGRs with prismatic fuel; liquid salt contains no nuclear fuel
Fuel cycle options, alternative	-
Design and regulatory status 1995	-
Design and regulatory status 2005	- Pre-conceptual design stage - AHTR is part of the U.S. Department of Energy Generation IV reactor programme and is being actively investigated - Detailed development plans are being prepared
Targeted deployment date 1995	-
Targeted deployment date 2005	If the AHTR is selected for large-scale development, the goal would to be to have an operating test reactor by 2012. A medium sized pre-commercial demonstration reactor would follow this
Design description in this TECDOC	Annex XXVI
Design descriptions in other TECDOCs	-
Similar or relevant SMRs	The development of the AHTR is tightly coupled to modular HTGRs because about 70% of the R&D is in common; this includes fuel development, materials development, and Brayton power cycles.

Electric output

A preferable choice for the electric output, as observed in 9 out of 13 water cooled SMR concepts, is around 300 MW (from 250 to 350 MW) excluding prototypes. The SCOR with its 630 MW(e) provides a notable exception towards higher outputs. The lowest output of 70 MW(th) is for the RUTA-70.

Non-electric applications

Except for the designs of RMWR, RUTA-70 and KAMADO, all SMRs provide for or do not exclude a cogeneration option with the production of potable water or heat for district heating, often flexible. The RMWR is designed for electricity generation only; it is a breeder reactor with the conversion ratio of more than 1.0. RUTA-70 is a dedicated reactor for district heating or potable water production; it has no energy conversion system. The KAMADO is proposed for electricity generation with simultaneous production of hydrogen with the use of high temperature process steam.

Design and regulatory status

Of the thirteen concepts and designs of water cooled SMRs, five (SMART, CAREM-25, VBER-300, VK-300, and AHWR) are at the detailed design stage; four (IRIS, MARS, CCR and RUTA-70) are at the basic (preliminary) design stage; others are at the conceptual or early conceptual (KAMADO) design stages.

Licensing pre-application has been made or preliminary licensability assessment was started for the IRIS, CAREM and AHWR.

Timeline of readiness for deployment

The targeted deployment dates, when specified by the designers, are generally between 2010 and 2015.

Progress achieved since previous IAEA publications

Progress in the design and regulatory status since previous IAEA publications (IAEA-TECDOC-881, 1995 [10]) can be assessed only for the CAREM, the MARS and the RUTA-70; other SMRs were either not addressed or not started in development at that time.

For the CAREM, the progress from a basic to the detailed design stage and licensing pre-application is observed, with the projected deployment date moved from 1995 to 2006 and beyond.

For the MARS, there is a slow progress towards completion of the basic design.

For the RUTA-70, the thermal output was changed from 20 to 70 MW(th), and the design stage has progressed from a conceptual to the preliminary (basic) design. Licensing pre-application for the new, higher output design has not been made so far.

User-related special features

An incremental plant capacity addition is offered by the modular designs of SMART, IRIS, CAREM, MARS, IMR, CCR, and KAMADO. The VBER-300 capacity can be increased by increasing the number of loops.

A floating, i.e. barge-mounted NPP option is elaborated in detail for the VBER-300 but foreseen or not excluded for the concepts of IRIS, CAREM and KAMADO. The VK-300 provides for using the afloat construction technology, with pontoons being welded to the assembled reactor compartment to transport it by water to a cargo ship and then, from a cargo ship to the point of mounting, assuming that both the factory and the site are located near a seaside.

Shop fabrication of certain components and modular approach to NPP construction are mentioned for the projects of SMART, IRIS, CAREM, MARS, and IMR. Shop fabrication of the complete reactor module is foreseen for the VBER-300.

Simplified operational licensing requirements are mentioned, as a target, for the IRIS and the SCOR. A reduced core power option to enable the operation without on-site refuelling (with long refuelling interval and whole core refuelling) is being reserved for the concepts of IRIS, VBER-300 and CCR.

An increased refuelling interval (operation cycle duration) within the conventional refuelling scheme is offered by the designs of SMART, IRIS, IMR, and CCR.

Fuel cycle options

Once-through fuel cycle with low-enrichment uranium dioxide fuel is foreseen as basic for most of the water cooled SMRs. The exceptions are (i) the RMWR, which is a breeder reactor designed to operate in a closed U-Pu fuel cycle; and (ii) the AHWR with the once-through Pu-Th-^{233}U fuel cycle specified as basic.

As an alternative, the concepts of IRIS, CAREM, SCOR, IMR, CCR, and KAMADO foresee an operation within a closed U-Pu fuel cycle. Regional or centralized reprocessing is specified for the IRIS; use of advanced PUREX processes is specified for the concepts of CAREM, IMR, CCR, and KAMADO. Pyro-metallurgical or, more generally, dry reprocessing is mentioned for the IMR and the CCR. A closed fuel cycle with the Pu-Th and ^{233}U-Th fuel is specified for the AHWR, which targets a future synergy with fast breeder reactors and the accelerator driven systems for transmutation of waste.

A once-through Radkowsky Thorium Fuel (RTF) cycle is selected as an alternative for the VBER-300 (for more details, see Annex VII).

Compatibility with innovative fuel types is mentioned for the MARS. The use of low enrichment uranium based cermet fuel is considered for the RUTA-70.

3.1.2. Gas cooled SMRs

Reactor types

All six gas cooled SMRs addressed in this report are high temperature gas cooled reactors (HTGRs). Of them, four are the designs with pebble bed fuel within the annular core (PBMR, HTR-PM, FAPIG-HTGR and ACACIA), and 2 are the designs with the prismatic, pin-in-block fuel (GT-MHR and GTHTR-300). Four designs, PBMR, GT-MHR, GTHTR-300 and FAPIG-HTGR use a direct gas turbine cycle; the HTR-PM is based on an indirect steam turbine cycle; and the ACACIA is based on an indirect gas turbine cycle or an indirect combined gas turbine and steam turbine cycle. Pebble-bed transport (continuous refuelling) is provided for in the designs of PBMR, HTR-PM, and FAPIC-HTGR; the ACACIA is a reactor with simplified on-site refuelling, which is the whole core refuelling after ~3 years of continuous operation, with no pebble transport.

Electric output

The electric output generally varies between 100 and 287 MW; the ACACIA has a lower electric output of 18 to 23 MW. The GT-MHR and the GTHTR have the highest electric outputs of 287 and 274 MW, respectively.

Non-electric applications

Except for the HTR-PM, all designs provide for a cogeneration option, often with multiple non-electric applications. The HTR-PM provides for electricity generation only. High

temperature process heat applications and, specifically, hydrogen production are the preferred options for all designs except the ACACIA. Low temperature process heat applications or potable water production are mentioned for the concepts of PBMR, GT-MHR, GTHTR-300 and ACACIA. Purposeful use of heat rejected in the basic thermodynamic cycle is foreseen for the GT-MHR. District heating is specified as an option for the GTHTR-300; process steam applications are specified for the GTHTR-300 and the ACACIA. The GT-MHR design developed for the Russian Federation has a specific goal of Pu utilization.

Design and regulatory status

The PBMR and the GTHTR-300 have entered detailed design stage; the GT-MHR is near entering the detailed design stage; the HTR-PM is at the conceptual design stage; the FAPIG-HTGR and the ACACIA are at a pre-conceptual design stage.

Pre-application licensing interactions have been started for the GT-MHR; pre-application licensing has been made for the PBMR.

Timeline of readiness for deployment

The targeted deployment dates, when specified by the designers, are generally between 2010 and 2020. The earliest dates for a demonstration prototype operation start-up are specified for the projects of PBMR, GTHTR-300, and HTR-PM; they are around 2010.

Progress achieved since previous IAEA publications

Progress in the design and regulatory status of HTGRs was assessed not only against the IAEA-TECDOC-881 of 1995 [10], but also against a more recent IAEA publication dedicated to the reactors of such type, the IAEA-TECDOC-1198 of 2001 [22].

Over the past four years, the PBMR has moved from a conceptual to the detailed design stage. The timeline of readiness for demonstration plant deployment was rescheduled from 2005 to 2010.

For the GT-MHR, the progress since 1995 appears uneven, which may be due to certain changes in the project implementation strategy. These changes have put forward the Russian Federation design for plutonium utilization. Since 2001, the basic design stage was ongoing for the new GT-MHR project, and it is near completion or has been completed at the time of this report. Three more years are said necessary to finalize the detailed design under favourable financing conditions. The timeline for demonstration plant deployment in the Russian Federation was shifted from 2009 to ~2015.

The ACACIA has shown little progress since 2001, remaining at a pre-conceptual design stage. Its design team has the intention to go on with further development; however, no further R&D is planned at the moment.

Other gas cooled SMRs (HTGRs) were not addressed in the abovementioned previous IAEA publications.

User-related special features

A multi-module NPP option is offered by all HTGR concepts considered in this report. In several design descriptions it is emphasized, that a several-module HTGR plant could be controlled from a single control room, due to simplicity in operation resulting from the exceptional heat removal capability and high margin to fuel failure typical of such reactors. Shop fabrication of major reactor components is mentioned for the PBMR. Simplified operational licensing requirements are mentioned possible for the GT-MHR and the FAPIG-HTGR. The ACACIA could be used as an autonomous energy source.

Flexible siting is specified for the GT-MHR. Maximum design standardization and modularization is targeted for the HTR-PM.

Fuel cycle options

Once-through fuel cycle with the uranium dioxide fuel in TRISO coated particles is identified as basic for all HTGR designs considered.

As an alternative, closed fuel cycles with MOX or hybrid U-Th fuel are considered for the GT-MHR. An option to apply fuel reprocessing is mentioned for the HTR-PM and the GTHTR-300; for the latter it has already been investigated. In case of TRISO coated particles, aqueous reprocessing methods (e.g., PUREX) cannot be applied directly. Specifically, the problem is with the SiC coating layers, which are not dissolved in mixtures of acids and, therefore, require a mechanical treatment to be removed.

3.1.3. Liquid metal cooled SMRs

Reactor types

Of the six liquid metal cooled SMRs, three are sodium cooled fast reactors (KALIMER, BMN-170 and MDP), and 3 are lead-bismuth cooled fast reactors (RBEC-M, PEACER-300/550, and Medium Scale Lead-bismuth Cooled Reactor). All designs implement indirect thermodynamic cycles. All sodium cooled SMRs incorporate intermediate heat transport systems (secondary sodium circuits to transport heat to a steam turbine circuit and to prevent the possibility of a contact of water with the primary sodium). All lead-bismuth cooled SMRs have no intermediate heat transport system. All designs use steam turbine power circuit.

All liquid metal cooled SMRs are pool type reactors with non-pressurized primary circuit. The designs of BMN-170, MDP, RBEC-M and Medium Scale Lead-Bismuth Cooled Reactor are specified as modular.

Electric output

The electric output varies between 170 MW and 710 MW. The KALIMER and the BMN-170 are towards the lower end with 150 and 170 MW(e), respectively. The MDP, the RBEC-M and one of the PEACERs have the capacities between 300 and 340 MW(e). Another version of the PEACER and the Medium Scale Lead-bismuth Cooled Reactor are towards the higher end with 550 and 710 MW(e), correspondingly.

Non-electric applications

The liquid metal cooled SMRs addressed in this report are being designed for electricity generation and, with a notable exception of the BMN-170 (Annex XXI), make no provision for energy products such as heat for district heating, potable water or hydrogen. At the same time, all of them are fast reactor capable of high conversion or fuel breeding.

The KALIMER has a breeding ratio (BR) of 1.05, which ensures a self-sustainable mode on fissile materials. The BMN-170 and the RBEC-M are fast breeder reactors with extended fuel breeding (BR>1); they are designed to ensure optimum balance of fissile materials in a multi-component nuclear energy system. The MDP design offers a breeding ratio of 1.16.

The PEACER is specified as high conversion reactor, not a breeder; it could also be used to incinerate (i.e. burn or transmute) the transuranic (TRU) and selected fission product components of LWR spent fuel.

The BMN-170 and the RBEC-M provide an option to breed ^{233}U in thorium blankets, for further use as a feed in reactors with thermal neutron spectrum, e.g. LWRs.

Design and regulatory status

The BMN-170, the RBEC-M, and the Medium Scale Lead-bismuth Cooled Reactor are within the conceptual design stages. For the KALIMER and the MDP, conceptual designs have been completed. The KALIMER project is ongoing with the development of key design technologies, with a link to the Generation IV International Forum programme. The PEACER project is based on the conceptual design of a liquid metal cooled reactor developed in 1998; the present R&D encompasses Pb-Bi coolant technology demonstration and the conceptual design of a PEACER version for power production and LWR spent fuel disposal.

Timeline of readiness for deployment

The targeted deployment dates, when specified or mentioned by the designers, are generally between 2025 and 2030, matching the anticipated deployment dates for the Generation IV systems. For the PEACER, a very optimistic timeline for the initiation of construction-related actions is specified, which is 2010, under favourable conditions of support from the industry and financing.

Progress achieved since previous IAEA publications

Except for the MDP, none of the considered liquid metal cooled SMRs were addressed in IAEA-TECDOC-881 of 1995 [10]. For the MDP, the design stage in 2005 is essentially the same as in 1995; at the time of this report, there was no financial support of further R&D.

User-related special features

An option of a multi-modular plant or an incremental plant capacity addition is offered by the designs of KALIMER, BMN-170, MDP, and Medium Scale Lead-bismuth Cooled Reactor. The concepts of MDP and Medium Scale Lead-bismuth Cooled Reactor provide for the deployment of four-module plants of 1300 MW(e) and 2840 MW(e), respectively.

High degree of prefabrication is mentioned for the BMN-170; shop fabrication of reactor components including the mono-block is specified for the RBEC-M.

The BMN-170 and the RBEC-M are designed for operation in a multi-component energy system with optimized nuclide flows (such system is described in Section XXIII-1.5 of Annex XIII).

Fuel cycle options

All liquid metal cooled SMRs are designed to operate in a closed nuclear fuel cycle providing for the use of non-aqueous reprocessing methods. The designs of KALIMER, MDP and PEACER make use of the ternary U-Pu-Zr or U-TRU-Zr fuel and pyro-metallurgical reprocessing. The RBEC-M and the Medium Scale Lead-bismuth Cooled Reactor are nitride fuel reactors. The BMN-170 offers flexibility in the selection of fuel with either oxide, or nitride, or metallic fuel being applicable.

An option of on-line refuelling is being considered for the KALIMER.

The recycle of all TRUs is foreseen for the concepts of KALIMER, RBEC-M and PEACER; the latter is also designed to recycle components of the external spent fuel.

On-site reprocessing option is specified for the KALIMER and the RBEC-M. For the PEACER, an option of an international energy park with the collocation of the PEACER reactors and closed fuel cycle facilities is being considered; see Section XXIV-1.5 of Annex XXIV.

Both, the transuranics recovered from spent LWR fuel (KALIMER) and an enriched uranium (PEACER) could be used as the initial (first) fuel load for liquid metal cooled SMRs.

3.1.4. Non-conventional SMRs

Reactor types

In this report, only one non-conventional SMR concept is presented, which is the AHTR.

The AHTR is a liquid salt cooled very high temperature reactor with HTGR type pin-in-block (prismatic) fuel. Different from molten salt reactors (MSRs), the liquid (molten) salt coolant of the AHTR contains no fuel. The reactor incorporates a liquid salt intermediate heat transport system (IHTS), which transports high temperature heat to process applications and to a multi-reheat helium or nitrogen Brayton power cycle. In this way, the AHTR design brings together the technologies of HTGRs (TRISO type coated particle fuel in graphite matrix), MSRs (liquid salt coolant), and sodium cooled fast reactors (integral, pool type design; IHTS; and non-pressurized primary coolant system).

Electric output

Several output options are provided, ranging from 300 to 1200 MW(e).

Non-electric applications

The AHTR is designed as a cogeneration plant to produce electricity and high temperature heat within the range of temperatures from 705 to 1000 ^{o}C; the heat may be used for hydrogen production or other high temperature process heat applications.

Design and regulatory status

The AHTR is at a pre-conceptual design stage.

Timeline of readiness for deployment

The AHTR is part of the U.S. DOE Generation IV reactor programme and is being actively investigated. If the AHTR is selected for large-scale deployment, the goal would be to have an operating test reactor by 2012.

Progress achieved since previous IAEA publications

The AHTR concept is relatively new; it was not addressed in previous IAEA publications.

User-appreciated special features

The AHTR design offers factory assembled modular units with ease of transport.

Fuel cycle options

The AHTR fuel cycle options are the same as for HTGRs with prismatic fuel.

3.2. Innovative SMRs not included in this report

As it was already explained in Section 1.3, this report does not address innovative small reactors without on-site refuelling. Space and navy propulsion reactors and dedicated systems for transmutation of waste were also excluded from the consideration.

This report does not address evolutionary SMRs. The descriptions of many of such designs, which are mostly water cooled reactors, can be found in [19, 20]. One could argue that some water cooled SMRs included in this report appear more like the evolutionary designs with innovative features but, as mentioned in Section 1.3, the decision to include them was made by their designers who also provided the justifications of why they rate their designs as innovative.

In some cases the designers made a selection of innovative SMRs for this report. For example, the Experimental Design Bureau of Machine Building (OKBM) of the Russian Federation has recommended only a few innovative designs for this report. More designs from this vendor are addressed in [7].

An agreement to submit design descriptions for this report was not reached with the designers of BREST-300 lead cooled fast reactor from RDIPE (NIKIET) of the Russian Federation and the designers of CANDU X NC reactor from AECL of Canada (the latter is a Generation IV system with supercritical light water coolant). A description of the BREST-300 can be found in reference [21].

4. SUMMARY OF THE TECHNICAL FEATURES

4.1. Introduction

This chapter gives a review of the design approaches and features that are definitive for innovative SMR performance in the areas of economics, sustainability (including resource utilization, waste management and environmental impacts), safety, proliferation resistance and physical protection. This review is based on the information provided in Sections 1.6.1 to 1.6.5 of the SMR design descriptions given in Annexes.

4.2. Technical features and technological approaches used to improve SMR economy

SMRs don’t benefit from the economics of scale; therefore, substantial efforts of their designers are targeted at the improvement of plant economy. When it comes to innovative SMRs, the designers often define very challenging economic goals, such as to approach the specific level of overnight capital costs typical of large capacity LWRs and to reduce the operation and maintenance costs significantly against current practices.

For all innovative SMRs the design approaches most commonly mentioned in conjunction with the improved plant economy are the following:

(1) Approaches to reduce capital costs and construction period:

- System simplification.
- Component modularization.
- Factory fabrication and direct site installation.
- Possibility of a staggered build of multiple modules.
- Standardization and construction in series.

(2) Approaches to reduce operation and maintenance costs:

- System simplification.
- Strong reliance on passive safety design options.
- Increased NPP service life.
- Extended operational cycle (infrequent refuelling).

(3) Approaches to reduce fuel costs:

- Improved neutron economy.
- High burn-up of fuel.

Strong reliance on proven technologies (with the system configurations only being innovative) and flexibility in siting and applications are mentioned by several designers as

factors that could minimize the investment risk and improve the competitiveness of an SMR based plant.

4.2.1. Water cooled SMRs

System simplification, which provides for the reduced number of structures, systems and components, could be achieved in different ways, such as the following:

- Integral design of primary circuit and the associated reduction of piping (essentially all designs except MARS, VBER-300, RUTA-70 and KAMADO).
- Natural circulation in primary coolant system, eliminating the main circulation pumps (CAREM-25, IMR, VK-300, CCR, RMWR, AHWR, RUTA-70).
- Modular loop type design of increased compactness with the reduced piping (MARS, VBER-300).
- Pool type design to eliminate excess pressure in the reactor pool and abandon the reactor pressure vessel (RUTA-70, KAMADO); etc.

The possibility of a staggered build of multiple modules is provided for by the designers of IRIS, CAREM, IMR, CCR and KAMADO; the VBER-300 provides for the capacity increase by increasing the number of loops.

Extended operational cycles are indicated for the designs of SMART, IRIS, IMR, and CCR.

Broad incorporation of passive safety features and passive systems is typical of all designs. The designers of MARS and SCOR emphasize the importance of finding an optimum combination between active and passive systems to achieve better economic competitiveness.

Pre-fabrication of components and modular approach to plant construction are foreseen for most of the water cooled SMRs; the VBER-300 provides for full factory fabrication.

4.2.2. Gas cooled SMRs

The designers of gas cooled SMRs (HTGRs) mention the following major factors as contributing to the improved nuclear power plant (NPP) economy:

- Strong reliance on the inherent and passive safety features common to all HTGRs, including large temperature margin of coated particle fuel, exceptional passive shutdown and decay heat removal capability, slow and stable response to transients caused by internal and external initiating events (due to large heat capacity of core graphite); the inherent and passive features of HTGRs contribute to achieving both, a reduced design complexity and the reduced operation and maintenance costs.
- High efficiency of energy conversion, achieved through the use of direct gas turbine Brayton cycles (PBMR, GT-MHR, GTHTR300, and FAPIG-HTGR) or the indirect steam turbine (HTR-PM) or gas turbine (ACACIA) cycle, and in all cases enabled by high core outlet temperatures; the economic characteristics of the plant are inversely proportional to the energy conversion efficiency.
- Flexible plant capacity achieved through modular reactor design; the concepts of PBMR, GT-MHR, GTHTR300 and FAPIG-HTGR provide for several-module plants as basic option; for all other designs a multi-module approach is not excluded; multi-modular HTGR plants provide for sharing of the common equipment and assume that the operation of all modules is performed from a single control room.
- Modular approach to plant construction with the pre-fabrication of certain standardized design units.

- High burn-up of fuel and high efficiency of energy conversion, altogether contributing to the reduction of fuel costs.

4.2.3. Liquid metal cooled SMRs

Sodium cooled SMRs need an intermediate heat transport system (IHTS) to prevent contacts of the power circuit water and steam with the primary sodium. Presence of an intermediate circuit generally increases the plant costs. To minimize this increase, the designs of the MDP and the BMN-170 use an adjacent configuration of the primary and intermediate circuits, with the space between main (primary) and safeguard (secondary) reactor vessels being filled with sodium and acting as an intermediate heat transport system. Different from sodium cooled reactors, lead-bismuth cooled SMRs incorporate no intermediate circuit.

For liquid metal cooled SMRs, the design features most commonly mentioned in conjunction with the improved plant economy are the following:

- System simplification, for example, resulting from integral design of the primary circuit or from adjacent configuration of the primary and intermediate circuits (BMN-170, MDP, RBEC-M, Medium Scale Lead-bismuth Cooled Reactor) or from a natural circulation based primary circuit (a gas lift system is used to facilitate natural circulation in the RBEC-M), or from the use of passive systems for reactor shutdown and decay heat removal; always resulting in the reduced number of structures, systems and components such as piping, main circulation pumps, active shutdown systems, etc.
- Multiple unit construction, allowing flexible plant capacity and a staggered build of multiple modules (all designs except the PEACER).
- Pre-fabrication of components; standardization and production in series; improved construction methods relying on modular approaches (KALIMER, BMN-170, and RBEC-M).
- Design compactness (MDP, RBEC-M).

4.2.4. Non-conventional SMRs

As mentioned before, the AHTR concept (Annex XXVI) brings together the technologies of HTGRs (high temperature fuel, gas turbine Brayton cycle), molten salt reactors (liquid salt that, in AHTR case, contains no fuel) and fast sodium cooled reactors (low pressure in the primary circuit, intermediate heat transport system). The designers of AHTR mention the following technical features as contributing to the improvement of NPP economy:

- System simplicity arising from low primary and intermediate coolant pressure and from the use of passive systems that, in the AHTR case, might be effective up to the unit power level of 600 MW(e).
- High efficiency of energy conversion resulting from high or very high temperature at the core outlet; good heat transfer properties of the liquid salt; and the use of a closed gas turbine Brayton cycle in power circuit.
- Design compactness, facilitated by high heat transfer coefficients of the liquid salt, and resulting in a relatively small size of the vessel and heat exchangers.

4.2.5. Summary of targeted economic characteristics

Table 3 summarizes the evaluations of capital costs, construction periods and electricity costs as provided by the designers of innovative SMRs for this report.

TABLE 3. PROJECTIONS FOR THE SPECIFIC CAPITAL COSTS, CONSTRUCTION PERIODS AND ELECTRICITY COSTS FOR INNOVATIVE SMRs

SMR NAME	ANNEX NUMBER	SPECIFIC CAPITAL COST, US$/ kW(e)	CONSTRUCTION PERIOD, MONTHS	ELECTRICITY COST, US$ cent/ kW-h
		Water cooled SMRs		
SMART	I	1714 (construction cost)	Less than 36	4.06
IRIS	II	1030-1240; first-of-a-kind plant	36 or less	3-4
CAREM	III	1050 – 1700; depending on the output, 300 or 125 MW(e)	48	-
MARS	IV	~2610*; for a 3-module plant	36 – 48	4.7* 1.7* (after 20 years)
SCOR	V	10% below the specific capital cost of a standard large loop-type PWR	~36	13% lower than for a classical PWR
IMR	VI	Equivalent to large scale reactors (target)	~30	-
VBER-300	VII	1084 –land based NPP; 820 – floating NPP	48 – 60	2.2 – land based NPP 1.8 – floating NPP
VK-300	VIII	865	-	1.0
CCR	IX	Equivalent to that of a large sized BWR	24	-
RMWR	X	-	-	-
AHWR	XI	1170; first-of-a-kind plant	72	-
RUTA-70	XII	279.6 per kW(th)	36	No electricity production
KAMADO	XIII	-	-	Comparable to large scale LWRs (target)
		Gas cooled SMRs		
PBMR	XIV	<1500 (construction cost)	30-34 – demonstration plant; 24 – commercial modules	-
GT-MHR	XV	1460; first-of-a-kind plant ~1000; N-th plant	Less than 36	3.1 (20-year levelized)
GTHTR300	XVI	<1750* (target)	-	3.5* (target)
HTR-PM	XVII	<1500 (target)	48	4.5 (target)
FAPIG-HTGR	XVIII	1200 (construction cost)	-	-
ACACIA	XIX	2700 – first-of-a-kind plant; 2200 – serial plant	24	3 – cogeneration mode; 6 –power generation only

SMR NAME	ANNEX NUMBER	SPECIFIC CAPITAL COST, US$/ kW(e)	CONSTRUCTION PERIOD, MONTHS	ELECTRICITY COST, US$ cent/ kW-h
Liquid metal cooled SMRs				
KALIMER	XX	2300 – first single module; 1400-1600 –multiple unit construction in series	-	-
BMN-170	XXI	-	-	-
MDP	XXII	-	31	-
RBEC-M	XXIII	1670 – 1750	-	-
PEACER	XXIV	2750	72	Competitive with alternative energy options available at the time (target)
Medium Scale Lead-bismuth Cooled Reactor	XXV	-	-	-
Non-conventional SMRs				
AHTR	XXVI	816 – intermediate temperature version 930 – high temperature version	-	-

* Converted from other currencies as of 5 October 2005

4.3. Provisions for effective resource utilization, waste management, and minimum adverse environmental impacts

A reduction of carbon emissions is one of the important incentives for future development of nuclear power [5]. By offering cogeneration options with flexible or multiple non-electric applications, many innovative SMRs could help minimize not only the emissions associated with electricity generation but also those arising from the heat and motive power production by fossil fuel combustion.

Energy conversion efficiency is an important factor that defines the specific (i.e. per unit of the useful energy produced) values of the resource consumption, emissions and discharges. These values are inversely proportional to the efficiency, so that, for example, gas cooled SMRs with direct Brayton power cycles (energy conversion efficiency ~50%) may offer a substantial reduction in the discharged (rejected) heat when compared to present day LWRs (energy conversion efficiency ~32%). Heat discharges could also be minimized by purposeful use of the rejected heat (see Annex XV).

Effective resource utilization may be a matter of many factors, such as material intensity of the reactor design, neutron economy, fuel burn-up, power density, energy conversion efficiency and, last but not least, the recycling.

4.3.1. Water cooled SMRs

The following technical features to minimize the production of wastes are commonly mentioned by the designers of innovative water cooled SMR:

- Soluble boron free core (all designs except IRIS, MARS, and VBER-300).
- Simplification of the primary circuit and reduction of the equipment, e.g. achieved by the use of the integral design, compact design or minimized connections and piping

and resulting in the reduced amount of activated or potentially contaminated structural materials (all innovative water cooled SMRs).

- Relatively low operating temperatures, contributing to the suppression of thermo-activated fission product release processes in fuel (MARS, SCOR, IMR, RUTA-70).
- Increased operational interval (SMART, IRIS, MARS, IMR), contributing to the reduced volume of high level waste.
- Technical provisions for easy decommissioning (MARS).
- Advanced methods of waste treatment and disposal (SMART).

Infrequent refuelling, integral design of primary circuit and reduced piping are also specified as factors contributing to the minimization of occupational doses (IRIS, SCOR, VBER-300) achieved though the resulting reduced maintenance.

To achieve improved fuel utilization and to broaden the available base of natural resources, the designers of many innovative water cooled SMRs foresee, as an alternative, the operation within a closed fuel cycle (IRIS, CAREM, SCOR, IMR, CCR, AHWR, and KAMADO) or the introduction of thorium fuel (VBER-300, AHWR). Two designs, the RMWR and the AHWR, target high conversion ratios initially. Specifically, the RMWR, which incorporates a tight lattice core, is designed to operate in a closed fuel cycle with MOX fuel and could achieve a breeding ratio of more than 1.0. The RMWR is also capable of recycling its own minor actinides (MAs), see Annex X. Similar option is reserved for the CCR. The AHWR achieves an increased conversion with the use of the Pu-Th fuel and on-line refuelling and also offers a minimized production of the highly radiotoxic long-lived minor actinides.

The designers of the AHWR consider an option of zirconium recycling.

4.3.2. Gas cooled SMRs

For gas cooled SMRs (HTGRs), the features commonly mentioned in conjunction with the effective resource utilization, minimized waste and reduced environmental impacts are:

- High efficiency of energy conversion, e.g. achieved through the use of direct (PBMR, GT-MHR, GTHTR300, FAPIC-HTGR) or indirect (ACACIA) Brayton cycles and through purposeful use of the rejected heat (GT-MHR).
- High fuel burn-up that, together with the efficient energy conversion, contributes to a more effective utilization of uranium as compared to presently operated LWRs.
- A proven radiological cleanness of TRISO coated particle fuel, resulting from the proven perfect confinement capability of such fuel at high temperatures.
- Reduced high level wastes, resulting from a high degree of depletion in the spent fuel;
- The absence of wastes in the form of activated metals (e.g. irradiated claddings).
- An option to use thorium fuel (GT-MHR) and an option to operate in a closed fuel cycle with the reprocessing of the TRISO coated particle fuel (GT-MHR, GTHTR300, HTR-PM); for the GTHTR300 it is indicated that the feasibility of TRISO fuel reprocessing has already been investigated.

The GTHTR300 design description mentions a problem of possible ^{14}C emissions in burning of the spent core graphite.

Although the reprocessing of fuel is considered for several HTGRs, it is noted that such reactors are not capable of the operation with high conversion or breeding, at least, as comes to the fuel cycles with the U-Pu MOX fuel.

4.3.3. *Liquid metal cooled SMRs*

For liquid metal cooled SMRs, the features commonly mentioned in conjunction with the effective resource utilization, minimized waste and reduced environmental impacts are:

- High conversion or fuel breeding and operation in a closed nuclear fuel cycle, which could ensure a self-sustainable operation mode on fissile materials (with the breeding ratio BR ~1.05) or the expanded breeding (BR>1) to produce fissile materials necessary for other, non-breeder reactors present in the system.
- High conversion or breeding are generally inherent to all reactors with the fast neutron spectrum; however, they could be improved by the use of dense metallic (KALIMER, BMN-170, MDP, PEACER) or nitride (RBEC-M, BMN-170, Medium Scale Lead-bismuth Cooled Reactor) fuel; for the nitride fuel, an option of nitrogen enrichment by ^{15}N is being considered (RBEC-M, Medium Scale Lead-bismuth Cooled Reactor).
- An option to recycle all self-produced transuranic (TRU) elements including minor actinides or to utilize spent nuclear fuel from other reactors, e.g. LWRs, is being examined (e.g., KALIMER, PEACER); both of the abovementioned approaches result in the reduction of long-lived high level waste.
- All innovative liquid metal cooled SMRs target the operation in a closed fuel cycle with the use of the advanced dry methods of fuel reprocessing with a potential to reduce the generation of wastes.

4.3.4. *Non-conventional SMRs*

The designers of AHTR identify high efficiency of energy conversion and advanced non-electric applications as major factors contributing to the effective resource utilization, minimized waste and reduced environmental impacts. Specifically, the reduced water consumption and the improved land use are mentioned, which could result from a dry cooling system considered for this concept.

4.4. Summary of innovative safety features

A short review given in Sections 4.4.1 – 4.4.4 of this Chapter establishes links between certain innovative safety features specified by the designers of SMRs in Annexes I – XXVI, and the innovation directions to enhance the levels of defence in depth as indicated in the IAEA-TECDOC-1434 [3]. For reference, the innovation directions [3] are summarized in Table 4.

TABLE 4. INNOVATION DIRECTIONS TO ENHANCE THE LEVELS OF DEFENCE IN DEPTH [3]

LEVEL OF DEFENCE IN DEPTH	OBJECTIVES	INNOVATION DIRECTION (INPRO)
1	Prevention of abnormal operation and failures	Enhance prevention by increased emphasis on inherently safe design characteristics and passive safety features.
2	Control of abnormal operation and detection of failures.	Give priority to advanced control and monitoring systems with enhanced reliability, intelligence and limiting features.

LEVEL OF DEFENCE IN DEPTH	OBJECTIVES	INNOVATION DIRECTION (INPRO)
3	Control of accidents within the design basis.	Achieve fundamental safety functions by optimized combination of active and passive design features; limit fuel failures; increase grace period to several hours.
4	Control of severe plant conditions, including prevention and mitigation of the consequences of severe accidents.	Increase reliability of systems to control complex accident sequences; decrease severe core damage frequency by at least one order of magnitude, and even more for urban-sited facilities.
5	Mitigation of radiological consequences of significant releases of radioactive materials	No need for evacuation or relocation measures outside the plant site.

4.4.1. Water cooled SMRs

The designers of all innovative water cooled SMRs considered in this report pursue an enhanced prevention or elimination of abnormal operation and failures (Level 1 in Table 4). The design features to achieve such elimination or prevention are specified as the following:

- Integral design of the primary circuit incorporating the steam generators and the pressurizer, providing for the elimination of large-diameter piping and large-diameter reactor vessel penetrations in order to prevent large-break loss of coolant accidents (all designs except MARS, VBER-300, RUTA-70 and KAMADO).
- The use of natural circulation in primary coolant system, to eliminate main circulation pumps and to prevent loss of flow accidents (CAREM-25, IMR, VK-300, CCR, RMWR, AHWR, RUTA-70).
- The in-vessel location of control rod drives to eliminate inadvertent control rod ejection and to prevent transient overpower accidents, as well as to reduce the number of reactor vessel penetrations (IRIS, CAREM, SCOR, IMR).
- Advanced, modular loop type designs with the reduced piping and physical connections between the primary coolant loop and auxiliary circuits (MARS, VBER-300), for the enhanced prevention of loss of coolant accidents.
- A pool type design with no excess pressure in the primary circuit and no reactor pressure vessel, for the enhanced prevention of loss of coolant accidents (RUTA-70).

Advanced control and monitoring systems with the enhanced reliability, intelligence and limiting features (Level 2 in Table 4) are mentioned for all designs that have reached the basic or the detailed design stage, see Table 2.

Regarding the control of accidents within the design basis (Level 3 in Table 4), all water cooled SMRs rely on certain inherent safety features (such as the reactor vessel penetrations located in the upper, steam part of the reactor pressure vessel to ensure that the leakage rate is low and the core is not uncovered in loss of coolant accidents[6]) and incorporate various combinations of passive and active systems. Most of the designs target an increased reliance on passive systems, as benefiting from smaller reactor size. In different water cooled SMRs, passive systems shoulder the functions of back up or main shutdown systems, emergency core

[6] This design approach is used in integral type PWRs and BWRs

cooling systems, decay heat removal systems and others. Providing the diversity, redundancy and independence of the systems and components important to safety is a common approach to accommodate common cause failures. The designers of CAREM, MARS and SCOR emphasize the importance of finding a reasonable combination of passive and active systems to achieve economic benefits in plant design.

The KAMADO concept targets a negligible probability of core melting (Level 4 in Table 4) by developing a pressure tube design with the separated fuel rods and pressure tubes being embedded in the graphite blocks submerged in a water pool with no excess pressure, Annex XIII.

Several designers of water cooled SMRs (e.g., MARS, SCOR, CCR) examine passive in-vessel retention of corium (Level 4 in Table 4).

Last but not least, the designers of many innovative water cooled SMRs target the reduced or eliminated off-site emergency planning (Level 5 in Table 4), pointing to very low (10^{-7} – 10^{-8} 1/year) evaluated or targeted core damage frequencies. Probabilistic safety assessments (PSAs) of a varying degree of detail are indicated for all designs except the SCOR, the IMR, the CCR, the AHWR and the KAMADO.

In presenting defence in depth (DID) approach, the designers of SMART, CAREM, CCR, RMWR and AHWR follow the recommendations of the IAEA safety standard NS-R-1 [15]; other designers describe the DID as a system of physical barriers. Specifically, the designers of IRIS introduce an alternative three-tier approach, Annex II.

Human actions of malevolent character are explicitly addressed in the design of VBER-300, Annex VII.

4.4.2. Gas cooled SMRs

The designers of all gas cooled SMRs (HTGRs) specify the outstanding fission product confinement capability of TRISO type coated particle fuel at high temperatures as very important inherent safety feature contributing to the overall DID concept. Proven in tests and operation, this capability is rated definitive for the prevention of the consequences of severe accidents and, therefore, it essentially belongs to DID Level 4 in Table 4. The role of this inherent barrier in Level 4 is so important that in many HTGR designs the mitigation measures in severe accidents are reduced to a minimum (Annex XVII).

However, the confinement capability of coated particle fuel is conditioned by the ability to shut down the nuclear installation and to transport decay heat from the fuel at any given time to ensure that the fuel will never overheat or be damaged (Annex XIV). Here, all designers refer to the enhanced passive shutdown and heat removal capabilities of HTGRs, which correspond to DID Levels 3 and 4 in Table 4. The design features of HTGRs mentioned in this context are the following:

- Strong negative temperature reactivity feedbacks over the whole temperature range and for all operational states.
- Passive decay heat removal that is effectively accomplished in many reactor states (including those with the completely lost helium coolant) by only natural processes of conduction, convection and radiation in the static structures and media.

Slow and stable response to transients caused by internal and external initiating events, due to large heat capacity of core graphite.

HTGRs with pebble bed fuel and continuous refuelling (PBMR, HTR-PM) provide a relatively small reactivity margin for fuel burn-up (Level 1 in Table 4), resulting in the reactor capacity to survive, without core damage, an unprotected transient overpower caused by the ejection of a control rod.

An impressive number of validations, tests and demonstrations performed for passive shutdown and decay heat removal capabilities of HTGRs (see Annexes XIV – XIX) provide evidences that the inherent and passive safety features of such reactors could play a definitive role in meeting the objectives of the DID Levels 1 and 3. Therefore, the designers of PBMR and GT-MHR argue that active systems, at least those requiring AC power, might actually be not required in the event of a major HTGR accident. Not requiring AC powered safety systems eliminates the need for complex active systems with sensors, controls, actuators, backup power, etc (DID Level 2 in Table 4).

De facto, the principles of redundancy, diversity and physical independence are incorporated in the designs of HTGRs, and various active systems or systems with actuators, such as shutdown or heat removal systems, as well as protection actions, such as stop of the primary helium blower, are provided for in all designs considered in this report.

Preliminary PSAs have been performed or are in progress for the designs of PBMR, GT-MHR and GTHTR300. The evaluated core damage frequencies are very low (10^{-8} 1/year), which motivates all designers to specify reduced or eliminated off-site emergency planning requirements (Level 5 in Table 4).

For DID approach, the recommendations of the IAEA safety standard NS-R-1 [15] are followed in only one design description (HTR-PM, Annex XVII). The designers of other HTGRs describe the DID as a system of physical barriers or provide no DID description at all.

All HTGRs described in this report incorporate no pressure containment of a LWR type. In the event of a substantial failure of the primary coolant pressure boundary, the reactor building is designed to release the low activity helium coolant and serve as a low pressure filtered confinement to retain the longer term (days) limited radionuclide releases calculated for the core under the heatup conditions resulting from the depressurization, Annex XV.

An underground or a half-underground location of the reactor cavity and modules is provided in the designs of PBMR, GT-MHR and GTHTR300 as an enhanced protection against the external events, including those of malevolent human-induced origin.

4.4.3. Liquid metal cooled SMRs

All fast reactors offer extended possibilities to 'build' the desired combinations of reactivity coefficients and effects by an appropriate selection of the design parameters of the core and reactor internals at the design stage. This possibility, resulting from a larger leakage rate of fast neutrons as well as from high conversion, can be effectively used to eliminate certain accidents at the design stage and to ensure the reactor self-control in a variety of unprotected transients.

Different from PWRs and BWRs, fast reactors with MOX fuel may have positive core void reactivity, which might, at least hypothetically, initiate a transient overpower in the reactors with relatively low coolant boiling points (which is the case with sodium cooled reactors). The void reactivity properties of such reactors could be adjusted appropriately at the design stage.

High conversion ratio of fast reactors results in the relatively small core reactivity changes with burn-up (low burn-up reactivity swing). The reactivity swing can be additionally

minimized (e.g., by the use of the advanced, dense fuel forms or heterogeneous core layouts) to achieve the values comparable to a single delayed neutron fraction, which could facilitate the exclusion of overpower transients caused by control rod ejections.

All sodium cooled reactor designs incorporate sodium based intermediate heat transport systems to prevent potential contacts of the power circuit water with the primary sodium, and to exclude the contamination of the primary circuit with sodium-water reaction products. The designs based on heavy lead-bismuth or lead coolants , which are chemically inert with water, incorporate no intermediate heat transport system but require a reliable system of primary coolant chemistry control to prevent the erosion and corrosion of claddings and other structural materials.

All liquid metal cooled SMRs have no excess pressure in the primary circuit (omit a hydrostatic pressure of the liquid metal coolant).

All innovative liquid metal cooled SMRs considered in this report target an enhanced prevention or elimination of abnormal operation and failures (Level 1 in Table 4). The design features specified in this connection are as follows:

- Integral design of the primary circuit with the use of a secondary guard vessel and location of the reactor module in a concrete silo, to prevent loss of coolant accidents (lead-bismuth cooled RBEC-M, Medium Scale Lead-bismuth Cooled Reactor); adjacent design of the primary and the intermediate circuits, with the space between main (primary) and safeguard (secondary) reactor vessels being filled with sodium and acting as an intermediate heat transport system, also to prevent loss of coolant events (sodium cooled BMN-170, MDP).
- A reduced burn-up reactivity swing in the core, achieved with the use of dense metallic or nitride fuel (all designs except the PEACER), to prevent transients with the ejection of control rods.
- A high degree of primary coolant natural circulation, ensured by the physical properties of lead-bismuth coolant that also exclude exothermic reactions with air and water (all lead-bismuth cooled SMRs).
- The use of a natural circulation based primary circuit, with natural circulation being facilitated by an active passive gas lift system, which shares the functions of lead-bismuth coolant chemistry control and prompt reactivity control, to ensure that the reactor is shut down immediately following any disruption of the corrosion control (RBEC-M).

Advanced methods of water leak detection (Level 2 in Table 4) are mentioned for the design of the MDP.

In all designs of liquid metal cooled SMRs, the fundamental safety functions related to control of accidents within the design basis (Level 3 in Table 4) are achieved by the redundant and diverse combinations of active and passive systems, with a strong role of the inherent safety features in reactor self-control. All design descriptions provide the results of safety analyses indicating that several unprotected transients, such as the unprotected transient overpower, the unprotected loss of flow, the unprotected loss of heat sink, or the unprotected total NPP blackout could be safely controlled by the appropriate combinations of reactivity effects only. In many cases, such accidents are included in the design basis of liquid metal cooled SMRs.

For all designs, active systems include a reactivity control and shutdown system based on the mechanical control rods. All designs incorporate passive air-cooled reactor vessel auxiliary cooling systems (RVACSs) or equivalents. Natural circulation based passive decay heat

removal is provided in all designs of lead-bismuth cooled SMRs. Other passive systems are diverse and may include gas expansion modules (GEMs, see Annex XX), the hydraulically suspended control rods (Annex XXI), etc.

Certain inherent safety features of innovative liquid metal cooled SMRs could contribute effectively to DID Levels 3 and 4, Table 4. A reduced or negative void reactivity effect (sodium cooled BMN-170); or the core reactivity maximum exactly matching the reactor operational state as achieved with the use of an active gas lift system (lead-bismuth cooled RBEC-M); prevent the propagation of accidents beyond the design basis or ensure an enhanced control of hypothetical severe plant conditions, when coolant is boiling or lost. In addition to this, high heat capacities of the primary or the adjacent primary and secondary circuits are specified as factors contributing to a slow and stable progression of transients (BMN-170, MDP, RBEC-M). High heat conductivity of the metallic fuel is identified in the design descriptions of the KALIMER, the MDP and the PEACER as contributing to lower core temperatures in anticipated transients without scram (ATWSs).

The designers of many liquid metal cooled SMRs argue that the design features within DID Levels 1-3 (see Table 4) could be sufficient to ensure no radioactivity release from fuel in any conceivable accidents. At the same time, the complete lists of design basis accidents (DBAs) and beyond design basis accidents (BDBAs) have been defined only for the design of KALIMER, Annex XX.

Guard or secondary reactor vessels to mitigate radiological consequence of the postulated severe accidents (DID Level 4 in Table 4) are provided in the designs of BMN-170, MDP, RBEC-M, and PEACER.

No PSAs have been so far performed for the innovative liquid metal cooled SMRs addressed in this report. Based on the analyses of the PSAs previously performed for other fast reactors, several designers of innovative liquid metal cooled SMRs specify $10^{-7} – 10^{-8}$ 1/year as a target for core damage frequency.

The designers of KALIMER suggest that no intervention in the public domain beyond the plant boundary as a consequence of any hypothetical core disruption accident would be required.

Regarding DID approach, the recommendations of the IAEA safety standard NS-R-1 [15] are followed in the design descriptions of KALIMER (Annex XX) and RBEC-M (Annex XXIII). The designers of other liquid metal cooled SMRs describe the DID as a system of physical barriers.

4.4.4. Non-conventional SMRs

As it was already mentioned, the AHTR concept (Annex XXVI) brings together the technologies of HTGRs (high temperature fuel, gas turbine Brayton power cycle), molten salt reactors (liquid salt coolant that, in the AHTR case, contains no fuel) and fast sodium cooled reactors (no excess pressure in the primary circuit; intermediate heat transport system). The AHTR uses forced circulation of the primary liquid salt coolant and its intermediate heat transport system is also based on liquid salt.

The safety concept of AHTR provides for the extended use of passive systems in design basis accidents and for a strong reliance on the inherent and passive safety features in beyond design basis accidents.

Regarding the prevention of abnormal operation and failures (DID Level 1 in Table 4), the important innovative feature of the AHTR is low (atmospheric) operating pressure of the primary liquid salt coolant, which, at operating conditions, has the heat transfer properties

similar to those of water. As the AHTR power circuit is based on a gas turbine Brayton cycle, an intermediate circuit becomes necessary to prevent the ingress of a pressurized power circuit medium (helium or nitrogen gas) to the primary circuit.

Regarding the control of accidents within the design basis (DID Level 3 in Table 4), the design of AHTR incorporates a mechanical reactivity control and shutdown system based on control rods with the external drives, and two diverse decay heat removal systems, of which one is passive and one is active. The reference AHTR design uses passive reactor vessel auxiliary cooling (RVAC) systems similar to that developed for decay heat removal in the General Electric sodium cooled S-PRISM reactor. Different from its prototype, the RVAC system of the AHTR relies not only on the processes of convection and conduction but on the radiation also.

Similar to gas cooled SMRs (see Section 4.3.2), the proven fission product confinement capability of TRISO type coated particle fuel at very high temperatures is specified as an important inherent safety feature for the prevention of consequences of severe accidents (DID Level 4 in Table 4). In addition to this, chemical inertness and low pressure of the liquid salt coolant eliminate the potential for damage to the confinement structure by rapid chemical energy releases or coolant vaporization. It is also mentioned that most fission products (excluding primary krypton and xenon) and all actinides escaping the fuel are soluble in the liquid salt and will remain in the liquid salt at very high temperatures.

As the AHTR concept includes a very high temperature reactor option (with the core outlet temperature of 1000°C), the additional design features to prevent the consequences of severe accidents are provided. The reactor has a second guard vessel and is located in an underground silo. The effectiveness of heat conduction to earth through the walls of the silo (an ultimate heat rejection in BDBA) is being examined.

In terms of passive decay heat removal systems, a major difference is noted between the liquid cooled AHTR and gas cooled reactors. As primary circuit depressurization and the loss of coolant resulting in core uncovery are eliminated for the AHTR by design, the plant could be built in very large sizes (>2400 MW(th)), while the maximum size of a gas cooled reactor with passive decay heat removal systems is limited to ~600 MW(th).

The DID concept and design basis accidents are not addressed in detail in the description of the AHTR, apparently because of an early (pre-conceptual) stage of this design.

4.5. Technical features and technological approaches that support proliferation resistance of innovative SMRs

All NPPs with innovative SMRs will provide for the implementation of the established safeguards verification procedures under the agreements of Member States with the IAEA. In addition to this, many innovative SMRs offer certain intrinsic proliferation resistance features to prevent the misuse, diversion or undeclared production of fissile materials and/ or to facilitate the implementation of safeguards. A short review of these features for different SMR types is given in Sections 4.5.1 – 4.5.2 below.

4.5.1 Water cooled SMRs

Most of the design descriptions of water cooled SMRs specify low enrichment uranium and once-through fuel cycle as basic options. Therefore, the features contributing to proliferation resistance of such SMRs are essentially similar to that of presently operated PWRs and BWRs. They include low enrichment of fresh fuel, low residual

enrichment, an unattractive isotopic composition of the plutonium in the discharged fuel, and radiation barriers provided by the spent fuel.

Several designs (SMART, IRIS, MARS, IMR) offer an increased duration of the operation cycle, which, in view of the reduced number of inspections, could be rated as a feature facilitating the implementation of safeguards.

Other features specified by the designers of individual water cooled SMRs are the following:

- Increased moderator-to-fuel ratio, contributing to a higher burn-up rate and the degraded composition of the secondary plutonium (IRIS).
- Restricted refuelling area and a remote seismic monitoring of the fuel assemblies' pool (CAREM).
- An option to use heterogeneous uranium-thorium fuel (Radkowsky Thorium Fuel) to reduce the specific plutonium production by 4-5 times (VBER-300), see section 1.5 of Annex VII.
- The available infrastructure for nuclear ship service and maintenance (for a floating NPP with the VBER-300 reactors).

The designers of the RMWR and the AHWR denote closed nuclear fuel cycles as basic. The technical features mentioned in conjunction with proliferation resistance are as follows:

- The use of a simplified PUREX process with low decontamination factor; the use of a process flowchart with no separation of uranium and plutonium at any fuel cycle stage (RMWR).
- Low excess reactivity preventing the undeclared production of fissile materials; high content of ^{232}U in ^{233}U (AHWR).

4.5.2. Gas cooled SMRs

The intrinsic proliferation resistance features common to all gas cooled SMRs (HTGRs) include high fuel burn-up (low residual inventory of plutonium, high content of ^{240}Pu), a difficult to process fuel matrix, radiation barriers, and a low ratio of fissile to fuel block/ fuel pebble mass.

Although several HTGRs, e.g. the GT-MHR, the GTHTR300 and the HTR-PM, make a provision for reprocessing of the TRISO fuel, the corresponding technology has not been established yet and, until such time as when the technology becomes readily available, the lack of the technology is assumed to provide an enhanced proliferation resistance, Annex XV.

In addition to this, the GT-MHR provides for the use of two types of coated particles, fissile and fertile, so that their sorting becomes necessary in any attempt of fresh fuel diversion.

The designers of the ACACIA suggest that a cartridge type design of the ACACIA core, with the full core replacement foreseen once in every three years, could facilitate nuclear material accounting and verification, see Annex XIX.

4.5.3. Liquid metal cooled SMRs

All liquid metal cooled SMRs are fast reactors that can ensure a self-sustainable operation on fissile materials or realize fuel breeding to feed other reactors present in nuclear energy systems. In both cases, and if the fuel cycle is closed, the need of fuel enrichment and relevant uranium enrichment facilities would be eliminated, which is reasonably mentioned as a factor contributing to enhanced proliferation resistance.

All liquid metal cooled SMRs addressed in this report provide for the operation in a closed fuel cycle with the use of the advanced methods of dry reprocessing of fuel. In this context, the intrinsic proliferation resistance features commonly mentioned by the designers of such SMRs are:

- Inherently low decontamination factor of fuel; the non-aqueous (dry) methods of fuel reprocessing could secure an incomplete removal of fission products and the separation of only curium from the fuel, which makes it possible to produce fresh fuel for fast reactors but prevents the use of such fuel for weapon programmes.
- No separation of plutonium and uranium at any fuel cycle stage.

Specifically, fuel breeding is mentioned as a factor that could help minimize the necessary number of fast reactors in nuclear energy systems. In turn, this might facilitate the location of fast reactors within a limited number of centres operated under an international control, see Annexes XXI, XXIII.

Other features specified by the designers of individual liquid metal cooled SMRs are the following:

- Low burn-up reactivity swing that complicates the undeclared production of fissile materials (KALIMER).
- Fuel assemblies with wide lattice pitch and central channels, to facilitate the verification of all fuel rods within each assembly (RBEC-M).
- Denaturing of the fissile materials, e.g., through the optimization of the core design to achieve a higher content of ^{238}Pu in the plutonium (PEACER).

4.5.4. Non-conventional SMRs

The intrinsic proliferation resistance features of the AHTR are essentially identical to that of the gas cooled reactors using graphite matrix coated particle fuel (see Section 4.5.2), because the same fuel and fuel cycle are used.

4.6. Technical features and technological approaches that facilitate physical protection of innovative SMRs

All NPPs with innovative SMRs provide for the implementation of standard or advanced security measures to assure physical protection of the plant against internal or external human actions of malevolent character. In additions to this, innovative SMRs may incorporate certain design features provided to resist such actions and/ or to facilitate security measures. A short review of these features for certain SMR types is given in Sections 4.6.1 – 4.6.4 below.

4.6.1. Water cooled SMRs

Most of the innovative water cooled SMRs addressed in this report incorporate the following technical features to support physical protection of the plant:

- The containment designed to resist extreme natural and human-induced external events.
- Inherent and passive safety features and passive systems incorporated in the design to prevent certain accidents from occurring, to ensure longer grace periods and to eliminate dependence on the external water and power supply and operator actions; these features include integral designs of the primary coolant systems (all designs except MARS, VBER-300, and AHWR), natural circulation based primary coolant

systems (CAREM-25, IMR,VK-300, CCR, RMWR, AHWR, RUTA-70), in-vessel control rod drives (IRIS, CAREM, SCOR, IMR), and natural circulation based residual heat removal systems (all designs).

Certain water cooled SMRs incorporate additional features to enhance physical protection of the plant, such as:

- A safeguard vessel, or an additional metallic enclosure of the primary loop, or a double containment (SMART, MARS, VBER-300, VK-300, AHWR).
- A compact containment with only half of it located above the ground level, thus leaving exposed only a relatively small area (IRIS); an option of an underground location of the NPP (RUTA-70, VK-300).
- Redundancy and separation of the air stacks of a passive decay heat removal system (SCOR).
- Protection of the water area by floating structures or objects (a floating NPP with the VBER-300).

4.6.2. Gas cooled SMRs

Most of the innovative gas cooled SMRs addressed in this report incorporate the following technical features to support physical protection of the plant:

- The inherent and passive safety features common to all HTGRs, including large temperature margin of coated particle fuel; an outstanding passive shutdown and decay heat removal capability; slow and stable response to transients caused by the internal and external initiating events (due to a large heat capacity of the core graphite); altogether, these features reduce or eliminate dependence on the external power supply and operator actions.
- The concrete reactor building (confinement) and the reactor cavity designed for protection against extreme external impacts.
- An underground or a half-underground location of the reactor cavity and modules (PBMR, GT-MHR, GTHTR300).

4.6.3. Liquid metal cooled SMRs

Most of the innovative liquid metal cooled SMRs addressed in this report incorporate the following technical features to support physical protection of the plant:

- The containment designed to resist extreme natural and human-induced external events.
- Strong reliance on the inherent and passive safety features and passive systems to prevent certain accidents from occurring, to ensure reactor self-control in unprotected transients, and to reduce or eliminate dependence on the external power supply and operator actions; these features include integral designs of the primary circuit or adjacent configurations of the primary and the intermediate circuit (BMN-170, MDP, RBEC-M, Medium Scale Lead-bismuth Cooled Reactor); the reduced or negative void reactivity effects and a minimized burn-up reactivity swing; and passive systems for the reactor shutdown and decay heat removal from the reactor vessel.

Certain liquid metal cooled SMRs incorporate additional features to enhance physical protection of the plant, such as:

- An underground location of the containment vessel (KALIMER).
- A double containment (PEACER).
- Large heat capacity of the primary circuit, resulting in slow transients (BMN-170, RBEC-M).

4.6.4. Non-conventional SMRs

The only design in this category – the AHTR – incorporates the features similar to that of the innovative gas cooled SMRs (HTGRs). Large thermal margins of coated particle fuel and the location of the reactor in an underground silo are mentioned in conjunction with physical protection of the plant.

4.7. Conclusion to Chapter 4

A review provided in this chapter points to many synergies in the designers' efforts to improve performance of innovative SMRs in the subject areas of economics, resource utilization, waste management, environmental impacts, safety, proliferation resistance, and physical protection. It is recognized that the designers of many innovative SMRs already apply an approach to incorporate several of the abovementioned subject areas in their original design concepts. Safety features of the innovative SMRs addressed in this report are well in line with the innovation directions to enhance the levels of defence in depth as indicated in the IAEA-TECDOC-1434 [3].

5. ENABLING TECHNOLOGIES THAT REQUIRE FURTHER R&D

5.1. Introduction

The lists of enabling technologies provided in each of the SMR design descriptions (Section 1.8 of Annexes I–XVII, XIX–XXIV, XXVI; Section 3 of Annexes XVIII and XXV) make it possible to identify the directions of necessary further R&D. Tables III.1 – III.9 of Appendix III present crosscuts of the R&D both ongoing and planned for the SMRs included in this report.

For convenience, Tables III.1 through III.9 follow the pattern used in the annexes to this report, i.e., water cooled SMRs appear first, followed by gas cooled, sodium cooled, lead-bismuth cooled and non-conventional SMRs. The order of certain designs within each of the abovementioned groups also follows the sequence used in the annexes, with the corresponding annex number being indicated against each SMR name.

Being arranged in the abovementioned way, Tables III.1 – III.9 of Appendix III appear to be self-standing; therefore, sections 5.2 – 5.10 below outline only those R&D areas that are common to several innovative SMR designs within each group. These sections are structured as the following:

- Reactor core and fuel design – Section 5.2 (Table III.1).
- Reactor internals and primary circuit – Section 5.3 (Table III.2).
- Power circuit – Section 5.4 (Table III.3).
- Safety concept, safety features and systems – Section 5.5 (Table III.4).
- Technologies for fabrication of fuel and materials, fuel reprocessing and waste disposal – Section 5.6 (Table III.5).

- Technologies of core refuelling, plant maintenance and in-service inspection – Section 5.7 (Table III.6).
- Issues of NPP licensing, construction, operation and decommissioning – Section 5.8 (Table III.7).
- Calculation technologies and data sets – Section 5.9 (Table III.8).
- Available or planned experimental facilities – Section 5.10 (Table III.9).

In addition to the abovementioned, Section 5.11 gives a summary of the technologies for potable water and hydrogen production considered for use or already incorporated in the designs of certain innovative SMRs.

5.2. R&D for reactor core and fuel design

Table III.1 of Appendix III collates the ongoing and planned R&D, relevant for reactor core and fuel design.

Achieving fuel burn-up increase through the development and use of advanced fuel designs is identified as a common objective for the majority of water cooled, gas cooled and liquid metal cooled SMRs.

For high temperature reactors, the focus is also on further acquisition of the data on carbon materials performance under increased temperatures and fuel burn-ups, including the anticipated accident conditions.

Further development of the technologies of nitride and metallic fuel is a common issue for the majority of sodium cooled and lead-bismuth cooled SMRs.

5.3. R&D for reactor internals and primary circuit

Table III.2 of Appendix III summarizes the ongoing and planned R&D of relevance to reactor internals and primary circuits.

Several of the innovative water cooled SMRs incorporate integral designs of primary circuits. The corresponding further R&D will include validation and testing of the performance of such integral circuits and their main components. Specifically, the need of further R&D on natural circulation is specified for many innovative water cooled SMRs.

For high temperature reactors, the focus is on testing of the reactor system components in high temperature and high pressure helium environment and on the material development programmes. Specifically, the need of major material development programmes is indicated for those designs that target a very high temperature option (the NGNP versions of the PBMR and the GT-MHR; the AHTR).

For lead-bismuth cooled SMRs, a common issue is the R&D on lead-bismuth coolant technology to ensure corrosion-free operation of the fuel element claddings and the primary circuit equipment.

5.4. R&D for power circuit

Table III.3 of Appendix III identifies the ongoing and planned R&D for power circuits.

The majority of inputs in this table are for the gas cooled SMRs (HTGRs) targeting the use of direct gas turbine cycles. The R&D programmes address the development of turbo-machinery design, the validation and testing of power conversion system components, and further development and integrated tests of the complete power conversion systems.

5.5. R&D for safety features and systems

Table III.4 of Appendix III provides a crosscut of the directions of further R&D for safety features and systems of the addressed innovative SMRs.

Regarding safety features, there is an emphasis on further R&D for certain inherent and passive safety features that may define the performance of SMRs in accidents. Such features include combinations of reactivity coefficients, low operating pressures, high thermal conductivities of fuel materials, and passive shutdown capabilities.

Regarding control and protection systems, several innovative water cooled SMRs target the use of control rod drives (CRDs) of the advanced designs. In this connection, the upper entry CRDs could be mentioned for BWRs, and the in-vessel CRDs for PWRs.

The majority of inputs for all SMRs correspond to the ongoing or planned R&D for a variety of different passive systems. Specifically, the in-vessel retention of corium by passive means is indicated as a design objective for three water cooled SMRs.

The designers of 3 SMRs of different types identify the need of further R&D on seismic design.

5.6. R&D for technologies of fuel and materials fabrication, fuel reprocessing and waste disposal

Table III.5 of Appendix III specifies the directions of further R&D on the technologies of fuel and materials fabrication, fuel reprocessing and waste disposal.

The R&D to establish a large scale manufacturing of qualified fuel elements is specified for the majority of high temperature SMRs, including both HTGRs and the AHTR. The R&D on fuel reprocessing is indicated for some HTGR designs.

Further development of dry reprocessing technologies, specifically for the nitride and metallic fuel, is identified for the majority of liquid metal cooled SMRs.

5.7. R&D for technologies of core refuelling, plant maintenance and in-service inspection

Table III.6 of Appendix III points to the ongoing and planned R&D relevant for the technologies of core refuelling, plant maintenance and in-service inspection.

Specifically, the R&D on advanced methods of in-service inspection is indicated for several SMR designs representing different reactor types.

5.8. R&D for technologies and issues of NPP licensing, construction, operation and decommissioning

Table III.7 of Appendix III collates the ongoing and planned R&D on the technologies and issues of NPP licensing, construction, operation and decommissioning. Though not always identified by the designers, such issues as improvement of the modular construction, reduction of the operating costs by plant operational life increase, development of the plant decommissioning programme and establishment of the licensing requirements for innovative SMRs are generally important for many, if not all innovative SMRs presented in this report.

5.9. R&D for calculation technologies and data sets

Table III.8 of Appendix III summarizes the ongoing and planned developments of the calculation technologies and data sets relevant for the SMR designs considered. The R&D directions cover the neutronic codes and nuclear data, the thermal-hydraulic and safety analyses codes, the uncertainty analysis codes, the depletion codes, the fuel performance codes, and the databases of material properties.

5.10. Available and planned tests facilities

Table III.9 of Appendix III indicates the available or planned test facilities in Member States with respect to further R&D programmes for innovative SMRs. The majority of facilities specified are for water cooled and gas cooled SMRs, which reflects a more advanced design status achieved by such reactors. At the same time, Table III.9 includes several facilities for lead-bismuth cooled reactors.

Regarding the potential to share design and technology development with reactors of other types, a remarkable example is provided by the AHTR, a pre-conceptual system that is part of the U.S. Department of Energy Generation IV reactor programme. About 70% of the R&D required for the AHTR is shared with that for helium cooled high temperature reactors. This includes fuel development, materials development, and Brayton power cycles, Annex XXVI.

5.11. Technologies for non-electric applications

Most of the innovative SMRs addressed in this report provide for one or more non-electric applications, see Table 2.

Whenever described in this report, the systems for district heating are always based on standard equipment and proven system configurations, with no further R&D specified.

Table 5 summarizes the technologies for potable water production identified in the design descriptions of those SMRs for which such options are foreseen.

At least one high temperature reactor design, the GT-MHR (Annex XV), includes the provisions to produce potable water using heat rejected in the power cycle, without reducing the efficiency of electric power generation.

TABLE 5. TECHNOLOGIES OF POTABLE WATER PRODUCTION SPECIFIED IN THE DESIGN DESCRIPTIONS OF SMRs

SMR NAME	ANNEX NO.	TECHNOLOGY OF POTABLE WATER PRODUCTION
Water cooled SMRs		
SMART	I	Multi-effect distillation with thermal vapour compressor (MED-TVC)
IRIS	II	Not specified
CAREM	III	Reverse osmosis
MARS	IV	Multi-effect distillation (MED) or thermo-compression
SCOR	V	Multi-effect distillation (MED) or thermo-compression
IMR	VI	Not specified
VBER-300	VII	Reverse osmosis
VK-300	VIII	Not specified
CCR	IX	Reverse osmosis
AHWR	XI	Low temperature multi-effect distillation (LT-MED)
RUTA-70	XII	MED

SMR NAME	ANNEX NO.	TECHNOLOGY OF POTABLE WATER PRODUCTION
	Gas cooled SMRs	
PBMR	XIV	Not specified
GT-MHR	XV	MED as an option, with the use of heat rejected in thermodynamic cycle
GTHTR300	XVI	Not specified
ACACIA	XIX	MSF
	Sodium cooled SMRs	
BMN-170	XXI	Not specified

None of the SMR design descriptions in this report identifies the need of further R&D on the technologies of seawater desalination. However, a certain amount of R&D on the selection and optimization of a potable water production process and the system configuration could be foreseen for the cases when such a selection has not been made.

Table 6 gives a summary of the technologies of hydrogen production as outlined in the design descriptions of SMRs in this report. The SMRs targeting hydrogen production as a high temperature process heat application are mostly HTGRs. It is noted that hydrogen production using high-temperature thermochemical processes is foreseen for one direct flow water cooled reactor and one non-conventional high temperature reactor concept.

TABLE 6. TECHNOLOGIES OF HYDROGEN PRODUCTION SPECIFIED IN THE DESIGN DESCRIPTIONS OF SMRs

SMR NAME	ANNEX NO.	TECHNOLOGY OF HYDROGEN PRODUCTION
	Water cooled SMRs	
KAMADO	XIII	Thermochemical process (to be defined) driven by high temperature (800^{o}C) process steam.
	Gas cooled SMRs	
PBMR	XIV	Sulphuric acid decomposition reactor and vaporizer and the HI decomposition reactor.
GT-MHR	XV	Several technologies have considered: • Electrolysis (net heat-to-hydrogen efficiency 36–38%) • Electrolysis at 900^{o}C (net efficiency 50%) • Thermochemical water splitting *net efficiency 50%) • The sulphur-iodine (S-I) thermochemical water-splitting cycle has been selected as one of the most promising (net efficiency more than 50% at 950^{o}C reactor outlet temperature)
GTHTR300	XVI	High temperature process heat applications (not specified)
FAPIG-HTGR	XVIII	High temperature process heat applications (not specified)
	Non-conventional SMRs	
AHTR	XXVI	High temperature process heat applications (to be defined)

6. NON-TECHNICAL FACTORS AND ARRANGEMENTS THAT COULD FACILITATE DEVELOPMENT AND DEPLOYMENT OF INNOVATIVE SMRs

6.1. Summary of the design descriptions

The non-technical factors and arrangements that, in view of the designers, could facilitate effective development and deployment of many innovative SMRs are summarized in Table 7.

Provisions for full-scope fuel cycle service agreements and simplified licensing requirements are mentioned by the designers of 9 SMRs representing different reactor types. Simplified licensing requirements are most often associated with the reduced or eliminated off-site emergency planning.

Options of NPP leasing or power purchase contracts with the plant being operated by a multinational generation company are rated important by the designers of 5 SMRs.

TABLE 7. NON-TECHNICAL FACTORS AND ARRANGEMENTS THAT COULD FACILITATE DEVELOPMENT AND DEPLOYMENT OF INNOVATIVE SMRs

NON-TECHNICAL FACTOR OR ARRANGEMENT	SMR TYPE SMRS FOR WHICH THIS PARTICULAR FACTOR OR ARRANGEMENT WAS SPECIFIED IN THE DESIGN DESCRIPTION (**SMR NAME**, ANNEX NO.)	TOTAL NUMBER OF INPUTS
Provisions for full-scope fuel cycle service agreements or fuel leasing	*Water cooled SMRs* **IRIS** (Annex II); **CAREM** (Annex III); **VBER-300** (Annex VII); **RMWR** (Annex X); **RUTA-70** (Annex XII) *Gas cooled SMRs* **GT-MHR** (Annex XV); **ACACIA** (Annex XIX) *Liquid metal cooled SMRs* **BMN-170** (Annex XXI); **RBEC-M** (Annex XXIII)	9
Simplified licensing requirements possible*	*Water cooled SMRs* **SMART** (Annex I); **IRIS** (Annex II); **CAREM** (Annex III); **MARS** (Annex IV); **VBER-300** (Annex VII); **AHWR** (Annex XI) *Gas cooled SMRs* **PBMR** (Annex (XIV)**; GT-MHR** (Annex XV); **HTR-PM** (Annex XVII)	9
Options for NPP leasing or power purchase contracts with the plant operated by a multinational generation company	*Water cooled SMRs* **CAREM** (Annex III); **VBER-300** (Annex VII); **RMWR** (Annex X) *Gas cooled SMRs* **GT-MHR** (Annex XV); **ACACIA** (Annex XIX)	5
International cooperation in the design and technology development for SMRs	*Water cooled SMRs* **IRIS** (Annex II) *Gas cooled SMRs* **GT-MHR** (Annex XV) *Liquid metal cooled SMRs* **KALIMER** (Annex XX)	3

NON-TECHNICAL FACTOR OR ARRANGEMENT	SMR TYPE SMRS FOR WHICH THIS PARTICULAR FACTOR OR ARRANGEMENT WAS SPECIFIED IN THE DESIGN DESCRIPTION (**SMR NAME**, ANNEX NO.)	TOTAL NUMBER OF INPUTS
Option of a turnkey contract for a NPP	*Water cooled SMRs* **SMART** (Annex I); **CAREM** (Annex III) *Gas cooled SMRs* **GT-MHR** (Annex XV)	3
Increased local participation in NPP construction	*Water cooled SMRs* **MARS** (Annex IV); **VBER-300** (Annex VII); **AHWR** (Annex XI)	3
Offer of training	*Water cooled SMRs* **SMART** (Annex I); **AHWR** (Annex XI) *Gas cooled SMRs* **GT-MHR** (Annex XV)	3
No information provided	*Water cooled SMRs* **SCOR** (Annex V); **KAMADO** (Annex XIII); *Gas cooled SMRs* **FAPIG-HTGR** (Annex XVIII) *Liquid metal cooled SMRs* **PEACER** (Annex XXIV); **MS-LBCR** (Annex XXV) *Non-conventional SMRs* **AHTR** (Annex XXVI)	6

* With a reference to both, Annexes I through XXVI of this report and the IAEA-TECDOC-1451 [7]

6.2. Addressing specific needs of certain markets at the design stage

It is remarkable to note that only 4 out of the 26 SMR design descriptions in this report mention explicitly the specific need of developing countries or a specific developing country perspective. Most of the innovative SMR designers apparently target many markets worldwide by offering the flexibility in NPP design, siting and applications; the sound and transparent safety concept based to a large extent on passive safety design options; the improved economy to meet the demands of markets with a limited nonrecurring investment capability; and certain design features that could improve the proliferation resistance and physical protection of SMR based NPPs.

There are increasing practices of periodical assessments of the competing innovative reactor programmes in Member States, as observed in several developed and developing countries [3, 4]. Therefore, the potential vendors are motivated to explore new market opportunities, sometimes starting from rather early design stages.

As an example, a preliminary economic feasibility study of a nuclear desalination plant with the SMART reactor in the Madura Island of Indonesia has been completed in 2003, see Annex I.

The PBMR was first analyzed from the perspective of its value to the nation of South Africa, and later it was recognized that the economic advantages of the PBMR would not be limited to the South African grid alone. In 2000, the PBMR (Pty) Ltd. company was formed with international investment partners to build and market PBMR-based power plants. The long term marketing approach taken by the partners of PBMR (Pty) Ltd is to sell plants comprised of multiple of modules. This has led the PBMR partners to develop early relationships with strategic suppliers for key equipment, see Annex XIV.

The IRIS borrows from the resources of the international consortium that includes members from 10 countries, Annex II. All consortium members are equal partners, and provide inputs regarding specific market requirements, that are ultimately reflected in the IRIS design.

The development and deployment of commercial GT-MHRs is based upon leveraging an ongoing international project to develop and deploy a multi-module GT-MHR designed to consume excess weapons grade plutonium in the Russian Federation, see Annex XV.

The designers of the VK-300 (Annex VIII) and the RUTA-70 (Annex XII) target certain sites in their country of origin, the Russian Federation. For the RUTA-70, cooperation was established with local authorities in the potential site area, and the design was adjusted accordingly.

The VBER-300 ensures an infrastructure support of the potential customers, Annex VII.

The PBMR, the GT-MHR, the GTHTR300 (Annex XVI) and the AHTR (Annex XXVI) target a very high temperature reactor option, which could make them competitive in future markets of non-electric applications, such as hydrogen production.
The KALIMER seeks cooperation with or within the Generation-IV International Forum programme, Annex XX.

The AHTR is part of the U.S. Department of Energy Generation IV reactor programme and is being actively investigated.

Most of the innovative SMRs addressed in this report require a prototype plant to be built to demonstrate reliable operation and qualify certain innovative features. Many of the considered SMRs are still at the conceptual design stage and would require multiple further R&D. Although there are examples of industry and utilities involvement in the design and technology development for innovative SMRs (e.g. PBMR, CCR, IMR, etc.), these are the commitments of governments that remain decisive for the progress in SMR development and deployment

7. PROGRAMMES FOR INNOVATIVE SMR DEVELOPMENT IN MEMBER STATES

7.1. Argentina

The main activities on innovative SMR development in Argentina are centred on the CAREM project (Annex III), which has the goal to develop, design and construct an innovative, simple and small nuclear power plant. This plant is based on an indirect cycle, integral type small PWR with some distinctive and characteristic features that simplify the design and contribute to a high safety level and the improved economics. The CAREM is a CNEA (Comisión Nacional de Energía Atómica) project developed co-jointly with the INVAP, an Argentine company. The R&D for CAREM is supported under a national programme. The project of a CAREM prototype, which is in the detailed design stage, is ongoing with the validation, testing and qualification of innovative components for the primary coolant system.

The CAREM research, design and demonstration (RD&D) costs necessary for safety acceptance are estimated as US $95 million, including the construction of a 100 MW(th) prototype (US $85 million) and other specific tests (US $10 million).

7.2. China

The HTR-PM is a modular high temperature gas cooled reactor (HTGR) plant being designed by the Institute of Nuclear and New Energy Technology (INET) of the Tsinghua University in China. The current HTR-PM design features a 160 MW(e) per module output. The HTR-PM is being promoted as an industrial demonstration plant.

The utilities and nuclear industry partners have confirmed their intention to participate in the HTR-PM project, which also has a support from the Government of China. Currently, a siting evaluation is being performed for the first demonstration plant and for the follow-up units. The HTR-PM design is now at the conceptual stage, with the activities on design optimization being underway. All participants of the project undertake great efforts, and the target is to have the HTR-PM demonstration plant constructed around 2010.

7.3. France

Since 2000, the Nuclear Energy Division of the Commissariat à l'Energie Atomique (CEA) develops a concept of the Simple Compact Reactor (SCOR). The SCOR is being developed in line with the European utility requirements for enhanced reliability and safety and improved economics for the next generation reactors, see Annex V.

The SCOR is a 2000 MW(th) integral type PWR being developed within the framework of a French programme on the innovation for light water reactors (LWRs). The work was partially supported by the AREVA / FRAMATOME-ANP.

The SCOR is at the conceptual design stage. Because of a limited volume of necessary R&D, it is estimated that the project could be implemented within the next 15 years.

7.4. India

Omit small reactors without on-site refuelling, the major activity for innovative SMRs in India is that for the Advanced Heavy Water Reactor (AHWR), which is a pressure tube type light water cooled heavy water moderated reactor of 300 MW(e) maximum electric output, see Annex XI. The AHWR makes use of the specific technologies of pressurized heavy water reactors (PHWRs) pertaining to pressure tubes and low-pressure moderator design. The differences are mainly related to the use of thorium based fuel with negative void coefficient of reactivity, the use of boiling light water in natural circulation mode and a strong reliance on passive safety design options to achieve a high safety level and the improved economy.

The R&D for AHWR is fully supported by the Government of India. The basic design of the reactor and the detailed design of its major nuclear systems have been completed. The RD&D for AHWR has been and is being performed at the Bhabha Atomic Research Centre (BARC). The Nuclear Power Corporation of India Ltd. (NPCIL) has completed a peer review of the design in September 2003. The Indian Atomic Energy Regulatory Board (AERB) has been approached for initially carrying out a pre-licensing safety appraisal of the AHWR.

7.5. Italy

Italy has no plans for nuclear power. Therefore, any design and technology development for innovative reactors can be performed only on the initiative and with the resources of interested research and academic institutions. The MARS design (Annex IV) was developed at the Department of Nuclear Engineering and Energy Conversion of the University of Rome "La Sapienza", see Annex IV. The MARS (Multipurpose Advanced Reactor, inherently Safe) is a 600 MW(th) single loop pressurized light water reactor (PWR). Finding a synergy

between plant safety and economic competitiveness was a challenge to the plant design team, which also had the support of experts from ENEA (the Italian Governmental Agency for Energy and Environment) and ENEL (the Italian Board for Electric Energy Production).

The basic design, the detailed design of main innovative mechanical components, and the technical specifications for the majority of the fluid systems of the MARS are targeted for the completion in 2006 will be completed. The final design of a MARS prototype could then be completed in two years, subject to the conditions of funding and support from the industry.

7.6. Japan

Japan has several ongoing R&D programmes for the innovative water cooled, gas cooled, sodium cooled and lead-bismuth cooled SMRs with conventional refuelling schemes.

Water cooled SMRs

Starting from 1999, a group of companies and universities led by the Mitsubishi Heavy Industries (MHI) and including also the Kyoto University, the Central Research Institute of Electric Power Industries (CRIEPI), and the Japan Atomic Power Company (JAPC) develops a concept of the integrated modular water reactor (IMR). The IMR is an integral primary system reactor with the reference output of 350 MW(e), see Annex VI. The design targets are to attain electricity generation costs comparable to those of a large-scale nuclear reactor and to eliminate the possibility of certain accidents by design. The Japan Ministry of Economy, Trade and Industry has been supporting the IMR conceptual design study and the feasibility tests of key technologies from 2001 to 2004. The IMR is in the conceptual design stage to be completed in 2005. The R&D for components and design methods, a certain amount of validation and testing, and basic design development are required before licensing. The target year to start licensing is 2011.

The compact containment boiling water reactor (CCR) is a modular boiling water reactor (BWR) developed by the Toshiba Corporation with the support from the Japan Atomic Power Company (JAPC) and, partially, the Agency of Natural Resources and Energy and the Ministry of Economy, Trade and Industry (METI). The current CCR design has an electric output of 300 MW per module, see Annex IX. The design goals are to provide economic flexibility for a variety of siting conditions and electricity demands, to mitigate investment risks, and to facilitate public acceptance. The conceptual design of the reactor and its major nuclear systems has been completed. It is expected that by mid-2010 the design will be sufficiently complete to enable the initiation of construction related actions, subject to the availability of funds and regulatory and other statutory clearances.

The reduced moderation water reactor (RMWR) of 300 MW(e) aims to achieve a high conversion ratio (over 1.0) with mixed oxide (MOX) fuel. The design is based on proven boiling water reactor (BWR) technology see Annex 10. High conversion ratio is attained by the reduction of neutron moderation, i.e., by the reduced water fraction in the reactor core, see Annex X. The RMWR is developed through collaboration between the Japan Atomic Energy Research Institute (JAERI), JAPC, Hitachi Ltd. and the Tokyo Institute of Technology (TITech) with partial support from the Government of Japan and the utilities. The current design stage is that of a conceptual design. Once adequate funding is available, the basic design of a 300 MW(e) RMWR could be developed in about one year, and the detailed design in about 2 years after it. The R&D costs needed to deploy the prototype are estimated at about US $10 million. The R&D costs needed to deploy a commercial NPP with the RMWR are estimated as US $500 million.

The KAMADO reactor concept was proposed in 2001 by the CRIEPI. The concept is based on a synthesis of the design approaches used in light water reactors, pool type research reactors, and the FUGEN heavy water reactor, see Annex XIII. The KAMADO is a direct flow pressure tube pool type light water reactor with superheated steam at core outlet; the design incorporates the graphite blocks that are located in a water pool and house the separated fuel rods and pressure tubes. The design objective of the KAMADO is to develop a nuclear reactor with a negligible possibility of core meltdown accidents. The KAMADO concept provides for a simple plant system design without the reactor pressure vessel, the emergency core cooling system (ECCS), the re-circulation systems, etc. Therefore, the construction cost per unit of the electric power generated is expected to be sufficiently low, comparable to that of conventional large scale LWRs. The preliminary conceptual design of the KAMADO is in progress. All activities on design and technology development for the KAMADO are performed and funded by the CRIEPI.

Gas cooled SMRs

Since 2001, the Japan Atomic Energy Research Institute (JAERI) has been developing an original concept of the high temperature reactor with direct gas turbine cycle, Gas Turbine High Temperature Reactor 300 (GTHTR300), Annex XVI. The innovative features of this simplified system include core design based on a newly proposed refuelling scheme named the sandwich shuffling; the use of conventional steel materials for the reactor pressure vessel; an innovative plant flow scheme and a horizontally-installed gas turbine unit. The principal stakeholder in the GTHTR300 project is JAERI.

The GTHTR300 project has completed the basic design stage, which included system and component design, safety evaluation, and economic assessment. A deployment roadmap was prepared for the GTHTR300, which comprises three successive phases. The first phase, currently ongoing in JAERI, would last until 2007 under an exclusive funding from the Ministry of Education, Culture, Sports, Science and Technology (MEXT) of Japan; it covers the prototype plant design and the associated basic R&D for technology development.

The second phase, called the utility prototype plant demonstration, will be carried out over a period of ten years (2008–2018) mainly in the private sector, with public funds added to cover the nonrecurring costs only.

A success of the prototype plant development and demonstration will mark an important milestone towards commercial deployment of the GTHTR300. The deployment during Phase III will exploit significant system advancement options. Through the development and introduction of a few performance-enhancing technologies relevant for the fuel, materials and power equipment, the GTHTR300 would be commercially deployed achieving more than 50% net plant efficiency and a further 10% reduction in the electricity cost. The system would be upgraded to deliver 950°C helium, which will contribute to the deployment of other attractive systems, such as the GTHTR300C, a cogeneration system for electricity and hydrogen production (around 2020).

A group of leading companies of the First Atomic Power Industry Group (FAPIG), including Fuji Electric Systems; Kawasaki Plant Systems, Ltd.; and Shimizu Corporation is involved in the development of a concept of a 100 MW(e) high temperature helium cooled reactor with pebble bed fuel and direct gas turbine cycle, the FAPIG-HTGR (see Annex XVIII). The design goals include high thermal efficiency and low cost power conversion system based on a high speed gas turbine system with a vertical single shaft rotor; an enhanced safety achieved through the use of a simple 3-vessel system and the decay heat removal being performed by a passive system utilizing natural circulation of the atmospheric air. The FAPIG-HTGR is at a pre-conceptual design stage.

Liquid metal cooled SMRs

The modular double pool fast breeder reactor (MDP), a sodium cooled fast reactor of 325 MW(e) per module output, has been designed to reduce the construction costs and improve the reliability by factory production of most the components, see Annex XXII. Specifically, the MDP is proposed for use within a 4-module plant of 1300 MW(e).The development of the MDP concept has been performed and funded by the CRIEPI. The double pool design is intended to reduce the distances in the intermediate heat transport system by installing steam generators and secondary pumps in the sodium filled annular space formed between the primary and secondary vessel. The preliminary conceptual design has been completed but, at the moment, there is no financial support for further R&D.

The Japan Nuclear Cycle Development Institute (JNC) is developing the medium scale lead-bismuth cooled reactor see Annex XXV. It is a tank type modular design without intermediate heat transport system. The module power is 710 MW(e), and the plant of 4 modules with 2840 MW(e) total electric output was selected as basic. The medium scale lead-bismuth cooled reactor is at the conceptual design stage.

7.7. The Republic of Korea

In the Republic of Korea, the R&D programmes addressing innovative SMRs include both water cooled and liquid metal cooled reactor designs.

Water cooled SMRs

Since 1997, the Korean Atomic Energy Research Institute (KAERI) has been developing the system-integrated modular advanced reactor (SMART), an advanced integral PWR of 330 MW(th), see Annex I. The SMART development has been conducted under a nuclear research and development programme supported by the Ministry of Science and Technology (MOST) of the Republic of Korea and, therefore, KAERI and MOST are the principal stakeholders. The SMART incorporates highly advanced design features contributing to a competitive economy and enhancing the plant safety, reliability, performance, and operability. The SMART development programme has been established with three phases, which are technology development, plant construction and plant commercialization. At the technology development phase, the fundamental technologies were developed and the conceptual design was completed. The basic design was finished in 2002 and the construction of a 65 MW(th) prototype plant (SMART-P) was launched to perform a comprehensive performance verification. Commercialization of the SMART desalination plant is scheduled to start in 2009. The validation and testing programme for the SMART has been established. Some of the tests were successfully completed during the basic design phase and some are currently underway. A research and development centre for the construction of the SMART-P was established in June 2002. The SMART-P construction project proceeds with the participation of many industrial companies and research organizations, such as the Doosan Heavy Industries and Construction Company (DSHIC), the Korea Power Engineering Company, Inc. (KOPEC), the Korea Institute of Nuclear Safety (KINS), and various universities.

Liquid metal cooled SMRs

The liquid metal cooled reactor (LMR) design technology development project was approved as a national long-term R&D programme in 1992 by the Korea Atomic Energy Commission (KAEC), which decided to develop and construct a LMR. Based upon the KAEC decision, KAERI has stared the development of the Korea Advanced Liquid Metal Reactor (KALIMER), Annex XX. The goal of the LMR design technology development project is to develop the LMR design technologies necessary for the efficient utilization of uranium

resources and the reduction of high level wastes. The design objectives of the KALIMER include enhanced safety, competitive economics, enhanced proliferation resistance and environmental friendliness. The R&D for the KALIMER is fully supported by the Government of the Republic of Korea. The design and development of this reactor are performed by KAERI.

The LMR design technology development project has been carried out in phases. During phases 1 and 2 (1997–2001) of the national fast reactor programme, the basic technologies and the conceptual design of a 150 MW(e) KALIMER-150 have been developed. As there are no plans for the construction of a prototype or a demonstration reactor, the basic key technologies and the advanced concept of a 600 MW(e) KALIMER-600 have been developed during phase 3, in 2002–2004. It was also suggested to foster international collaboration including the Generation IV International Forum programme. Furthermore, as a result of the nuclear technology roadmap activities in the Republic of Korea, a sodium cooled fast reactor was selected as one of the two reactor options deployable by 2030.

The concept of the PEACER (Annex XXIV), a fast lead-bismuth cooled reactor of 300 or 550 MW(e) for electricity generation and waste transmutation, was proposed in 1998. The design and technology development for the PEACER is carried out by the Nuclear Transmutation Reactor Engineering Center Korea (NuTRECK) of the Seoul National University, with the full financial support from the Korean Ministry of Commerce, Industry and Energy. The conceptual design of the PEACER and the basic design of its major nuclear systems have been completed. The present research encompasses technology development for a 3D virtual reality, Pb-Bi coolant technology demonstration and the ongoing conceptual design of a PEACER version for power production and LWR spent fuel disposal.

7.8. The Netherlands

The Netherlands made a decision to phase out nuclear power; therefore, there are no national programmes on design and technology development for innovative reactors.

The ACACIA (Advanced Atomic Co-generator for Industrial Applications) is a concept of a 60 MW(th), 23 MW(e) nuclear power plant with the pebble bed high temperature gas cooled reactor and an indirect Brayton cycle, see Annex XIX. This concept is well suited as an autonomous energy source but also allows for a multiple-module plant construction. Different from other pebble bed reactors, the ACAIA provides for a whole core refuelling performed once in every 3 years. The ACACIA is at a pre-conceptual design stage. A principal stakeholder for the ACACIA is the Nuclear Research and Consultancy Group (NRG) of the Netherlands. No further R&D is planned at the moment, but the design team intends to pursue further development of the ACACIA design.

7.9. South Africa

In 1993, the pebble bed modular reactor (PBMR) was identified by ESKOM, the electric utility of South Africa, as a leading option for the installation of new generating capacity to their electric grid.

This innovative nuclear power plant of 165 MW(e) capacity incorporates a closed cycle primary coolant system utilizing helium to transport heat energy directly from the modular pebble bed reactor to a recuperative power conversion unit with a single-shaft turbine/compressor/generator, see Annex XIV. This replacement of the steam cycle that is common in present nuclear power plants (NPP) with a direct gas cycle provides the

benefits of simplification and a substantial increase in overall system efficiency with the attendant lowering of capital and operational costs.

The principal shareholders in the PBMR are incorporated within the South African company, PBMR (Pty) Ltd. These shareholders include:

- ESKOM, National South Africa Electric Utility.
- Industrial Development Corporation of South Africa, a national development financial institution.
- British Nuclear Fuel plc, a global nuclear fuel cycle company, (parent of Westinghouse nuclear) solely owned by the government of the United Kingdom.

Initial PBMR development is focused on completion of the detailed design and engineering for a demonstration unit to be located at the Koeberg NPP site north of Cape Town, South Africa. The full cost of the demonstration plant is estimated at ~US $1 billion.

The South African government has designated the PBMR a national strategic project with a cabinet level committee appointed in February 2004. The demonstration plant site preparation is scheduled to begin at the Koeberg NPP site in the first quarter of 2007 with fuel loading anticipated for mid-2010. The commercial acceptance by ESKOM is scheduled for early 2011.

Further development targets achieving the Generation IV goals with a very high temperature PBMR version.

7.10. The Russian Federation

The Russian Federation has multiple R&D programmes on design and technology development for innovative SMRs with conventional refuelling schemes. A short summary given below covers the R&D programmes for SMRs addressed in this report, see Section 1.3. The information on several other developments can be found in IAEA-TECDOC-1451 [7].

Water cooled SMRs

The VBER-300 is a modular PWR of 295 MW(e) developed on the basis of marine reactor technologies, see Annex VII. In many ways, the plant design appears as a scaled-up version of the marine modular reactors, which have a solid design and operation experience in the Russian Federation. The VBER-300 is developed as a small-to-medium power source for both land based NPPs or cogeneration plants and floating NPPs or desalination complexes. The VBER-300 is well suited for autonomous operation in the immediate proximity to the customer. The principal stakeholders are Russian research and design organizations, including the OKB Mechanical Engineering (OKBM, Nizhny Novgorod), the Russian Research Centre “Kurchatov Institute” (RRC “Kurchatov Institute”, Moscow), the Scientific-Research and Design Institute “Atomenergoproekt” (NIAEP, Nizhny Novgorod), and the Public Company “Lazurit” (Nizhny Novgorod).

A preliminary (basic) design of the VBER-300 has been completed in 2002. At present, the design is undergoing an expertise in the Rosatom of the Russian Federation. The phase that included the optimization of separate design features and schemes is near completion, and the development of the detailed design is underway. A land based nuclear cogeneration plant with the VBER-300 could be deployed in 2013, and a floating NPP with two VBER-300 reactors could be deployed in 2012.

The VK-300 is a 250 MW(e) simplified boiling water reactor with natural circulation of the coolant and many passive systems, see Annex VIII. Several options for the core arrangement

and reactor design were considered at the development stage of the VK-300 to enhance its safety and economic efficiency. Specifically, of all options considered a principle of passive operation of the main safety systems was retained to achieve an optimum balance between an enhanced safety and the improved economic characteristics. The VK-300 was developed by Russian research and design organizations, including the Research and Development Institute of Power Engineering (RDIPE, also known as NIKIET), the Russian Research Centre “Kurchatov Institute” (RRC KI), and the Institute of Physics and Power Engineering (IPPE, Obninsk).

As of 2005, the detailed design of the VK-300 reactor for a nuclear cogeneration plant was completed; the basic regulations for a typical nuclear cogeneration plant using the VK-300 reactors were developed; and the substantiation of investments to the construction of a nuclear cogeneration plant in the Arkhangelsk region of the Russian Federation has been performed. Research and development activities are currently underway for further validation of the design approaches adopted in the VK-300 design. It is estimated to take 2–3 years to implement these activities. The first power unit could then be deployed around 2012.

The RUTA-70 is a single-purpose nuclear heat plant of 70 MW(th) for district heating, see Annex XII. The design is based on natural convection of the primary coolant sand is characterized by the absence of excess pressure in the primary circuit, which is a reactor pool. Nuclear heat plants with such reactors are characterized by inherent safety features and could be located in the immediate proximity of the heat users. The RUTA-70 project is performed by Russian research and design organizations, including the RDIPE as a leader, the IPPE, the VNIPIET, the MI KRC RAS, and the ‘Atomenergoproekt’ (AEP).

In 2001, the IPPE proposed conducting a feasibility study on the upgrading of the district heating in Obninsk, Kaluga region, by constructing a nuclear heat plant with the RUTA reactor. This proposal was approved by a decision of the board of the programme of Obninsk development as a ‘science town’. The RUTA-70 nuclear heat plant does not require support by national R&D programmes because proven technical solutions and proven equipment are used to the maximum extent. At the moment, the RUTA-70 is supported under the Obninsk development programme. The time needed for the deployment of the RUTA-70 is estimated as 3 years in the Russian Federation and 4–5 years outside the Russian Federation. At present, the RUTA-70 is at the preliminary (basic) design stage.

Gas cooled SMRs

The GT-MHR couples a modular high temperature gas cooled reactor (HTGR) with a Brayton power conversion cycle to produce electricity at high efficiency, with a potential for high temperature process heat applications, see Annex XX.

The development of a multi-module Gas Turbine – Modular Helium Reactor (GT-MHR) for excess weapons grade plutonium consumption in the Russian Federation in ongoing within an international project that pools together major efforts of the Russian Federation and the USA. The institutions involved in the R&D, design and deployment of the plutonium consumption GT-MHR include the following:

- The Russian Federation – Rosatom, OKBM, RRC KI, VNIINM Bochvar, SPA “Lutch”, SCC, VNIPIET, NIIAR, SNTC, ISTC.
- The United States – DOE/NNSA, EPRI, General Atomics, ORNL.
- The European Union and Japanese participation via ISTC.

From a technology development standpoint, the path forward for deployment of the GT-MHR technology is necessarily a demonstration project, because of a number of

heretofore-unproven characteristics embodied in the design. The most prominent of these include items such as the safety design approach, fuel operating conditions (burn-up, fluence, temperature), power conversion system design (vertical shaft, magnetic bearing suspension), and pressure vessel design (size, operating temperatures).

The GT-MHR for plutonium consumption is at the preliminary design stage. A schedule produced indicates that its prototype could begin full power operation nine years after completion of the preliminary design. This would include the following elements:

- Complete design and development in the Russian Federation – 3 years.
- Russian regulatory review (in parallel with above) – 4 years.
- Prototype construction in the Russian Federation – 4 years.
- Fuel load, ascent to power and demonstration testing – 1 year.

Funding support for the development of the plutonium consumption version of the GT-MHR is continuing through the DOE NNSA in the United States and Rosatom in the Russian Federation, with additional technology development support from the EU and Japan through ISTC.

Liquid metal cooled SMRs

The BMN-170 is a modular nuclear power plant of 170 MW(e) with a sodium cooled fast reactor, see Annex XXI. Its concept was developed in 1990s by Russian research and design organizations, including the OKBM, the Sankt Peterburg Atomenergoproekt (SPb AEP) and the IPPE. The BMN-170 is designed to ensure the economically effective generation of electricity or the co-generation of heat and power in autonomous power systems. By offering fuel breeding, the concept opens the possibility of using the BMN-170 plants in a multi component structure of future nuclear energy systems. At the stage of conceptual design development, several options of the BMN-170 core arrangement and reactor module design were considered to enhance safety and economic effectiveness. During recent years (1997–2003), the activities for BMN-170 were stimulated by the exchange of scientific and technical information with companies in Russia and abroad. Among them, the Ministry of Atomic Industry of Kazakhstan could be mentioned. At the moment, conceptual investigations for the BMN-170 project are being performed on the initiative of OKBM specialists (Nizhny Novgorod, Russia).

The RBEC-M is a lead-bismuth cooled fast reactor with a high level of primary coolant natural circulation and a gas lift system in the primary circuit to provide the supply of an inert gas (e.g. argon) in the coolant under the core, see Annex XXIII. This concept is developed with an insight of future multi-component nuclear energy systems, where it might be used for breeding or the adjustment of fissile material flows. Conceptual studies for the RBEC-M are performed in the Russian Research Centre “Kurchatov Institute” (Moscow, Russia).

Recently, the work towards elaboration of the RBEC-M concept was stimulated by the exchange of scientific and technical information with the domestic and foreign organizations developing new reactors with liquid heavy metal coolants. In particular, the French Commissariat à l’Energie Atomique (CEA) and the Japan Nuclear Cycle Development Institute (JNC) could be mentioned in this context.

7.11. The United States of America

The USA maintains several ongoing R&D programmes for innovative SMRs with conventional refuelling schemes.

Water cooled SMRs

The International Reactor Innovative and Secure (IRIS) is a modular, integral-type, pressurized, light water cooled, medium power reactor of 1000 MW(th), see Annex II. The IRIS concept addresses the top-requirements for next generation reactors, i.e., enhanced safety, reliability, positively proliferation resistance, and improved economics.

The IRIS "safety-by-design"™ philosophy is a systematic approach that aims—by design—at eliminating altogether the possibility for an accident to occur, i.e., to eliminate accident initiators, rather than having to design and implement systems to deal with the consequences of the accident.

The IRIS development was initiated with U.S. DOE support through the Nuclear Energy Research Initiative (NERI). Currently, the institutions involved in IRIS R&D are members of the IRIS consortium and include Westinghouse Electric Co. (USA), BNFL (UK), Ansaldo Energia (Italy), Ansaldo Camozzi (Italy), ENSA (Spain), NUCLEP (Brazil), Bechtel (USA), OKBM (Russia), ORNL (USA), CNEN (Brazil), ININ (Mexico), LEI (Lithuania), Polytechnic of Milan (Italy), MIT (USA), Tokyo Institute of Technology (Japan), University of Zagreb (Croatia), University of Pisa (Italy), Polytechnic of Turin (Italy), University of Rome (Italy), TVA (USA), Eletronuclear (Brazil). All team members are stakeholders.

At the moment, IRIS is at the preliminary (basic) design stage. It is expected that the first-of-a-kind IRIS will be deployed in the 2012–2015 timeframe.

Gas cooled SMRs

The development and deployment of commercial GT-MHRs is based upon leveraging an ongoing international project to develop and deploy a multi-module GT-MHR designed to consume excess weapons grade plutonium in the Russian Federation, see Section 7.10. The commercial GT-MHR design would utilize the technology development conducted in support of the plutonium consumption version, with the majority of additional development focused on fabrication and qualification of LEU fuel.

Many of the institutions involved in the R&D on the GT-MHR for plutonium consumption would also be involved in transfer of technology from the plutonium consumption version to the commercial GT-MHR. In addition, the GT-MHR Utility Advisory Board (UAB) has been actively supporting the commercialization of the GT-MHR.

Efforts in support of the commercial version of the GT-MHR have resulted in the production of licensing and deployment plans. Planning for a possible demonstration plant in the US (the next generation nuclear plant – NGNP-project), begun in 2003, may result in a restructuring of the GT-MHR commercialization strategy.

The commercial GT-MHR schedule for a US deployment would parallel the plutonium consumption version schedule summarized in Section 7.10, lagging the deployment in the Russian Federation by about one year. Pre-application licensing interactions with the US Nuclear Regulatory Commission began in 2001, including submittal of a Licensing Plan.

Funding support for pursuing early site permits that include the GT-MHR as an option is provided by the DOE under the NP 2010 initiative and by participating generating companies.

The potential benefits of the GT-MHR for the generation of electricity coupled with the potential for efficient production of hydrogen provide significant incentives for a government-sponsored demonstration programme such as the proposed Idaho National Laboratory (INL) NGNP demonstration project, see Annex 15.

Non-conventional SMRs

The Advanced High Temperature Reactor (AHTR) is a new reactor concept that combines four existing technologies in a new way:

- Coated particle graphite matrix nuclear fuels (traditionally used for helium cooled reactors).
- Brayton power cycles.
- Passive safety systems and plant designs from liquid metal cooled fast reactors.
- Low pressure liquid salt coolants with boiling points far above the maximum coolant temperature (the coolant, however, it contains no fuel).

The AHTR is designed to produce electricity and/or high temperature heat. The reactor concept is being developed in the USA co-jointly by Oak Ridge National Laboratory, Sandia National Laboratories, and the University of California at Berkeley. Several commercial reactor vendors are currently evaluating the concept. The design stage is that of a pre-conceptual design.

The AHTR is part of the U.S. Department of Energy Generation IV reactor programme and is being actively investigated, specifically, as a very high temperature reactor (VHTR) option. At the same time, commercial reactor vendors are conducting parallel studies. Detailed development plans are being prepared. If the AHTR is selected for large-scale development, the goal would to be to have an operating test reactor by 2012. A medium sized pre-commercial demonstration reactor would follow this.

Appendix I

OUTLINE FOR SMR DESIGN DESCRIPTION (FULL DESCRIPTION)

I.1. General information, technical features, and operating characteristics

I.1.1. Introduction

- Full and abbreviated name of the nuclear installation with an innovative SMR
- Historical technical basis: identify plants/test facilities, and R&D previously performed that support this innovative SMR design
- List of principal stakeholders

I.1.2. Applications

Including: electricity generation/co-generation, district heating, seawater desalination, hydrogen production, process steam production etc, or a combination thereof

I.1.3. Special features

Such as: floating NPP option, option to use NPP as an autonomous energy source (nuclear cell), modular approach allowing for incremental capacity increase, option of prefabrication, transportability, etc.

I.1.4. Summary of major design and operating characteristics[1]

- Installed capacity (thermal and electric)
- Mode of operation (basic, load follow)
- Load factor/ availability (specify targets)
- Summary of major design characteristics: types of fuel, fuel enrichment, types of coolant/moderator, types of structural materials, core type/characteristic dimensions, vessel type/characteristic dimensions, cycle type (direct/indirect), number of circuits
- Simplified schematic diagram of the nuclear installation with an innovative SMR
- Neutron-physical characteristics (temperature and coolant density reactivity effects, void reactivity effect and burn-up reactivity swing,), power flattening (peaking factors, approaches to reduce them)
- Reactivity control mechanism (burnable poisons, control rods, liquid boron, spectral shift, movable reflector, etc or a combination thereof), number of independent active reactor control and protection (RCP) systems, cumulative worth for each RCP system
- Cycle type (direct or indirect) and thermodynamic efficiency
- Thermal-hydraulic characteristics (circulation type: natural/ forced, inlet/outlet coolant temperatures, flow rates and pressures in circuits, temperature limits for fuel/claddings and other ‘critical’ structural materials, maximum/average temperatures of fuel and structural materials in normal operation, DNBR, if any)

[1] Any other relevant parameters could be added by the designer

- Maximum/average discharge burn-up of fuel (% FIMA)
- Fuel lifetime/period between refuellings in effective full power days (EFPD)
- Mass balances/flows of fuel and non-fuel materials (on an annual basis per unit of thermal and equivalent electric energy produced) with a short description of how they were obtained (best estimate, reference etc)
- Design basis lifetime for reactor core, vessel and structures
- Design and operating characteristics of systems for non-electric applications, including process type, ranges for sharing energy production between different applications and specific production rate per unit of thermal and equivalent electric energy
- Economics (capital costs, estimated construction period, O&M costs, fuel costs, final product costs for a prototype and relevant projections for a final version of the nuclear installation with an innovative SMR)

I.1.5. Outline of fuel cycle options

- Standard fuel cycle (specify once-through, closed etc. for basic type of fuel)
- Alternative fuel cycle options (specify once-through, closed etc. for alternative types of fuel)
- Suggested fuel reprocessing method, if any
- Provision for fuel cycle organization (such as centralized, regional, or on-site reprocessing, etc.), if any
- SNF management and disposal planning, if any

I.1.6. Technical features and technological approaches that are definitive for nuclear installation performance in particular areas

I.1.6.1. Economics and maintainability

- Targeted markets and their specifics (for example, low labour costs and limited resources for investments in some developing countries etc.)
- Provisions for reduced capital and construction costs, such as design standardisation, option of factory fabrication, transportability etc.
- Provisions for low O&M costs, such as elimination of the need for on-site refuelling, long-lived core etc.
- Provisions for low fuel reload costs (such as low enrichment, particular fuel or fuel cycle type etc.)

I.1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

- Provisions for low consumption of non-renewable resources, including the degree of fuel utilization
- Provisions for minimum generation of wastes at the source
- Provisions for acceptable or reduced dose limits
- Provisions for low SNF and waste management costs (such as particular fuel forms, minimized specific production of waste etc.)

I.1.6.3. Safety and reliability

- Safety concept and design philosophy
- Provisions for simplicity and robustness of the design
- Active and passive systems and inherent safety features
- Structure of the defence-in-depth
- Design basis accidents and beyond design basis accidents
- Provisions for safety under seismic conditions
- Probability of unacceptable radioactivity release beyond the plant boundaries
- Measures planned in response to severe accidents

I.1.6.4. Proliferation resistance

- Technical features to reduce the attractiveness of nuclear material for nuclear weapon programmes, such as isotopic content, chemical form and radiation properties
- Technical features to prevent the diversion of nuclear material
- Technical features to prevent the undeclared production of direct-use material
- Technical features to facilitate nuclear material accounting and verification

I.1.6.5. Technical features and technological approaches used to facilitate physical protection of the NPP with an innovative SMR

Such as: features that ensure enhanced protection against external impacts and sabotage due to intentional actions of the personnel, etc.

I.1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of reactor installation

Such as:

- Taking into account (at the design stage) market demands and the specific needs of different market, in particular, needs of developing countries
- Provisions for leasing of fuel and/or full-scope fuel cycle service agreements
- Options for NPP leasing

I.1.8. List of enabling technologies relevant to the nuclear installation with an innovative SMR and status of their development

I.1.9. Status of R&D and planned schedule

Including:

- Information on whether R&D for this particular SMR are supported by national (State) R&D or NPP deployment programmes
- Companies/Institutions involved in the RD&D and design
- Estimate of an overall time frame within which the design could be implemented
- Information on main RD&D and licensing stages and their duration

- R&D costs needed to deploy the prototype, R&D costs needed to deploy final version of the NPP with an SMR (assume all R&D are performed in a specified country)
- Financial information, if any, including the status of funding

I.1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed,

i.e. why this particular SMR could be rated as innovative

I.1.11. List of other similar or relevant SMRs for which the design activities are ongoing

I.2. Design description and data for each nuclear installation with an innovative SMR

I.2.1. Description of the nuclear systems

- Reactor core and fuel design, including core map and cross section of fuel
- Main heat transport system, with specification of heat removal path in normal operation and in accidents
- Intermediate circuit, if any

I.2.2. Description of the turbine generator plant and systems (details to be given if innovative equipment is to be used)

I.2.3. Systems for non-electric applications

- Outline of the circuits, systems, and processes for non-electric applications
- Table of basic design data

I.2.4. Plant layout

- General philosophy governing plant layout
- Reactor building and containment layout
- Plant plot, if available

References

Appendix II

OUTLINE FOR SMR DESIGN DESCRIPTION (SHORT DESCRIPTION)

II.1. Basic summary, including:

- Full and abbreviated name of the concept, principal stakeholder(s)
- Core design summary
- Plant design summary
- Safety concept summary

II.2. Major design and operating characteristics

II.2.1. Table of major reactor (core) characteristics

- Fuel type and enrichment
- Fuel assembly type and number
- Type of structural materials
- Fuel burn-up and cycle length
- Core dimensions; reflectors, if any
- Approach to power flattening
- Average power density
- Major reactivity effects
- Breeding ratio, if applicable
- Decay heat removal systems
- Other characteristics suggested by the designer

II.2.2. Table of major plant characteristics

- Reactor type
- Thermal and electric output
- Plant efficiency
- Cycle type (direct or indirect), secondary coolant (intermediate coolant, if any)
- Circulation type (natural or forced)
- Core inlet/outlet temperature, primary circuit pressure, and primary coolant flow rate
- Turbine inlet temperature and pressure
- Containment system
- Other characteristics suggested by the designer

Figure(s) illustrating plant design scheme

II.3. List of enabling technologies and status of their development

References

Appendix III

CROSSCUT TABLES OF ONGOING AND PLANNED R&D; AVAILABLE OR PLANNED TEST FACILITIES

TABLE III.1. FURTHER R&D FOR REACTOR CORE AND FUEL DESIGN

SMR type **SMR name** (annex number corresponding to design description in this report) • Directions of R&D
Water cooled SMRs
IRIS (Annex II) • Irradiation testing, qualification and licensing of fuel for an operation cycle increased from 4 to 8 years **CAREM** (Annex III) • Hydrodynamic and structural tests of fuel assemblies • Validation of mechanical design (structural, dynamic, seismic, etc.) of the core **IMR** (Annex VI) • Physical and chemical conditions and durability of the materials: more data are required to validate the integrity of fuel cladding under boiling conditions **VK-300** (Annex VIII) • Study of fuel assembly characteristics at higher fuel burn-ups **CCR** (Annex IX) • Design study to reduce the number of fuel assemblies by applying a wider lattice pitch **RMWR** (Annex X) • New stainless steel cladding to secure reliable fuel rod operation at increased fuel burn-ups • In-depth analytical studies and irradiation experiments to examine the degradation of fuel pellet thermal conductivity under increased burn-up, swelling of the fission product gas pores generated around Pu-rich spots, and fuel rod deformation behaviour **RUTA-70** (Annex XII) • Justification and study of the possibility to use alternative fuel, e.g., cermets fuel **KAMADO** (Annex XIII) • Research, design and demonstration for innovative fuel elements (graphite blocks housing fuel rods and water and steam pipes)
Gas cooled SMRs
PBMR (Annex XIV) • Separate effect tests of heat transfer mechanisms in annular pebble bed • Investigation of oxidation of hot graphite cores by oxygen with natural circulation following air ingress events • Enhancement of fuel performance for PBMR-VHTR development (maximum accident fuel temperature 2000 ^{o}C, 200 GW day/t burn-up; improved resistance to oxidation) **GT-MHR** (Annex XV) • Data are required for performance of the graphites specified for the GT-MHR to address multi-axial strength, fatigue strength, mechanical properties, irradiation-induced dimensional change, irradiation induced creep, thermal properties, fracture mechanics, corrosion and oxidation, and coke source qualification • Data are also needed for other ceramic components, including carbon/carbon composite materials for control rods (if used) and hard ceramic insulation used under the graphite core support structure

TABLE III.1. (Continued)

SMR type **SMR name** (annex number corresponding to design description in this report) • Directions of R&D
Gas cooled SMRs (continued)
GTHTR300 (Annex XVI) • Irradiation tests and data for high burn-up fuel **FAPIG-HTGR** (Annex XVIII) • Additional irradiation experiment for coated particle fuel is required for burn-ups over 100 GW d/t
Sodium cooled SMRs
BMN-170 (Annex XXI) • Operational validation of a core employing fuel elements with nitride and metallic fuels • Experimental validation of operability of fuel elements with nitride and metallic fuel • Study of options to increase fuel burn-up **MDP** (Annex XXII) • Acquisition of irradiation data for fuel licensing (metallic fuel) • Development of cladding materials (for fuel elements with metallic fuel)
Lead bismuth cooled SMRs
RBEC-M (Annex XXIII) • New solutions for high burn-up fuel • Deeper knowledge is required on the properties of fuel composition and on the fuel pellet-cladding interaction (FPCI) for fuel rods with nitride fuel and ferritic-martensitic stainless steel cladding • Additional tests are required to study thermal stability of (U-Pu)N and to obtain the precise ultimately allowable temperature for mixed nitride fuel • Cost estimates for nitrogen enrichment by ^{15}N for fuel rods with nitride fuel and ferritic-martensitic stainless steel claddings **PEACER** (Annex XXIV) • Metal fuel design • Core design optimization, including fuel management scheme **MS-LBCR*** (Annex XXV) • Nitride fuel technology • Corrosion resistant operation of claddings in lead bismuth coolant
Non-conventional SMRs
AHTR • Optimization of core design • Development of a more radiation resistant graphite to reduce graphite swelling with time and thus reduce the costs associated with the periodic replacement of graphite in the reactor core

* The acronym MS-LBCR is used in TablesIII.1 – III.9 to denote medium scale lead-bismuth cooled reactor described in Annex XXV.

TABLE III.2. FURTHER R&D FOR REACTOR INTERNALS AND PRIMARY CIRCUIT

SMR type

SMR name (annex number corresponding to design description in this report)
- Directions of R&D

Water cooled SMRs

SMART (Annex I)
- Experiment for natural circulation in the integral arrangement of the reactor system
- Experiments on wet thermal insulation, to determine the insulating effects for the low operating temperature pressurizer design, and to derive heat transfer coefficients
- Performance tests for key parts of control element driving mechanism (CEDM) and main circulation pump
- A high temperature, high-pressure integrated-system thermal-hydraulic test is underway

IRIS (Annex II)
- Improved instrumentation for the integral reactor vessel layout, where all primary components are located within the vessel

CAREM (Annex III)
- Modelling and qualification of the reactor core coolant system, based on natural circulation
- Validation of mechanical design (structural, dynamic, seismic, etc.) of the reactor internals

SCOR (Annex V)
- Confirmation of the possibility to build fully integrated primary spool type pumps with immerged coils to be operated in hot conditions
- Assessment of primary circuit hydraulic behaviour and the venturi performance (mock-up test will be necessary to validate hydraulic performance of the primary circuit, since the water flow path in the vessel with the venturi bypass is different from that used in standard PWRs)

IMR (Annex VI)
- A strategy of water chemistry must be developed

VK-300 (Annex VIII)
- Validation and testing of thermal-hydraulic characteristics of the in-vessel natural circulation circuit under normal and emergency operating conditions

CCR (Annex IX)
- R&D to validate the use of natural convection for core cooling

KAMADO (Annex XIII)
- Prove the reliability of the primary cooling system for a direct flow reactor with fuel elements including fuel rods, graphite blocks, water and steam pipes, and reactor water pool

Gas cooled SMRs

PBMR (Annex XIV)
- Testing of reactor system components in high temperature, high pressure helium environment
- Demonstration of sphere transport system
- For PBMR-VHTR: high temperature qualification of primary system materials and components (1200^{o}C core outlet temperature)

GT-MHR (Annex XVI)
- Major reactor system components, including the reactor internals and hot duct, fuel handling equipment and reactor service equipment require detailed design and validation through testing of scale models and assemblies, and in some cases demonstration testing of prototypical components
- Metallic materials must be developed and/or qualified for the GT-MHR service conditions for use in the reactor vessel, reactor internals, etc.

TABLE III.2. (Continued)

SMR type **SMR name** (annex number corresponding to design description in this report) • Directions of R&D
Gas cooled SMRs (continued)
GT-MHR (Annex XVI) – continued
• For the reactor vessel, internals and hot duct, qualification will require data on performance of the materials under irradiation conditions representative of GT-MHR service over the component design life • For next generation nuclear plant (NGNP), higher temperature reactor vessel material may be required because of the higher core outlet helium temperature (1000°C); specifically, alternative materials (e.g. carbon-carbon composites) may be required in place of thermal barrier metallic materials
Sodium cooled SMRs
MDP (Annex XXII) • Evaluation of high temperature and in-sodium characteristics of the electromagnetic pump (EMP) coil • Evaluation of irradiation characteristics of the EMP coil • Evaluation of flow characteristics in the EMPs • Demonstration of EMP structural integrity and performance in a large-scale model • Design and element tests for the upper internal structure (UIS) and variable arm type in-vessel transfer machine (IVTM) • Mock-up tests for the UIS and IVTM
Lead-bismuth cooled SMRs
RBEC-M (Annex XXIII) • Regarding gas lift application, experiments are necessary to study flow regimes, hydrodynamics, gas void and processes for separating the mixtures of lead-bismuth and inert gas under the conditions typical for normal and abnormal regimes of operation • Regarding gas lift application, experimental study of thermal regimes of the fuel rods in a two-phase flow of lead-bismuth and inert gas is necessary **PEACER** (Annex XXIV) • R&D on Pb-Bi coolant technology • Design and technology development for submersible electromagnetic pumps of the primary circuit **MS-LBCR** (Annex XXV) • Oxygen control system to protect structural materials operating in Pb-Bi from corrosion
Non-conventional SMRs
AHTR (Annex XXVI) • Reactor vessel insulation system • Minimizing corrosion caused by impurities in helium and liquid salts in the materials for vessels, pipes, and heat exchangers • Selection of the preferred liquid fluoride salt • For a 1000°C core outlet temperature option, major material development programmes are required (development of high temperature structural materials for vessels, pipes, and heat exchangers)

TABLE III.3. FURTHER R&D FOR POWER CIRCUIT

SMR type **SMR name** (annex number corresponding to design description in this report) • Directions of R&D
Water cooled SMRs
SMART (Annex I) • Boiling heat transfer characteristics of the helically coiled steam generator tube
Gas cooled SMRs
PBMR (Annex XIV) • Gas turbine power conversion system feasibility issues • Investigate magnetic and catcher bearing reliability in high pressure, high temperature helium environment • Validation of power conversion unit operational performance (physical model for validation of primary thermal-hydraulic code) **GT-MHR** (Annex XV) • Major power conversion system components require detailed design and validation. These components include the turbo machine (helium turbo compressor and generator), recuperator, and pre-cooler/intercooler. • An integrated test of the power conversion system (PCS) is needed to confirm the performance of prototype components under normal operation and plant transient conditions • Materials must be developed and/or qualified for the GT-MHR service conditions for use in the turbine and recuperator. **GTHTR300** (Annex XVI) • R&D to confirm basic performance of the magnetic bearing for turbo-machinery • Development of the turbo-machinery aerodynamic design • Demonstration of control and operation for the non-intercooled conversion cycle **FAPIG-HTGR** (Annex XVIII) • High-speed large power generator • Low cost and compact frequency converter • High-speed gas turbine system with vertical single shaft rotor

TABLE III.4. FURTHER R&D FOR SAFETY FEATURES AND SYSTEMS

SMR type **SMR name** (annex number corresponding to design description in this report) • Directions of R&D
Safety features
Water cooled SMRs
MARS (Annex IV) • Experimental testing of new coupling of components **SCOR** (Annex V) • Low operating pressure design • Boron free core **AHWR** (Annex XI) • Validation of negative void reactivity coefficient (achieved by the use of a scatterer cum absorber component within fuel cluster)

TABLE III.4. (Continued - 3)

SMR type **SMR name** (annex number corresponding to design description in this report) • Directions of R&D
Gas cooled SMRs (continued)
GT-MHR (Annex XV) • Major reactor system components, including neutron control components, safety instrumentation, the shutdown cooling system circulator and heat exchanger, the reactor cavity cooling require detailed design and validation through testing of scale models and assemblies, and in some cases demonstration testing of prototypical component
HTR-PM (Annex XVII) • Passive decay heat removal (safety demonstration experiments are being conducted within the HTR-10 test reactor)
Sodium cooled SMRs
KALIMER (Annex XX) • Optimization of the use of passive decay heat removal system **BMN-170 (**Annex XXI) • Validation of a passive cool down concept by performing complex experiments to investigate modes of sodium flow in the primary circuit and air circulation outside the reactor module, as well as sodium and air hydrodynamics under conditions typical of normal and emergency operation • Experimental investigations of thermal regimes with an irregular distribution of airflow on the reactor vessel perimeter
Non-conventional SMRs
AHTR (Annex XXVI) • Further development of passive decay heat removal systems to allow higher temperature operations
Systems for severe accidents mitigation / management
Water cooled SMRs
SCOR (Annex V) • In-vessel cooling of corium **CCR** (Annex IX) • R&D on in-vessel retention (IVR) by passive means **AHWR** (Annex XI) • Steam driven poison injection system is planned to be demonstrated in a large integral test facility (ITL)
Seismic design
Water cooled SMRs
MARS (Annex IV) • Seismic qualification of innovative mechanical solutions
Sodium cooled SMRs
MDP (Annex XXII) • Development and evaluation of seismic isolation technology • Development of a reliability evaluation procedure regarding seismic isolation • Establishment of guidelines for seismic isolation technology
Lead-bismuth cooled SMRs
MS-LBCR (Annex XXV) • Three-dimensional seismically isolated reactor building

TABLE III.5. FURTHER R&D ON TECHNOLOGIES FOR FABRICATION OF FUEL AND MATERIALS, FUEL REPROCESSING AND WASTE DISPOSAL

SMR type **SMR name** (annex number corresponding to design description in this report) • Directions of R&D
Water cooled SMRs
AHWR (Annex XI) • R&D on dry reprocessing of fuel • Laser isotopic denaturing of zirconium • Reconstitution of fuel
Gas cooled SMRs
PBMR (Annex XIV) • R&D on fuel cycles • Fuel qualification, specifically the TRISO coated fuel **GT-MHR** (Annex XV) • Fuel development and demonstration programme is required to develop or validate a fuel fabrication process and a qualified fuel product for GT-MHR service conditions **GTHTR300** (Annex XVI) • Demonstration of fuel recycling technology in a small-scale test facility • Development of an economical waste storage system for graphite blocks discharged from the core **HTR-PM** (Annex XVII) • Large scale manufacturing of qualified fuel elements
Sodium cooled SMRs
KALIMER (Annex XX) • Remote fabrication technologies for metal fuel • Dry reprocessing of metal fuel **BMN-170** (Annex XXI) • Mastering fabrication of the nitride and metallic fuel and fuel elements **MDP** (Annex XXII) • Demonstration of technical feasibility of the metallic fuel reprocessing and fabrication technologies
Lead-bismuth cooled SMRs
PEACER (Annex XXIV) • Conceptual design of pyro-processing plant • Optimization of pyro-process design
Non-conventional SMRs
AHTR (Annex XXVI) • Development of a large scale manufacturing of high quality HTGR type fuel

TABLE III.4. (Continued - 3)

SMR type **SMR name** (annex number corresponding to design description in this report) • Directions of R&D
Gas cooled SMRs (continued)
GT-MHR (Annex XV) • Major reactor system components, including neutron control components, safety instrumentation, the shutdown cooling system circulator and heat exchanger, the reactor cavity cooling require detailed design and validation through testing of scale models and assemblies, and in some cases demonstration testing of prototypical component
HTR-PM (Annex XVII) • Passive decay heat removal (safety demonstration experiments are being conducted within the HTR-10 test reactor)
Sodium cooled SMRs
KALIMER (Annex XX) • Optimization of the use of passive decay heat removal system **BMN-170 (**Annex XXI) • Validation of a passive cool down concept by performing complex experiments to investigate modes of sodium flow in the primary circuit and air circulation outside the reactor module, as well as sodium and air hydrodynamics under conditions typical of normal and emergency operation • Experimental investigations of thermal regimes with an irregular distribution of airflow on the reactor vessel perimeter
Non-conventional SMRs
AHTR (Annex XXVI) • Further development of passive decay heat removal systems to allow higher temperature operations
Systems for severe accidents mitigation / management
Water cooled SMRs
SCOR (Annex V) • In-vessel cooling of corium **CCR** (Annex IX) • R&D on in-vessel retention (IVR) by passive means **AHWR** (Annex XI) • Steam driven poison injection system is planned to be demonstrated in a large integral test facility (ITL)
Seismic design
Water cooled SMRs
MARS (Annex IV) • Seismic qualification of innovative mechanical solutions
Sodium cooled SMRs
MDP (Annex XXII) • Development and evaluation of seismic isolation technology • Development of a reliability evaluation procedure regarding seismic isolation • Establishment of guidelines for seismic isolation technology
Lead-bismuth cooled SMRs
MS-LBCR (Annex XXV) • Three-dimensional seismically isolated reactor building

TABLE III.5. FURTHER R&D ON TECHNOLOGIES FOR FABRICATION OF FUEL AND MATERIALS, FUEL REPROCESSING AND WASTE DISPOSAL

SMR type **SMR name** (annex number corresponding to design description in this report) • Directions of R&D
Water cooled SMRs
AHWR (Annex XI) • R&D on dry reprocessing of fuel • Laser isotopic denaturing of zirconium • Reconstitution of fuel
Gas cooled SMRs
PBMR (Annex XIV) • R&D on fuel cycles • Fuel qualification, specifically the TRISO coated fuel **GT-MHR** (Annex XV) • Fuel development and demonstration programme is required to develop or validate a fuel fabrication process and a qualified fuel product for GT-MHR service conditions **GTHTR300** (Annex XVI) • Demonstration of fuel recycling technology in a small-scale test facility • Development of an economical waste storage system for graphite blocks discharged from the core **HTR-PM** (Annex XVII) • Large scale manufacturing of qualified fuel elements
Sodium cooled SMRs
KALIMER (Annex XX) • Remote fabrication technologies for metal fuel • Dry reprocessing of metal fuel **BMN-170** (Annex XXI) • Mastering fabrication of the nitride and metallic fuel and fuel elements **MDP** (Annex XXII) • Demonstration of technical feasibility of the metallic fuel reprocessing and fabrication technologies
Lead-bismuth cooled SMRs
PEACER (Annex XXIV) • Conceptual design of pyro-processing plant • Optimization of pyro-process design
Non-conventional SMRs
AHTR (Annex XXVI) • Development of a large scale manufacturing of high quality HTGR type fuel

TABLE III.6. FURTHER R&D ON TECHNOLOGIES OF CORE REFUELLING, PLANT MAINTENANCE AND IN-SERVICE INSPECTION

SMR type **SMR name** (annex number corresponding to design description in this report) • Directions of R&D
Water cooled SMRs
IRIS (Annex II) • Advanced diagnostics and prognostics are needed due to the integral configuration and to support the 4-year maintenance cycle and the 4-year refuelling cycle **IMR** (Annex VI) • Methods of control rod drive mechanism (CRDM) and rod position indicator (RPI) maintenance should be improved • In-vessel steam generator (SG) inspection devices should be improved (tube inspection is available from the secondary side of the SGs)
Gas cooled SMRs
PBMR (Annex XIV) • Maintainability of primary system components to assure ALARA • Demonstration of the effectiveness of sphere handling for the reactor unit • Validation of capability to measure fuel burn-up in each sphere
Sodium cooled SMRs
KALIMER (Annex XX) • Provisions for on-line refuelling **MDP** (Annex XXII) • Development of in-service inspection (ISI) methods • Establishment of guidelines for ISI • Establishment of post-accident maintenance and repair methods
Non-conventional SMRs
AHTR (Annex XXVI) • Refuelling and maintenance operations in the reactor vessel at 350 to 500°C

TABLE III.7. FURTHER R&D ON ISSUES OF NPP LICENSING, CONSTRUCTION, OPERATION AND DECOMMISSIONING

SMR type **SMR name** (annex number corresponding to design description in this report) • Directions of R&D
Water cooled SMRs
MARS (Annex IV) • Decommissioning programme of the plant **CCR** (Annex IX) • Improvement of module construction methods to reduce the construction period **RUTA-70** (Annex XII) • Reduction of operating costs by changeover to operating life of 100 years
Gas cooled SMRs
HTR-PM (Annex XVII) • Licensing requirements need to be established

TABLE III.8. FURTHER R&D FOR CALCULATION TECHNOLOGIES AND DATA SETS

SMR type **SMR name** (annex number corresponding to design description in this report) • Directions of R&D
Water cooled SMRs
SMART (Annex I) • Verification of the capability and modification of the analytic models of the design method, including design and analysis codes • Improvement and enhancement of the accuracy and capabilities of the design/analysis methods through the reduction of uncertainties **AHWR** (Annex XI) • Acquisition of nuclear data for nuclides important for the thorium cycle **RUTA-70** (Annex XII) • Verification and certification of the computer codes
Gas cooled SMRs
PBMR (Annex XIV) • Validate primary thermal-hydraulic code • Validation of PBMR codes and models to specific benchmark problems and test facilities such as HTTR and HTR-10 **GT-MHR** (Annex XV) • Development and qualification of fuel performance and fission product transport analysis methods and codes • Reactor physics and thermal hydraulics analysis methods that have been developed for earlier HTGR designs will need to be updated and qualified for application to the GT-MHR **GTHTR300** (Annex XVI) • Irradiation data for high burn-up fuel
Sodium cooled SMRs
BMN-170 (Annex XXI) • Refining of a database on properties of nitride and metallic fuel
Lead-bismuth cooled SMRs
PEACER (Annex XXIV) • Validation and testing of the REBUS neutronic and depletion analysis code system • Code development with respect to transient thermal-hydraulic and fuel performance analysis (DSNP, MATRA codes) • Development of a database of materials' properties

TABLE III.9. AVAILABLE OR PLANNED TEST FACILITIES IN MEMBER STATES

SMR type **SMR name** (Member State, annex number corresponding to design description in this report) • Facilities mentioned in relation to certain R&D
Water cooled SMRs
SMART (The Republic of Korea, Annex I) • A scaled-down test facility has been established; comprehensive tests using this facility will produce data on the whole system interaction, performance behaviour of the self-controlled pressurizer (PZR), indirect performance effect of passive residual heat removal system (PRHRS), natural circulation effects, etc. • High temperature, high-pressure integrated system thermal-hydraulic test facility • Separate effect test facilities for testing programmes mentioned in other tables of this Section **IRIS** (International consortium led by Westinghouse, USA; Annex II)) • A range of testing facilities is available through the IRIS consortium member organizations, which will enable effective conduct of the testing programme (see Table II-1 in Annex II) **CAREM** (Argentina, Annex III) • Related to fuel assembly design, hydrodynamic and structural tests are planned at two (low and high) pressure rigs • Related to the reactor core coolant system, modelling and qualification are boosted by the testing performed in a high pressure natural circulation rig (CAPCN) • Related to mechanical design (structural, dynamic, seismic, etc.) of the core and other RPV internals, different mock-up facilities are under construction • Qualification tests for the second shutdown system and, specifically, the hydraulic control rod drives (CRDs) are planned in a high pressure rig for CRD tests (CAPEM); the first series of tests (completed) was performed in a cold pressure rig (CEM) **AHWR** (India, Annex XI) • Large experimental facility (Integral Test Loop, ITL) is under construction; the tests will include validation of the isolation condensers, large passive heat sink located within containment, passive valves, steam driven poison injection, passive ECCS of enhanced effectiveness, passive containment isolation, etc. • Critical facility is under construction to acquire nuclear data for nuclides important for the thorium cycle, to validate negative void reactivity coefficient, etc.
Gas cooled SMRs
PBMR (South Africa, Annex XIV) • HTTR (Japan) and HTR-10 (People's Republic of China) test reactors data are used for the validation of PBMR codes and models • ASTRA critical facility (The Russian Federation) is used to investigate neutron physics of the PBMR pebble bed reactor • Burn-up measurement system prototype facility • Natural convection in core with corrosion (NACOK) facility to investigate oxidation of hot graphite cores by oxygen with natural circulation following air ingress events • Helium test facility (HTF) for testing of reactor system components in high temperature, high pressure helium environment • Heat transfer test facility (HTTF) to perform separate effects tests of heat transfer mechanisms in annular pebble bed core • Air test loop to demonstrate sphere transport system • Core unloading device to demonstrate effectiveness of sphere handling for the reactor unit • Control rod drive test facility to evaluate behaviour including torque and friction in CRDM **GT-MHR** (Annex XV) • Detailed description as of 2001, is provided in reference [19]

TABLE III.9. (Continued)

SMR type **SMR name** (Member State, annex number corresponding to design description in this report) Facilities mentioned in relation to certain R&D
Gas cooled SMRs (continued)
GTHTR300 (Japan, Annex XVI) • Small-scale test facility for demonstration of fuel recycling technology **HTR-PM** (People's Republic of China, Annex XVII) • Safety demonstration experiments are being conducted within the HTR-10 test reactor to demonstrate intrinsic reactor shutdown
Lead-bismuth cooled reactors
PEACER (The Republic of Korea, Annex XXIV) • Technology loop for Pb-Bi coolant technology analysis • Demonstration loop for the technologies ensuring reliable core operation • Corrosion test loop to define and validate a technology for the protection of structural materials from corrosion / erosion in a flow of Pb-Bi coolant

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DESIGN DESCRIPTIONS OF WATER COOLED SMRs

ANNEX I

SYSTEM-INTEGRATED MODULAR ADVANCED REACTOR (SMART)

KAERI, The Republic of Korea

I-1. General information, technical features and operating characteristics

I-1.1. Introduction

Since the Kori nuclear power plant unit 1, the first nuclear power plant unit ever dedicated in Korea, began commercial operations with a generating capacity of 587 MW in 1978, much research and development has been conducted in the nuclear industry. In the middle 1980s, the Korean standard nuclear power plant (KSNP) was first developed under the "nuclear power promotion plan" promulgated by the government with reference to system 80 of ABB-CE of the USA. Applying indigenously accumulated technologies and up-to-date design standards from both home and abroad, the initial KSNP project began with the construction of the Younggwang NPP units No. 3 and 4. In addition, the Korea Atomic Energy Research Institute (KAERI) designed and constructed a high performance multipurpose research reactor based on experience in the operation of previous reactors and accumulated nuclear technology [I-1]. Timed with completion of construction in April 1995, the reactor was named HANARO (high-flux advanced neutron application reactor), which, in Korean means, "uniqueness".

In the middle of the 1990s, research and development was launched related to small and medium sized reactors (SMRs) to promote the utilization of nuclear energy. SMRs are under development worldwide for various purposes such as district heating, seawater desalination, nuclear ship propulsion, as well as electricity production. Generally, modern SMRs for power generation are expected to have greater simplicity of design, economy of mass production, and reduced capital costs. Many SMRs also have advantages of reactor safety and economics by implementing advanced design concepts and technology [I-2]. Since 1997, KAERI has been developing the system-integrated modular advanced reactor (SMART), an advanced integral pressurized water reactor (PWR). The SMART is a promising, advanced SMR and has an integral type reactor with a rated thermal power of 330 MW. All major primary components, such as reactor core, steam generator (SG), main coolant pump (MCP) and pressurizer (PZR), are installed in a single reactor vessel assembly (RVA). The conceptual and basic designs of SMART with a desalination system were completed in March of 1999 and March of 2002, respectively [I-3, I-4]. SMART development has been conducted under the nuclear research and development programme supported by the Ministry of Science and Technology (MOST) of the Republic of Korea and thus KAERI and MOST are the principal stakeholders.

The SMART design focuses on the enhancement of safety and improvement of the reliability as well as the economics. For these purposes, highly advanced design features enhancing the safety, reliability, performance, and operability were introduced into the SMART design. Advanced design features should be proven or qualified by experience, testing, or analysis and, if possible, the equipment should be designed according to approved standards. Some fundamental thermal-hydraulic experiments were carried out during the design concept development to assure the fundamental behaviour of major concepts of the SMART systems. Various thermal-hydraulic and mechanical tests are in progress and planned. In addition, overall SMART performance will be demonstrated through the SMART pilot plant construction and operation.

I-1.2. Applications

The application of SMART is dual-purpose: electricity generation and seawater desalination. A concept of an integrated nuclear desalination plant coupled with SMART has been established and basic design and analysis were performed. The integrated plant aims to produce 40,000 m^3/day of potable water with a multi-effect distillation (MED) process and generates about 90 MW(e) of electricity. Preliminary analysis estimates that the amount of potable water and electricity produced are sufficient for a population of about 100,000. Design for optimal coupling of the MED system with SMART focuses on the economic use of energy and system safety.

I-1.3. Special features

The SMART is an advanced reactor for dual purposes, such as electricity generation and seawater desalination and can be used to supply electricity and fresh water to isolated areas where the main grid is not interconnected. The SMART has a daily load following capacity, such as 100% ~ 50% ~ 100% and electricity can be finely controlled by combining with the amount of seawater desalination. Since safety enhancement has been achieved by passive safety systems and radiation protection in the SMART design, these systems should enhance safety without relying on offsite power.

The SMART adopts a three (3) year refuelling cycle, which is two or three times longer than those of conventional NPPs, and soluble boron-free operation. These two design features can reduce the amount of liquid waste dramatically compared with a conventional PWR. The SMART is designed as a land-based NPP.

I-1.4. Summary of major design and operating characteristics

The SMART is a nuclear power plant to supply energy for seawater desalination and electricity generation. A high safety level is emphasized. Enhancement of system reliability and the exclusion of probable human errors are key design principles for securing a high level of safety [I-5].

The SMART is designed as an integral type PWR; one reactor pressure vessel contains the major primary components, such as modular once-through helically coiled tube steam generators, canned motor main coolant pumps, self-pressurizing pressurizer, etc. Figure I-1 shows the structural configuration of the SMART reactor. Four (4) main coolant pumps are installed vertically at the top of the reactor pressure vessel (RPV). The reactor coolant flows upward through the core and enters into the shell side of the steam generator from the top of the SG. The SGs are located at the circumferential periphery between the core support barrel and the RPV above the core. This design excludes the possibility of the large-break loss of coolant accident (LBLOCA) by the elimination of coolant loops. The integral reactor type feature also reduces the fast neutron fluence on the RPV. Additional innovations include the canned motor pumps, which remove the necessity of pump seals and the possibility of the small-break LOCA (SBLOCA) associated with pump seal failure, and the passive pressurizer that does not have an active spray and heater. This pressurizer design eliminates complicated control and maintenance requirements and reduces the possibility of malfunction. Advanced man-machine interface systems using digital technologies and equipments will reduce the human error.

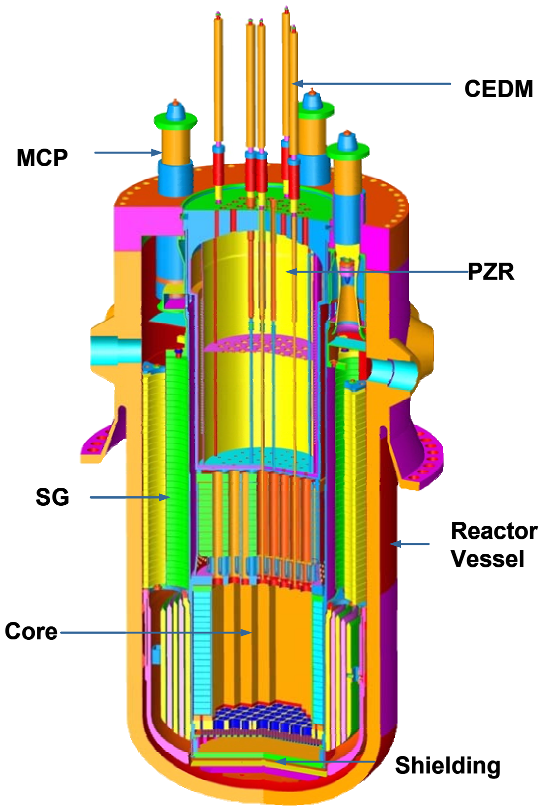


FIG. I-1. SMART reactor.

Figure I-2 shows the simplified schematic diagram of the SMART nuclear steam supply system (NSSS) and exhibits the safety systems and the primary system as well as auxiliary systems. The engineered safety systems designed to function passively on demand consist of a reactor shutdown system, passive residual heat removal system, emergency core cooling system, safeguard vessel and reactor overpressure protection system.

The major auxiliary systems of SMART consist of a component cooling system (CCS), purification system and make-up system. The function of the CCS is to remove heat generated in the main coolant pumps (MCPs), control element drive mechanisms (CEDMs), pressurizer (PZR), and the internal shielding tank. Feedwater supplied from the condensate pump of the turbo-generator is used as the coolant to remove heat. The purification system purifies the primary coolant and controls water chemistry to provide reliable and safe operation of the reactor core and all equipment in any mode of operation. The make-up system fills and makes-up the primary coolant in case of a primary system leak and supplies water to the compensating tanks for the PRHRS; it consists of two independent trains, each with one positive displacement makeup pump, a makeup tank, and piping and valves.

The turbine bypass system and condenser, in conjunction with the power cutback system, can accommodate a 100% load rejection without a reactor trip and without lifting either primary or secondary safety valves.

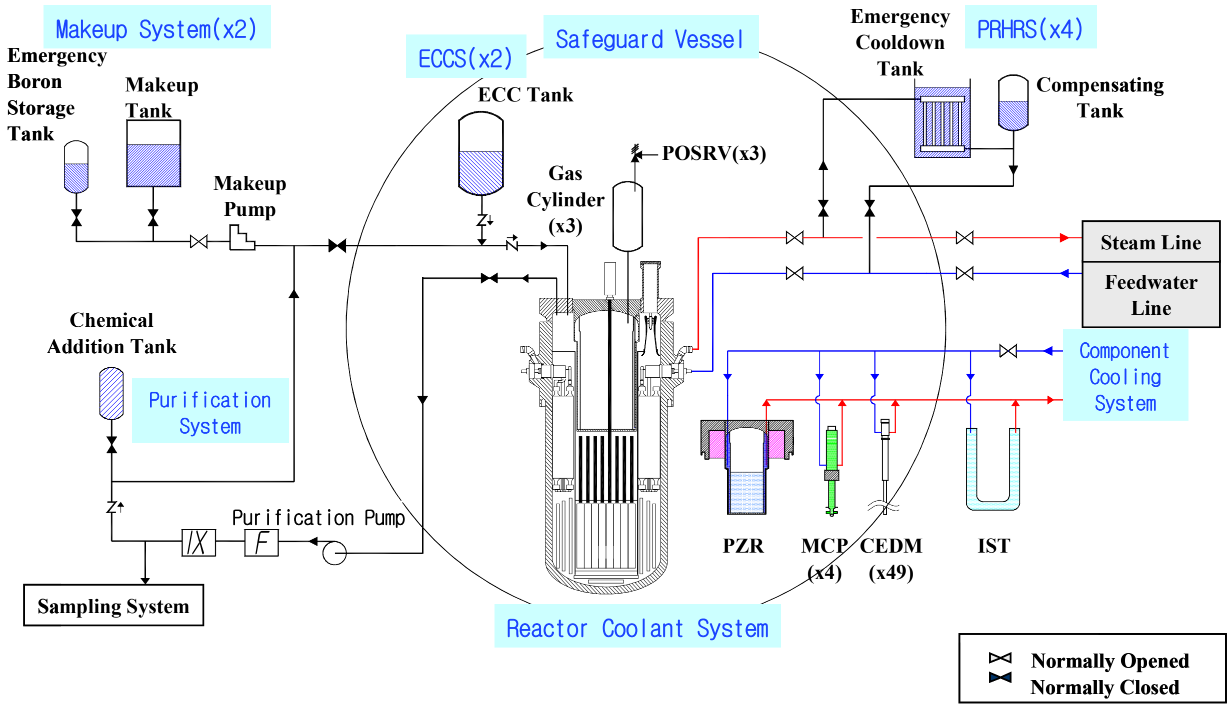


FIG. I-2. Simplified schematic diagram of the SMART NSSS.

Installed capacity:

Power plant output: Electricity 90 MW(e) and 40,000 tons of fresh water /day

Reactor thermal output: 330 MW(th)

Mode of operation: basic and/or load follow operation

Availability factor: more than 90%

Summary of major design characteristics:

Fuel material	Sintered UO_2
Enrichment	4.95 weight % ^{235}U
Rod array	Square, 17×17
Type of coolant	Light water
Type of moderator	Light water
Core type	57 square fuel assemblies
Core characteristics	Soluble boron free
	Low power density
Core dimension:	
Active core height	2.0 m
Equivalent core diameter	1.832 m
Type of reactor vessel:	
Cylindrical shell inner diameter	4072 mm
Wall thickness of cylindrical shell	264 mm
Cycle type	Indirect (Rankine cycle)
Number of circuits	3 (Primary, secondary, and condenser cooling system)

Neutron physical characteristics:

Excess reactivity at cold (20°C) condition, BOC	14.8 %$\Delta\rho$	
Reactivity defects		
Xenon worth	1.9 %Δ	
Power defect (HFP to HZP[1])	1.4 %$\Delta\rho$	
Temperature defect (HZP to CZP)	8.1 %$\Delta\rho$	
Reactivity coefficients at HFP		
Moderator temperature coefficient (MTC)	$-72 < MTC < -42$	pcm/℃
Fuel temperature coefficient	$-4.52 < FTC < -2.54$	pcm/℃
Maximum peaking factor (HFP)	3.29	

Reactivity control mechanism:

Type of control rod drive mechanism (CRDM):	Linear pulse motor
Number of CRDMs	49
Absorber rods per control assembly	24
Absorber material	Ag-In-Cd
Control Rod Worth:	
HFP Scram rod worth (from critical)	30 %$\Delta\rho$
CZP[2] total bank worth	25 %$\Delta\rho$
CZP bank worth at stuck rod condition	20 %$\Delta\rho$
Burnable absorber material	Al_2O_3-B_4C and Gd_2O_3-UO_2
Soluble neutron absorber	
Number of independent active reactor controls:	
Power manoeuvring	CRDM and negative MTC
Emergency shutdown	CRDM and emergency boron injection

Cycle type and thermodynamic efficiency:

Type	Indirect super heated steam Rankine cycle
Power plant efficiency	~30 %

[1] HZP: Hot Zero Power
[2] CZP: Cold Zero Power

Thermal-hydraulic characteristics:

Circulation type	Forced circulation
Number of coolant loops	Integral type
Reactor operating pressure	15 MPa
Coolant inlet temperature, at RPV inlet	270°C
Coolant outlet temperature, at RPV outlet	310°C
Mean temperature rise across core	40°C
Primary circuit volume, including pressurizer	56.27 m^3
Steam flow rate at normal conditions	152.5 kg/s
Feedwater flow rate at nominal conditions	152.5 kg/s
Steam temperature/pressure	≥ 274/3.0°C / MPa
Feedwater temperature/pressure	180.0/5.2°C / MPa
Core average heat flux	402 kW/m^2
Core average linear heat generation rate	12.0 kW/m
The design limit DNBR	1.41
Minimum operating thermal margin	15%

Maximum/average discharge burn-up of fuel:

Average discharge burn-up (nominal)	26,200 MW·d/t
Maximum discharge burn-up (nominal)	31,000 MW·d/t

Fuel lifetime:

Cycle length	990 Effective full power days

Mass balances/flow of fuel:

Fuel inventory	12.47 t U
Annual consumption of uranium	13,930[3] kg/GW(th) year
Annual consumption of natural uranium	143,850[4] kg/GW(th) year

Design basis lifetime of reactor core, vessel and structures:

Design basis lifetime for vessel and structures	60 years
Steam generators	15 years

[3] Total uranium inventory 12.47 t U

$$\text{annual consumption of uranium} = \frac{12{,}470 kgU * 365 D / year}{990 EFPD * 330 MWD * 0.001 GW / MW} = 13{,}930 kgU / GWtyear \cdot$$

[4] Feeding natural uranium enrichment: 0.71 weight %.
Tail uranium enrichment: 0.25 weight %
Annual consumption of uranium = 13,930 kg U/GW(th) year

Design and operating characteristics of system for non-electric applications:

Seawater desalination process type	MED
Number of effects	4
Evaporator type	horizontal tube falling film type
Performance ratio	19.6 kg of distillate / 2326 kJ

Operating characteristics:

Step turbine power changes of ±10% in the 20%–100% power range;
Power ramp of 5% per minute in the 20%–100% power range;
100% load rejection without a reactor trip.

Economics[5]:

Electricity generation cost	4.06 cent/kW h
Water generation cost	1.04 US $/m^3
Specific construction cost	1714 US $/kWe

I-1.5. Outline of fuel cycle options

The standard fuel cycle for the SMART utilizes conventional low enriched uranium in a once-through single batch fuel cycle without reprocessing. The cycle average burn-up of a single batch operation is 26.2 MW·d/kg U. The SMART reactor core design can accommodate an optional 1.5 batch reload cycle. The average core burn-up of the 1.5 batch reload cycle is about 31 MW·d/kg U. The conventional UO_2 ceramic fuel manufacturing companies would supply the SMART fuel.

I-1.6. Technical features and technological approaches that are definitive for SMART performance in particular areas

I-1.6.1. Economics and maintainability

Major economic improvements for SMART can be summarized as system simplification, component modularization, factory fabrication, direct site installation of components and reduced construction time. The integral arrangement of the primary reactor systems requires only a single pressurized vessel and removes large-sized pipes connecting primary components. The adoption of simplified passive systems provides a net reduction in the number of safety systems and drastically reduces the number of valves, pumps, wirings and cables, pipes, etc.

A simplified modular design approach is applied to all SMART primary components. Optimized and modularized small-sized components allow easy factory fabrication and direct installation at the site, leading to a shortened construction time and schedule. These features allow a construction period of less than three (3) years from first concrete to fuel load. The compact and integral primary system also eliminates complex and extra components associated with conventional loop-type reactors.

The SMART uses advanced on-line digital monitoring and protection systems that increase system availability and operational flexibility. The adoption of advanced man-machine

[5] Economic assessment study has been carried out for co-production of electricity and potable water [I-6]. The desalination economic evaluation program (DEEP) was used to evaluate the economics of desalination [I-7]. A 330 MW(th) integral reactor, SMART, has been taken as a nuclear energy option for seawater desalination.

interface technology leads to the reduction of human errors and to a compact and effective control room design with respect to minimizing staff requirements. The availability factor of the SMART plant is 95%, and the occurrence of unplanned automatic scram events is less than one per year.

The passive mechanisms of the major engineered safety systems largely contribute to simplification of the associated systems and components; simplification of the system is achieved mostly through the elimination or reduction in tanks, valves, and pumps. Table I-1 briefly compares the level of simplification of the major systems between the SMART and a 1000 MW(e) loop-type standard Korean PWR. The soluble boron-free design is an important design feature that largely contributes to system simplification by allowing the removal of associated systems and components required for boric acid processing, chemical volume and control systems. The passive safety design features can also reduce the number of active components. Such system simplifications lead and contribute to the improvement of the economy and system reliability by reducing construction time and cost, maintenance-related human errors and the probability of system failure, etc.

TABLE I-1. COMPARISON OF THE LEVEL OF SIMPLIFICATION OF MAJOR SYSTEMS

SYSTEM COMPONENT	1000 MW(e) CONVENTIONAL PWR	SMART
Chemical and Volume Control System		
Heat exchanger	2	1
Pump	13	6
Tank	7	4
Residual Heat Removal System		
Heat exchanger	2	4
Pump	2	2
Tank	0	8
Emergency Core Cooling System		
Heat exchanger	0	0
Pump	4	4
Tank	4	2
Valves	~1,100	~180

Maintainability criteria require that the SMART provides easier means to perform preventive and operational maintenance, the available space, ready access and devices to carry out replacement of the main components. In the basic design stage, a preliminary assessment of the maintainability and the ability to inspect the reactor pressure vessel and its internal components was performed and a programme for maintenance and inspection has been established.

The application of the SMART as an energy source for the dual purpose of electricity generation and seawater desalination, promises a new era of nuclear energy utilization and offers benefits only achievable with small-sized reactors. The SMART can be used not only for a dual-purpose but also for single-purpose applications based on the user's demands and can be effectively utilized to supply electric power to isolated areas not connected to the main grid and to the relatively small-sized industrial complexes needing high quality electricity.

I-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

The solid and liquid wastes produced in SMART must be stored and disposed of in ways safeguarding human health and protecting the environment. The overall strategy is to reduce the active nuclides in gaseous and liquid wastes to a solid form, reduce the volume of the solids, solidify loose material, pack in drums, and ship to a disposal site.

For low and intermediate level liquid wastes the general approach is to collect them in tanks and measure the activity. Active nuclides are removed through filters and ion exchange columns and after processing; the remaining activity is within the allowable site release limit. After liquid wastes are evaporated, the resulting sludge is solidified and then processed in the solid waste system, along with the filters and ion exchange resins. The solid waste treatment process chosen depends on the physical nature of the wastes; wastes could be classified as combustible or non-combustible but compressible, or non-combustible and non-compressible. The general approach to low level waste solid waste is volume reduction, followed by solidification where necessary and packing, usually in drums. Drums can be temporarily stored at site buildings for 10–20 years or shipped to a disposal repository.

High level waste management comprises activities related to irradiated or spent fuel after discharge from the reactor and thus includes storage and disposal, with or without reprocessing. The technology solution is to leave spent fuel or waste at surface level, to cool and decrease in radioactivity for some 30–60 years, and thereafter, to safely isolate them from the biosphere by deposit in deep and stable geological formations with a number of containment barriers.

Underground disposal is founded on a system of multiple, relatively independent barriers designed to ensure that toxic radionuclides in the spent fuel remain isolated from human beings and their environment. The barriers have three main components, the near field, the geosphere and the biosphere. The near field consists of stable wastes and some corrosion-resistant packaged wastes combined with the immediate engineered barriers incorporated in the repository. The geosphere comprises the barriers offered by the host geological media. A key factor is the ability to restrict the flow of groundwater, hence the relative impermeability of low regimes such as clay, salt and crystalline rock are considered. The biosphere may not constitute a barrier in the strict sense of the word but would serve to dilute radioactivity. An understanding of pathways through the biosphere is also important for the prediction of the eventual fate of any radionuclides.

Since the SMART does not use soluble boron in the primary coolant, the total amount of liquid waste generation would be minimized. This feature simplifies the liquid waste processing system.

I-1.6.3. Safety and reliability

The safety approach of the SMART is based on the defence-in-depth concept, inherent safety features and use of passive engineered safety systems. The SMART design combines firmly established commercial reactor design technologies with advanced technologies. Thus, substantial parts of the design features of the SMART have already been proven in industry. The new and advanced features of the SMART design provide significant safety enhancements classified into three major categories as follows:

- Application of the defence-in-depth concept,
- Inherent safety design features, and
- Passive engineered safety systems.

Application of defence-in-depth concept

In the SMART design, adopting and implementing safety features for all levels implement the defence-in-depth concept.

- *1st level: prevention of abnormal operation and failures.* This is achieved by the advanced design of the SMART, which is implemented in the first three barriers as the fuel matrix design, fuel rod cladding, and integral design of the primary coolant boundary to confine the radioactive materials. This level has been also considered in the implementation of other levels of defence.
- *2nd level: control of abnormal operation and detection of failures.* It includes control and response to abnormal operation and detection of system failures by means of the core monitoring system (SCOMS) and the SMART core protection system (SCOPS).
- *3rd level: control of accidents within the design basis.* It is provided by SMART engineered safety features, which are highly reliable, designed to function passively on demand and the reactor protection systems. The reactor protection system consists of 2 independent systems: control rods and an emergency boron injection system. This level is implemented as the fourth barrier in the form of vessel safeguards and enhanced by a steel-structure containment within a concrete building.
- *4th level: control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents*. This level is realized as accident management to preserve the integrity of confinement. This has been considered in the analysis of responses of the SMART safety features to accidents and transients.

Inherent safety characteristics

The SMART contains major primary components such as a core, twelve Steam Generators, a Pressurizer, four Main Coolant Pumps, and forty-nine Control Element Drive Mechanisms (CEDMs) in a single PRV. The integral arrangement of the primary system removes large size pipe connections between major components and thus, fundamentally eliminates the possibility of LBLOCA.

Canned motor MCPs eliminate the need for an MCP seal, and basically eliminate a potential for SBLOCA associated with seal failure. The modular type once-through SGs is located relatively high above the core to provide a driving force for natural circulation flow. This design feature along with low flow resistance enables the capability of the system to have natural circulation with the maximum power of 25%.

The system pressure is self-controlled by the partial pressures of steam and nitrogen gas filled in the PZR in accordance with the variation in pressure and temperature of the reactor coolant system. A large volume of passive PZR can accommodate a wide range of pressure transients during system transients and accidents.

Soluble boron-free operation is an evolving design characteristic of the SMART core along with the low core power density design. The very fine-step maneuvering of CEDM compensates for the reactivity change due to the fuel burn-up. Due to the soluble boron-free core, a substantially large negative moderator temperature coefficient (MTC) increases the inherent power stability and resistance to transients.

Passive systems

Besides the inherent safety characteristics of SMART, its safety is further enhanced by highly reliable engineered safety systems. These are designed to function passively on demand and consist of a reactor shutdown system, passive residual heat removal system (PRHRS), emergency core cooling system (ECCS), safeguard vessel, reactor overpressure protection system (ROPS) and containment overpressure protection system (COPS).

Active systems

The ECCS is designed to provide a coolant to the RPV passively during an early stage of SBLOCA. When an initiating event occurs, the primary system is depressurized, the valve in the line of the ECCS automatically opens and water immediately comes into the core by gas pressure. However, when the upper annular cavity water level reaches a low-level setpoint during the final stage of SBLOCA, the active makeup pump actuates to compensate for the loss of primary coolant inventory and keep the coolant level well above the top of the core for a long term cooling period.

Design for high reliability

High reliability is achieved in the SMART design by considering the potential for common cause failure and by including single failure criteria and fail-safe design. Common cause failures are accommodated through the principles of diversity, redundancy and independence for systems and components important to safety. Diversity is achieved with two distinctly independent systems for shutting down the reactor. Redundant and independent trains and components are installed in the safety systems such as the passive residual heat removal system (PRHRS), emergency boron injection system, emergency core cooling system (ECCS), and reactor overpressure protection system (ROPS). Single failure criteria are applied in the SMART safety analysis to show the capability of achieving emergency core reactivity control, emergency core and containment heat removal, and containment isolation, integrity and atmospheric clean-up. Diversity and redundancy are also included in the instrumentation and control system and structure. The principle of fail-safe design is incorporated by such means as dropping of the CEDM after loss of electrical energy and valves designed to be fail-open or closed in the secondary, auxiliary, and safety systems.

Design basis

Safety related design basis events (SRDBE) are selected in accordance with ANSI/ANS-51.1-1983 (R1988 [I-8]) for the safety analysis and safety review plan (SRP [I-9]). The SRDBE defines transients and accidents postulated in the SMART safety analysis, classifying unplanned occurrences to be accommodated by the SMART design and mitigated by the reactor protection system, engineered safety features or operator intervention. The SRDBE consists of 35 internal events; external events are not yet considered in the basic design stage. Due to the integral characteristics of SMART, the postulated major accident in the DBE having major consequences is initiated by the small break LOCA. For the DBE requiring the ECCS actuation, failure leads to severe accident occurrences. To anticipate severe accidents, SMART safety systems are equipped with the severe accident mitigation system (SAMS), preventing the egress of molten corium from the containment with a combination of safeguard vessel, internal shielding tank and containment.

Safety analysis

In the deterministic safety analysis, it is confirmed that operational limits are in compliance with the assumptions and intent of design for normal operation of the SMART. The safety analysis is performed on initiating events listed in SRDBE that are appropriate for the

SMART design. The initiating events result in event sequences that are analyzed and evaluated, for comparison with radiological and design limits as acceptance criteria. Up to the basic design stage, over 1500 simulations have been performed to demonstrate that the management of anticipated operational occurrences and DBA is possible by automatic responses of the safety systems. For non-LOCA initiating events, the safety analysis is supported with TASS/SMR [I-10] and MATRA [I-11] computer codes using a digital analysis method compatible with the digital protection and monitoring system of SMART. For LOCA initiating events, MARS/SMR [I-12] computer code is utilized for nodalization of SMART using a conservative method based on 10 CFR 50 Appendix K [I-13].

A steam line break (SLB) is a typical limiting accident belonging to heat removal capacity reducing accidents. A break size of 0.022 m^2 (100 mm in diameter) inside the containment is assumed and a loss of power concurrent with a turbine trip is considered. As the MCPs coast down following a reactor trip at 115% of the nominal power, the departure from the nucleate boiling ratio (DNBR) rapidly decreases to reach the minimum value and then rises abruptly due to the increased decay heat removal with the fully-established natural circulation as shown in Figure I-3. The primary and secondary pressures are well below the safety criteria of 110% of the design pressure, 18.7 MPa [I-14].

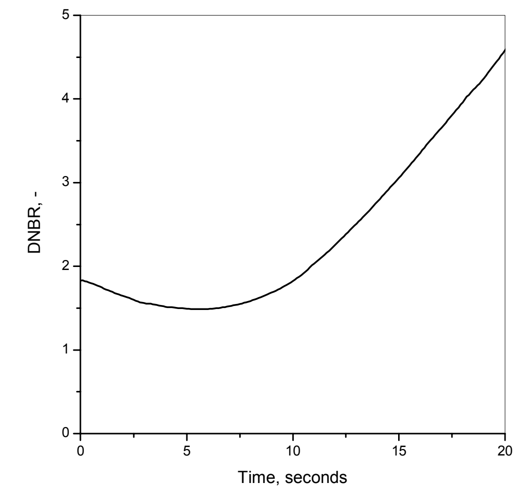


FIG. I-3. DNBR for the total loss of flow (TLOF) accident.

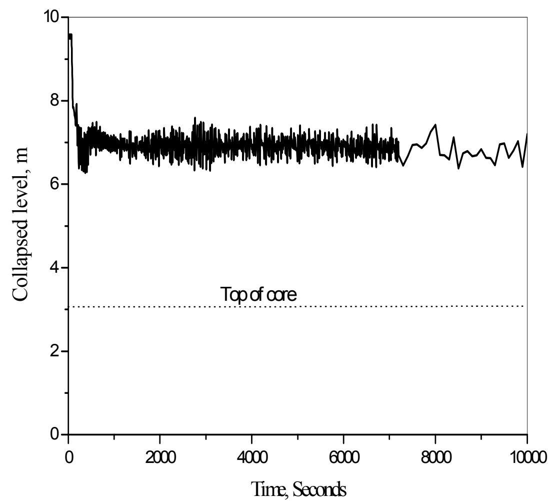


FIG. I-4. RPV collapsed water level for the SBLOCA.

The instantaneous guillotine rupture of the pipeline (20mm in diameter) connecting the PZR end cavity and the gas cylinder was considered for the analysis of the SBLOCA. The rapid discharge of N_2 gas through the break leads to a rapid decrease in system pressure and then the reactor trip occurs. The PRHRS starts to remove the decay heat and the further decrease of the pressure below 10 MPa causes the ECCS to actuate. The continuous discharge of steam to the safety guard vessel (SGV) eventually leads to actuation of the makeup system, which compensates for the loss of primary coolant inventory and keeps the coolant level well above the top of the core. Analysis shows that the collapsed water level is maintained at about 3 m above the top of the core as shown in Figure I-4, to prevent core damage [I-14].

Probabilistic safety assessment

The core damage frequency (CDF) of the SMART reactor system is evaluated by the level-1 full power probabilistic safety assessment (PSA) using the computer code KIRAP (KAERI integrated reliability analysis code package) [I-15]. The level 2 and 3 PSA, external PSA and

the low power / shutdown PSA will be performed in the final design stage. Reliability data for the components and initiating events is developed mostly from the Korean standard nuclear power plant (KSNP). PSA and data for common cause failure and human error are assumed by a conservative judgment on the basis of the KSNP PSA.

A total of 10 groups of initiating events (general transients, loss of feedwater, loss of off-site power, SBLOCA, steam line break inside the SG, SG tube rupture, large secondary side break, rod ejection, ATWS, and CEA bank withdrawal) are selected for the PSA on the basis of the SMART design and events considered in the design basis event analysis. The total CDF of the SMART design is evaluated as 8.6×10^{-7}/reactor-year. The SBLOCA event turned out to be the most dominant event contributing to 53% of the total CDF. The dominant sequence is attributed to common cause failure of the check valves in the makeup system following the event.

I-1.6.4. Proliferation resistance

Some of the important technical features of SMART, which reduce the attractiveness of spent nuclear fuel material for use in any nuclear weapons programme, are the following:

- Long refuelling cycle: SMART refuelling cycle is 3-year, and thus, the fissile content of plutonium in discharged fuel is too low to be used for weapon programmes.
- Low enriched uranium: SMART utilizes 4.95 weight % slightly enriched uranium oxide for the fuel ingredient.

I-1.6.5. Technical features and technological approaches used to facilitate physical protection of SMART

The safeguard vessel is a leak-tight pressure-retaining steel vessel intended to accommodate all primary reactor systems including the reactor assembly, pressurizer gas cylinders, and associated valves and piping. The primary function of the safeguard vessel is to contain the radioactive fission products within the vessel and to protect against primary coolant leakage to the containment. The containment building is therefore required to possess only a limited confinement function in case of failure of the safety vessel integrity or beyond design basis accidents.

I-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of SMART

The SMART is an advanced integral reactor developed in the Republic of Korea for the dual purpose of seawater desalination and electricity generation. Many organizations have participated in the project to develop SMART. Korea can supply a complete SMART nuclear plant with the integrated desalination plant under a turnkey contract. Korea has a large infrastructure for manpower training, which has been used and will continue to be available, to provide specialized training in nuclear related areas to personnel from several IAEA member states under IAEA programmes.

I-1.8. List of enabling technologies relevant to SMART and status of their development

A list of enabling technologies relevant to SMART based nuclear power plant is given in Table I-2.

TABLE I-2. ENABLING TECHNOLOGIES RELEVANT TO SMART

DESIGN	SAFETY IMPLICATION	TYPICAL DBE	EFFECT
Integral reactor	No large primary piping	Large LOCA	Physically eliminated LBLOCA
Reactor internal layout	Natural circulation	Loss of flow accident	Mitigation of accident consequence
No soluble boron	Soluble boron-free core	Boron dilution accident (BDA)	Physically eliminates BDA
Large shut down margin	Cold shutdown by control rods	Steam line break (SLB)	No possibility of return-to-power during SLB
High design pressure of SG and secondary system.	Feed / steam system designed for reactor coolant system pressure. No SG safety valve.	SG transient	Section isolation mitigates accident consequences
Helically coiled once-through SG	Shell side: primary Tube side: secondary	SG transient	Reduced possibility of accident occurrence
Self controlled pressurizer	Self pressure control	Pressurizer control system malfunction	The possibility of control system malfunction is physically eliminated
PRHRS	Closed loop passive residual heat removal	Fuel failure events	Elimination of release path
Large water inventory	Slow transient	SBLOCA	Mitigation of accident consequence
Active safety injection	No core uncover		

I-1.9. Status of R&D and planned schedule

Figure I-5 shows the SMART development programme; it was divided into three phases, technology development, plant construction and plant commercialization. Since 1997, the Korean Government has been supporting the SMART technology development. During this period, fundamental technologies were developed and the conceptual design was performed. After conceptual design, the basic design was finished in 2002. After the technology development phase, SMART plant construction with a power capacity of 65 MW(th) was launched for comprehensive performance verification. Commercialization of the SMART desalination plant will be introduced beginning in 2009.

The existing proven PWR technologies are basically utilized for the SMART design. However, it also adopts new and innovative design features and technologies that must be proven through tests, experiments, analyses, and/or the verification of design methods. For the design verification, a wide-spectrum programme was developed and implementation is currently underway. This programme includes basic thermal-hydraulic experiments, separate effect tests for major design features and components and an integrated thermal-hydraulic test for the SMART system [I-16].

At the early stages of development, several basic thermal-hydraulic experiments were established and conducted for the components and system concepts. The main purpose of these experiments was two-fold: to understand the thermal-hydraulic behaviour of the specific design concepts and to obtain fundamental data to be used, in turn, for further concept design. Among the experiments conducted, the current SMART design-related experiments are as follows:

- Boiling heat transfer characteristics in the helically coiled steam generator tube.
- Experiment for natural circulation in the integral arrangement of the reactor system.
- Performance tests for key parts of CEDM and MCP.
- Tests for the instrumentation and control systems.

Analytical models developed using data from these experiments and tests are implemented into the related design codes and then used for conceptual design of the related SMART system.

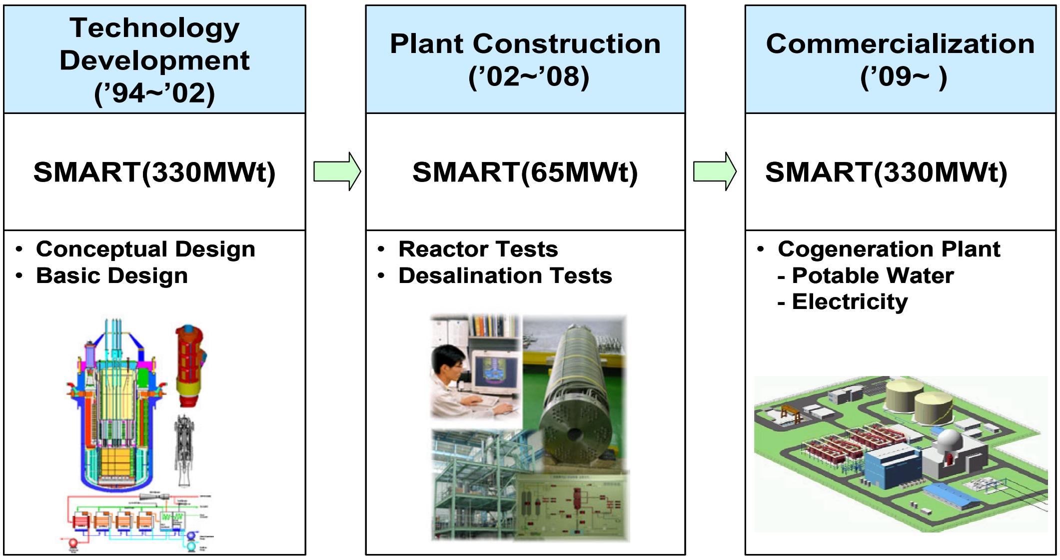


FIG .I-5. SMART development programme.

From concepts developed during basic design of the SMART system, a test and verification programme focusing on tests and experiments was established. Some of them were successfully completed during the basic design phase and some are currently underway. Valuable data produced from the tests and experiments were directly utilized to upgrade and modify the analytical models incorporated in the design codes and utilized for design and safety analysis. Representative tests conducted during the SMART basic design phase were:

- Two-phase critical flow tests with non-condensable gases to investigate the thermal-hydraulic phenomena of critical flow with the existence of non-condensable gases,
- Critical heat flux measurement for SMART-specific UO_2 fuel rod bundles,
- Water chemistry and corrosion tests at a loop facility to examine the corrosion behaviour and characteristics of fuel cladding, internal structural materials and steam generator tube materials at reactor operating conditions, and
- Upgraded performance tests for key components of the CEDM and MCP.

Other on-going separate effect test results to be used in modelling for the performance and safety analysis as well as design optimization include:

- Experiments on wet thermal insulation, to determine the insulating effects for the low operating temperature PZR design and to derive a heat transfer coefficient for the design;

- Experiments on phenomena and characteristics of heat transfer through the condensing mechanism of the heat exchanger inside PRHRS tanks.

In addition to these tests and experiments for the specific design concept and features, a high temperature, high-pressure integrated-system thermal-hydraulic effect test is also underway. A scaled-down test facility has been established. Comprehensive tests using the facility will produce valuable and practical information on the whole system interaction, performance behaviour of the self-controlled PZR, indirect performance effect of the PRHRS, natural circulation effects, etc. Most of the information from the tests will be used to verify the capability and to modify the analytic models of the design method including design and analysis codes. Performance tests of major SMART components such as CEDM, MCP, and SG are planned and will be carried out soon.

Along with the design verification of hardware, efforts to improve and enhance the accuracy and capabilities of the design/analysis methods are also being continuously pursued through the reduction of uncertainties. The government of the Republic of Korea decided to construct a one-fifth scale pilot plant for demonstration of the overall SMART performance by 2008 [I-17]. The six-year-long, 2nd-phase project of development started in July 2002 with the objective of verifying the integral performance of system technologies and confirming industrial applications through construction and operation of the pilot plant, the SMART-P.

A research and development centre for construction of the SMART-P was launched in June 2002. Many industries and universities in the Republic of Korea, such as Doosan Heavy Industries and Construction Company (DSHIC), the Korea Power Engineering Company, Inc. (KOPEC), Korea Institute of Nuclear Safety (KINS) and various universities participate in the SMART-P construction project.

I-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

The overall performance and industrial applications of the SMART can be confirmed and operational information and data can be obtained through SMART-P construction and operation.

I-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

CAREM (Argentina) [I-18]

IRIS (USA) [I-19]

MRX (Japan) [I-20]

I-2. Design description and data for SMART

I-2.1. Description of the nuclear systems

The descriptions in this section provide only selected design data for SMART; more details on SMART design and systems can be found in [I-5].

Reactor core and fuel design

The SMART core consists of fifty-seven (57) fuel assemblies based on the 17×17 KOFA[6] designed by KAERI/Siemens-KWU and used in the 900 MW(e) Westinghouse-type Korean

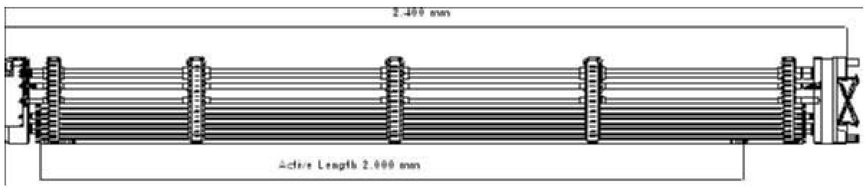
[6] KOFA: Korean standard fuel assembly

PWRs. The active fuel height of the SMART is 200 cm. Using a 4.95 weight % enrichment of U-235; the core can be operated for 3 years without refuelling. The core design is characterized by a long cycle operation with a single or modified single batch reload scheme, low core power density, soluble boron-free operation, enhanced safety with a large negative moderator temperature coefficient (MTC) at any time during the fuel cycle, a large thermal margin, inherently-free from xenon oscillation instability and minimum rod motion for the load follow with coolant temperature control.

In Figure I-6, the SMART fuel assembly and core loading pattern are presented for the 3 fuel types. Since neutron fluxes are higher in the core central region than in the peripheral region, more burnable absorber rods with a higher concentration are used in fuel type C than in fuel type B. The number and concentration of the burnable absorber rods in each fuel type are selected so that reactivity of each assembly can be as flat as possible. Therefore, the power distribution and critical control rod position do not change much during the cycle.

Main coolant pump (MCP)

The SMART MCP is a canned motor type pump that eliminates the problems of conventional seals and associated systems. In other words, canned motor type pump eliminates a small break loss of coolant accident (SBLOCA) associated with a pump seal failure. Four (4) MCPs are installed vertically on the RPV annular cover. An MCP is an integral unit consisting of a canned asynchronous, 3-phase motor and an axial flow single-stage pump. The motor and pump are connected through a common shaft rotating on three radial and one axial thrust bearings. The impeller draws the coolant from above and discharges downward directly to the SG. This design minimizes the pressure loss of the flow.



J	H	G	F	E	D	C	B	A	
			A	B	A				1
		B	B	B	B	B			2
	B	B	C	C	C	B	B		3
A	B	C	C	C	C	C	B	A	4
B	B	C	C	C	C	C	B	B	5
A	B	C	C	C	C	C	B	A	6
	B	B	C	C	C	B	B		7
		B	B	B	B	B			8
			A	B	A				9

A	4.95w/o U-235 / 28 Al_2O_3-B_4C Shim / 12 Gd_2O_3-UO_2
B	4.95w/o U-235 / 20 Al_2O_3-B_4C Shim / 4 Gd_2O_3-UO_2
C	4.95w/o U-235 / 24 Al_2O_3-B_4C Shim / 4 Gd_2O_3-UO_2

FIG. I-6. SMART fuel assembly and core loading pattern.

Safety systems

Enhanced safety is accomplished in the SMART with highly reliable engineered safety systems. These systems are designed to function passively on demand and consist of a reactor shutdown system, passive residual heat removal system, emergency core cooling system, safety vessel, and containment overpressure protection system. Additional engineered safety systems include the reactor overpressure protection system and the severe accident mitigation system.

Figure I-7 shows the schematic diagram of the safety systems. A detailed description of the SMART safety systems is given in [I-21].

Main heat transport system and path of heat removal in normal operation and in accidents

The scheme of the main heat transport system specifying the path of heat removal in normal operation and in accidents is presented in Fig. I-8.

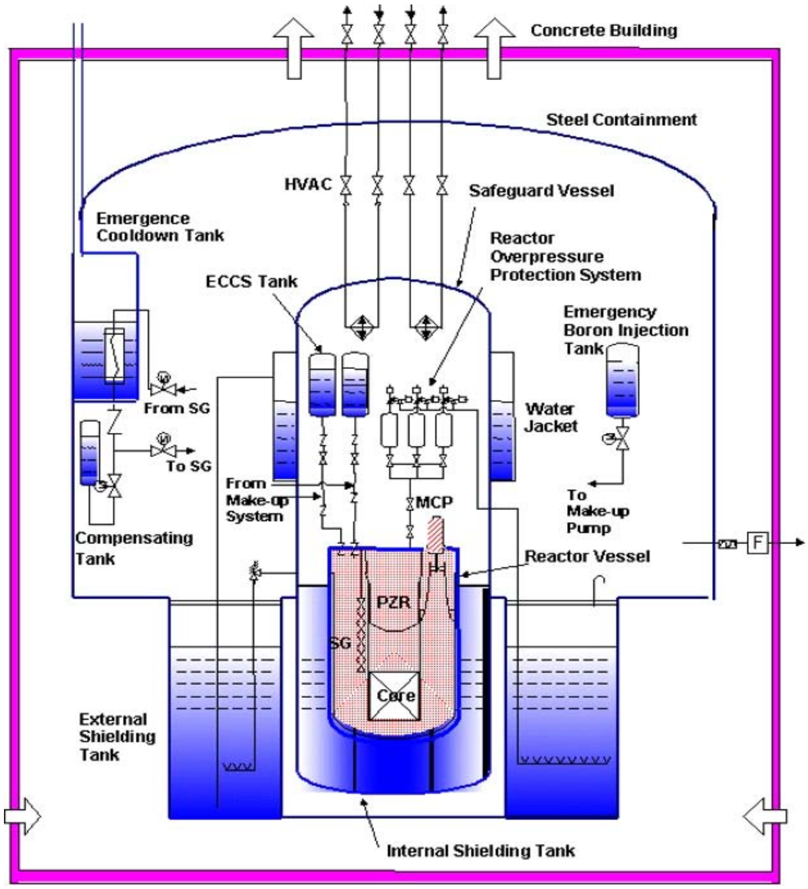


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FIG. I-7. Schematic diagram of the SMART safety systems.

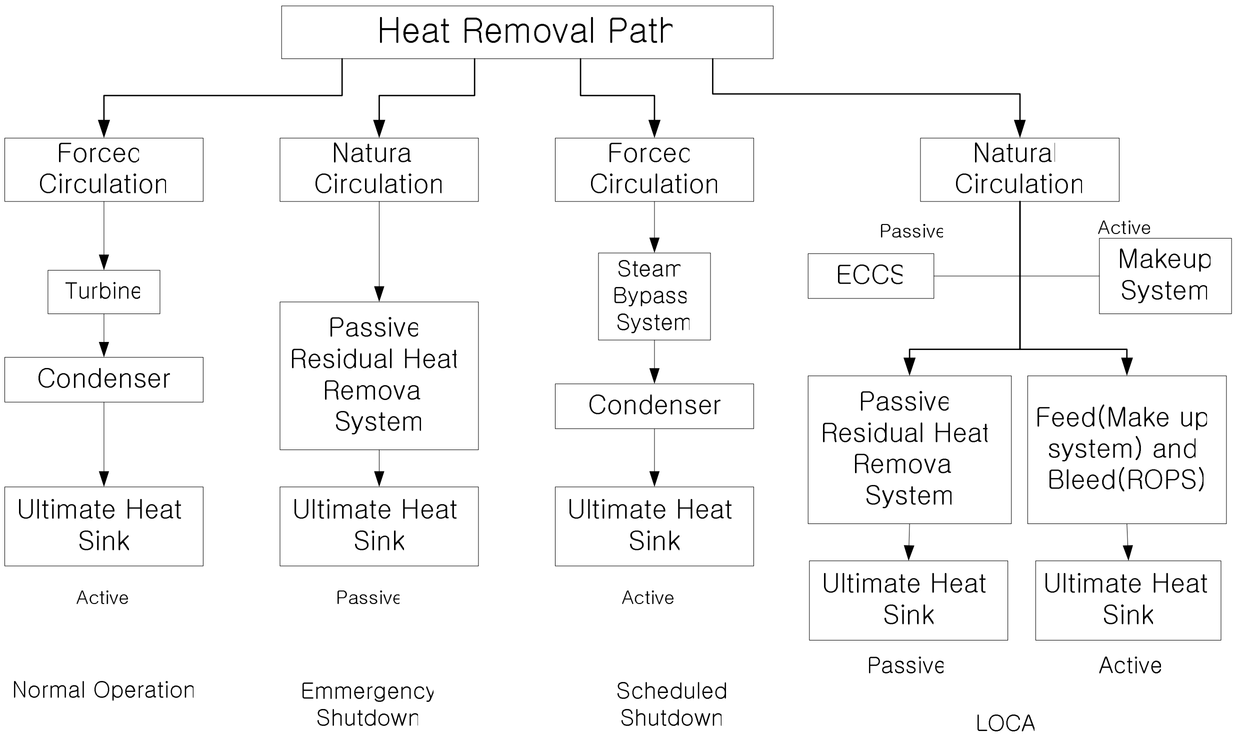

FIG. I-8. Heat removal paths of SMART.

Electrical systems

The concepts of the SMART electrical systems are basically the same as that of the PWRs presently operating in Korea. These systems include the main generator, main transformer, unit auxiliary transformers, stand-by auxiliary transformers, diesel generators, and batteries. The electrical systems, including the Class 1E and non-class 1E, are based on a “two train” approach.

The main generator is connected to the grid via the main transformer with auxiliary transformers connected between them. The stand-by auxiliary transformers (the off-site power source) receive electrical power from the grid. The unit auxiliary transformers and/or the stand-by auxiliary transformers supply electrical power for start-up, normal operation and shutdown.

If normal electrical power is unavailable, the diesel generators (non-class 1E, class 1E) act as a back-up source. In the event of a station blackout (loss of off-site and on-site AC power supply), the alternate AC diesel generator (class 1E) supplies power to the class 1E loads to maintain the reactor in a safe shutdown condition. The batteries have adequate capacity to supply DC power to perform required functions in an accident assuming a single failure. To ensure the safety of the reactor, the electric power supply for the safety related systems is designed as a highly reliable power source (class 1E). Two physically separate power sources are provided to the safety related system.

I-2.2. Description of the turbine generator plant and systems

The reference concept of the turbine plant has been developed including a coupling system for seawater desalination. The overall design is similar to that of a present-day power plant. The turbine plant receives superheated steam from the NSSS, it uses most of the steam for electricity generation, seawater desalination and to provide heat supply to the pre-heaters.

The SMART and MED-TVC units are connected through the steam transformer. The steam transformer produces the motive steam using steam extracted from a turbine and supplies it to the desalination plant. It also prevents the contamination of water produced by hydrazine and radioactive material of the primary steam. More details about the turbine generator systems can be found in [I-21].

I-2.3. Systems for non-electric applications

The integrated SMART desalination plant consists of 4 units of MED combined with thermal vapour compressor (MED-TVC) as shown in Figures I-9 and I-10. Each unit can produce 10,000 m^3/day of distilled water in 24-hour operation at the maximum brine temperature of 65°C and the supplied seawater temperature of 33°C. MED process coupled with the SMART incorporates the falling film, multi-effect evaporation with horizontal tubes and steam jet ejector.

One significant advantage of the MED-TVC is its ability to use the pressure energy in steam. Thermal vapour compression is very effective where the steam is available at higher temperatures and pressures than that required in the evaporator. A thermal vapour compressor boosts the low-pressure waste steam to a higher pressure, effectively reclaiming the available energy. Using the ejector, compression of steam flow can be achieved with no moving parts. The MED-TVC unit is designed with a performance ratio (PR) of 15 and the motive steam to load ratio of one.

The PR and steam to load ratio were determined based on the results of thermodynamic analysis and economic evaluation for the water production capacity of 40,000 m^3/day and electricity generation of 90 MW(e).

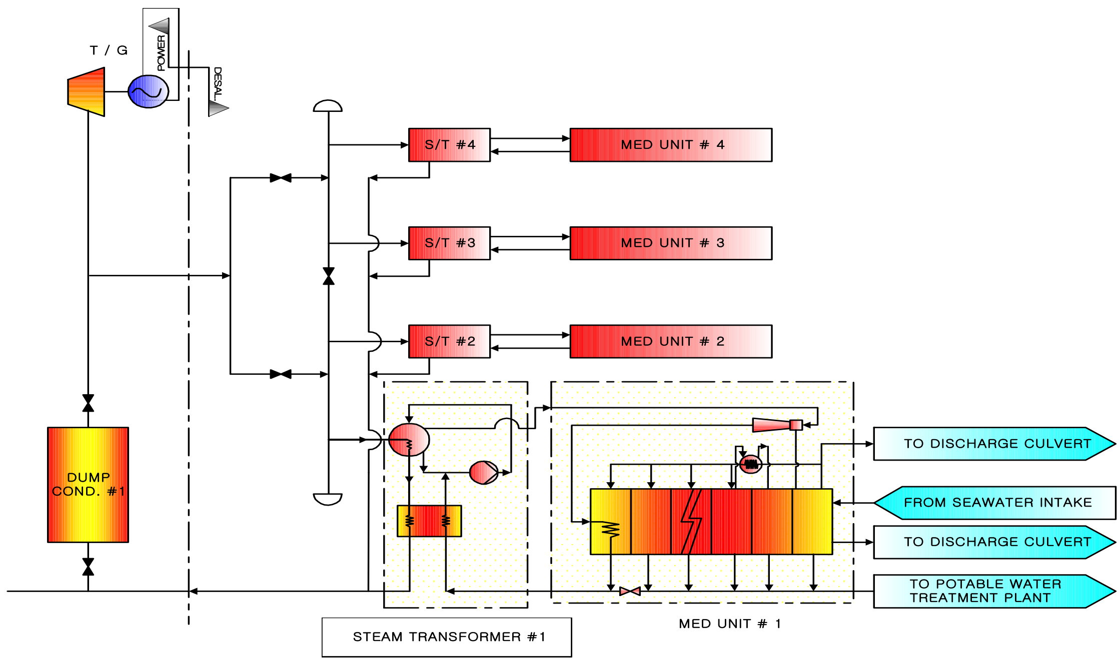

FIG. I-9. Coupling concept between SMART and desalination plant.

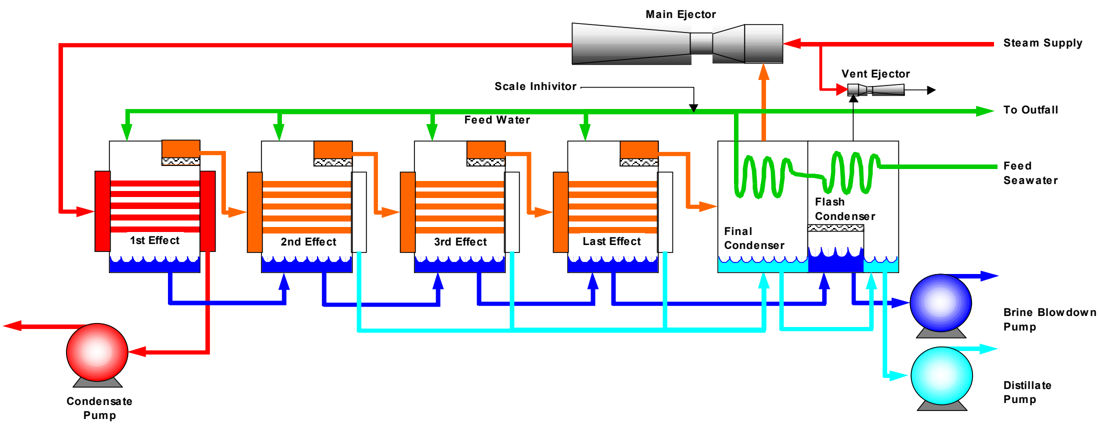

FIG. I-10. Typical flow diagram of the MED-TVC.

The SMART and MED-TVC units are connected through the steam transformer. The steam transformer produces the motive steam using steam extracted from the turbine and supplies process steam to the desalination plant. A steam transformer also prevents contamination of the produced water by hydrazine and radioactive material of the primary steam. The steam transformer is made of horizontal tube bundles; the primary steam flow is condensed inside the tubes at its saturation temperature. The feed brine is sprayed outside of the tube bundles by the recycling pump. Part of the sprayed water is evaporated and the produced steam is used as the motive steam for the thermo-compressor of the evaporator. Part of the condensate in the first cell of the evaporator is used as make-up for the steam transformer and this makeup water is preheated by the condensate of the primary steam before being fed into the steam transformer. The pre-heater is a plate-type heat exchanger made of welded titanium.

An important safety concern in using nuclear thermal energy for desalination is prevention of the radioactivity carry-over into the product water from the nuclear reactor. In the integrated nuclear desalination plant using the SMART, two mechanisms for protection are provided to avoid radioactive contamination of the product water. Figure I-11 shows a schematic diagram of the steam transformer coupling the SMART and the desalination system. Two barriers, the steam generator and the brine heater, along with the pressure reversal between the energy supply and the desalination system, act as protection mechanisms in the coupling. In addition, a continuous radioactivity monitoring system will be installed in the water production system to check for symptoms of contamination, with an immediate system reaction to follow in case of the detection of radioactivity. Additional monitoring may also be performed in an intermediate loop where the concentration of contaminate is higher than the water plant.

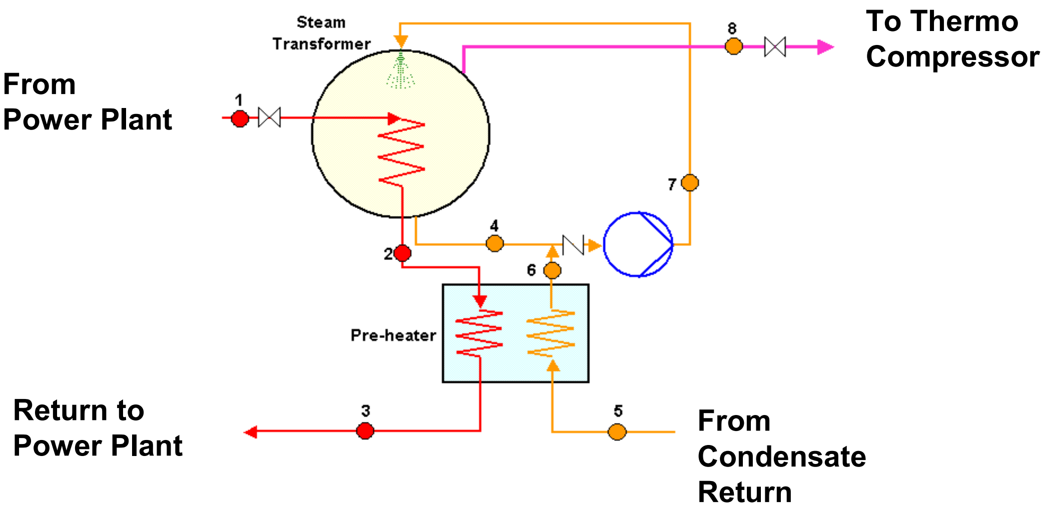

FIG. I-11. Schematic diagram of steam transformer.

I-2.4 Plant layout

The plant layout of the SMART is optimized to meet functional needs, safety, radiation zoning, and access and security considerations. Figure I-12 shows the general arrangement of a SMART plant. The building arrangement is designed to consider simplicity and convenience of maintenance. The approximate sizes of the some major buildings are given in Table I-3. More detailed information on the SMART reactor buildings is provided in [I-21].

TABLE I-3. APPROXIMATE SIZES OF SOME BUILDINGS

Type / Size	Type	L × W (m)	H (m)	Area (m^2)	Volume (m^3)
Containment building	Steel/Concrete double containment	18 ×16	45	576	14,079
Turbine building	Steel structure	41 × 28	27.5	4,100	31,570
Auxiliary building	Reinforced concrete structure	45 × 34	34	5,598	38,556
Utility building	Steel structure	52 × 24	8	1,872	9,984
Desalination building	Steel structure	52 × 24	8	1,872	9,984

For site-dependent internal or external hazards, design requirements will be similar to advanced evolutionary reactors, particularly with respect to:

- Earthquake,
- Aircraft crash,
- Explosion pressure wave,
- Internal hazards,
- Radiation protection aspects such as accessibility, shielding, ventilation, etc

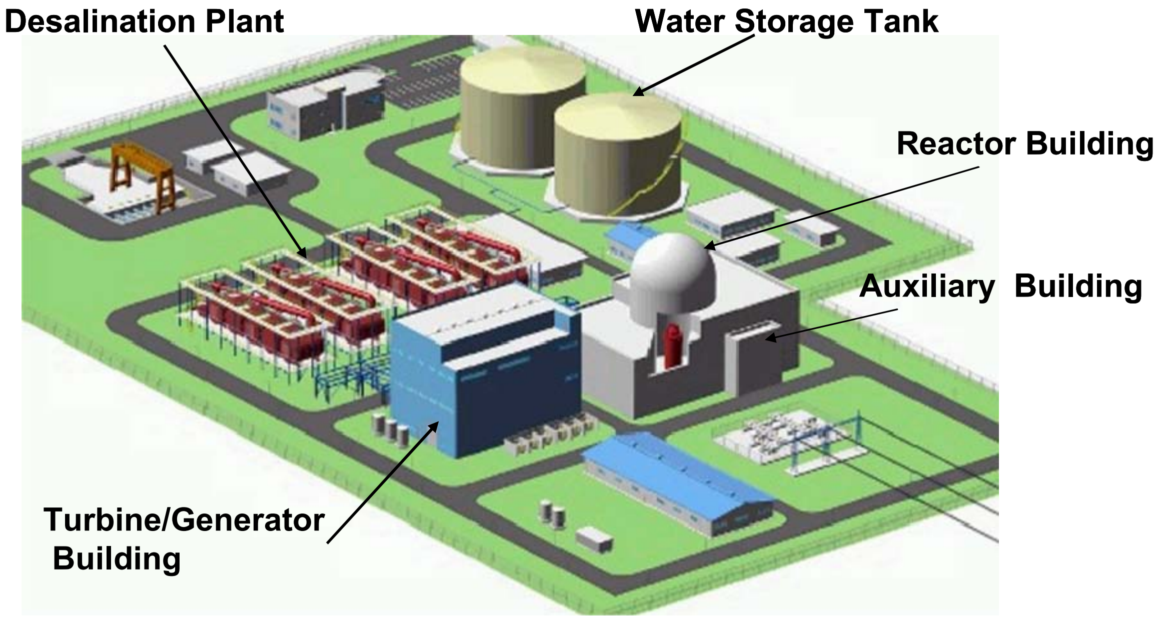

FIG. I-12. SMART desalination and power generation plant.

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INTERNATIONAL REACTOR INNOVATIVE AND SECURE (IRIS)

International Consortium led by Westinghouse Electric Company, United States of America

II-1. General information, technical features and operating characteristics

II-1.1. Introduction

The International Reactor Innovative and Secure (IRIS) is a modular, integral-type, pressurized, light water cooled, medium power (1000 MW(th)) reactor. The U.S. Department of Energy (DOE) as part of the Nuclear Energy Research Initiative (NERI) programme initiated the IRIS development programme under sponsorship. IRIS is classified as an International Near Term Deployment (INTD) reactor, within the Generation IV International Forum activities [II-1]. The IRIS concept addresses the top-requirements for next generation reactors, i.e. enhanced safety, reliability, positively proliferation resistance, and improved economics.

IRIS is based on proven LWR technology, with new engineering to improve safety and operating characteristics. IRIS employs an integral primary circuit configuration, rather than the loop configuration used in current power LWR reactors. IRIS is an innovative design because of its integral configuration and many novel design and safety features. However, since it does not represent a new technology, but rather new engineering, IRIS does not require a prototype, and in this aspect it differs from the strict interpretation of the IAEA definition of an innovative reactor.

For most systems/components, IRIS relies on the established PWR and partly BWR reactor technology, with its extensive operational experience, thus requiring none or only limited testing. Additionally, integral configuration water cooled reactors have been previously built and successfully operated (e.g. Otto Hahn ship), thus providing a historical technical basis for the IRIS integral configuration. Other integral concepts, such as SIR [II-2] and ISIS [II-3], have been investigated and various relevant tests performed (for example the ISIS steam generator testing is fully applicable to IRIS). There is also a wealth of experience with design and operation of liquid metal cooled integral reactors.

IRIS requires a testing and/or qualification programme for novel components (e.g., internal steam generators, internal fully immersed reactor coolant pumps), and some systems. A range of testing facilities is available through the IRIS consortium member organization, which will enable effective conduct of the testing program.

IRIS is being developed by an international consortium (led by Westinghouse Electric Company), which includes 21 organizations (including industry, vendors, national laboratories, academia and power producers) from 10 countries. Consortium members and their primary responsibilities are listed in Table II-1. All team members are stakeholders.

In a wider sense, developing countries that need affordable, smaller/medium NPPs early in the next decade may be considered as stakeholders as well, and there is already sufficient evidence of the interest by developing countries.

TABLE II-1. IRIS CONSORTIUM

INDUSTRY		
Westinghouse	USA	Overall coordination; leading core design, safety analyses and licensing
BNFL	UK	Commercialization and fuel cycle
Ansaldo Energia	Italy	Steam generators design
Ansaldo Camozzi	Italy	Steam generators fabrication
ENSA	Spain	Pressure vessel and internals
NUCLEP	Brazil	Containment
Bechtel	USA	Balance of plant, architect engineering
OKBM	Russia	Testing, desalination and district heating co-generation
LABORATORIES		
ORNL	USA	Instrumentation and control (I&C), probabilistic risk assessment (PRA), desalination, shielding, pressurizer
CNEN	Brazil	Transient and safety analyses, pressurizer, desalination
ININ	Mexico	PRA, neutronics support
LEI	Lithuania	Safety analyses, PRA, district heating co-generation
UNIVERSITIES		
Polytechnic of Milan	Italy	Safety analyses, shielding, thermal hydraulics, steam generators design, advanced control system
MIT	USA	Advanced cores, maintenance
Tokyo Inst. of Technology	Japan	Advanced cores, PRA
University of Zagreb	Croatia	Neutronics, safety analyses
University of Pisa	Italy	Containment analyses, severe accident analyses, neutronics
Polytechnic of Turin	Italy	Source term
University of Rome	Italy	Radwaste system, occupational doses
POWER PRODUCERS		
Electronuclear	Brazil	Developing country utility perspective
TVA	USA	Maintenance, utility perspective
ASSOCIATED U.S. UNIVERSITIES (NERI PROGRAMMES)		
University of California Berkeley	USA	Neutronics, advanced cores
University of Tennessee	USA	Modularization, I&C
Ohio State University	USA	In-core power monitor, advanced diagnostics
Iowa State University (and Ames Laboratory)	USA	On-line monitoring
University of Michigan (and Sandia Laboratories)	USA	Monitoring and control

II-1.2. Applications

The basic nuclear power plant (NPP) design employs each IRIS module to produce ~335 MW(e) of electricity. Co-generation NPP designs are being developed for desalination, district heating and process steam.

II-1.3. Special features

The reference design is a standard medium-power (335 MW(e)) land-based NPP.

Additional design options were considered and developed up to a certain stage, but are not actively pursued at this time. These include:

- A barge-mounted option.
- A reduced power (~50 MW(e)) version of IRIS with full natural circulation, no on-site refuelling, for deployment in remote locations as an autonomous energy source, with more than 15 years core lifetime.

Special features, some of which are further discussed subsequently, include:

- The IRIS safety-by-design™ philosophy improves safety and offers the technical basis for licensing with reduced or eliminated off-site emergency planning zone.
- Simplified design improves reliability and reduces construction cost and staffing level.
- Modularity enables adding new generating capacity in increments of 335 MW(e), to fit the electricity demand growth, and optimize cash flow.
- Pre-fabrication of components and systems is employed to reduce the construction time.

II-1.4. Summary of major design and operating characteristics

Major design characteristics of IRIS are given in Table II-2. A simplified schematic diagram of the IRIS nuclear installation is presented in Figure II-1. Further design and operating parameters are listed in Table II-3. Additional design information is provided in section II-2 as well as in the list of references.

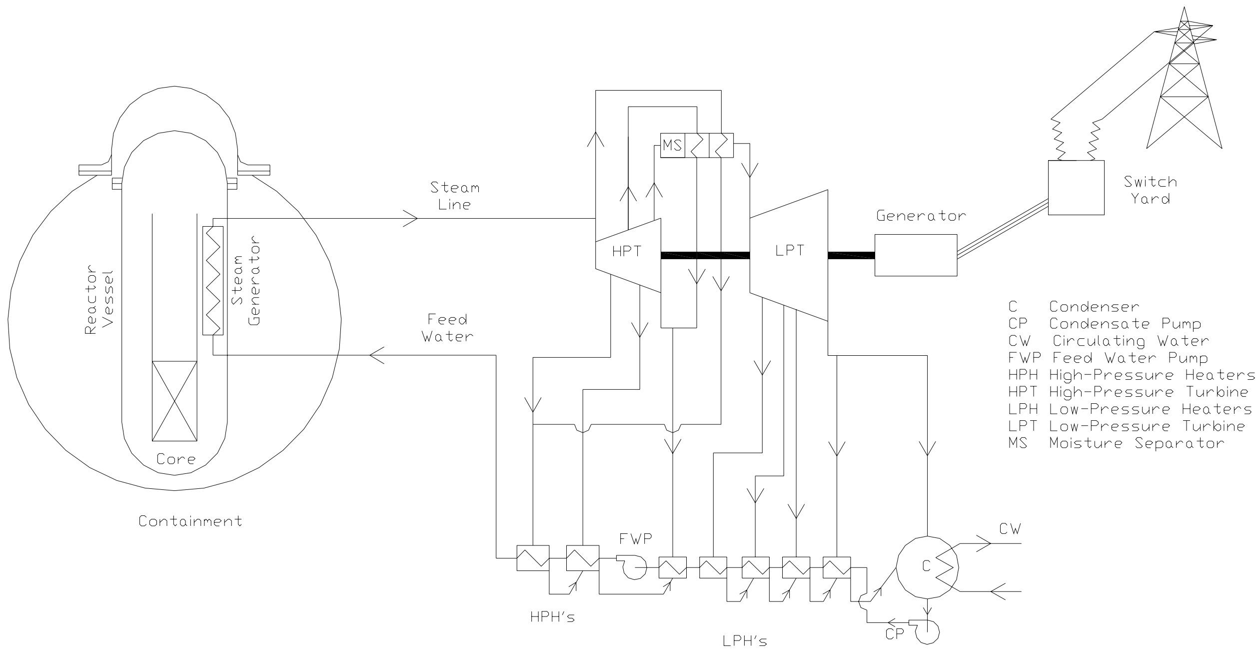

FIG. II-1. Simplified schematic diagram of the IRIS nuclear installation.

TABLE II-2. MAIN IRIS DESIGN CHARACTERISTICS

GENERAL PLANT DATA (SINGLE UNIT)	
Core thermal power	1000 MW
Power plant output, net	335 MW(e)
Mode of operation	Base load operation standard Enhanced load follow mode with MSHIM
Load factor Availability factor	Target >96% over the plant lifetime Target >98%
Plant design lifetime	Over 60 years (reactor vessel and structures)
FUEL	
Fuel material	Sintered ceramic UO_2 / MOX pellets
Enrichment	Up to 4.95 weight % ^{235}U fuel readily available Option for infrequent refueling requires ~7-10% fissile content
COOLANT/MODERATOR	
Coolant and moderator	Light water, subcooled
Number of coolant loops	Integral primary system
REACTOR CORE	
Equivalent diameter	2.41 m
Active core height	4.267 m
Fuel inventory	48.5 t U
REACTOR PRESSURE VESSEL	
Type	Cylindrical, low carbon steel
Cylindrical shell inner diameter	6.21 m
Wall thickness of cylindrical shell	285 mm
Total height	21.3 m
Design basis vessel lifetime	60 years (due to very low fast neutron fluence, lifetime over 60 years is possible)
CONTAINMENT	
Type	Pressure suppression, steel
Geometry	Spherical, 25 m diameter
Design pressure and temperature	1300 kPa, 200ºC

TABLE II-3. IRIS DESIGN AND OPERATING PARAMETERS

NEUTRON-PHYSICAL CHARACTERISTICS		
Reactivity Feedback	MTC negative over the whole cycle and power operating range	
Burn-up reactivity swing	Depends on the specific core design and use of burnable absorbers. Typical swing for a 42-month refueling cycle is about 8%.	
Peaking factors	$F_{\Delta H}$=1.7, F_q=2.65 (max. values)	
Power flattening approach	Use of burnable absorbers Selection of fuel loading pattern	
Reactivity control	Soluble boron, burnable absorbers, control rods	
Shut-down systems	Control rods, Emergency boration system	
REACTOR COOLANT SYSTEM		
Number of coolant loops	Integral primary system	
Primary circulation	Forced circulation, 8 in-vessel fully immersed pumps	
Primary coolant flow rate	4700 kg/s	
Reactor operating pressure	15.5 MPa	
Core inlet / outlet temperature	292°C / 330°C	
NUCLEAR STEAM SUPPLY SYSTEM		
Cycle type	Indirect	
Thermodynamic efficiency	34.9% (site dependent)	
Steam temperature and pressure	317°C, 5.8 Mpa	
Feedwater temperature and pressure	224°C, 6.4 Mpa	
FUEL CYCLE		
Fuel cycle options	Near-term deployment (fuel licensable today)	Mid-term deployment (requires fuel irradiation testing)
Equilibrium cycle length (period between refueling)	30-48 months	Up to 96 months
Average discharge burnup	• Up to 60 GWd/t U with the current U.S. limit on lead rod (62 GWd/t U) • Up to 70 GWd/t U with increased limit on lead rod (75 GWd/t U)	Up to 120 GWd/t heavy metal
Fuel enrichment	Up to 4.95 weight % ^{235}U	~7-10% fissile content
Annual consumption of natural uranium (t U_{nat}/GW(e) year)	169 (based on 60 GWd/t U discharge burn-up)	
ECONOMICS		
Construction time	3 years or less	
Capital cost	1030-1240 US \$/kW(e) – N^{th} of a kind plant (overnight cost, including interest/financing)	
Cost of electricity	30-40 US \$/MW/h	

IRIS employs a compact, spherical, steel containment vessel, about ~25 m diameter; its layout is depicted in Fig. II-2. This relatively small containment size results in a reduced cost, small plant footprint, and a high maximum allowed design pressure (not operational pressure).

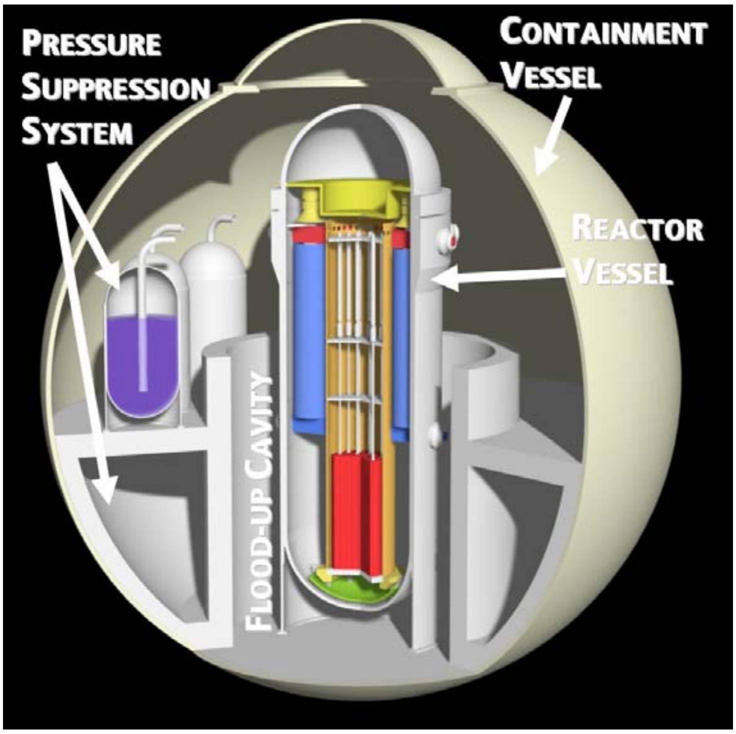

FIG. II-2. IRIS containment vessel layout.

IRIS uses an integral primary circuit, which is contained within the integral reactor vessel, as shown in Fig. II-3. The IRIS integral reactor vessel [II-4] includes all the major primary circuit components: the nuclear fuel and control rods (core); internal control rod drive mechanisms [II-5], eight small, fully immersed reactor coolant pumps [II-6], eight modular, helical-coil, once-through steam generators [II-7], neutron reflector; and, a pressurizer [II-8] located in the upper vessel head. As is shown in Fig. II-3, coolant flows upward through the core and then riser region, formed by the extended core barrel. Near the top of the riser, the coolant is directed laterally into the upper plenum where the suction of the pumps is located. The flow of each pump is directed downward through a corresponding steam generator module. The flow path continues downward through the annular downcomer region, reaching the lower plenum, where it is redirected back into the core, completing the circuit.

The core is composed of 89 fuel assemblies, each with a standard 17×17 lattice, as detailed in section II-2. Reactivity control is achieved by standard PWR means, i.e., a combination of soluble boron, control rods and burnable absorbers. Soluble boron concentration is kept limited to below about 1000 ppm, to maintain negative moderator coefficient and improve response in transients.

Another important design feature is the cold vessel because of the internal shielding (thick water annulus downcomer), intrinsically provided by the integral configuration. Its positive impact is described in more detail in section II-1.6.2.

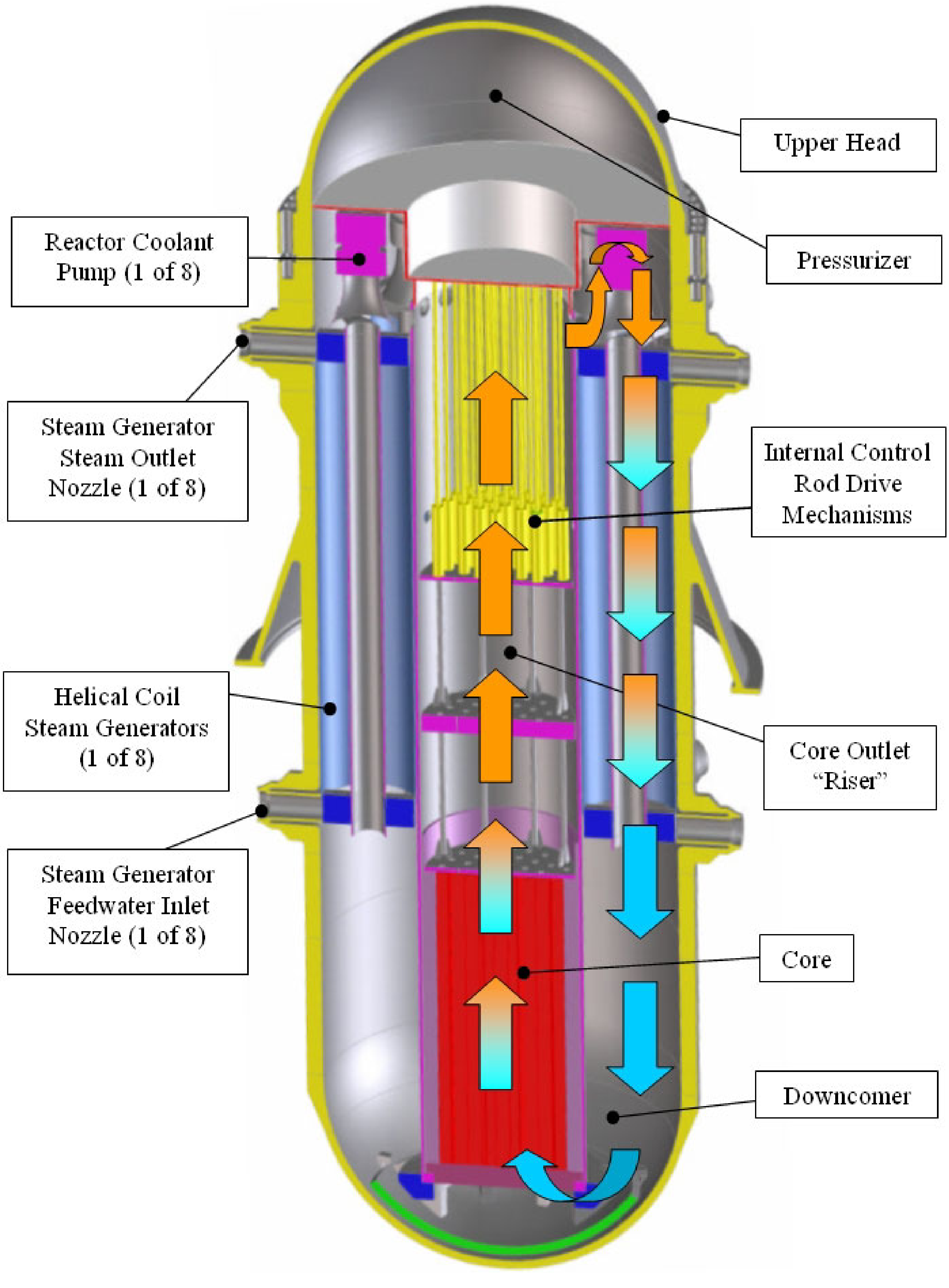

FIG. II-3. IRIS integral reactor vessel (integral primary circuit).

II-1.5. Outline of fuel cycle options

IRIS employs oxide fuel, UO_2 or MOX, with the associated once through or reprocessing, depending on specific market preferences.

IRIS targets near-term deployment and therefore it is not practical or feasible to immediately implement all the foreseen innovative features. This refers in particular to fuel cycle, where the long-term objective for infrequent refuelling (not more frequently than each 8 years) requires fuel irradiation testing to demonstrate high burn-up capability. Such testing takes time, the fuel must be licensed, and is thus not compatible with the near-term deployment schedule.

Therefore, the following phased approach is used to accomplish the fuel cycle objectives.

Near Term. Initial IRIS deployment is targeting the period 2012–2015. All the current licensing activities are focused on this deployment and rely on the current, proven and licensable fuel technology. This means that UO_2 fuel enrichment is limited to below 5% wt% ^{235}U, with additional limit on the maximum discharge burn-up (62 GWd/t U for the lead rod in the USA). This naturally limits the achievable cycle length. Nevertheless, IRIS has been designed to achieve up to a 4-year refuelling period, a significant extension of the current PWR cycle (12–24 months) that has positive economic and proliferation resistance implications. This extension may prove adequate for effective proliferation resistance, but even if further extension is eventually desired, it will provide a bridge while more advanced and distant technologies are being developed.

Mid Term. IRIS fuel cycle objective for the 2020s is to further extend the cycle length, resulting in refuelling each 8 years (or even less frequently). The corresponding fuel/core design is not part of the current licensing efforts, but it is pursued as part of separate, longer-term R&D. Beside the already mentioned irradiation testing, it would require increasing fissile content to ~8% for UO_2 fuel, and to ~10% for MOX fuel. The variable moderation approach is introduced to enhance Pu utilization in this case.

Long Term. Beyond 2030, even more innovative solutions are envisioned, including epithermal lattice and exotic fuel forms to further extend the refuelling period.

Out-of-core fuel cycle activities

An option is considered to provide front-end and back-end fuel cycle activities, as well as the ultimate SNF disposal, by the IRIS consortium, either on a regional or centralized basis. As the consortium includes members from 10 countries, it is ideally positioned to implement such approach, should it prove to be economically beneficial and supported by the international community.

II-1.6. Technical features and technological approaches that are definitive for IRIS performance in particular areas

II-1.6.1. Economics and maintainability

IRIS targets electricity markets in both developed and developing countries; therefore it needs to be competitive in all world market segments. Economic analyses performed so far have confirmed its economic viability worldwide [II-9]. Additionally, with its simple design, medium size and moderate cost of each unit, IRIS offers an acceptable option for smaller markets, electric grids, or countries with limited financial resources, where introduction of large power plants is not feasible due to technical or financial reasons.

IRIS offers several provisions for reduced capital, construction and O&M costs, including:

- Simplified design reduces construction costs.
- Production in series enables standardization and pre-fabrication of building modules and subsystems, with accompanying cost reduction.
- Relatively short construction time (3 years or less) reduces financing costs.
- Cash flow is improved (i.e. debit level is reduced) due to possible staggered build of multiple modules.
- Potential for licensing with reduced emergency planning zone offers prospects for reduced infrastructure costs, i.e. reduced capital investment, as well as reduced operating costs.
- Simplified design, infrequent refuelling, and optimized maintenance lead to reduced O&M costs, as well as reduced staffing level.
- Significantly reduced activation of reactor vessel leads to reduced doses to personnel in operation, maintenance, and decommissioning, with associated cost reduction.
- The approach to initially use fuel of the current PWR type (UO_2 with enrichment below 5%) provides a readily available, low cost solution, with already established fuel performance, and no *ad hoc* testing or licensing costs. More advanced fuel and fuel cycle may be introduced when fully developed, licensed, and if justified by market conditions, however IRIS deployment does not depend on these developments.

II-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

In the near-term option, using current fuel technology, IRIS improves fuel utilization by enhanced moderation, and thus extends refuelling to up to 4 years. The mid-term option with infrequent-refuelling (each 8 years), further reduces the volume of high-level waste.

A significant advancement is achieved in the area of dose reduction. The IRIS integral configuration and integral reactor vessel include a wide downcomer region (~1.7 m radially), which, together with the radial neutron reflector, provides significant shielding and results in a very large neutron flux attenuation. Thus, neutron flux at the reactor vessel and in the cavity, together with the corresponding activation of materials (vessel, cavity liner, concrete), is reduced by over 5 orders of magnitude, compared to a typical loop-type PWR reactor and over 3 orders of magnitude compared to a typical BWR. As a result, the vessel outer surface and structures outside the vessel are practically “cold”, thus reducing the dose in operation, maintenance, as well as the final decommissioning activities. Infrequent refuelling additionally contributes to the dose reduction during O&M activities.

Additionally, reactor vessel embrittlement due to irradiation is for all practical purposes eliminated. This not only extends the vessel lifetime (not limited by irradiation), but also eliminates the need for a surveillance program, further reducing the dose to personnel and the associated costs.

II-1.6.3. Safety and reliability

The overall approach to safety, which is a uniquely, defining characteristic of IRIS [II-10] may be represented by the following three-tier approach:

1. The first tier is the safety-by-design™, which aims at eliminating by design the possibility for an accident to occur, rather than dealing with its consequences. By eliminating some accidents, the corresponding safety systems (passive or active) become unnecessary as well.
2. The second tier is provided by simplified passive safety systems, which protect against the still remaining accidents and mitigate their consequences.
3. The third tier is provided by active systems, which are not required to perform safety functions and are not considered in deterministic safety analyses, but will contribute to reducing the core damage frequency (CDF).

First Tier. Nuclear power plants consider a range of hypothetical accident scenarios. The IRIS "safety-by-design"™ philosophy is a systematic approach that aims—by design—at eliminating altogether the possibility for an accident to occur, i.e., to eliminate accident initiators, rather than having to design and implement systems to deal with the consequences of the accident. It should be noted that the integral configuration is inherently more amenable to this approach than a loop-type configuration, thus enabling safety improvements not possible in a loop reactor. To give only the most obvious example, loss of coolant accidents caused by a large break of external primary piping (LBLOCA) are eliminated by design since no large external piping exists in IRIS. Additionally, in cases where it is not possible or practical to completely eliminate potential initiators of an accident, safety-by-design™ aims at reducing the severity of the accident's consequences and the probability of its occurrence. As a result of this systematic approach, the eight Class IV design basis events (potentially leading to most severe accidents) that are usually considered in LWRs, are reduced to only one in IRIS, with the remaining seven either completely eliminated by design, or their consequences (as well as probability) reduced to a degree that they are no longer considered Class IV events. Further discussion (with a description of specific design solutions) may be found in [II-10].

Second Tier. Elimination of the possibility for some accidents to occur enables simplifications of IRIS design and passive safety systems, resulting simultaneously in enhanced safety, reliability, as well as economics. In other words, the increased safety and improved economics support each other in the IRIS design.

Third Tier. The third tier has been addressed within the PRA/PSA (Probabilistic Risk Assessment/Probabilistic Safety Assessment) framework. In fact, PRA was initiated early in the IRIS design, and was used iteratively to guide and improve the design safety and reliability (thus adding "reliability by design"). The PRA has suggested modifications to the reactor system layout, resulting in reduction of the predicted CDF. After these modifications, the preliminary PRA level 1 analysis [II-11] estimated the CDF due to internal events (including anticipated transients without scram, ATWS) to be about 2×10^{-8}, more than one order of magnitude lower than in advanced LWRs. A subsequent evaluation [II-12] of the LERF (Large Early Release Frequency) also produced a very low value, of the order of 6×10^{-10}, which is more than one order of magnitude lower than in advanced loop LWRs, and several orders of magnitude lower than in present LWRs.

Specific features of the design that resulted from implementing the safety-by-design™ philosophy are summarized in Table II-4, while Table II-5 provides an overview of how these safety-by-design™ features will impact the typical design basis events. A detailed description of IRIS response to each of the eight Class IV events is provided in reference [II-10].

TABLE II-4. IMPLICATIONS OF SAFETY-BY-DESIGN™ IRIS PHILOSOPHY

IRIS DESIGN CHARACTERISTIC	SAFETY IMPLICATION	ACCIDENTS AFFECTED
Integral Layout	No large primary piping	- LOCA
Large, tall vessel	Increased water inventory Increased natural circulation Can accommodate internal CRDMs	- LOCA - Decrease in heat removal - Various events - RCCA ejection, eliminate head penetrations
Heat removal from inside the vessel	Depressurizes primary system by condensation and not by loss of mass Effective heat removal by SG/EHRS	- LOCA - LOCA - All events for which effective cool down is required
Reduced size, higher design-pressure containment	Reduced driving force through primary opening	- LOCA
Multiple coolant pumps	Decreased importance of single pump failure	Locked rotor, shaft seizure/break
High design-pressure steam generator system	No SG safety valves Primary system cannot over-pressure secondary system Feed/steam system piping designed for full RCS pressure reduces piping failure probability	- Steam generator tube rupture - Steam line break - Feed line break
Once-through steam generator	Limited water inventory	- Steam line break - {Feed line break}*
Integral pressurizer	Large pressurizer volume/reactor power	- Overheating events, including feed line break - ATWS

* The only accident, which is potentially affected in a negative way.

TABLE II-5. IRIS RESPONSE TO PWR CLASS IV EVENTS

Class IV design basis event		IRIS Design characteristic	Results of Safety-by-Design™ IRIS philosophy
1	Large break LOCA	Integral RV layout – no loop piping	Eliminated by design
2	Steam generator tube rupture	High design pressure once-through SGs, piping, and isolation valves	Reduced consequences, simplified mitigation
3	Steam system piping failure	High design pressure SGs, piping, and isolation valves. SGs have small water inventory	Reduced probability, reduced (limited containment effect, limited cooldown) or eliminated (no potential for return to critical power) consequences
4	Feedwater system pipe break	High design pressure SGs, piping, and isolation valves. Integral RV has large primary water heat capacity.	Reduced probability, reduced consequences (no high pressure relief from reactor coolant system)
5	Reactor coolant pump shaft break	Spool pumps have no shaft	Eliminated by design
6	Reactor coolant pump seizure	No DNB for failure of 1 out of 8 RCPs	Reduced consequences
7	Spectrum of RCCA ejection accidents	With internal CRDMs there is no ejection driving force	Eliminated by design
8	Design basis fuel handling accidents	No IRIS specific design feature	No impact

To complement its safety-by-design™, IRIS features limited and simplified passive systems as shown in Fig. II-4. They include:

- A passive emergency heat removal system (EHRS) made of four independent subsystems, each of which has a horizontal, U-tube heat exchanger connected to a separate steam generator (SG) feed/steam line. These heat exchangers are immersed in the refuelling water storage tank (RWST) located outside the containment structure. The RWST water provides the heat sink to the environment for the EHRS heat exchangers. The EHRS is sized so that a single subsystem can provide core decay heat removal in the case of a loss of secondary system heat removal capability. The EHRS operates in natural circulation, removing heat from the primary system through the steam generators heat transfer surface, condensing the steam produced in the EHRS heat exchanger, transferring the heat to the RWST water, and returning the condensate back to the SG. The EHRS provides both the main post-LOCA depressurization (depressurization without loss of mass) of the primary system and the core cooling functions. It performs these functions by condensing the steam produced by the core directly inside the reactor vessel. This minimizes the break

flow and actually reverses it for a portion of the LOCA response, while transferring the decay heat to the environment.

- Two full-system pressure emergency boration tanks (EBTs) to provide a diverse means of reactor shutdown by delivering borated water to the RV through the direct vessel injection (DVI) lines. By their operation these tanks also provide limited gravity feed makeup water to the primary system.

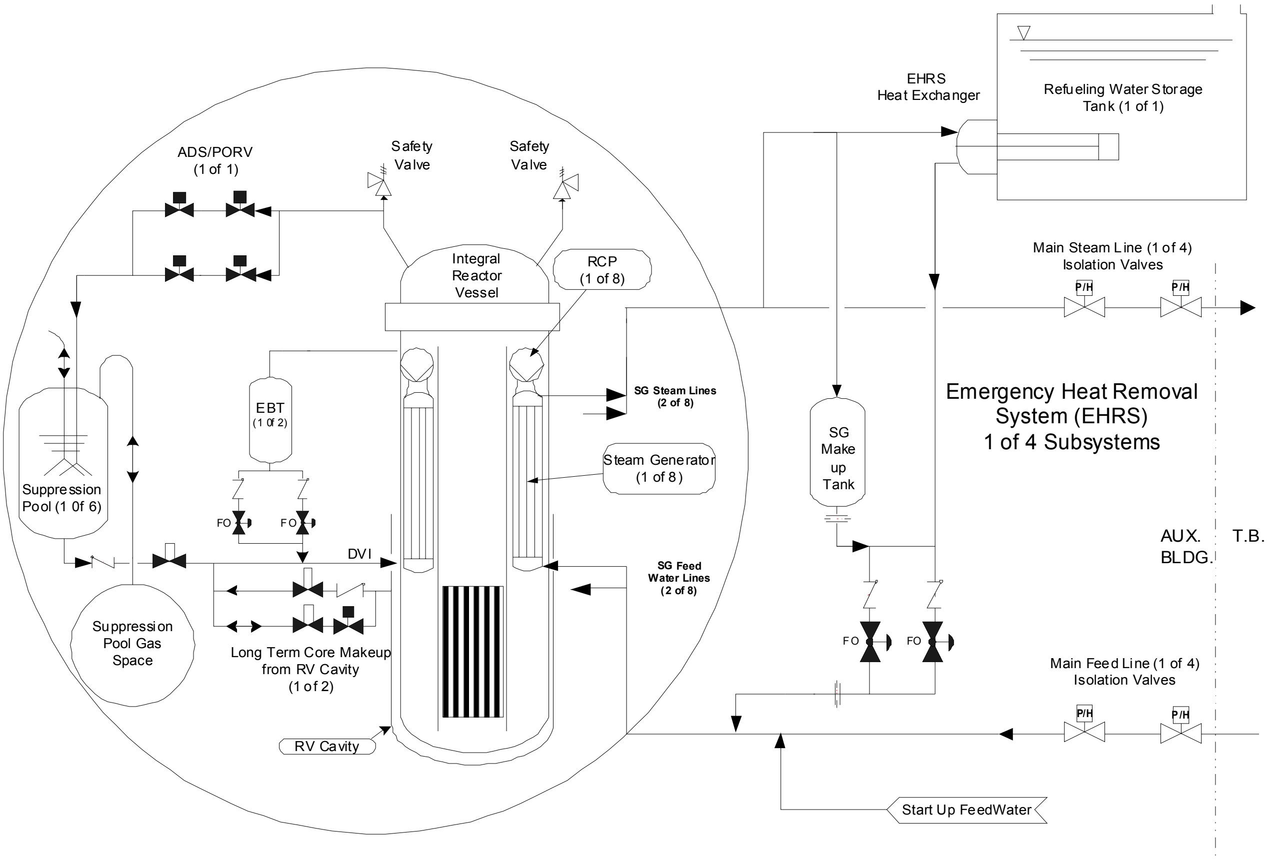

FIG. II-4. IRIS passive safety system schematic.

- A small automatic depressurization system (ADS) from the pressurizer steam space, which assists the EHRS in depressurizing the reactor vessel when/if the reactor vessel coolant inventory drops below a specific level. This ADS has one stage and consist of two parallel 4 in. lines, each with two normally closed valves. The single ADS line downstream of the closed valves discharges into the pressure suppression system pool tanks through a sparger. This ADS function ensures that the reactor vessel and containment pressures are equalized in a timely manner, limiting the loss of coolant and thus preventing core uncovering following postulated LOCAs even at low RV elevations.

- A containment pressure suppression system (PSS), which consists of six water tanks and a common tank for non-condensable gas storage. Each suppression water tank is connected to the containment atmosphere through a vent pipe connected to a submerged sparger so that steam released in the containment following a loss of coolant or steam/feed line break accident is condensed. The suppression system limits the peak containment pressure, following the most limiting blowdown event, to less than 1.0 MPa, which is much lower than the containment design pressure. The suppression system water tanks also provide an elevated source of water that is available for gravity injection into the reactor vessel through the DVI lines in the event of a LOCA.

- A specially constructed lower containment volume that collects the liquid break flow, as well as any condensate from the containment, in a cavity where the reactor vessel is located. Following a LOCA, the cavity floods above the core level, creating a gravity head of water sufficient to provide coolant makeup to the reactor vessel through the DVI lines. This cavity also assures that the lower outside portion of the RV surface is or can be wetted following postulated core damage events.

As in the AP600/AP1000, the IRIS safety system design uses gravitational forces instead of active components such as pumps, fan coolers or sprays and their supporting systems.

The safety strategy of IRIS provides a diverse means of core shutdown by makeup of borated water from the EBT in addition to the control rods; also, the EHRS provides a means of core cooling and heat removal to the environment in the event that normally available active systems are not available. In the event of a significant loss of primary-side water inventory, the primary line of defence for IRIS is represented by the large coolant inventory in the reactor vessel and the fact that EHRS operation limits the loss of mass, thus maintaining a sufficient inventory in the primary system and guaranteeing that the core will remain covered for all postulated events. The EBT is actually capable of providing some primary system injection at high pressure, but this is not necessary, since the IRIS strategy relies on “maintaining” coolant inventory, rather than “injecting” makeup water. This strategy is sufficient to ensure that the core remains covered with water for an extended period of time (days and possibly weeks). Thus, IRIS does not require and does not have the high capacity, safety grade, high-pressure safety injection system characteristic of loop reactors.

Of course, when the reactor vessel is depressurized to near containment pressure, gravity flow from the suppression system and from the flooded reactor cavity will maintain the RV coolant inventory for an unlimited period of time. However, this function would not be strictly necessary for any reasonable recovery period since the core decay heat is removed directly by condensing steam inside the pressure vessel, thus preventing any primary water from leaving the pressure vessel.

The IRIS design also includes a second means of core cooling via containment cooling, since the vessel and containment become thermodynamically coupled once a break occurs. Should cooling via the EHRS be defeated, direct cooling of the containment outer surface is provided and containment pressurization is limited to less than its design pressure. This cooling plus multiple means of providing gravity driven makeup to the core provide a means of preventing core damage and ensuring containment integrity and heat removal to the environment that is diverse from the EHRS operation.

IRIS is designed to provide in-vessel retention of core debris following severe accidents by assuring that the vessel is depressurized, and by cooling the outside vessel surface. The reactor vessel is cooled by containing the lower part of the vessel within a cavity that always will be flooded following any event that jeopardizes core cooling. Also, like in AP1000, the vessel is covered with standoff insulation that forms an annular flow path between the insulation and the vessel outer surface. Following an accident, water from the flooded cavity fills the annular space and submerges and cools the bottom head and lower sidewalls of the vessel [II-13]. A natural circulation flow path is established, with heated water and steam flowing upwards along the vessel surface, and single-phase water returning downward along the outside of the vessel insulation, to the bottom of the flood-up cavity. AP1000 testing has demonstrated that this natural circulation flow is sufficient to prevent corium melt-through. Application of AP1000 conditions to IRIS is conservative, due to the IRIS much lower core power to vessel surface ratio. The design features of the containment ensure flooding of the vessel cavity

region during accidents and submerging the reactor vessel lower head in water since the liquid effluent released through the break during a LOCA event is directed to the reactor cavity. The IRIS design also includes a provision for draining part of the water present in the PSS water tanks directly into the reactor cavity.

The superb safety characteristics of IRIS summarized in Table II-6 can conceivably lead to more "relaxed" licensing regulations with significant socio-economic advantages. In fact, the defence-in-depth provided by the safety-by-design™ as the first step, which results in the elimination or lessening of Class IV events (only one left out of the eight typically considered), combined with the orders of magnitude improvement in CDF and LERF, and risk-informed licensing could allow IRIS to attain ambitious licensing objectives, such as licensing with no requirements for off-site emergency response planning (i.e. emergency planning zone equal to the site exclusion zone).

This objective, also declared by the IAEA's international project on innovative nuclear reactors and fuel cycle (INPRO) as one of the top-level goals for advanced reactors [II-14] would have quite a significant positive impact. Economically, the utility/plant operator will not be required to plan for emergency evacuation, allowing a larger choice of sites, and avoiding the expenses of physically preparing the site and conducting planning for emergency response. The possibility of siting the plant closer to urban developments allows better implementation of co-generation (district heating, industrial steam, desalination) and reduction of transmission costs. Public acceptance will be greatly improved, because essentially IRIS is declared to be no different from other power producing plants and industrial facilities.

TABLE II-6. SAFETY FEATURES OF IRIS

CRITERION	ADVANCED PRESENT LWRS	IRIS
Defence-in-depth (DID)	Traditional barriers; emergency response	One additional safety layer BEFORE the traditional DID layers enables elimination of accidents' initiators; possibly could be substituted for the emergency response
Class IV design basis events	8 typically considered	Only 1 remains Class IV (fuel handling accident)
Core damage frequency (CDF)	$\sim 10^{-6}$—10^{-7}	$\sim 10^{-8}$
Large early release frequency (LERF)	$\sim 10^{-6}$—10^{-8}	$\sim 10^{-9}$

Licensing without the off-site emergency response requires elaboration of a new licensing framework, which the IRIS project will explore with the U.S. NRC during the currently on-going pre-application licensing process scheduled to be completed by spring 2006, as well as through a collaboration with the IAEA, as it shares this top-level objective with the INPRO goals [II-14].

II-1.6.4. Proliferation resistance

The infrequent refuelling supports proliferation resistance in IRIS. Initially, with the current PWR fuel (enriched below 5%), IRIS achieves cycle length of up to 4 years, at least doubling the interval encountered in present PWRs. The foreseen mid-term fuel and core upgrade aims to further extend the refuelling interval to 8 years [II-15].

Moreover, fuel isotopic content is well within the LEU boundary (still only ~10% for 8-year refuelling). In the MOX fuel option, IRIS uses highly degraded reactor-grade plutonium. Increased moderation achieved through the increased moderator-to-fuel volume softens the spectrum, ensuring that fuel isotopic evolution promotes proliferation resistance. In fact, IRIS is a good plutonium burner, better than advanced LWRs, and may be used for plutonium disposition, if desired. The long cycle option facilitates the nuclear material inspection, accounting and verification.

II-1.6.5. Technical features and technological approaches used to facilitate physical protection of IRIS

IRIS has a compact, spherical containment with a 25 m diameter. Only half of the containment vessel is located above the ground, thus leaving exposed only a relatively small area (compared to large present LWRs) that may be additionally protected, if desired, fairly easily and at an acceptable cost.

The safety-by design™ inherently improves IRIS response to accidents, whether due to technical causes or external impacts, thus facilitating its physical protection.

II-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of IRIS

IRIS is ideally positioned for deployment worldwide [II-16] through the resources of the IRIS consortium that includes members from 10 countries. All consortium members are equal partners, and provide inputs regarding specific market requirements, that are ultimately reflected in the IRIS design. About half of the countries represented in the consortium are developing countries; therefore, IRIS is well aware of and responsive to specific needs of the developing countries. Issues related to specific needs, such as co-generation with desalination or district heating, are addressed by member countries having first-hand experience in those areas, i.e. Brazil and Mexico for desalination, and Lithuania for district heating. Also, regional collaboration (e.g., in Central Europe and Latin America) may strongly promote prospects for IRIS deployment.

Option for full scope of fuel cycle services is one possibility that may be effectively implemented within the international IRIS consortium, certainly easier than by any single organization, but its actual realization would first require political agreements of all involved entities and the international community. This is an area where active involvement of the IAEA would be especially beneficial.

II-1.8. List of enabling technologies relevant to IRIS and status of their development

No major technology development is required for IRIS, in particular for the initial near-term deployment. However, certain components or systems designs need to be developed, improved, or qualified, including:

- Improved instrumentation is needed for the integral reactor vessel layout, where all primary components are located within the vessel. This work is in progress, and no showstoppers are expected.
- Advanced diagnostics and prognostics are needed due to the integral configuration and to support the 4-year maintenance cycle [II-17, II-18] and the 4-year refuelling cycle. This work is in progress, and no showstoppers have been identified.
- Development and qualification of internal control rod drive mechanisms (CRDMs) is required [II-5]. Significant progress has been achieved both within the consortium and outside (e.g. several Japanese organizations), and it is expected that this development will be completed in time for the first IRIS deployment. However, if this CRDM development is for some reason delayed, there is a fall back position, employing standard external CRDMs.

The mid-term deployment (in the 2020s) where the refuelling interval would be increased from 4 to 8 years, requires irradiation testing, qualification and licensing of fuel that would support the corresponding fuel residence time and discharge burn-up. This will not be an isolated IRIS effort, but is related to similar activities by many organizations worldwide.

II-1.9. Status of R&D and planned schedule

The IRIS development was initiated with U.S. DOE support through the Nuclear Energy Research Initiative (NERI). Currently, IRIS is identified as a member of the International Near-Term Deployment (INTD) group, within the Generation-IV International Forum (GIF).

Institutions involved in IRIS R&D are members of the IRIS consortium (listed in Table II-1) and include: Westinghouse Electric Co. (USA), BNFL (UK), Ansaldo Energia (Italy), Ansaldo Camozzi (Italy), ENSA (Spain), NUCLEP (Brazil), Bechtel (USA), OKBM (Russia), ORNL (USA), CNEN (Brazil), ININ (Mexico), LEI (Lithuania), Polytechnic of Milan (Italy), MIT (USA), Tokyo Institute of Technology (Japan), University of Zagreb (Croatia), University of Pisa (Italy), Polytechnic of Turin (Italy), University of Rome (Italy), TVA (USA), Eletronuclear (Brazil).

It is expected that the FOAK IRIS will be deployed in the 2012–2015 timeframe [II-16]. A more detailed schedule with development and licensing stages is given in Table II-7. It is worth mentioning that so far all the milestones were reached on or ahead of schedule.

TABLE II-7. IRIS PROJECT SCHEDULE AND MILESTONES

MILESTONE	TARGET DATE
Assess key technical and economic feasibility (completed)	End 2000
Perform conceptual design (completed)	End 2001
Preliminary cost estimate (completed)	End 2001
Initiate licensing pre-application (completed)	Fall 2002
Develop licensing plan (completed)	Fall 2002
Outline path to commercialization (completed)	Early 2003
Perform preliminary design	2002–2005
Complete licensing pre-application	Spring 2006
Obtain final design approval	2010
First-of-a-kind deployment	2012–2015

II-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

IRIS introduces significant innovative features and the integral design represents a radical change in system configuration from existing loop reactors that justify its classification as an innovative SMR. However, while it is innovative engineering, it relies on proven LWR technology and thus it only requires engineering and confirmatory testing. A prototype or demonstration plant is not required for obtaining final design approval and design certification, a project position which has not been objected by NRC.

II-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

Several other organizations are considering SMRs of integral type that share some design features with IRIS, within the IPSR group (Integral Primary System Reactors) of the INTD/GIF.

IRIS shares with INPRO [II-14] the goal of revising the emergency planning requirements.

II-2. Design description and data for IRIS

II-2.1. Description of the nuclear systems

The IRIS core and fuel assemblies (Fig. II-5 and II-6; Table II-8) are similar to those of a loop type Westinghouse PWR design. Specifically, the IRIS fuel assembly design is similar to the Westinghouse 17×17 XL Robust Fuel Assembly design and AP1000 fuel assembly design. An IRIS fuel assembly consists of 264 fuel rods with a 0.374 inches outer diameter in a 17×17 square array. The central position is reserved for in-core instrumentation, and 24 positions have guide thimbles for the control rodlets. Low-power density is achieved by employing a core configuration consisting of 89 fuel assemblies with a 14-ft (4.267 m) active fuel height, and a nominal thermal power of 1000 MW. The resulting average linear power density is about 75% of the AP600 value. The improved thermal margin provides increased operational flexibility, while enabling longer fuel cycles and thus increased overall plant capacity factors.

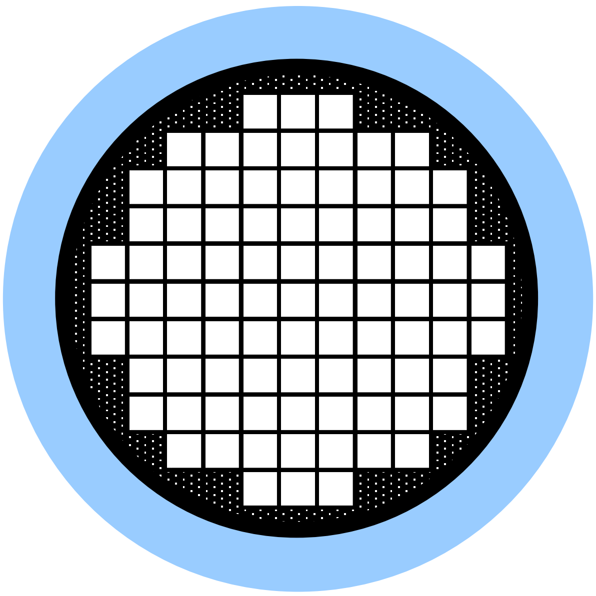

FIG. II-5. IRIS core configuration.

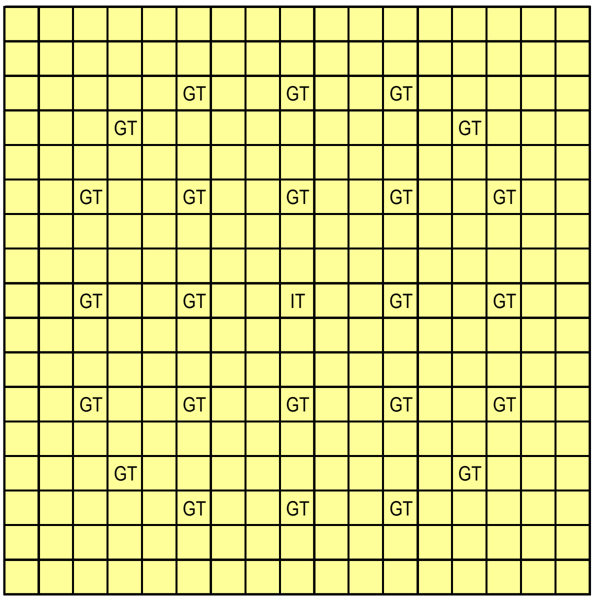

FIG. II-6. IRIS fuel assembly, 17×17 lattice.

TABLE II-8. IRIS CORE MAIN PARAMETERS

Equivalent diameter	2.41 m
Active core height	4.267 m
Fuel inventory	48.5 t U
Average linear heat rate	10.0 kW/m
Number of fuel assemblies	89
Number of fuel rods/assembly	264
Outer diameter of fuel rods	9.5 mm

The initial, reference IRIS core [II-19] will use UO_2 fuel, enriched to 4.95 w/o in ^{235}U, with lower enrichment in the axial blankets and at the core periphery. The fission gas plenum length is increased (roughly doubled) compared to current PWRs, thus eliminating potential concerns with internal overpressure. The integral RV design permits this increase in the gas plenum length with practically no penalty, because the steam generators mainly determine the vessel height.

Reactivity control is accomplished through solid burnable absorbers, control rods, and the use of a limited amount of soluble boron in the reactor coolant. The reduced use of soluble boron makes the moderator temperature coefficient more negative, thus increasing inherent safety. The initial core is designed for a three- to three-and-half-year cycle with half-core reload to optimize the overall fuel economics while maximizing the discharge burn-up. In addition, a four-year straight burn fuel cycle can also be implemented to improve the overall plant availability, but at the expense of a somewhat reduced discharge burn-up.

Also, IRIS core designs capable of 8-year refuelling period have been developed using UO_2 or MOX fuel with fissile content increased to 7–10% [II-15]. This is facilitated by the “variable moderation approach”, whereas the moderator-to-fuel ratio is increased with the increased fissile content, to achieve adequate neutron thermalization.

Main parameters of the IRIS steam generators and reactor coolant pumps are given in Table II-9.

TABLE II-9. IRIS STEAM GENERATORS AND COOLANT PUMPS MAIN PARAMETERS

STEAM GENERATOR	
Type	Once-through with superheated steam
Tubes	Helical coil tube bundle, primary outside the tubes
Number	8
Thermal capacity (each SG)	125 MW
REACTOR COOLANT PUMPS	
Type	Axial (propeller) type pumps, fully immersed
Number	8
Pump head	19.8 m

Description of the Emergency Heat Removal System (EHRS) is provided in section II-1.6.3. A simplified scheme of heat removal paths in normal operation and accident conditions is given in Fig. II-7.

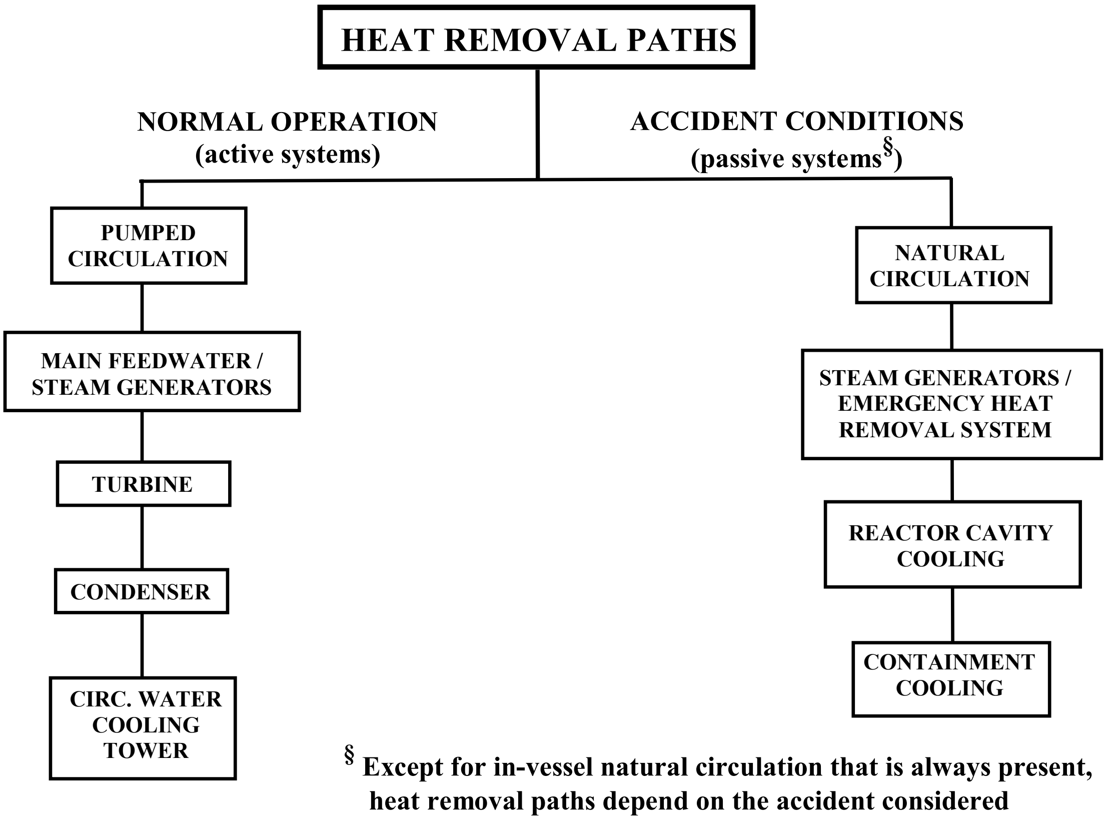

FIG. II-7. Simplified heat removal paths (details omitted because of proprietary considerations).

II-2.2. Description of the turbine generator plant and systems

Use of conventional equipment is planned in the turbine generator plant and systems.

II-2.3. Systems for non-electric applications

Desalination, district heating and industrial steam co-generation designs are being developed at this time [II-20]. Detailed description will be included in a future edition of this document.

II-2.4. Plant layout

A nuclear installation with IRIS may employ either a single IRIS module, or multiple single modules, or multiple twin modules, thus providing great flexibility regarding the range of the generating capacity (from 335 MW(e) to several thousand MW(e)).

Based on interest in IRIS expressed by several electric power utilities, IRIS has developed two alternative site layouts with corresponding site requirements. The first option shown in Fig. II-8 presents a multiple single-unit site layout, while the second option shown in Fig. II-9 presents a multiple twin-unit site layout.

In the first option, shared systems and structures are minimized. Units are constructed in a “slide-along” manner, with the first unit put into operation while subsequent units are under construction. Such arrangement minimizes construction time and provides generating capacity (and revenue) as soon as possible, thus minimizing the required financial resources. It also maximizes workforce efficiency and significantly shortens construction time of subsequent units.

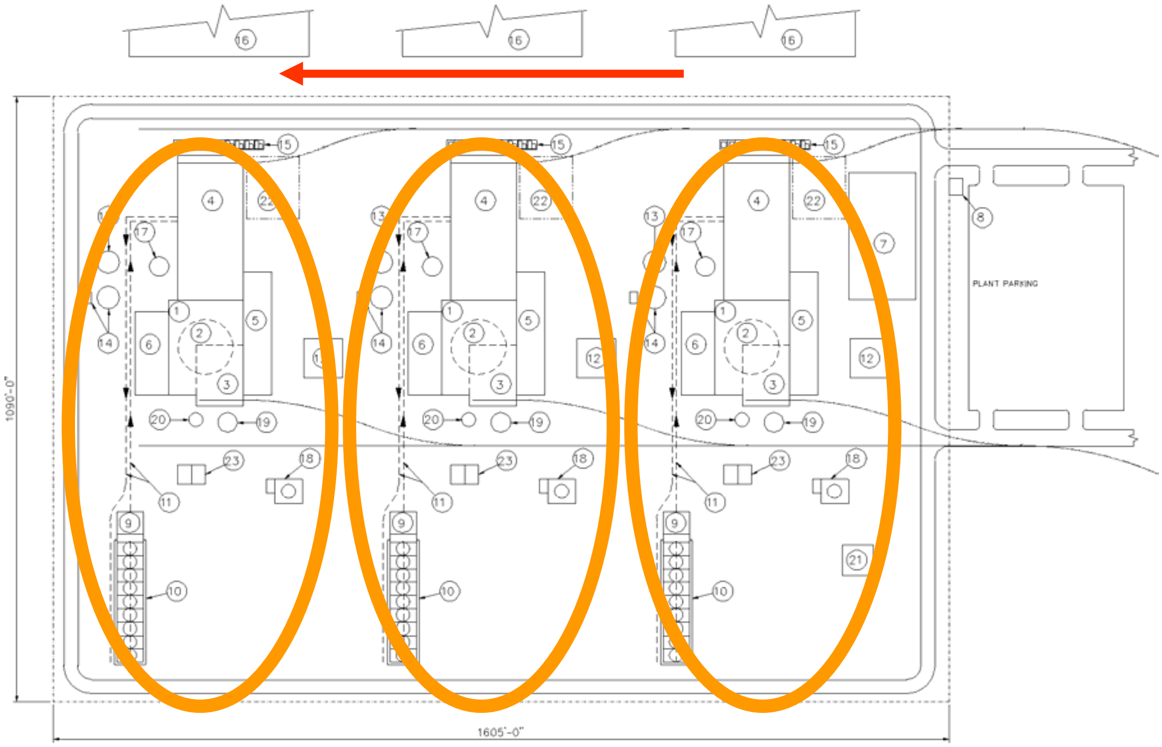

FIG. II-8. IRIS multiple single-unit site layout.

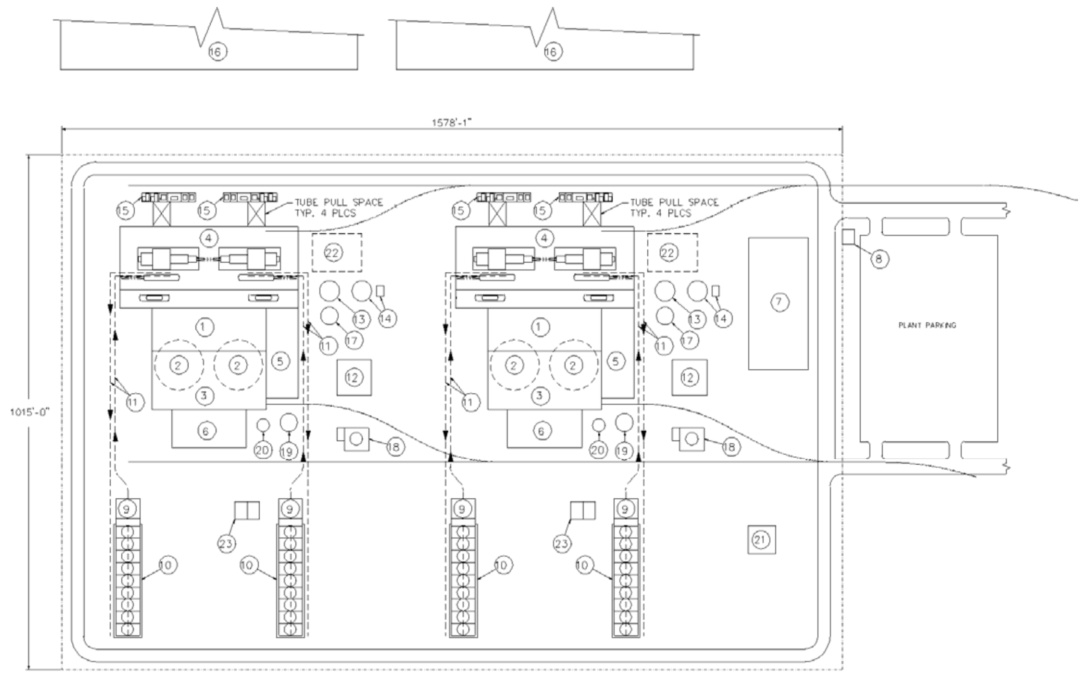

FIG. II-9. IRIS multiple twin-unit site layout.

In the second option, shared systems and structures (including fuel handling and spent fuel pool, support systems in auxiliary building) are maximized. Twin-units share control rooms, but have separate safety and protection systems. Twin-units are also constructed in "slide-along" manner, with the same advantage as for the first option, plus maximization of

shared equipment and workforce, but they require adding generating capacity in 670 MW(e) increments. A perspective view of the site arrangement with two twin units is shown in Figure II-10.

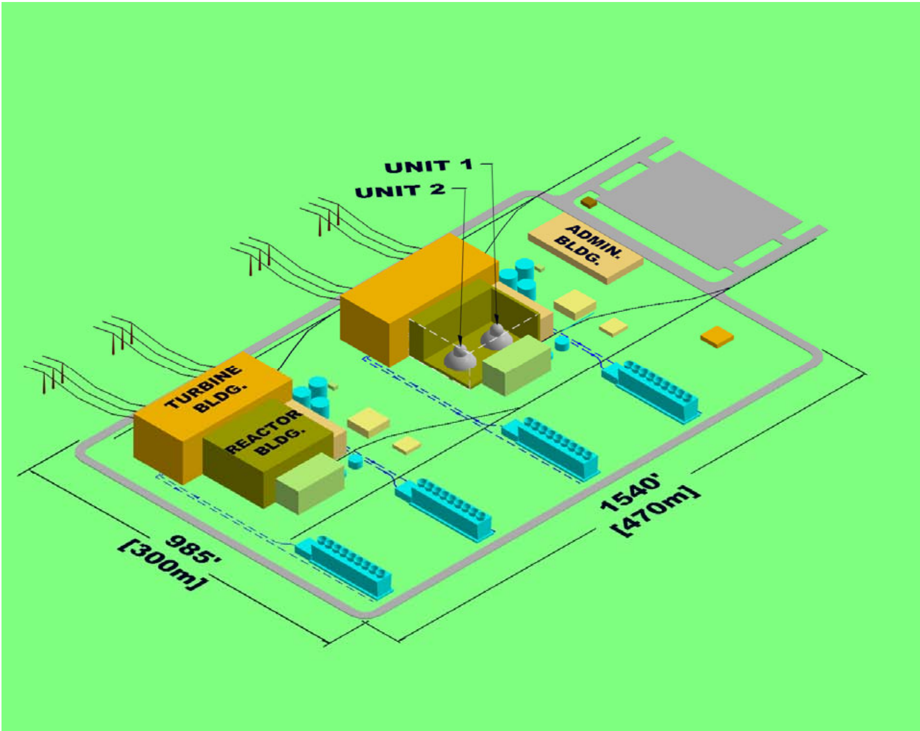

FIG. II-10. Perspective view of IRIS multiple twin-unit site layout.

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CENTRAL ARGENTINA DE ELEMENTOS MODULARES (CAREM)

CNEA, Argentina

III-1. General information, technical features and operating characteristics

III-1.1. Introduction

CAREM (Central ARgentina de Elementos Modulares) is an Argentine project to develop, design and construct an innovative, simple and small nuclear power plant (NPP). This plant has an indirect cycle reactor with distinctive and characteristic features that greatly simplify the design and contribute to a high safety level. Some of the high-level design characteristics of the plant are: an integrated primary cooling system; self-pressurized primary system and safety systems relying on passive features. CAREM is a CNEA (Comisión Nacional de Energía Atómica) project, which has been jointly developed with INVAP, an Argentine company.

The CAREM concept was first presented in March 1984, in Lima, Peru, during the IAEA's conference on small and medium sized reactors. Chronologically CAREM was one of the first of the present new generation of reactor designs. The first step of this project is the construction of the prototype of about 27 MW(e) (CAREM-25). This project allows Argentina to sustain activities in nuclear power plant design, assuring the availability of updated technology in the mid-term [III-1]. The design basis is supported by the cumulative experience acquired in research reactor design, construction and operation and pressurized heavy water reactor (PHWR) operation, as well as development of advanced design solutions [III-2].

The Generation IV International Forum (GIF) [III-3] recognized CAREM as an international near term deployment (INTD) reactor.

III-1.2. Applications

The CAREM concept is designed for competitive electric power production with low power modules.

The CAREM power plant has a potential use for non-electric applications such as nuclear desalination. In this application, this plant could be used either as a heat source or electrical energy supply. Several studies were performed to analyze the potential use of CAREM as the energy supply for seawater desalination plants [III-4] and it appears to be an appropriate option. This reactor has many inherent safety features: no large loss-of-coolant accidents (LOCA) and long characteristic times in the event of transient or severe accidents, due to the large coolant inventory and use of passive safety systems. These factors plus simple operation make CAREM a realistic option in countries with limited nuclear development. For economic reasons, it is convenient to locate the reactor and desalination plant near cities. However, this is only possible with a high level of nuclear safety. On the other hand, to minimize specialized labour on the construction site, complex operations associated with reactor assembly should be reduced. In this field, CAREM has advantages over traditional designs in quality control, construction schedules and costs (less difficult welding on the construction site, off-site assembly of systems, etc.) [III-5].

III-1.3. Special features

The power range of CAREM modules allows sequencing of additional capacity to more closely match the demand.

The use of passive safety systems facilitates deployment as an autonomous system, like floating NPP.

The CAREM nuclear power plant is small and has such advantages as a potential for design standardization, series production and shop fabrication of equipment.

III-1.4. Summary of major design and operating characteristics

Mode of operation

CAREM is designed to be operated in base load, but it also has some load-follow capability.

Load factor/ Availability

CAREM is designed to have an availability factor of 90% or higher.

Some major design characteristics of CAREM are given in Table III-1.

TABLE III-1. SUMMARY OF MAJOR DESIGN CHARACTERISTICS

CHARACTERISTIC	DESIGN PARTICULARS
Type of fuel	PWR type fuel assembly with low enriched UO_2
Fuel enrichment	About 3.5%
Moderator	Light water
Coolant	Light water
Structural materials	Barrel: SS-304L Core grids and envelope: SS-304 Steam generator shell: SS-304L Steam generator tubes: Inconel 690 (SB 163 N06690)
Core	Fuel assemblies of hexagonal cross section. Each fuel assembly contains 108 fuel rods of 9 mm outer diameter, 18 guide thimbles and 1 instrumentation thimble. The core of CAREM-300 has 199 fuel assemblies having about 2.85 m active length. The core of CAREM-25 has 61 fuel assemblies having about 1.40 m active length.
Reactor vessel	Vessel material: SA508 Grade 3 Class 1 Lining material: SS-304L For CAREM-25 vessel the main dimensions are Height: 11 m Inner diameter: 3.16 m Wall thickness: 0.135 m

Installed capacity

Different CAREM concept power modules are available; the higher power module, CAREM-300, is designed to produce 900 MW(th), generating 300 MW(e).

For power modules below 150 MW(e), the flow rate in the primary reactor system is achieved by natural circulation; a 100 MW(th) and 27 MW(e) prototype will be constructed for demonstration purposes.

Simplified schematic diagram of CAREM plant

The CAREM nuclear power plant design is based on a light water integrated reactor. The whole primary system (core, steam generators, primary coolant and steam dome) is contained in a single pressure vessel, Fig. III-1.

In Figure III-2, a simplified schematic diagram of CAREM reactor is shown.

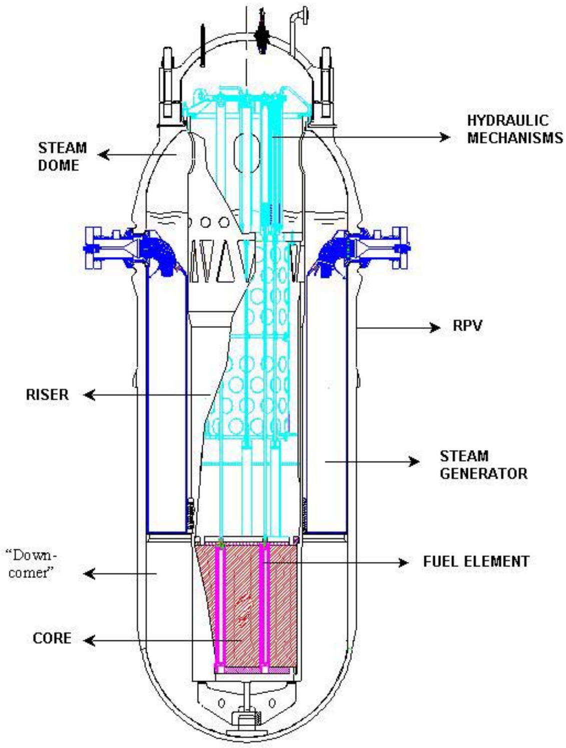


FIG. III-1. Reactor prototype pressure vessel.

The main neutron-physical characteristics of CAREM-25 are given in Table III-2.

TABLE III-2. NEUTRON-PHYSICAL CHARACTERISTICS OF CAREM

CHARACTERISTIC	VALUE
Fuel temperature reactivity coefficient	< -2.1 pcm/°C
Coolant temperature reactivity coefficient	< -40 pcm/°C in normal operation < -4 pcm/°C in cold shutdown
Coolant void coefficient	< -147 pcm/% in normal operation < -43 pcm/% in cold shutdown
Burn-up reactivity swing	3600 pcm
Maximum Peaking Factor	2.7

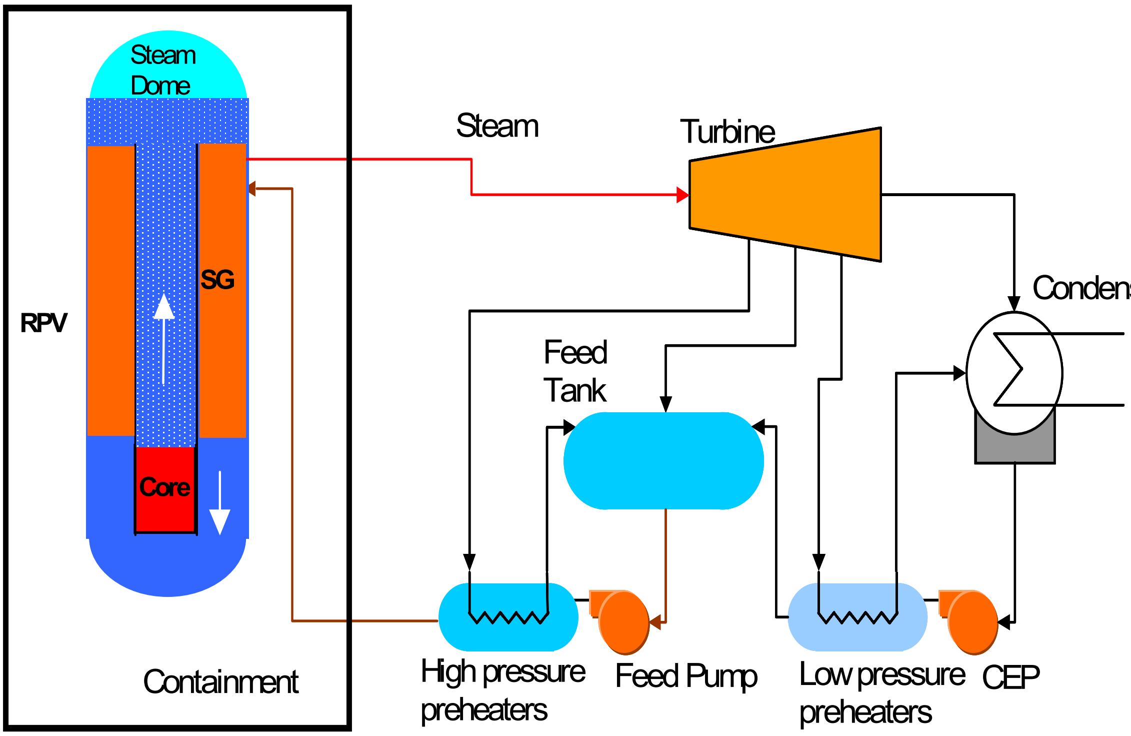


FIG. III-2. Simplified schematic diagram of CAREM NPP.

Reactivity control mechanism

The fuel is enriched UO_2. Core reactivity is controlled by the use of Gd_2O_3 as burnable poison in specific fuel rods and movable absorbing elements belonging to the adjust and control system. Liquid chemical compounds are not used for reactivity control during normal operation.

Each absorbing element (AE) consists of a cluster of rods linked by a structural element (namely, “spider”), so the cluster moves as a single unit. Absorber rods fit into the guide tubes. The absorber material is the commonly used Ag-In-Cd alloy. Absorbing elements (AE) are used for reactivity control during normal operation (adjust and control system) and to produce a sudden interruption of the nuclear chain reaction when required (fast shutdown system).

The shutdown system is diversified to fulfil Argentine regulatory-body requirements.

The first shutdown system (FSS): this system consists of gravity driven neutron-absorbing elements. In CAREM-25, this system provides a total negative reactivity at cold shutdown of 6880 pcm, with all rods inserted.

Many of the elements are for the fast shutdown system; during normal operation they are kept in the upper position. They are designed to obtain a minimal dropping time, so that it takes only a few seconds to completely insert absorbing rods inside the core. In CAREM-25, this system has a minimum worth of 3500 pcm, with one rod unavailable.

The second shutdown system (SSS) is a gravity-driven injection device of borated water at high pressure. In CAREM-25, this system provides a total negative reactivity at cold shutdown of 5980 pcm, assuming single failure.

Cycle type and thermodynamic efficiency

CAREM is an indirect cycle reactor with a standard steam cycle of simple design.

A thermodynamic efficiency of 33% is estimated for CAREM-300.

The main thermal-hydraulic characteristics of CAREM are given in Table III-3.

TABLE III- 3. THERMAL-HYDRAULIC CHARACTERISTICS OF CAREM

PARAMETER	CHARACTERISTICS
Primary system configuration	Integrated
Circulation type	Natural circulation for normal operation as well as hot shutdown for low power modules (below 150 MWe). Forced circulation for full power operation and natural circulation for hot shutdown for high power modules (over 150 MW(e)).
Coolant conditions	Self-pressurization of the primary system in the steam dome is the result of the liquid-vapour equilibrium. Due to self-pressurization, bulk temperature at core outlet corresponds to saturation temperature at primary pressure. For CAREM-25 the core outlet temperature is 326°C and the core inlet temperature is 284°C.
Primary flow rate and pressure	For CAREM-25 the coolant mass flow through the core is 410 kg/s and the operating coolant pressure is 12.25 MPa.
Steam conditions	For CAREM-25 the steam mass flow to the turbine is 175.32 t/hr and the steam pressure and temperature are 4.7 MPa and 290°C.
Fuel temperature	For CAREM-25 the maximum fuel centre line temperature is 950°C.
MDNBR	For CAREM-25 is larger than 1.7.

For CAREM-300, the average discharge burn-up is about 35 000 MW·d/Mt U and the maximum discharge burn-up is about 45 000 MW·d/Mt U.

The fuel cycle can be tailored to customer requirements, with a reference design of 330 full-power days. In CAREM-300, 1/3 of the core is refuelled and 1/2 is refuelled in CAREM-25.

CAREM-300 has about 200 tones of natural U (feed)/GW(e) per year based on a 3.5% initial enrichment, 35,000 MW·d/Mt U of average discharge burn-up, 33% thermodynamic efficiency, 0.25% of enrichment tail and 90% of load factor.

The design life for the pressure vessel is 40 years. Steam generators and reactor internals are easily replaceable during shutdown.

The CAREM power plant has a potential use for non-electric applications such as nuclear desalination. This plant can be used either as a heat source or electrical energy supply for a Reverse-Osmosis-based desalination plant.

Economics

The CAREM concept economy was analyzed using IREP code [III-6, III-7]. Below 150 MW(e), the natural convection option is preferred because the estimated costs are similar and the present version of IREP results more representative for this configuration. Over that power level, the size and cost of the RPV are outside the acceptable range so the forced convection option is preferable [III-8]. The capital cost for a natural convection module of 125 MW(e) is US $1700/kWe and for CAREM-300 is US $1050/kWe. The estimated O&M + fuel costs for CAREM-300 are US $12.5/MW(e)-hr.

The estimated construction period is four years.

The construction cost of the prototype of about 27 MW(e), CAREM-25, is US $85 million.

III-1.5. Outline of fuel cycle options

CAREM is designed for a once-through (OT) fuel cycle. For this standard fuel cycle, a deep geological repository for final disposal of high level waste after surface intermediate storage in horizontal natural convection silos is considered [III-9].

CAREM can be adapted to use MOX fuel as an alternative fuel cycle option. In this case, advanced PUREX methods could be used to obtain higher purity levels, to avoid fissile material contamination and to have more time for material utilization.

III-1.6. Technical features and technological approaches that are definitive for CAREM performance in particular areas

III-1.6.1. Economics and maintainability

There is a significant need for small nuclear power plants suitable for developing countries and for the utilities of small or medium-developed countries. The CAREM concept is conceived to offer nuclear options to this market, with a US $1000 /KW h overnight cost for a 300 MW(e) nuclear power plant.

Technical and economic advantages are achieved with the CAREM design compared to traditional design:

- To simplify the design, the whole high-energy primary system (core, steam generators, primary coolant and steam dome) is contained in a single pressure vessel. This reduces considerably the number of pressure vessels needed and simplifies the layout.
- Due to the absence of large diameter piping associated with the primary system, no large LOCA needs to be handled by the safety systems. The elimination of large LOCA considerably reduces the needs for ECCS components, AC supply systems, etc.
- Due to self-pressurization, the elimination of an active pressurizer (heaters and sprinklers) results in lower costs and advantages in maintenance and availability.
- Eliminating primary pumps in low power modules results in lower costs, added safety, and advantages in maintenance and availability.
- The development of an innovative hydraulic mechanism located completely inside the reactor pressure vessel eliminates the possibility of control rod ejection accidents. Furthermore, the hydraulic control rod drive mechanism has a significantly lower cost compared with current PWR control rod drive mechanisms.

- A large coolant inventory in the primary circuit results in greater thermal inertia and longer response time in case of transients or accidents.
- The large water volume between the core and the wall leads to a very low fast-neutron dose over the RPV wall.
- The design and fabrication is based on modularity. Reactor modules can be fabricated in a factory and readily transported to the site, reducing expensive on-site assembling/welding and ultimately, construction time.
- Shielding requirements are reduced by the elimination of gamma sources from dispersed primary piping and parts.
- The ergonomic design and layout make maintenance easier. Maintenance, like steam generator tube inspection, does not compete with refuelling because it is conducted from outside the vessel.
- The use of less active components increases both plant availability and load factor.
- The reduction of staff and maintenance reduce the cost of power generation.

III-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

CAREM design is within the boundaries of current requirements in relevant areas. However, the use of borated water as a long-term reactivity control is eliminated, resulting in less waste.

III-1.6.3. Safety and reliability

Safety requirements and design philosophy

Since genesis of the design, emphasis has been on the prevention of core degradation accidents by using passive safety features, thus obviating the necessity for active systems or operator actions for a period of several days. By design, a proper balance is assured to avoid jeopardizing economic competitiveness of the reactor.

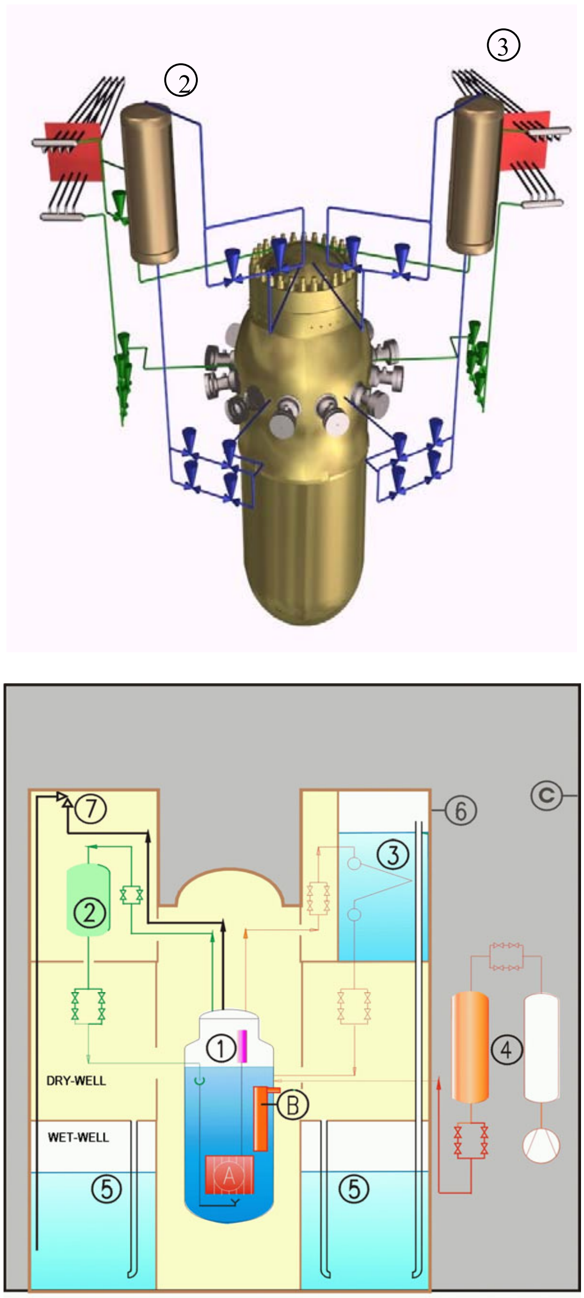
The CAREM project uses an innovative method for reactor design by balancing safety and economics at the conceptual engineering stage. The key is to consider safety aspects in design optimization, where the variables are balanced to obtain a better *figure of merit* relative to economic performance of the reactor. The design-parameter effect on characteristic or critical safety variables, chosen from known reactor behaviour during accidents (*safety performance indicators*), is synthesised in *Design Maps*. These maps allow the comparison of observations with limits determined by design criteria or regulations and transfer of those limits to the design variables or parameters. Therefore, dynamic responses of the reactor during transients or accidents and safety aspects are integrated using additional rules to those necessary for steady state dimensioning, to the neutronic, thermal-hydraulic and mechanical areas, in the conceptual engineering stage of the design. With this method, the design can be simplified at an early stage, avoiding complexities resulting when these concepts are introduced in a later engineering stage or as a “patch”.

This method allows balance and optimization of the reactor and the safety system at an early engineering stage, to cost-efficiently internalize safety issues based on the defence in depth approach and considering appropriate conservative assumptions and safety margins. This achieves a balance between inherent capability of the reactor and the safety systems to cope with the postulated initiating events.

Finally, this balanced design avoids the problem where the search for economic performance causes less safe reactors and assures design competitiveness in spite of unavoidable safety costs.

Active and passive systems and inherent safety features

CAREM safety systems are based on passive features obviating the need for actions to mitigate accidents during a long period (Fig. III-3). They are duplicated to fulfil redundancy criteria. The shutdown system should be diversified to fulfil Argentine regulations.



1: First shutdown system
2: Second shutdown system
3: Residual heat removal system
4: Emergency injection system
5: Pressure suppression pool
6: Containment
7: Safety valves
A: Core B: Steam generators C: Reactor building

FIG. III-3. Containment and safety systems.

The first shutdown system (FSS) is designed to shut down the core when an abnormality or a deviation from normal operation occurs and to maintain the core as subcritical during all shutdown states. This is achieved by dropping neutron-absorbing elements into the core through gravity. Each neutron-absorbing element is a cluster composed of a maximum of 18 individual rods together in a single unit. Each unit fits well into the guide tubes of each fuel assembly.

The hydraulic control rod drive (CRD) avoids passing mechanical shafts through RPV or the extension of the primary pressure boundary and since the whole device is located inside the RPV, eliminates any possibility of large loss of coolant accidents (LOCA). This design is an important element in the CAREM concept [III-10]. Many CRD (simplified operating diagrams are shown in Fig. III-4) are in the fast shutdown system. During normal operation, they are kept in the upper position, where the piston partially closes the outlet orifice and reduces water flow to a leakage into the RPV dome. The CRD of the adjust and control system is a hinged device controlled in steps and fixed in position by pulses over a base flow, designed so that each pulse produces only one step.

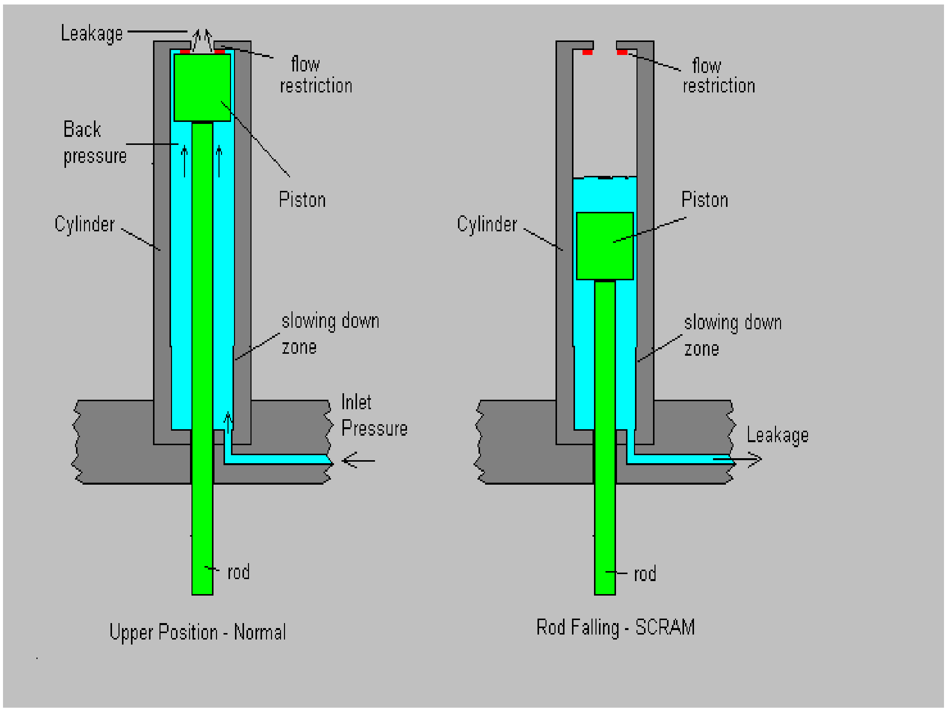


FIG. III- 4. Simplified operating diagram of a hydraulic control rod drive (Fast shutdown system).

Both types of devices perform the SCRAM function by the same principle: “rod drops by gravity when flow is interrupted”, so malfunction of any powered part of the hydraulic circuit (i.e., valve or pump failures) causes immediate shutdown of the reactor. CRD of the fast shutdown system is designed with a large gap between piston and cylinder to obtain a minimum dropping time of a few seconds to insert absorbing rods completely in the core. The CRD manufacturing and assembling allowances are stricter and clearances are narrower for the adjust and control system, but there is no stringent requirement on dropping time.

The second shutdown system (SSS) is a gravity-driven injection device of borated water at high pressure. It acts automatically when the reactor protection system detects failure of the FSS or in case of LOCA. This system consists of two tanks located in the upper part of the containment. Each of them is connected to the reactor vessel by two piping lines; one is from the steam dome to the upper part of the tank and the other is from a position below the reactor water level to the lower part of the tank. When the system is triggered, the valves open automatically and the borated water drains into the primary system by gravity. The discharge of a single tank produces the complete shutdown of the reactor.

The r*esidual heat removal system (RHRS)* has been designed to reduce pressure on the primary system and to remove the decay heat in case of loss of heat sink. It is a simple and reliable system that operates by condensing steam from the primary system in emergency condensers. The emergency condensers are heat exchangers consisting of an arrangement of parallel, horizontal U-tubes between two common headers. The top header is connected to the reactor vessel steam dome, while the lower header is connected to the reactor vessel at a position below the reactor water level. These condensers are located in a pool filled with cold water inside the containment building. The inlet valves in the steam line are always open, while the outlet valves are normally closed and therefore, the tube bundles are filled with condensate. When the system is triggered, the outlet valves open automatically; water drains from the tubes and steam from the primary system enters the tube bundles and condenses on the cold surface of the tubes. The condensate is returned to the reactor vessel forming a natural circulation circuit. In this way, heat is removed from the reactor coolant. During the condensation process the heat is transferred to the water of the pool by a boiling process. This evaporated water is then condensed in the suppression pool of the containment.

The e*mergency injection system* prevents core exposure in case of LOCA. The system consists of two redundant accumulators with borated water connected to the RPV. The tanks are pressurized, so that during a LOCA, when the pressure in the reactor vessel becomes relatively low, rupture disks break and the flooding of the RPV starts, preventing uncovering of the core for a long period. The RHRS is also triggered to help depressurize the primary system when the area of breakage is small.

Three *safety relief valves* protect the reactor pressure vessel against over-pressurization in case of strong differences between the core power and the power removed from the RPV. Each valve is capable of 100% of the necessary relief. The blow-down pipes from the safety valves are routed to the suppression pool.

The primary system, the reactor coolant pressure boundary, the safety systems and the high-pressure components of the reactor auxiliary systems are enclosed in the primary containment, a cylindrical concrete structure with an embedded steel liner. The primary containment is a pressure-suppression type with two major compartments: a drywell and wet well. The drywell includes the volume that surrounds the reactor pressure vessel and the second shutdown system rooms. A partition-floor and cylindrical wall separate the drywell from the wet well. The lower part of the wet well is filled with water that acts as the condensation pool and the upper part is a gas compression chamber.

A summary of functions to cover and the available safety systems is shown in Table III-4.

TABLE III-4. SAFETY FUNCTIONS AND SAFETY SYSTEMS

SAFETY FUNCTION	SAFETY SYSTEM
Reactivity control	First shutdown system: Safety control rods
	Second shutdown system: Boron injection
Primary pressure limitation	Safety relief valves
	Residual heat removal system
Primary depressurization	Residual heat removal system
Primary water injection	High pressure: Second shutdown system
	Low pressure: Emergency injection system
Secondary pressure limitation	Relief valves
Residual heat removal	Residual heat removal system

Structure of the defence-in-depth

The first line of defence-in-depth in the CAREM approach is elimination of initiators that could lead to core damage.

Level 1: Prevention of abnormal operation and failures:

- Due to the absence of large diameter piping in the primary system, large LOCAs are eliminated.
- Natural circulation core cooling in low-power modules eliminates loss-of-flow accidents.
- Innovative hydraulic mechanisms completely inside the reactor pressure vessel eliminate control rod ejection accidents.

Level 2: Control of abnormal operation and detection of failures:

- Improvements in the reliability of control and protection systems with "real time" computerized systems of distributed design.
- A large coolant inventory in the primary circuit results in greater thermal inertia and longer response time in case of transients or accidents.

Level 3: Control of accidents beyond the design basis:

- The reactor protection system was designed with the most advanced technology for nuclear power plants, the defence-in-depth principle and early failure detection, with the objective of avoiding the occurrence of accidents beyond the design basis.
- The reactor protection system has two independent subsystems: the first, responsible for generation of the first shutdown system trip signal, consists of a combination of hard logic and digital processing modules. The second subsystem, responsible for the generation of the second shutdown system trip signal, is based on hard logic technology to fulfil the principle of diversity for the first and second shutdown systems.
- The reactor protection system has four independent and redundant channels with voting and protective logic of a dynamic type. This results in high reliability and availability.
- The CAREM safety systems are based on passive features and obviate the necessity for actions to mitigate accidents during an extended period.

Level 4: Control of severe plant conditions:

- The CAREM reactor prototype core melt frequency is 9.2×10^{-7}/year. This value is based on a level III PSA of CAREM-25 NPP and an expert judgement of changes introduced in the design to increase safety.

Level 5: Mitigation of radiological consequences of significant release of radioactive materials.

The following features contribute to mitigation of the severe consequences of accidents:

- Provisions for injection of water into the reactor cavity from the refuelling water storage tank to cool the RPV from the outside and enhance cooling of the core debris, taking advantage of the high relationship between RPV lower bottom head area to core mass, characteristic of integral type reactors.
- When core exposure is supposed, only for analytic proposes, low fuel elements heat-up rates in the exposed part are predicted when the geometry is still intact. Therefore, core-melt characteristic time is long, eventually preventing temperature excursion due the metal water reaction, which in turns limits hydrogen generation rate.
- Reduction of the hydrogen concentration in the containment by catalytic recombiners and if necessary, selectively located igniters.
- Sufficient floor space for molten debris cooling.
- Extra layers of concrete to avoid containment basement exposure directly to debris.
- The suppression pool type containment provides a good physical mechanism for retention of fission products by water.

Design basis accidents and beyond design basis accidents

For CAREM-25, accident analysis of several initiating events was performed [III-11], Fig. III-5-8:

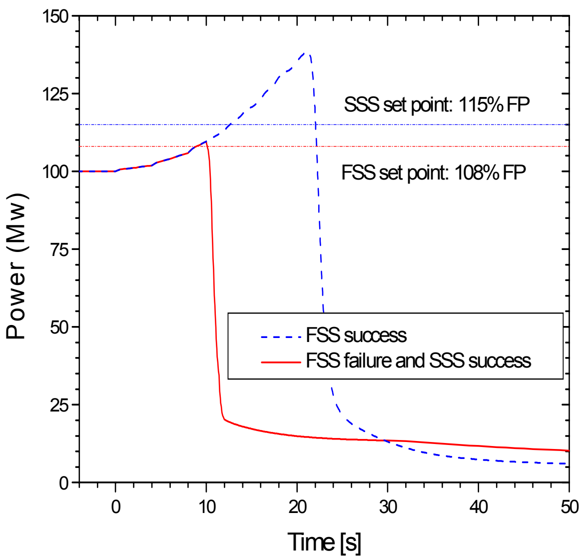


FIG. III-5. RIA power evolution.

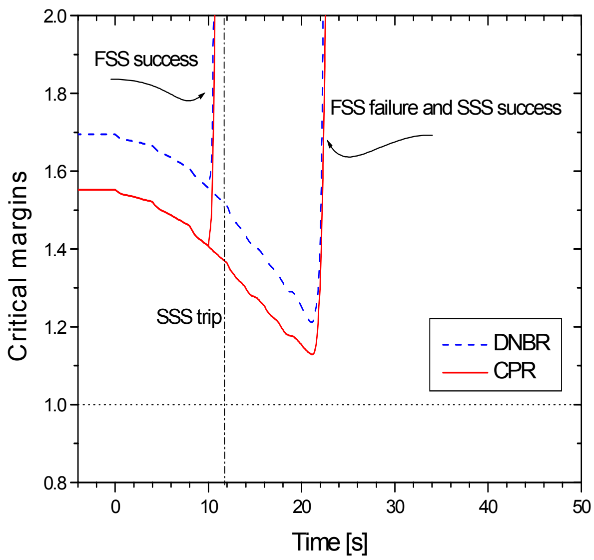


FIG. III-6. RIA DNBR and CPR margins.

Reactivity insertion accident (RIA): since the innovative hydraulic control drive for the FSS and the adjust and control system is located inside the RPV, the control rod ejection accident is avoided; only inadvertent control rod withdrawal transients are postulated. Two scenarios considering FSS success and FSS failure with SSS actuation were modelled, assuming a conservative hypothesis. The results of simulation show that safety margins are well above critical values (DNBR and CPR - critical power ratio) and no core damage is expected. Moreover, as there is no boron in the coolant, boron dilution as a reactivity-initiating event is precluded.

Loss of heat sink: in case of a total loss of feedwater to the steam generators, the RHRS is demanded, cooling the primary system, reducing reactor pressure to values lower than those of hot shutdown. In case of the hypothetical failure of FSS, reactor power is reduced due to the negative reactivity coefficients, without compromising the fuel elements. The SSS will guarantee medium and long-term reactor shutdown.

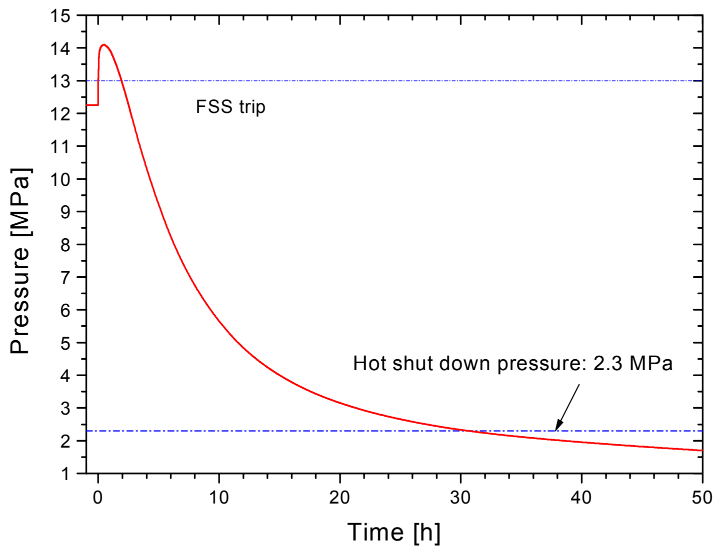


FIG. III-7. LOHS long-term: pressure evolution.

Total loss of flow: In natural circulation modules, with power lower than 150 MW(e), there are no primary pumps, therefore this initiating event is excluded. In high power modules with forced circulation, natural circulation is enhanced intrinsically by the integral-type reactor layout.

Loss of coolant accident: RPV penetration of a maximum diameter is limited by design. Therefore, no large LOCA is possible and there is no need for a high-pressure injection system. In case of LOCA the FSS, SSS, RHRS are demanded and when pressure decreases, the emergency injection system discharges water to keep the core covered for several days. Since the design obviates active systems, the secondary system is not considered to cool and depressurize the primary system in safety evaluations. However, if it is available and in case of need, it could be used as part of the accident management strategy. Moreover, by design a broken pipe is not considered as an injection line (steam coming into the RPV from the containment in case of high depressurization of the primary system due to the use of the steam generators). Examples of a collapsed water level inside the RPV are shown in Figure 8 for the different break diameters analyzed.

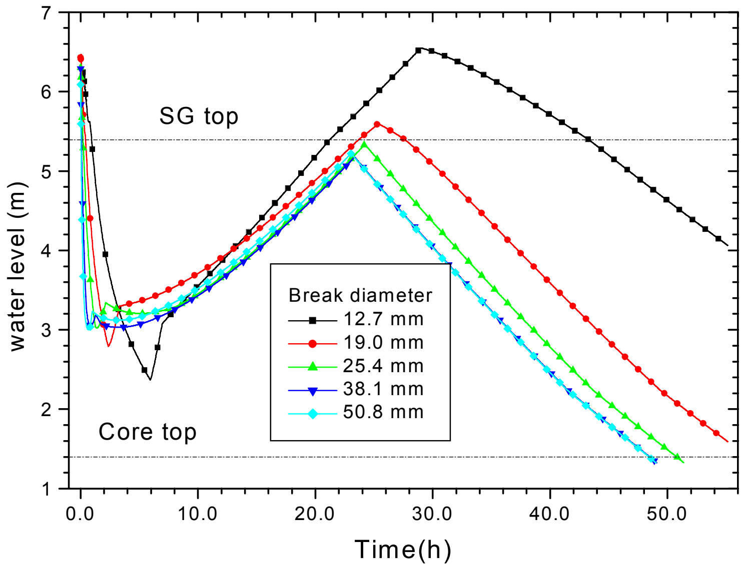


FIG. III-8. RPV collapsed water level (without operator actions nor active systems actuation).

The inherent response of the reactor to LOCA was also analyzed, considering FSS success and failure of all safety systems related to core cooling. Due to a large water inventory over the core and small penetration diameters through the RPV, the core is uncovered only after several hours.

Steam generator tube rupture: this accident is mitigated by isolating the group of steam generators affected, closing both steam and feedwater lines. The secondary side of the steam generators reaches thermal equilibrium with the primary circuit, equalizing pressure with this system. Eventually the reactor could continue operating at 50% of power.

Steam line break accident: sudden depressurization of the secondary side of the steam generators increases heat removal from the primary system with the consequent core overpower. Reactor shutdown (FSS and SSS) and the residual heat removal system are demanded and the reactor reaches a safe condition. In case of an hypothetical failure of both shutdown systems, reactor overpower does not compromise critical safety values (DNB and CPR) because primary total heat removal by the steam generators is intrinsically limited by the reduced tube-side water inventory.

Blackout: This is an event with a major contribution to core meltdown probability in a conventional light water reactor. In CAREM, extinction and cooling of the core and decay heat removal are guaranteed without electricity, by the passive safety systems. Loss of electrical power causes the interruption of feedwater supply to the hydraulically driven CRDs and results in the insertion of absorbing elements into the core. Nevertheless, in the case of failure of the first and second shutdown systems (both passive) in CAREM, feedback coefficients cause self-shutdown of the fission reaction without compromising safety related variables. The decay heat is removed by the RHRS with autonomy of several days.

As a general conclusion, it can be said that due to the large coolant inventory in the primary circuit, the system has great thermal inertia and a long response time in case of transients or accidents.

The CAREM concept greatly enhances accident prevention and mitigation by simplicity, reliability, redundancy and passivity. Nevertheless, in case of the extremely low probability of

failure of the passive safety systems (both redundancies) or no recovery actions after the design period to be covered by the safety systems (grace period, several days), a severe accident could be postulated to occur. Several features are considered to protect the confinement and address hypothetical severe accidents, also allowing optimum use of all process systems for the primary system cooling and containment recovery after the grace period. The severe accident prevention and mitigation features were mentioned above, when discussing Level 5 of the defence-in-depth.

Probability of unacceptable radioactivity release beyond the plant boundaries: The large release probability of the CAREM reactor prototype is 5.2×10^{-8}/year, which may make it possible to simplify or abandon off-site emergency planning requirements.

III-1.6.4. Proliferation resistance

The CAREM design includes several features facilitating and reducing the costs of safeguard implementation. For example, all refuelling will progress in a unique reactor hall, where storage space is allotted for removal of various components, while the RPV internals will be allocated in the auxiliary pool. The reactor hall was designed for remote monitoring of nuclear material handling. The entrance-exit and interfaces allow counting items during their movement [III-12]. The fuel assemblies' pool can be sealed and remote seismic monitoring included to detect perimeter violation.

Additionally, the small size means a small fissile material inventory and reduced proliferation risk. This is similar to the small-source term that reduces the overall risk in the safety analysis.

III-1.6.5. Technical features and technological approaches used to facilitate physical protection of CAREM

The CAREM nuclear island is located inside a pressure suppression containment system. The building surrounding the containment is in a single reinforced concrete foundation mat. It supports all structures with the same seismic classification in one block, allowing the integration of the RPV, the safety and reactor auxiliary systems, the fuel elements pool and other related systems. This building acts as a secondary containment.

The CAREM also provide somewhat better resistance to sabotage because it makes extensive use of multiple and independent passive safety systems. Most of these systems are difficult to hinder or disable from outside the containment building.

III-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of CAREM

Options for CAREM deployment could include purchase of a turnkey plant with standardized acceptance criteria and maintenance and operational requirements, leasing the plant with ongoing operation and maintenance or a power purchase contract with plant operation by a multilateral generating company.

Fuel supply or fuel manufacturing facilities leasing and full scope fuel cycle service agreements can be arranged according to the needs of individual customers.

III-1.8. List of enabling technologies relevant to CAREM and status of their development

Hydraulic control rod drives avoid passing mechanical shafts through RPV or extension of the primary pressure boundary and, since the whole device is located inside the RPV, eliminate any possibilities of large (LOCA). Their design is an important development in the CAREM concept. An important experimental plan is underway; the first series of tests have been conducted in the cold low pressure rig (CEM); a second series of qualification tests are planned in the high pressure rig for CRD test (CAPEM).

Natural circulation and self-pressurization are also important enabling technologies for CAREM. A high pressure natural convection loop (CAPCN) was constructed and operated to produce data to verify the thermal hydraulic tools used in design of the CAREM reactor, mainly its dynamic response. This was accomplished by validation of the calculation procedures and codes for the rig working in states very close to the operating states of the CAREM-25 reactor.

Passive safety systems are also very important in CAREM design; many well established technologies for PWR or BWR are used. Design tools were validated against other proven tools or available experimental data.

A cost effective safety approach is considered in the CAREM concept. An innovative method has been developed in reactor design to balance safety and economics at the conceptual engineering stage.

III-1.9. Status of R&D and planned schedule

The CAREM is a CNEA project and is supported by national R&D.

The CAREM research, development and design activities already performed, underway or planned in the near future are:

- Related to the Reactor Core Coolant System, modeling and qualification are boosted by the testing performed in a high pressure natural circulation rig (CAPCN), covering thermal hydraulics, reactor control and operating techniques. Several sets of experiments were conducted at nominal and ex-nominal conditions. The CAPCN facility may also test the second shutdown system and some in-vessel instrumentation probes.
- Related to core design, neutronic modelling needs have been covered by benchmark data available worldwide.
- Related to fuel assembly design, hydrodynamic and structural tests are planned at two (low and high) pressure rigs. The fuel assembly, the low-pressure test section and the pressure drop tests have been completed.
- Related to mechanical design (structural, dynamic, seismic, etc.) of the core and other RPV internals, different mock-up facilities are under construction. They represent sections of the core and include one vertical full-scale model of a control rod drive with supporting barrel and kinematics chain, to test the design and integration with structures and core. Evaluation of manufacturing and assembly processes for SGs was done using a mock-up.
- An experimental plan is underway for the first shutdown system and more specifically, the control rod drive mechanism (CRD). The first series of tests have been conducted in the cold low pressure rig (CEM) and second series of qualification tests are planned in the high pressure rig for CRD test (CAPEM).

The CAREM RD&D costs necessary for safety acceptance are US $95 million, including construction of a 100 MW(th) prototype (US $85 million) and other specific tests (US $10 million).

III-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

Integral primary system reactors used classical PWR or BWR technologies but this configuration is a new approach that needs demonstration. Demonstration is the final step in the RD&D and system verification strategy and it should be performed unless other strategies are possible and convenient due to, for example, economic reasons. RD&D costs for safety acceptance of different options should be compared. A CAREM reactor prototype will be constructed because it is the cheaper strategy in the context of CAREM.

The next step of this project is the construction of a prototype of about 27 MW(e) (CAREM-25). The conceptual engineering of this prototype reactor was completed.

III-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

There are other integral primary system reactors under development. Three of them are IMR (Mitsubishi, Japan), IRIS (Westinghouse, USA) and SMART (KAERI, Rep. of Korea) [III-13].

III-2. Design description and data for CAREM

III-2.1. Description of the nuclear systems

The CAREM is an indirect cycle reactor with some distinctive features greatly simplifying the design and contributing to a high safety level. Some of the high-level design characteristics are:

- Integrated primary cooling system;
- Self-pressurization;
- Safety systems relying on passive features; and
- Balanced and optimized design with a cost-effective internalization of safety.

Primary circuit and its main characteristics

The CAREM nuclear power plant design is based on a light water integrated reactor. Figure III-9 shows a diagram of the natural circulation of the coolant in the primary system of CAREM-25. Water enters the core from the lower plenum, after heating, the coolant exits the core and flows up through the riser to the upper dome. In the upper part, water leaves the riser through lateral windows to the external region. Then it flows down through modular steam generators, decreasing its enthalpy. Finally, the coolant exits the steam generators and flows down through the downcomer to the lower plenum, closing the circuit. The driving forces obtained by the differences in the density along the circuit are balanced by the friction and form losses, producing adequate flow rate in the core to bring a sufficient thermal margin to critical phenomena. Natural circulation of the reactor coolant results from location of the steam generators above the core. The coolant also acts as a neutron moderator.

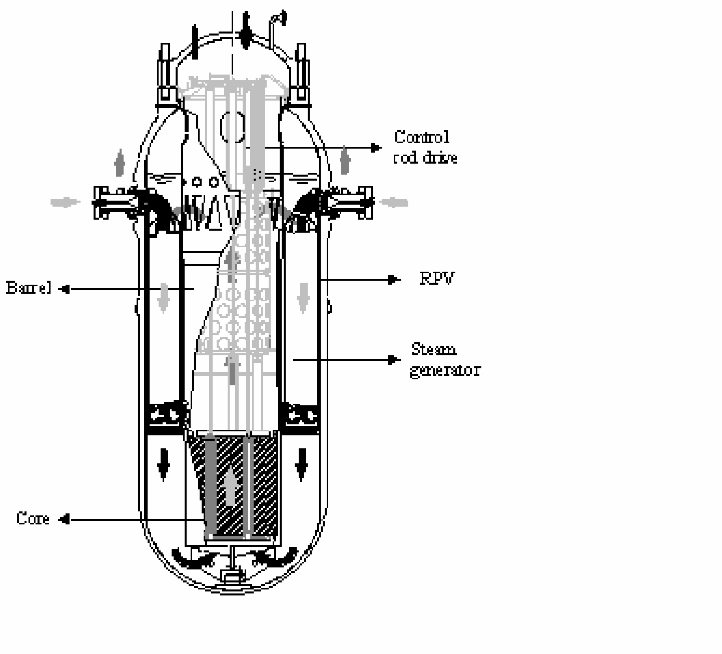


FIG. III-9. Reactor pressure vessel and circulation path.

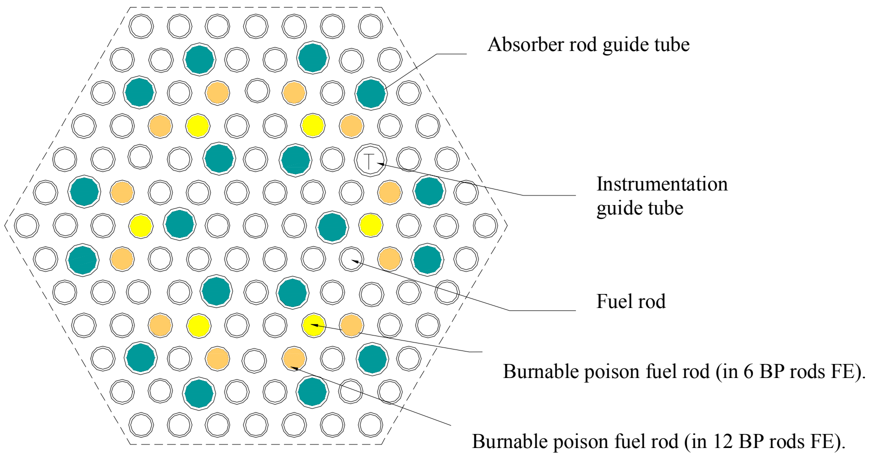


FIG. III-10. Fuel assembly diagram. Fuel rods, guide thimbles and instrumentation thimble distribution.

For power modules over 150 MW(e) pumps achieve flow rates needed to operate at full power.

For low power modules, the natural circulation of coolant produces different flow rates in the primary system according to the power generated (and removed). Under different power transients, a self-correcting response in the flow rate is achieved [III-14].

Due to self-pressurizing of the RPV (steam dome), the system maintains the pressure very close to saturation. At all operating conditions this has proved sufficient to guarantee a remarkable stability in the RPV pressure response. The control system is capable of keeping the reactor pressure practically at the operating set point through different transients, even in case of power ramps. The negative reactivity feedback coefficients and large water inventory of the primary circuit combined with the self-pressurization features make this possible with minimum control rod motion. It concludes that the reactor has an excellent behaviour under operational transients.

Reactor core and fuel design

The core has fuel assemblies (FA) of a hexagonal cross section shown in Fig. III-10. This design is typical of PWR fuel assemblies.

Primary components

Twelve identical 'mini-helical' vertical steam generators, of the once-through type are placed equidistant from each other along the inner surface of the reactor pressure vessel (RPV), see Fig. III-11. They are used to transfer heat from the primary to the secondary circuit, producing superheated dry steam at 47 bar.

The secondary system circulates upwards within the tubes, while the primary is in counter-current flow. An external shell surrounding the outer coil layer and adequate seal form the flow separation system. It guarantees that the entire stream of the primary system flows through the steam generators.

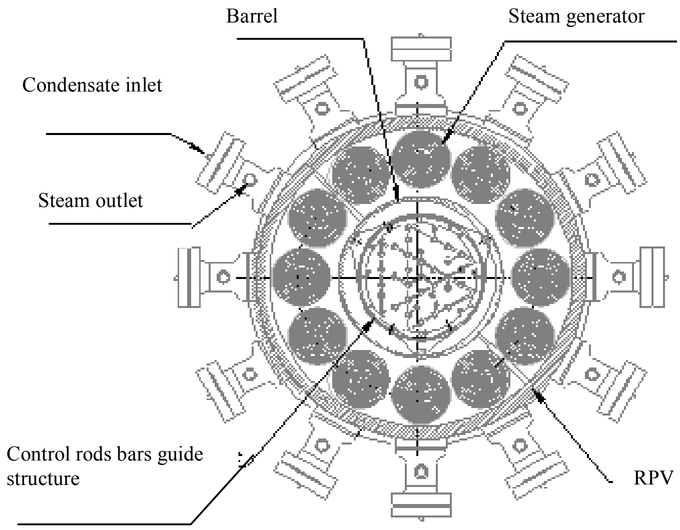


FIG. III-11. Steam generation layout.

To achieve a rather uniform pressure-loss and superheating on the secondary side, the length of all tubes is equalized. For safety reasons, steam generators are designed to withstand the primary pressure without pressure in the secondary side and the live steam system is designed to withstand primary pressure up to isolation valves (including the steam outlet / water inlet headers) in case of SG tube breakage.

Main heat removal paths

The heat removal paths of CAREM-25 under various operational states and LOHS are shown in Figure III-12.

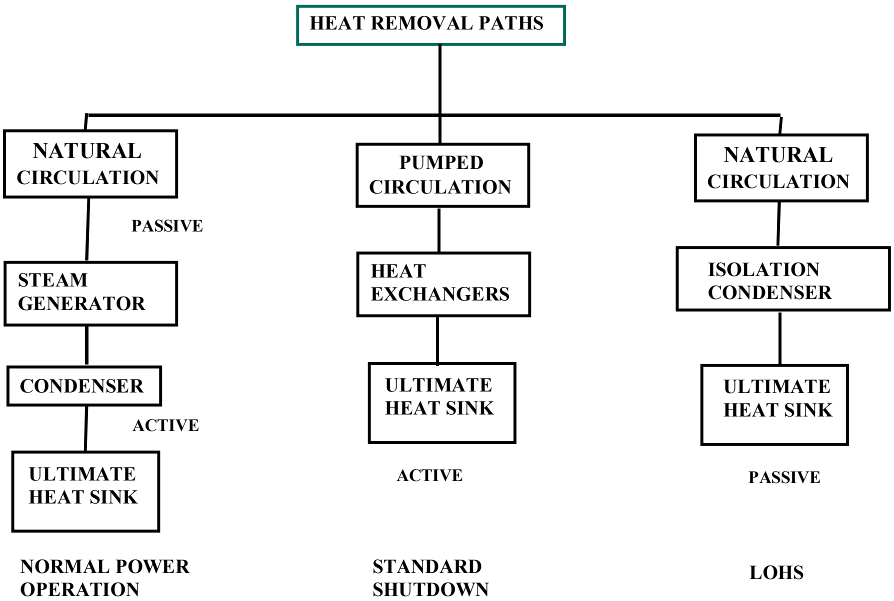


FIG. III-12. CAREM-25 heat removal paths.

III-2.2. Description of the turbine generator plant and systems

The CAREM commercial plants will use a two-stage turbine with re-heater and exhaust steam at low pressure is condensed in a water-cooled surface condenser. The CAREM prototype uses a single turbine.

III.2.3. Systems for non-electric applications

The CAREM can be used either as a heat source or electrical energy supply for water desalination.

A technology selection was made according to the special characteristics of the Puerto Deseado site, world tendencies and technological advances in desalination processes and economic data. According to that, reverse osmosis is the technology considered the most convenient for this site [III-4].

III-2.4. Plant layout

Buildings and structures

The CAREM nuclear island is inside a pressure suppression containment system, which contains the energy and prevents fission product release in the event of accidents.

The building surrounding the containment is in a single reinforced concrete foundation mat. It supports all structures with the same seismic classification, allowing the integration of the RPV, the safety and reactor auxiliary systems, the fuel elements pool and related systems in one block.

The plant building (Fig. III-13), is divided in three main areas:

- Nuclear module;
- Turbine module;
- Control module.

More detailed data of CAREM prototype (CAREM-25) are presented in [III-13].

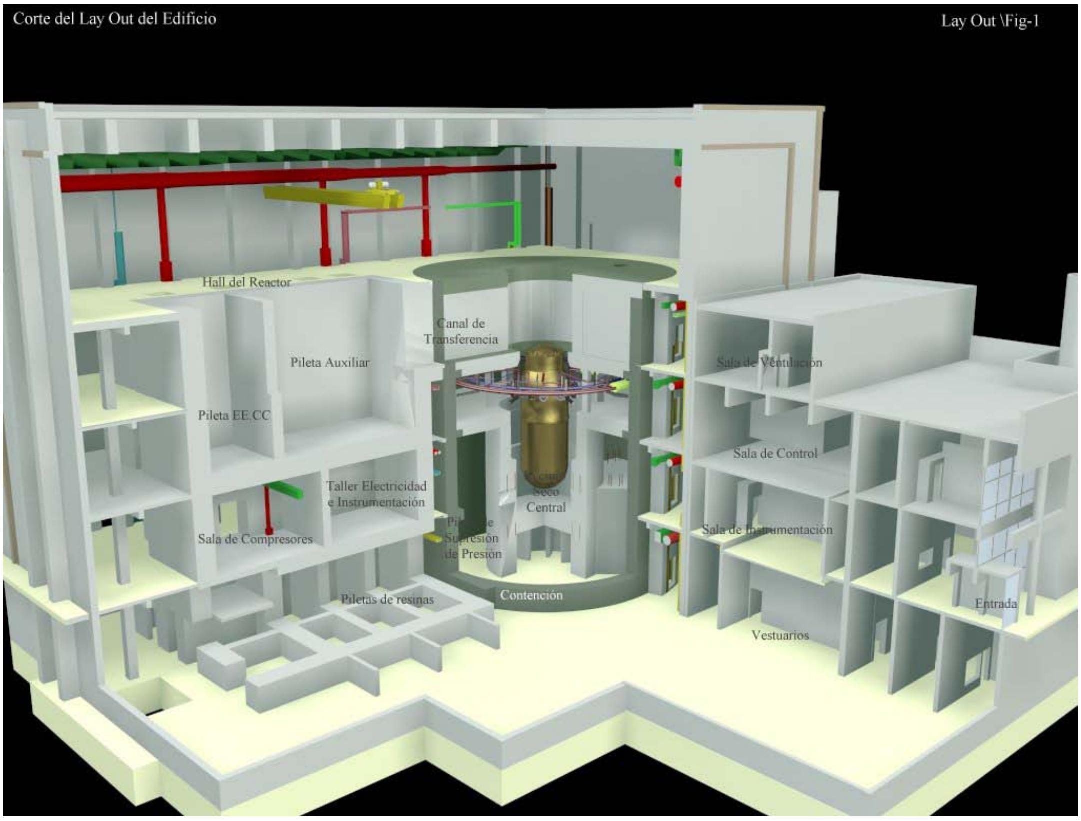


FIG. III-13. Plant layout.

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ANNEX IV

MULTIPURPOSE ADVANCED REACTOR, INHERENTLY SAFE (MARS),

University of Rome “La Sapienza”, Italy

IV-1. General information, technical features and operating characteristics

IV-1.1. Introduction

The MARS (Multipurpose Advanced Reactor, inherently Safe) is a 600 MW(th), single loop, pressurized light water reactor (PWR). The design was developed at the Department of Nuclear Engineering and Energy Conversion of the University of Rome “La Sapienza” [IV-1].

Design of the plant began in 1983 and was originally focused on a multipurpose reactor for high population density areas, for industrial heat production and especially, for water desalination and district heating. These uses were the main reason for an emphasis on safety requirements, to avoid subjecting the population to radiological hazards. At the same time, for commercial appeal, the new plant had to be economically competitive with traditional nuclear and fossil fuel power plants.

The compromise between exacting safety levels and economic competitiveness was a challenge to the plant design team, which had the support of experts from ENEA (the Italian Governmental Agency for Energy and Environment) and ENEL (the Italian Board for Electric Energy Production).

The MARS nuclear power plant (NPP) design uses well-established technology and the operational experience of PWRs but also incorporates innovative features that keep the cost per KWh competitive without sacrificing a high level of safety. Extensive use of passive safety, in-depth plant simplification and decommissioning-oriented design were the main guidelines for design development.

Dedicated experimental test facilities were built, as listed in the next paragraphs.

IV-1.2. Applications

The MARS NPP is designed to produce electric energy and/or industrial heat; the most efficient utilization of such a plant is definitively cogeneration.

Heat produced in a MARS NPP is typically for low temperature uses of hot water or low pressure steam; among these, the following utilizations were analyzed:

- Water desalination using low temperature processes (as thermo-compression or multiple effects).
- District heating.
- Food industry (conservation industry).

IV-1.3. Special features

The MARS NPP:

- Provides for incremental capacity increases through a modular approach.
- Can benefit from extended factory production and from assembling main piping bolted-segments to provide fast construction, easy component substitution, system

modification following new technological achievements, special plant life extension, easy and “clean” decommissioning and production cost cutting.

IV-1.4. Summary of major design and operating characteristics

Installed capacity

The capacity characteristics of NPPs with the MARS modules are summarized in Table IV-1.

TABLE IV-1. CAPACITY CHARACTERISTICS OF MARS NPPs

Reactor rated thermal power, MW	600
Rated electric power (one module), MW	150
Rated electric power (suggested cluster of 3 modules), MW	450
Suggested rated electric power in co-generation configuration (electricity + desalinated water/district heating)	300

The MARS reactor is moderated and cooled by pressurized light water (PWR). The reference rated core thermal power is 600 MW. In case of only electric energy production, 150 MW(e) of gross power (146 MW(e) net power) are produced in a 600 MW(th) plant, with 25% gross efficiency (24.5% net). In case of cogeneration, electric energy production strongly depends on the thermodynamic requirements of the hot water or steam produced. Cogeneration cycles are designed to produce electric power between 80 and 100 MW(e) and hot water or steam at temperature around 100°C.

Mode of operation

For the nuclear core described herein, the suggested mode of operation is base load; load following is possible with limited power variation rates.

Load factor/ Availability

With the nuclear design referred to in this chapter (further improvement is definitely possible), an availability factor of 95% is quite realistic and for the base load operation mode, the load factor could be 95% as well.

Summary of major design characteristics

The fuel is low-enrichment uranium dioxide (in the fuel loading strategy described here, the ^{235}U enrichment is 2.8%); the core includes 89 “standard” PWR fuel assemblies. The assemblies are Zircaloy-cladded with a rod array 17×17, including 264 fuel rods and 25 positions for zircaloy guide tubes for control rods (black, Ag-In-Cd; grey, stainless steel) or for burnable poisons (borosilicates). The fuel rod pitch is 1.26 cm and the active length is 260 cm.

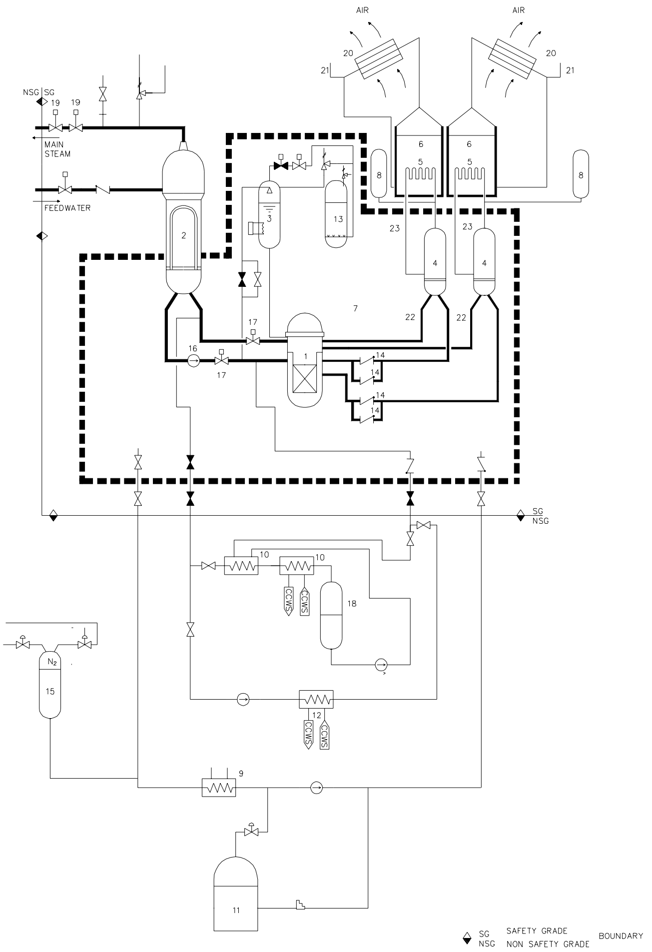
Light water acts as the coolant/moderator, flowing in a single cooling loop. The average core coolant temperature is 234°C.

The reactor internals are AISI 304 made.

The reactor vessel internal diameter is 3000 mm; the overall height of the assembled vessel is 11 091 mm.

Simplified schematic diagram

The reactor cooling system and some of its components are shown in Figures IV-1, IV-2 and IV-3. The scheme of steam-turbine circuit is presented in Figure IV-18.



Legend

1. Reactor
2. Steam generator
3. Pressurizer
4. Heat exchanger (reactor coolant/intermediate coolant)
5. Heat exchanger (intermediate coolant/ final heat sink
6. Water reservoir
7. Pressurized containment for primary loop protection (CPP)
8. Intermediate loop pressurizer
9. Heat exchanger (primary containment water cooling system)
10. Chemical and volumetric control system heat exchangers
11. Water storage tank
12. Residual heat removal system heat exchanger
13. Pressurizer relief tank
14. Safety core cooling system check valve
15. Primary containment pressure control system pressurizer
16. Main coolant pump
17. Primary loop on/of valve
18. Volumetric control system (VCS) tank
19. Steam line on/off valve
20. Ultimate heat sink condenser
21. Communication path with the atmosphere
22. Safety core cooling system primary loop
23. Safety core cooling system intermediate loop

FIG. IV-1. Reactor cooling system (RCS) and main auxiliaries.

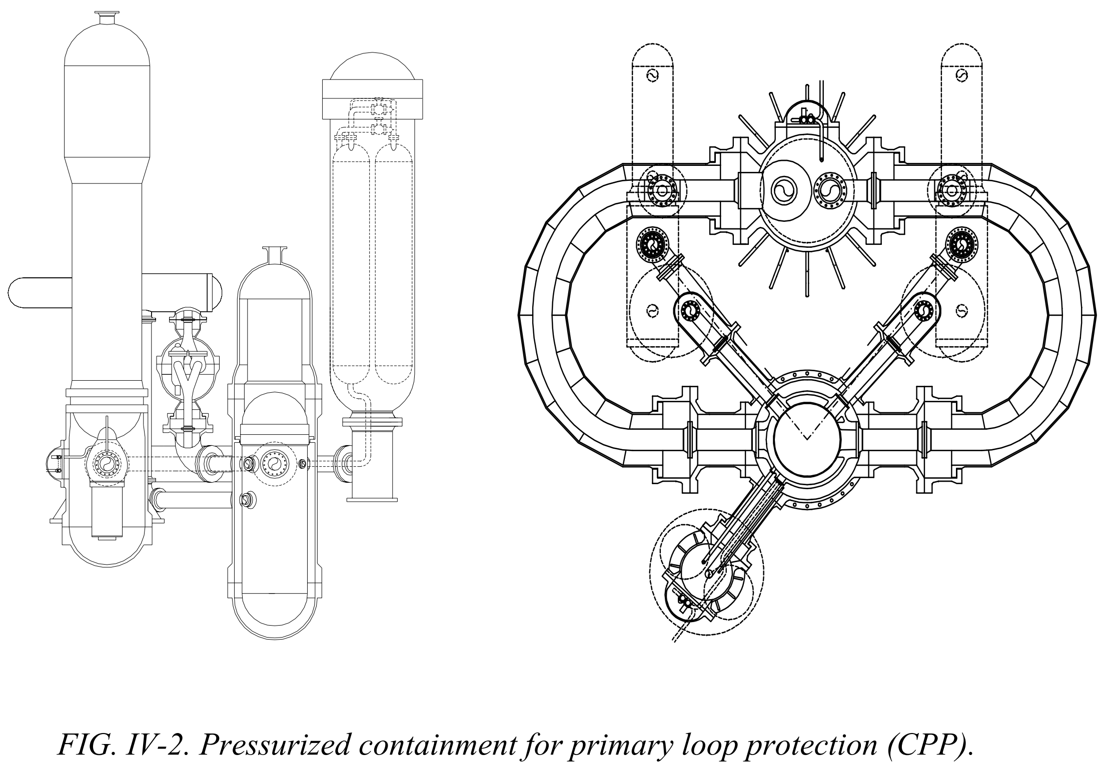


FIG. IV-2. Pressurized containment for primary loop protection (CPP).

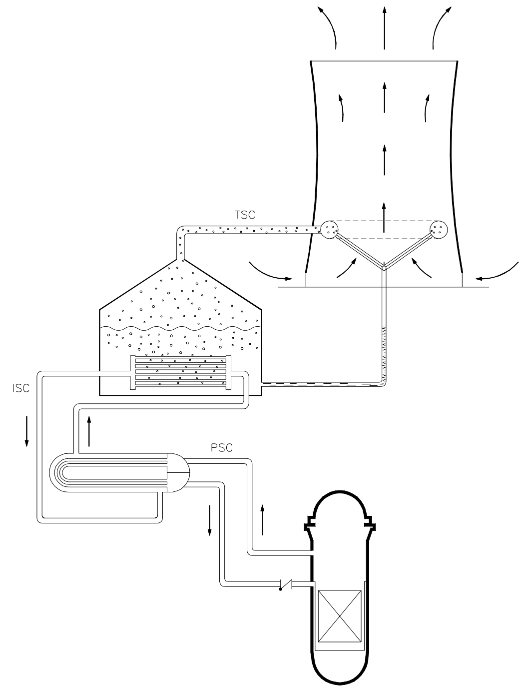


FIG. IV-3. Scheme of the safety core cooling system (SCCS).

Some characteristics of the MARS design are similar to well-known PWRs (primary loop type, core geometry and materials, reactor control type, etc.). Along with them, the MARS design incorporates the following innovative solutions:

- A passive-type, quasi-static emergency core cooling system, based only on natural circulation of cooling fluids and using external air as the ultimate heat sink.
- An additional passive-type scram system based on a two-metal core temperature sensor and operated by gravity.
- Full enclosure of the primary-coolant boundary in a pressurized containment filled with low enthalpy water (primary loop jacket).

The primary cooling system (Figure IV-1) includes only one loop, with 25" I.D. pipes, one vertical-axis U-tube steam generator and one canned rotor pump connected to the steam generator outlet nozzle.

The safety core cooling system (SCCS) is connected to the reactor vessel. A vapour-bubble pressurizer controls the pressure inside the primary cooling system.

On/off valves in the primary loop main isolation system (MIS) are installed in the primary cooling loop, to isolate if necessary, the steam generator (SG) and primary pump (i.e., in the event of an SG tube rupture).

The primary cooling system and the SCCS are inside a pressurized containment, which is filled with water at the same pressure as the primary coolant but at a lower temperature (70°C). This is called the CPP (pressurized containment for primary loop protection, Figure IV-2), which allows reduction (or even the elimination) of primary stresses on the primary coolant boundary and provides an intrinsic defence to loss of coolant.

Cooling of the MARS core in emergency conditions is provided by the SCCS, Figure IV-3. It is designed to transfer the core decay heat directly from the reactor pressure vessel to the external air, without the intervention of any energized system or component.

The system operating principle relies on fluid density differences, due to temperature differences between vertical fluid columns, causing the fluid circulation.

The presence of multiple circuits (primary safety cooling loop [PSC], the intermediate safety cooling loop [ISC], and pool and condenser loop [third safety cooling loop or TSC]) in a cascading operation chain provides redundant barriers between the activated reactor coolant and the external environment.

The SCCS includes two trains; each train may remove 100% of the core decay power. In an accident causing the reduction of the core coolant flow (such as station black-out or primary pump trip), activation is automatic (without intervention either by the operator or by the control and supervision system, because the PSC interception valves are kept in a closed position by forces from the primary coolant flow and start opening when this flow decreases below a set-point value); the operation of the system is completely passive.

The SCCS operation relies on special check valves that open automatically without operator intervention and without the need for energized systems, when conditions require additional core cooling. These valves have a completely innovative design (Figure IV-16). They are kept in a closed position by pressure differences between the reactor vessel inlet and outlet (that is roughly proportional to the square of the coolant flow-rate); when flow-rate through the core goes to zero, the pressure difference decreases and when it is no longer sufficient to sustain the weight of the valve plug, this falls and a complete flow area is opened with a very low hydraulic resistance. Two valves, each with 100% capacity, are inserted in each SCCS train and, to increase the system availability to values that make failure incredible, the two additional valves (in each loop, the second valve is of traditional design) are different in typology and mechanical construction.

When any of the four check valves is opened, after a short transient phase the flow in the PSC is assured by a difference in level of about 7 m between the vessel outlet nozzle and the primary heat exchanger and by the difference between inlet and outlet vessel temperatures. A horizontal-axis, U-tube heat exchanger (Figure IV-17) transfers heat from the PSC to the ISC.

Pressure in the ISC loop is slightly higher than 75 bar (thanks to a dedicated pressurizer); this value guarantees sub-cooled water conditions of the fluid during any accidental situation or transient; the difference in level for natural circulation in the ISC loop is about 10 m. A second heat exchanger transfers the heat from the ISC circuit to the water of a reservoir.

The steam produced in the reservoir is mixed with air initially present in the dome over the pool; pressure in the dome rises and this causes a flow of the air-steam mixture towards a small connection path with the atmosphere. An inclined-tube heat exchanger is placed between the pool dome and the connection path with the atmosphere, where steam is partially condensed thanks to the action of external air drawn by a chimney.

The above mentioned choices introduced some constraints in the plant design; the first limit in particular, imposed by the functional requirements of the special emergency core cooling system regarding the rated thermal power, that cannot exceed approximately 1000 thermal MW and in the solution herein described, has been chosen equal to 600 MW(th). Another characterizing parameter is the pressure in the primary system, chosen equal to 75 bar, which is different from the pressure values usually adopted in PWRs for the production of electric power (150–170 bar). This choice leading to a loss in thermodynamic efficiency of the plant because of the limitation of the higher isotherm in the steam cycle, has nevertheless allowed the adoption of the pressurized containment for primary loop protection (CPP, the pressurized boundary that envelopes the primary cooling system and the emergency core cooling system), substantially eliminating the possibility of any type of loss-of-coolant accidents, including control rod ejection.

Inclusion of the primary coolant system (average operating temperature: 234°C) inside the low-enthalpy water filled pressurized containment (CPP, at a temperature of 70°C) requires thermal insulation to reduce heat losses from the primary coolant system. An insulating system has been designed on the external side of the primary coolant boundary (only the lower head of the reactor vessel is thermally insulated in the internal part) through matrices of stainless steel wiring that cause the presence of semi-stagnant water which resists high pressure and fast pressure gradients, with acceptable shape modifications. The system limits heat losses to about 0.3% of the reactor thermal power.

Neutron-physical characteristics

At full power, the inlet temperature reactivity coefficient (pcm/°C) is -7.6, BOC and -20, EOC. The coolant density reactivity coefficient (pcm/ % density variation) is +40.1 BOC and +117.5 EOC.

The average fuel burn-up per cycle (3 irradiation cycles; 5 assemblies are irradiated for 4 cycles) is about 11,300 MW·d/t.

The peak factor changes on increase of the fuel burn-up are shown in Figure IV-4.

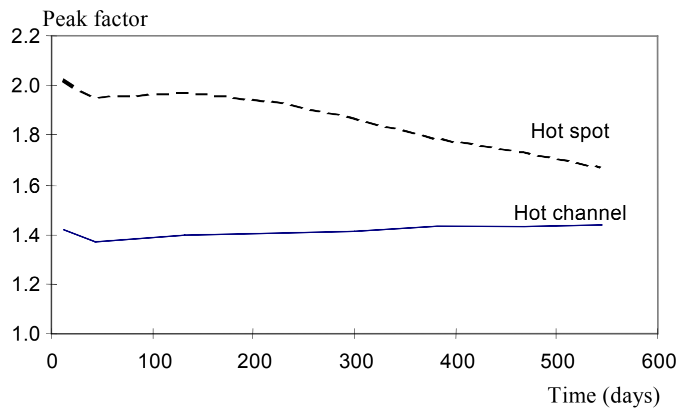


FIG. IV-4. Peak factors vs. irradiation time.

Reactivity control mechanism

The core is cooled and moderated by pressurized light water containing a boron solution. Boron (Figure IV-5) and burnable poisons compensate the excess reactivity during the irradiation cycle.

The core is equipped with two different control rod systems. The first is an active type, quite similar to classic PWR control rod systems and is divided into four different banks. At the beginning of cycle (BOC), with maximum boron concentration (711 ppm), with all active control rod clusters in, the multiplication factor is 0.90435. This figure changes to 0.99335 without boron.

The second control rod system foreseen in the MARS reactor is a passive type and causes control rods insertion into the core when the core coolant temperature reaches a selected set value. The operation of this system (called ATSS, Figure IV-6) is based on the differential thermal expansion of a bimetallic sensor located inside the fuel assembly; the differential displacement, due to coolant temperature increase, causes the release of a traditional-type control rod cluster.

At the BOC, Keff with 711 ppm boron in the core coolant becomes 0.84920 if all control rods, active plus passive type, are in the core.

The control rod drive mechanisms of the first control rod system (traditional, active type) are placed on the vessel top head; the passive control rod system is within the reactor vessel. A core map with the indication of control rod positions is shown in Figure IV-7, the worth of control rods is given in Table IV-2.

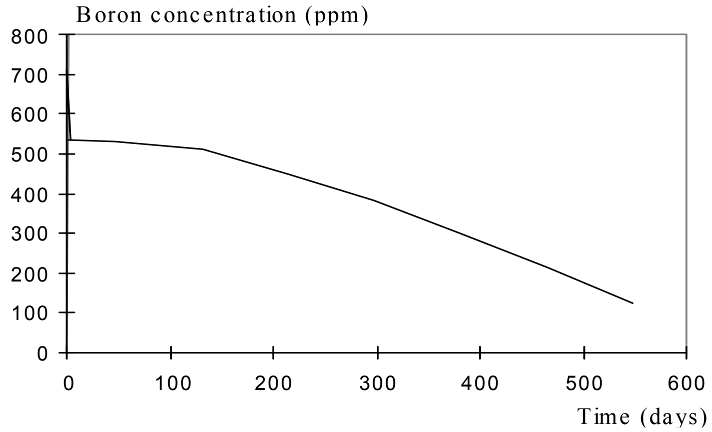


FIG. IV-5. Critical boron concentration vs. time.

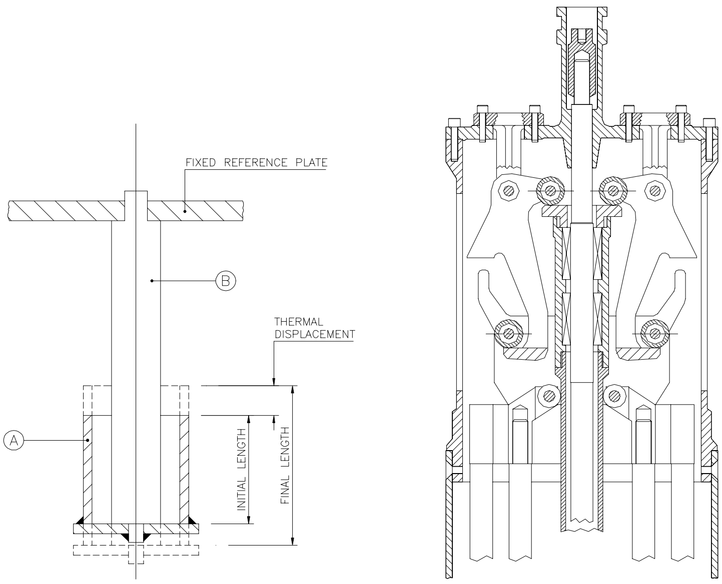
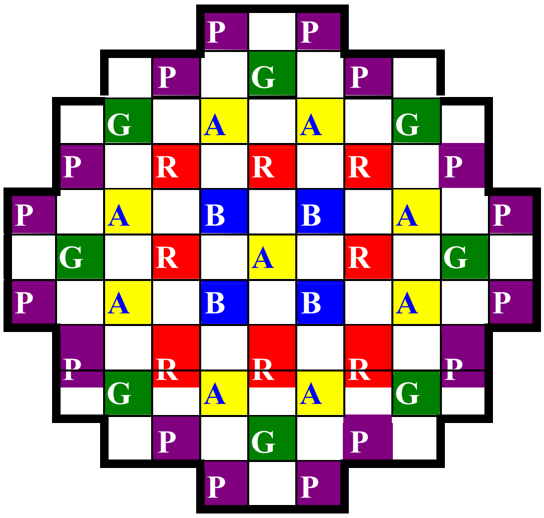


FIG. IV-6. Operating scheme and self-releasing head of ATSS.



A Passive rods
B Blue rods
G Green rods
P Purple rods
R Red rods

FIG. IV-7. Control rod cluster distribution.

TABLE IV-2. ROD CLUSTER WORTH, BOC (IN BRACKETS, BORON CONCENTRATION)

CLUSTERS	K (711 ppm)	K (0 ppm)
4 blue clusters	0.99615	1.10460
8 red clusters	0.98100	1.08735
8 green clusters	0.99185	1.10700
16 purple clusters	0.99360	1.10915
9 yellow clusters (passive)	0.97360	1.08010

Cycle type

The thermodynamic cycle is indirect (as in traditional PWRs). The thermodynamic efficiency is only 25%, due to constraints imposed by the possibility of relying only on unalterable, totally passive safety systems.

Thermal-hydraulic characteristics

The thermal-hydraulic characteristics of MARS are summarized in Table IV-3.

TABLE IV-3. THERMAL-HYDRAULIC CHARACTERISTICS

Primary coolant flow rate (forced flow) (kg/s)	3227
Total RCS internal volume (m^3)	130
Pressurizer heaters power (kW)	800
Steam flow rate (kg/s)	277
SG steam pressure (bar)	18.8
Temperatures (°C)	
• Reactor vessel outlet	254
• Reactor vessel inlet	214
• Steam generator steam outlet	209
• Steam generator feed-water inlet	150

Among basic thermal-hydraulic core design criteria, fully satisfied in the safety analyses is: the minimum departure from nucleate boiling ratio (DNBR) must be always higher than 3.0, for Level 1 (normal events), Level 2 (unplanned events; probability higher than 3.10^{-2}

events/year) and Level 3 (rare events; probability between 3.10^{-2} and 1.10^{-3} events/year) plant conditions; it must be higher than 2.0 for all Level 4 (unexpected events; probability between 1.10^{-3} and 1.10^{-4} events/year) plant conditions.

Average discharge burn-up of fuel: 35 000 MW·d/t.

Period between refuellings: 600 effective full power days.

Mass balances/flows of fuel materials

The fuel loading strategy reported here envisions loading 28 fresh fuel assemblies per irradiation cycle, for 3 cycles and 33 fresh fuel assemblies in the 4th cycle. Each fuel assembly contains 439.4 kg of UO_2. Since each irradiation cycle lasts 600 days and 28×3 assemblies are irradiated for 3 cycles and 5 assemblies irradiated for 4 cycles, allowing 30 days for refuelling, each fuel assembly is "blocked" in the reactor for an average of 2068 days. One fuel assembly produces 13 556 MW-day of thermal energy. The 89 fuel bundles in the reactor produce 1251 GW-h of electric energy per year; as an average value, every year 6.90 t of UO_2 are loaded, i.e. 6.08 t of 2.8% enriched U. The corresponding natural U consumption is 24 t/year. The specific natural uranium consumption is 172 000 kg/(GW(e) year).

Design basis lifetime for reactor core, vessel and structures

Fuel has a reference permanence time in the core of 5.5 years (average value); the lifetime for the reactor vessel is 50 years; the lifetime for most plant systems, that can be removed, is in the range of 50 years (technology improvement suggesting substitution); the structures have a lifetime higher than 100 years (as in hydroelectric plants, long-term maintenance may lead to a lifetime of centuries); due to its quick substitution feasibility, no component limits the lifetime of the plant.

Design and operating characteristics of systems for non-electric applications

Non-electric applications studied, possibly proposed in a cogeneration scheme, include desalinated water and district heating.

No limitation exists for application, provided the user requires heat below some 190°C.

Economics

The provisional total investment cost for a 3-module prototype NPP of 450 MW(e) is €978 million (€829 million for direct costs and €149 million for indirect costs), i.e. €2173/kW(e).

The construction period is between 3 and 4 years.

The total provisional O&M costs are €47 million/year, i.e. €0.014/kW·h (referred conservatively to a load factor of 85%).

With an interest rate for debt service of 5%/year, the investment cost at the beginning of operation is €1,061 million; if repayment of the debt service in 20 years is assumed, a production cost of €0.039/kW·h results, in the first 20 years. After that period, the value of €0.014 /kWh applies.

A series-production MARS NPP would allow cost savings in the range of 10–15%.

IV-1.5. Outline of fuel cycle options

The MARS fuel cycle is a standard, once-through cycle, typical of PWRs. Innovative fuels are compatible for use in the MARS NPP.

IV-1.6. Technical features and technological approaches that are definitive for MARS performance in particular areas

IV-1.6.1. Economics and maintainability

The main characteristics of the MARS design with high impact on economic features of the plant are:

- Extensive plant simplification.
- Extensive use of proven technologies.
- Extensive in-shop pre-manufacturing of components (including main, large components of the primary coolant loop).
- Wide adoption of metallic instead of concrete structures.It should be noted that the whole design of the MARS NPP is aimed at strongly simplifying the plant layout, the component construction and their assemblage on the site to reduce construction time and costs (to make the plant competitive with larger, traditional ones) and to make plant decommissioning easy, fast and cheap.

In particular, the following aspects are relevant for circuit simplification:

- The selected plant characteristics (specific power, temperature, pressure, thermal inertia, etc.) make it possible to simplify all non safety-related auxiliary systems, thanks to the reduced performances required.
- The extensive use of passive systems to assure plant safety eliminate some traditional safety-related auxiliary systems (e.g. injection systems) or strongly simplify others, reducing the number of redundant components (e.g. boron emergency shutdown).
- The selected plant power reduces the size of the main components, making them easy to construct, to transport, to assemble and to disassemble.
- The low boron content in the primary coolant simplifies boron systems.
- The adoption of an innovative design for all large components (SG, Pressure Vessel) allow easy, fast construction, assembling and disassembling, and even substitution during reactor operating life.
- The adoption of flanged connections for components and piping, also in the primary loop, simplifies plant erection, shop fabrication, and achieves construction time reduction; a large amount of work is removed from the site with a part transferred to factories, an important change in the approach of the nuclear industry.
- The adoption of metal structures to support components, for working floors and the biological shield, limits the amount of concrete in the plant, simplifying and making the construction phase faster and obviously, speeding the final decommissioning as well.

In nuclear plants, construction cost is strongly dependent on the technology selected, on the seismic characteristics of the site and on the local rules and regulations affecting the type and cost of active safeguard systems and passive protection systems. For these reasons, every attempt at cost evaluation when not aimed at a specific design solution is rather approximate.
As a result of this effort, the number of main components (pumps, valves, tanks, etc.) in the MARS plant is reduced to about 50%, with respect to traditional PWRs of the same rated power.

The plant construction time is reduced to less than 4 years.

An evaluation of construction, operation and maintenance costs has been performed with reference to a site with ”standard” characteristics, to Italian laws and regulations and to the production of electric energy [IV-2].

With reference to a 450 MW(e) power station equipped with three MARS units, the direct construction costs listed in Table 4 are expected. The total direct investment cost is €753 429 000, including a 10% overestimation to account for contingencies during construction, €828 772 000 has been considered, corresponding to a unit direct cost of €1841 /kW(e).

The total indirect construction cost (for the same power station equipped with three 150 MW(e) MARS reactors) has been assessed as €149 179 000 (18% of direct cost), corresponding to €331 /kW(e) installed.

The fixed costs of operation and maintenance for the three-units plant, including personnel and other fixed costs, has been estimated as €17 756 000 /year (€39.4 /kW(e)-year).

The variable operation cost, in case of operation of the plant for base service (rated power without operation in load-following, with a yearly production of 3 942 000 000 kW·h), has been evaluated as €4 444 000, corresponding to €0.0011 /kW·h.

Some uncertainty margins exist in the fuel costs mainly because available references apply to plants operating in countries different from Italy and conditions vary from one plant to another. A full core for each of the three reactors of the reference power station has a cost that may be assumed to range between some 39 000 000 and some €51 000 000.

TABLE IV-4. COST OF A 450 MW(e) POWER STATION WITH 3 MARS UNITS

SYSTEM	COST (€1000)
Buildings	68 638
HVAC systems	11 528
Closed circuit water cooling system and main condenser	40 826
Control rod systems	9 144
Fuel handling and storage system	8 869
Reactor coolant system (RCS)	125 502
Pressurized containment for primary loop protection (CPP)	66 232
Safety core cooling system (SCCS)	13 216
Main RCS auxiliaries	10 272
CPP auxiliaries	4 808
Reactor auxiliaries	9 889
Containment building safeguards	929
Radwaste system	2 653
Turbine	110 313
Condensate system	30 820
Feedwater system	14 389
Main steam system	13 837
Electric power station	53 527
Protection systems	1 962
Control systems	9 636
Plant supervising system	10 221
Environmental monitoring system	522
Plant monitoring system	2 047

SYSTEM	COST (€1000)
Electric boards and panels	7 176
Neutron monitoring system	2 278
Demineralized water system	359
Auxiliary steam system	286
Instrumentation air system	889
BOP fire protection system	137
Elevators and lighting system	6 956
Common services common buildings	14 273
Condenser cooling water system	48 461
Minor RCS auxiliaries	15 912
Turbo-alternator lubricating system	162
Plant electrical system	20 252
Other plant auxiliaries	16 510
TOTAL	753 429

This core can produce more than 6 200 000 000 kW·h but conservatively we assume, for each equilibrium core, a production of 5 913 000 000 kW·h (corresponding to the hypothesis that only three irradiation cycles are applicable for all the elements). In this conservative hypothesis, the specific cost of the electric energy produced, defined by the nuclear fuel only, ranges between 0.0066 and €0.0087 kW·h. Just as a reference and considering all items contributing to the definition of the fuel cost, as well as its limited incidence on the overall cost of produced kW·h, in this analysis, we refer to a cost of kW·h for fuel, intermediate between the two preceding values and equal to €0.0074 kW·h.

The overall production cost for electric energy is affected by the investment financing conditions and the load factor.

Assuming the realistic hypothesis of an interest rate equal to 5% and a repayment period of 20 years, with the conservative hypothesis of a load factor of 85%, the cost of kW·h produced by the plant with MARS reactors is, therefore, globally, €0.039 kW·h during the first 20 years of operation and €0.014 kW·h during the following years.

IV-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

The quite good neutron economy allows use of low initial enrichment in comparison with traditional water-cooled reactors, guaranteeing better utilization of the primary energy source.

The MARS plant is designed to guarantee very low radioactive contamination of the primary coolant and an extremely low production of radioactive wastes.

In fact, the low temperature of the fuel rods in the MARS core allows strong retention of fission products within the rods themselves (thanks to the huge reduction in fission products diffusion inside the fuel matrix); furthermore, the low operating temperature of the primary coolant limits stress on fuel cladding and eventual damage is highly reduced, in comparison to traditional PWRs.

The selected structural materials (extensive use of stainless steel, elimination of cobalt alloys, etc.) limit coolant contamination; in addition, the lower coolant operating temperature also reduces the oxidation of all materials that physically interface with the reactor coolant.

Finally, the physical impossibility of fast thermal transients limits thermal stresses on materials and their oxide coating, so that crud detachment is strongly reduced.

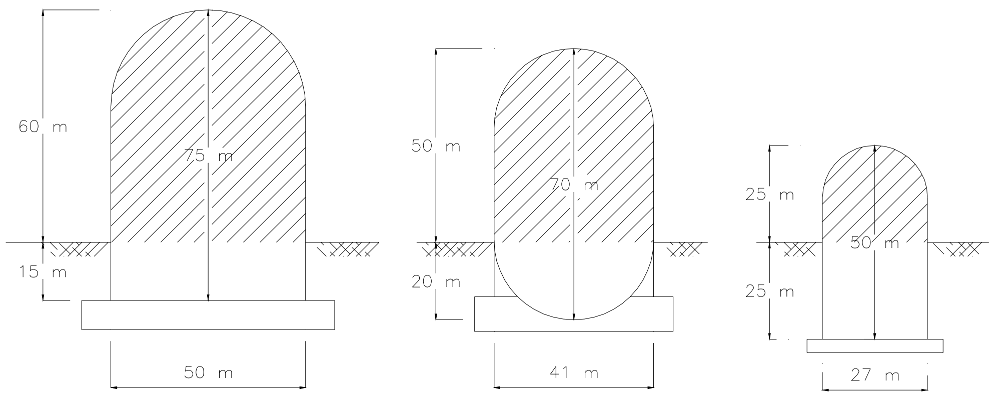
The MARS plant has been developed with the main design criteria of minimizing radiological risks at the site due to accidents, radiological releases to the site during normal operation and minimizing nuclear waste production; other relevant design criteria concern minimizing the impact of the plant on the environment for other reasons including land use requirements and visual impact.

The quantity of radioactive streams treated in the MARS plant is reduced to 80% of the corresponding value in traditional PWRs (with equal thermal power), while the average contamination of fluid streams is reduced to only 5% of the value in traditional PWRs (at equal thermal power).

Solid wastes produced in the plant have a volume reduced by a factor 10 with respect to traditional PWRs (obviously, referred to the same thermal power).

Regarding land use, the MARS nuclear island requires an area covered by buildings of about 1500 m^2 for a 600 MW(th) unit, corresponding to a land use density of 2.5 m^2/MW(th), against traditional values of 4.5 m^2/MW(th); referring to electric power production, the land utilization density is of 8.5 m^2/MW(e) for the MARS plant, against a reference value of 12.5 m^2/MW(e) for other plants (see Figure IV-8).

Contrarily, due to the lower thermal efficiency of the steam cycle achievable with a MARS plant dedicated to electric energy production, the thermal impact on the cooling water is higher for the MARS plant: 3.0 MW(th) released/MW(e), against 1.8 MW(th) released/ MW(e) for traditional PWRs.



Traditional 1000 MW(e) PWR
Total volume = 130 000 m^3

Evolutionary 600 MW(e) PWR
Total volume =74 000 m^3

MARS
Total volume =24 500 m^3

FIG. IV-8. Comparison between reactor buildings of MARS and other PWRs.

The design effort addressing simplification of plant construction and erection produces as a parallel but absolutely not secondary result, a huge simplification for decommissioning.

The most relevant feature of the MARS plant affecting decommissioning concerns the “easy” disassembling of components, thanks to the adoption of flanged connections in the primary loop. Each component (even SG and pressure vessel) may be disassembled for maintenance or substitution in an operation basically the reverse of the assembly. Normal precautions regarding dose limits to personnel are the main differences.

In the MARS plant decommissioning, use of special, complicated techniques or special equipment is unnecessary; in particular, cutting very thick, highly activated or contaminated components is not required, hugely simplifying disassembly operations and reducing wastes.

In addition, the reduced size of all components (the largest may be also disassembled into transportable sub-components) makes them easily removable. The follow-on conditioning operation in a dedicated area of the power station is much simplified and definitely "cleaner".

General plant simplification causes a great reduction in the number of contaminated or activated components (up to 50%, compared with a same size traditional plant), and a corresponding reduction in quantity of radioactive materials. In addition, the selected structural materials, together with use of a cleaner primary coolant, reduces total and specific radioactivity of contaminated materials.

Last, but not least, the sole use inside the MARS containment building of steel structures (working floors, shields, etc., made with steel frames) allows dismantling the plant completely within the reactor building by removing steel components, leaving a "clean" building to be demolished or reused. Most of the metal pieces would be radiologically clean and their free release will be possible.

In Figure IV-9, the disassembling of the steam generator is shown; the possibility of a sub-disassembly of the SG can be noted. First the upper shell is removed and transferred outside the reactor building and then the lower shell and the tube bundle may be removed.

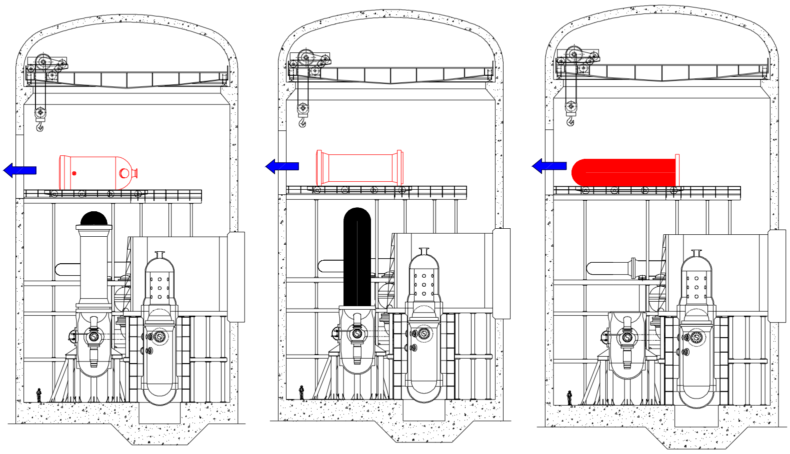


FIG. IV-9. Steam generator disassembling.

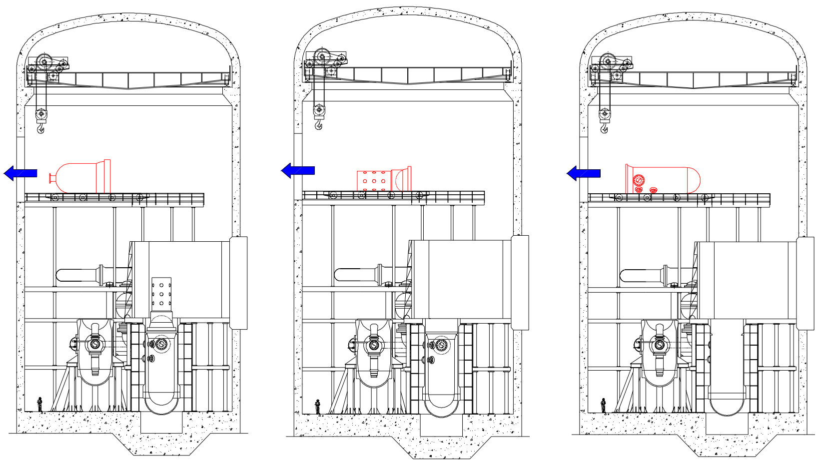


FIG. IV-10. Pressure vessel disassembling.

Figure IV-10 shows the dismantling phases for the pressure vessel and other components within the reactor building.

A detailed analysis of MARS decommissioning costs is in progress but preliminary results indicate the possibility of keeping direct costs of decommissioning lower than 10% of direct construction cost, instead of 15–20% computed by OECD for traditional nuclear power plants.

IV-1.6.3. Safety and Reliability

Safety concept and design philosophy

The principal features of the MARS safety design are:

(1) The use of passive components (most of them, static) to perform safety functions.
(2) Adoption of low fuel temperatures and temperature gradients (both operationally and as maximum conceivable values).
(3) Adoption of low core heat fluxes and of high values for minimum DNBR (both in operation and as expected values in the most drastic design accident scenarios).
(4) A drastic reduction in the number of physical connections between the primary coolant loop and auxiliary circuits.
(5) A drastic reduction of maximum conceivable pressurization of the reactor building.
(6) A reduction of human factors affecting safety systems.
(7) Easy testing of all safety systems.

Provisions for simplicity and robustness of the design

The design of the MARS was fully simplified to utilize few components in simple circuits. The design itself is quite simple and no uncertainties exist regarding possible unknowns arising in the future.

Active and passive systems and inherent safety features

The inherent safety features are: negative reactivity coefficients in all power and coolant temperature ranges; decay heat removal relying on natural fluid circulation only; additional scram system relying on gravity and on thermal elongation of metallic bars.

The passive safety systems are: emergency core cooling system totally passive, with only one non-static component (400% redundant, 200% + 200% in two independent trains, each having two redundant components [each with 100% capacity] manufactured under different conceptual designs).

The active safety systems are: traditional scram system.

Emergency safety functions are conducted only by passive components. Only one non-static component is relevant to plant safety, the check valve (passive component) of the SCCS, which operates with 200% component redundancy in each of the two redundant trains.

Structure of the defence-in-depth

The traditional PWR design approach has been adopted but higher barriers are provided by design (lower fuel temperature; higher resistance margins of fuel cladding and of the primary coolant pressure boundary) to radioactivity release. The primary coolant pressure boundary is enclosed in a pressurized, low-temperature containment, filled with water (CPP).

Design basis accidents and beyond design basis accidents

A complete safety analysis of the MARS nuclear plant has been performed to verify the capability of the plant to guarantee fulfilment of safety objectives and to confront any accidental condition with a frequency of occurrence higher than 1×10^{-7} year^{-1} [IV-3].

This analysis was extended to ascertain the ability of the plant to handle accidental conditions with an even lower frequency but involving severe consequences (severe accidents).

This design is intended to prevent the harmful release of radioactive products into the environment from fuel assemblies; since such a release is possible only if fuel is damaged and fuel damage is possible only if core cooling is jeopardized, all possible situations (accidental sequences) leading to the failure of core cooling were identified.

An analysis was performed of possible transients of thermal-hydraulic parameters in the reactor core by first identifying initiating events leading to variations from standard operating values and then analyzing possible sequences of events resulting from the initiating events, up to identification of combinations of those events finally leading to fuel damage.

This approach considered the unique aspects of the MARS reactor plant with respect to traditional PWRs. So, the HAZOP method was used to identify initiating events, the "fault tree" technique was used to evaluate the probability of failure of non-traditional components or systems and the "event tree" method was used to identify possible evolutions of accidental sequences. Twenty-eight different initiating events, grouped in 8 main categories, were identified and their evolutions were analyzed. The results of the probabilistic safety analysis are summarized in Figure IV-11.

The highest probability of core damage is 2.1×10^{-8}. This figure is lower than the probability of ultra-catastrophic natural events such as, for example, the impact of a meteorite on a large city (such as New York) causing 1 million deaths, was evaluated as 1×10^{-7} yr^{-1}. For this reason, since a common mode failure depending on ultra-catastrophic natural events may be recognized, the core damage probability for the MARS plant should be assumed to be 1×10^{-7} yr^{-1} [IV-4].

The sequence group regarding the loss of electric supply to primary pumps causing core damage has a probability of 3.65×10^{-12} yr^{-1} provided the initiating event is followed by failure of both the automatic and the additional scram systems.

The sequence group regarding LOCAs through the pressurizer safety/relief valves causing core damage has a probability of 2.73×10^{-9} yr^{-1} provided the initiating event is followed by failure of the safety core cooling system through failure of the primary loop interception and the primary pump stop.

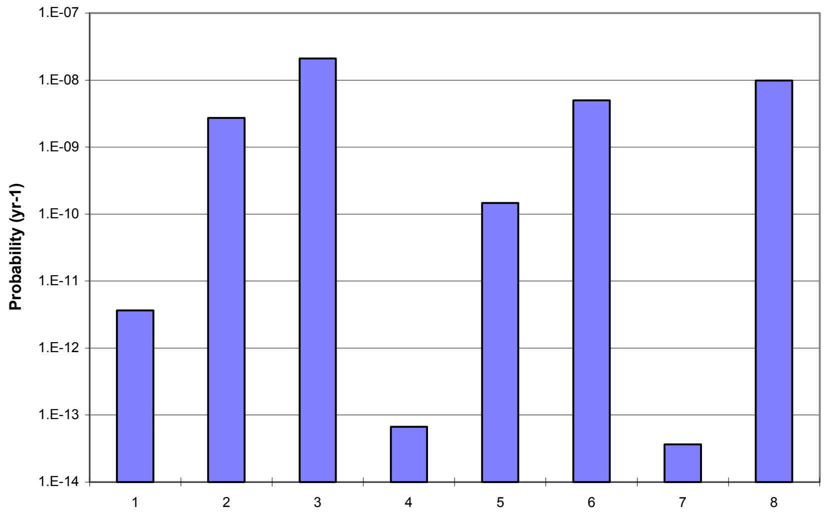
The sequence group mainly regarding the loss of steam generator feedwater as the initiating event and causing core damage has a probability of 2.1×10^{-8} yr^{-1}, if the initiating event is followed by the failure of the safety core cooling system through failure of the primary loop interception and the primary pump stop.

The sequence group regarding the loss of on/off site power causing core damage has a maximum probability of 6.67×10^{-14} yr^{-1} if the initiating event is followed by failure of the safety core cooling system through a failure of check valves or by simultaneous failure of the automatic and the additional scram systems.

The sequence group regarding the loss of coolant from connections with auxiliary systems causing core damage has a probability of 1.46×10^{-10} yr^{-1}, if the initiating event is followed by failure of the primary loop isolation valves and the primary loop interception system.

The sequence group regarding a steam generator tube rupture causing core damage has a probability of 5×10^{-9} yr^{-1} if the initiating event is followed by failure of the safety core cooling system, through failure of the special check valves.

The sequence group regarding the primary pump trip causing core damage has a probability of 3.65×10^{-14} yr^{-1} if the initiating event is followed by the failure of both the automatic and the additional scram systems.



1: Primary pump stop	3: SG exchanged power degradation (loss of SG feedwater)	5: Loss of coolant from auxiliary systems	7: Primary pump trip
2: Relief/safety valves stuck open	4: Loss of on/off site power	6: SG tube rupture	8: Steam line break

FIG. IV-11. Results of probabilistic analysis.

Provisions for safety under seismic conditions

Safety-related components will resist seismic loads under the reference site design. Initiating events for accident scenarios in the area include the crash of military aircraft at the site.

Probability of unacceptable radioactivity release beyond the plant boundaries

Results of the PRA analysis are described above (core fuel melting probability lower than 1×10^{-7}).

Measures planned in response to severe accidents

Even if the probabilistic safety assessment of the MARS plant shows a core damage probability lower than the probability of ultra-catastrophic natural events, core melting has been considered to evaluate the capability of the plant to confront it.

In particular, thermal-mechanical analyses show that the reactor vessel of the MARS plant can guarantee the in-vessel retention and cooling of corium produced by the melting of 60% of core and vessel internals if the following conditions are met:

- Water is present in the CPP with pressure equal to 1 bar and temperatures lower than 100°C.
- The heat transfer coefficient between the external wall of the vessel and the water in the CPP is higher than 1800 $W/m^2\,K$ (to avoid partial melting of the vessel wall itself).

The first condition may be met simply by a depressurizing system for the containment for primary loop protection that, on the other hand, is already foreseen to follow the primary loop pressure in case of depressurization.

The second condition is guaranteed if boiling conditions are reached because heat transfer coefficients on the order of 10 000 W/m^2 K may be reached. Boiling conditions are surely reached if the pressure is reduced to 1 bar, with an acceptable temperature of the external vessel wall.

Therefore, the two requirements above are actually reduced to the first one and fulfilling it requires the adoption of a reliable, possibly passive, depressurization system for the CPP.

As a consequence of the extremely low probability of core damage and the capability of the MARS concept to confront even severe accidents while maintaining the reactor vessel integrity, licensing of a MARS NPP does not require any off-site emergency planning.

IV-1.6.4. Proliferation resistance

A MARS NPP equipped with traditional PWR fuel as described above has no special feature for proliferation resistance. The low enrichment of UO_2 fresh fuel, the residual fissile enrichment of the spent fuel and the total quantity of fuel stored in the plant are aspects that discourage the use of MARS fissile materials for military aims.

Nevertheless, the MARS NPP may easily accept an innovative fuel identified but not described in this paper and characterized by longer irradiation times, that could permit plant management by a skilled contractual vendor who brings the fuel to the site for energy production and then takes it back at the end of the irradiation.

IV-1.6.5. Technical features and technological approaches used to facilitate physical protection of MARS

The MARS plant has been designed to face external natural and human-originated events.

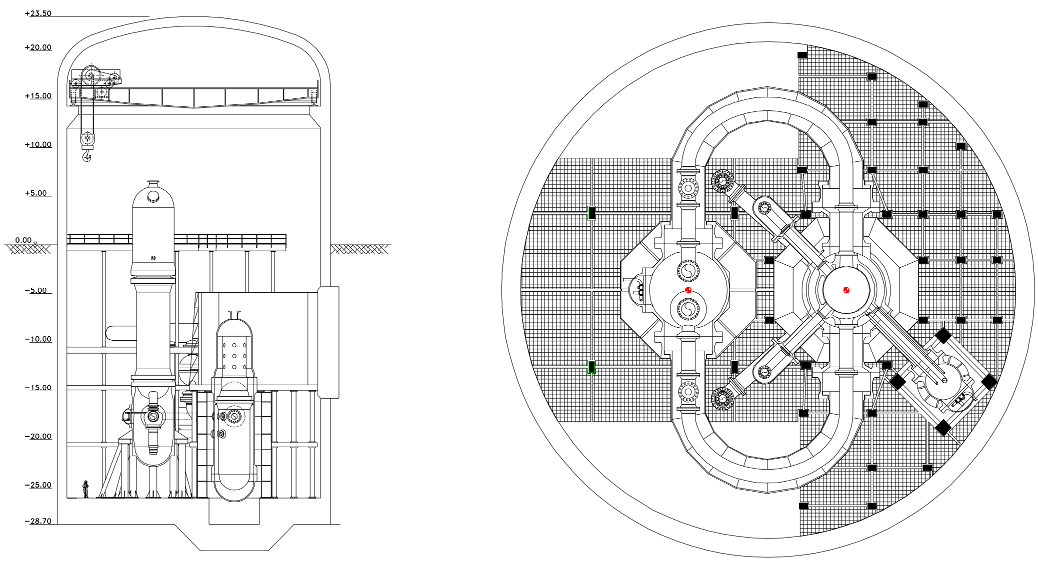


FIG. IV-12. MARS reactor building.

According to Italian and European regulations, a containment building (reactor building) to protect the nuclear island is envisioned; in the case of MARS (see Figure IV-12), a large

portion (about one half the height) of the reactor building is located under the ground level, thus minimizing the surface over the ground level and improving resistance.

The reactor building can face all external natural events such as earthquakes or tornadoes and impacts generated from them (mainly missiles caused by a tornado).

The following human events potentially interacting with the reactor building have been considered:

- A plane pressure wave due to external explosions.
- Aircraft impact.

In the MARS plant, an additional barrier is envisioned between the core and the external environment: it is the CPP, the metallic enclosure of the primary loop. It is constructed of thick steel to improve physical protection of the primary loop.

Regarding protection against internal events, the entire plant is designed to protect personnel and the external environment against radiological hazard. To summarize, the main technical features to protect the plant are:

- Effective in-depth defence against radiological hazard for personnel and population.
- Capability of the plant to manage any kind of accident, including severe accidents.
- Reliance on physical laws to not only detect failures but perform safety actions, which means extensive use of “inherent” and “passive” safety features.
- “Insensibility” of plant safety design to human errors and faulty actions.
- Extremely low doses to personnel.
- Very low production of nuclear wastes.

IV-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of MARS

The following consequences may be recognized, that could facilitate the deployment of MARS:

- No requirements for special, sophisticated manufacturing technology: the technology for the nuclear design is available in many industrial organizations; the technologies for plant systems are widely disseminated.
- The manufacturing capability of industries involved in a MARS NPP construction is already available in thousands of companies through the world; many industries in developing countries may have access to the business of plant construction.
- The energy produced is cheap; operation of the plant is easy and safe: it is “human-error-resistant”.
- The load factor is quite high; the plant life is long; the plant can accommodate continuous nuclear technology development over the years.
- Plant behaviour in accidental conditions is easy to understand; the proposed prototype provides a record of the behaviour of the plant safety systems; licensing authorities may easily acquire a complete confidence with the plant. No emergency plan is needed.
- Component behaviour certification may be obtained in shop, under optimal operating conditions.
- The plant minimizes waste production and personnel doses.
- The final decommissioning of the plant is easy, with a “green field” obtained in a short time.

IV-1.8. List of enabling technologies relevant to MARS and status of their development

MARS plants undoubtedly belong to the "generation III+" family for the market niche of small electric networks and cogeneration. The technologies utilized are basically proven and refer to NPPs widely used in the world (PWRs). The technological aspects of the innovative solutions should be analysed "hand-in-hand" with industrial companies to identify the best economic manufacture of plant components.

The innovations consist mainly of new component coupling and utilization of simple, passive-type circuits. For all of these, special experimental testing was conducted or is in progress.

Dedicated experimental facilities were built to check:

- The innovative, passive-type, quasi-static emergency core cooling system, based on natural circulation of cooling fluids and using external air as the ultimate heat sink.
- The innovative, additional, passive-type scram system based on a two-metal core temperature sensor and operated by gravity.
- Complete enclosure of the primary-coolant boundary in a pressurized containment filled with low enthalpy water (primary loop jacket); this experimental activity is being carried out at the date of the issue of this report.

The systematic utilization of inherent and passive safety features had made it possible to greatly simplify the MARS NPP design, bringing the following consequences:

- Drastic reduction in the number of physical connections between the primary coolant loop and auxiliary circuits (the large number of ECCS trains typical of traditional PWRs was eliminated).
- Drastic reduction of the reactor building maximum conceivable pressurization (apart from the reduction in the probability of core melt, the quantity of cold water in the CPP limits reactor building pressurization to vanishing values).
- Reduction of human factors affecting safety systems (a few components are involved in plant operation and safety management; safety-related component performance cannot be jeopardized by human error).

IV-1.9. Status of R&D and planned schedule

A summary of activities already performed and to be completed in the overall design of the MARS plant follows: The nuclear design of the core and of the reactivity control systems has been completed.

- The design of the primary coolant system has been completed.
- The design of the passive-type emergency core cooling system has been completed.
- The mechanical design of the additional, passive-type scram system has been completed.
- The design of main NSS auxiliary systems has been completed.
- The mechanical design of the advanced solutions proposed for traditional components has been completed.
- The design and verification of the reactor building and internal supporting structures have been completed.
- The analysis of wastes produced has been completed.

- The HAZOP analysis and the probabilistic safety assessment of the plant have been completed.
- The safety analysis regarding any type of nuclear accidents has been completed.
- The cost analysis of the produced energy has been completed.
- The study of coupling of the NSS system to a cogeneration system including desalination has been completed.
- The decommissioning program of the plant will be completed.
- A preliminary safety assessment report (PSAR) was officially submitted to the Italian Nuclear Safety Authority.
- The seismic qualification of the innovative mechanical solutions will be completed in a couple of years.
- The design of minor components or systems (such as the biological shield) will be completed by the end of 2005.

By the end of 2006, the MARS basic design, the detailed design of the main innovative mechanical components and the technical specifications for most of the fluid systems will be completed on the basis of the only work forces of the Department of Nuclear Engineering and Energy Conversion of the University of Rome “La Sapienza”.

Furthermore, under favourable financing and with the cooperation of a suitable industrial entity, the final design for a prototype could be completed in two years.

The construction of a prototype plant will require less than four additional years.

IV-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

The MARS concept relies to a high degree on experimentally and analytically proven design solutions and may not require a substantial amount of further R&D.

Many experimental facilities, both small and large, have been used to validate behaviour and performance of the components assuring passive safety or involving innovative features.

Most experimental facilities even at the research laboratories were designed, manufactured and operated according to “industrial promotion program procedures”, making them applicable as industrial tests.

Among them, the following deserve mention:

- MORIS facility, built at the ENEA research centre at Casaccia at the start of the plant design to simulate the general thermal-hydraulic behaviour of the MARS primary cooling system and of the emergency core cooling system.
- CIVAP facility, built at the ENEA research centre of Casaccia to validate a half-scale prototype of the SCCS innovative check valve.
- QUSCOS facility, built at the Department of Nuclear Engineering and Energy Conversion, University of Rome “La Sapienza” to evaluate the effect of the shape and material characteristics on the performance of the innovative heat exchanger located inside the SCCS pool.
- COTINCO facility, built at the same Department to evaluate the effect of non-condensables on the performance of the atmospheric condenser of the SCCS.
- ITASCO facility, built at the same Department to evaluate the effect of tube characteristics on the performance of the atmospheric condenser of the SCCS.

- NICOLE plant, a large-scale (about 15 m height) facility built at the ENEA research centre of Casaccia to analyse phenomena related to natural circulation in the primary loop and boiling and condensation in the pool loop and to validate the design computer codes.

An experimental facility to qualify innovative mechanical solutions of the MARS plant from a dynamic, and in particular seismic, point of view is being built at the Department of Nuclear Engineering and Energy Conversion.

Though the authors do not deem construction of a prototype as necessary, it should be noted that because of innovations in the MARS concept related to strong reliance on the inherent and passive safety design options, there actually may be a need to construct a prototype to demonstrate reliability of a MARS NPP operation before licensing it as a commercial project.

IV-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

No information was provided.

IV-2. Design description and data for MARS

IV-2.1. Description of the nuclear systems

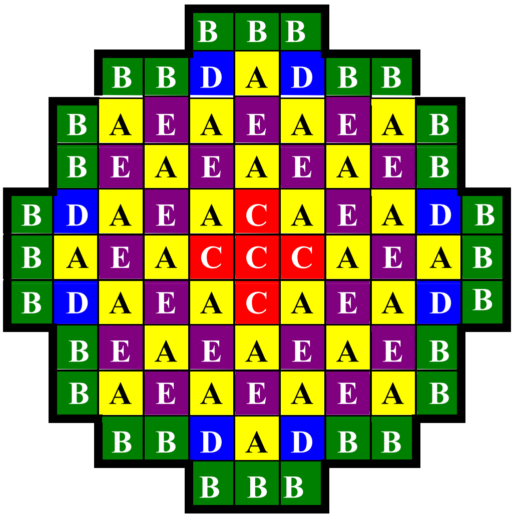
Reactor core and fuel design

Proven technological features of the most widely operated PWRs (geometry and materials of fuel rods and assemblies; layout of the primary cooling system; geometry and materials of control rods and drive mechanisms; core internals; etc.) were chosen for the basic design of the (primary) reactor coolant system (RCS). The core and fuel design data are presented in Table IV-5 and Figure IV-13.

TABLE IV-5. MARS REACTOR CHARACTERISTIC DATA

Fuel bundles	89
Fuel rod array	17×17
Fuel rods per fuel bundle	264
Fuel rod external diameter (cm)	0.95
Fuel rod active length (cm)	260
Fuel rod cladding thickness (mm)	0.63
Fuel rod pitch (cm)	1.26
Control rod guide tube diameter (cm)	1.224
Control rod guide tube thickness (mm)	0.4
Control rod diameter (cm)	0.978
Core heat transfer surface (m^2)	1823
Core average linear power density (W/cm)	98.2
Core average heat flux (W/cm^2)	32.9
Core average volumetric power density (kW/litre)	56.5

A 1/3 core loading strategy for more than 18 months is compatible with fresh fuel enrichment with ^{235}U lower than 2.8% (20 and 24 burnable poison locations) with a maximum boron concentration (without xenon) lower than 711 ppm.



A 1.6 % (0 burnable poisons)
B 2.2 % (0 burnable poisons)
C Recovery 1.6 % (8 burnable poisons)
D 2.8 % (20 burnable poisons)
E 2.8 % (24 burnable poisons)

FIG. IV-13. Fuel loading strategy.

Primary coolant system

The innovative steam generator of the MARS NPP is shown in Figure IV-14. The special design allows the easy substitution of the full tube bundle.

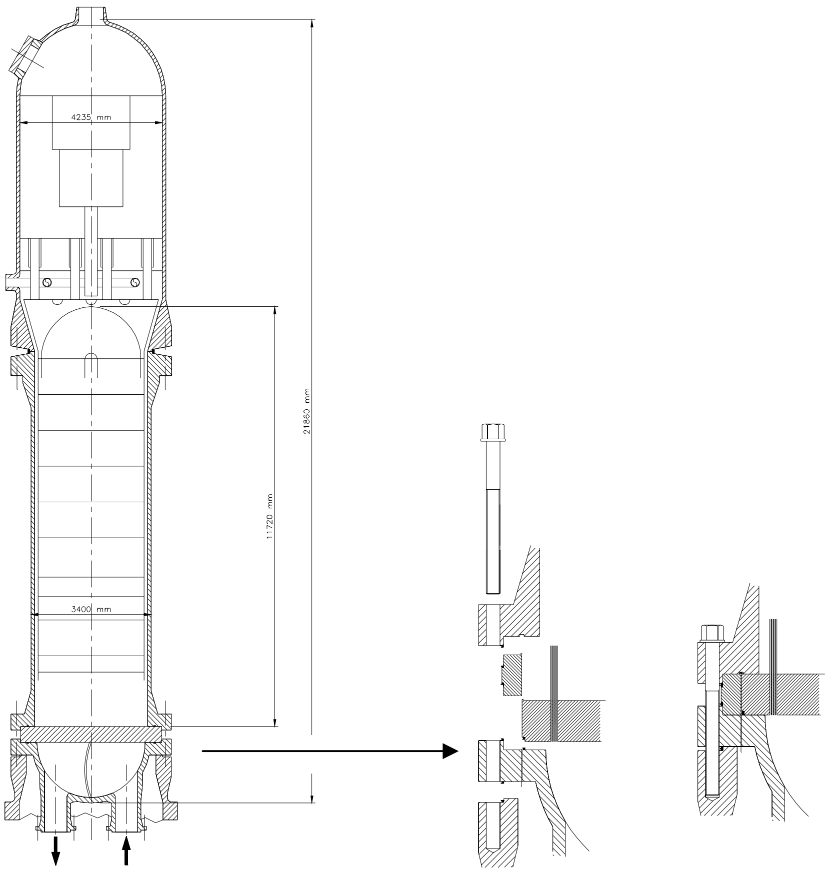


FIG. IV-14. Steam generator design.

Detailed indications of materials selected and specifications of reactor pressure vessel, steam generator and pressurizer parameters are listed in Table IV-6 through IV-9.

TABLE IV-6. REACTOR VESSEL DATA

Vessel material	Low alloy carbon steel
Cladding material	Stainless steel
Design temperature	290°C
Internal diameter of the shell	3000 mm
Internal radius of the bottom head	1500 mm
Internal radius of the upper head	1410 mm
Internal design pressure	83 bar
Length of the cylindrical shell	8056 mm
Upper head thickness	80 mm
Bottom head thickness	80 mm
Nominal cladding thickness	5 mm
Operating pressure	75 bar
Operating temperature	254°C
Overall length of the assembled vessel	11 091 mm
Shell thickness	120 mm
Total weight (approximate; dry)	88 000 kg

TABLE IV-7. STEAM GENERATOR DATA (PRIMARY SIDE)

Overall height	20 800 mm
Upper shell outer diameter	4235 mm
Lower shell outer diameter	3400 mm
Operating pressure, tube side	75 bar
Design pressure, tube side	83 bar
Design temperature, tube side	280 °C
Full load pressure, shell side	18.8 bar
Design pressure, shell side	83 bar
Reactor coolant flow rate	3227 kg/s
Reactor coolant inlet temperature	254 °C
Reactor coolant outlet temperature	214 °C
Tube type	U-TUBES 3/4" BWG 18
Number of tubes	5800
Tube bundle total height	11 720 mm
Tube bundle diameter	3000 mm
Tube material	INCONEL 600

TABLE IV-8. PRESSURIZER DATA

Overall height	11 968 mm
Inside diameter	1688 mm
Shell thickness	65 mm
Total internal volume	25 m^3
Water volume	15 m^3
Steam volume	10 m^3

TABLE IV-9. MATERIALS USED IN THE MARS PRIMARY COOLANT SYSTEM COMPONENTS

ITEM	MATERIAL
Reactor vessel	
Upper head, shell plates and lower head	SA 533 Grade B Class 1
Flanges and nozzle forgings	SA 508 Class 2
Closure studs, nuts, washers, inserts	SA 540 B23 Class 3
Cladding	AISI 308/309
Lower and upper core plate	AISI 304
Neutron and thermal shields	AISI 304
Core barrel	AISI 304
Baffle	AISI304
Steam generator	
Pressure plates	SA533 Grade A Class 2
Nozzles and tube sheet	SA508 Class 2
Bolts and nuts	SA540 B23 Class 3
Tube bundle	Inconel 600
Pressurizer	
Pressure plates	SA533 Grade A Class 2
Nozzles	SA508 Class 2
Bolts and nuts	SA540 B23 Class 3
Reactor coolant pump	
Pressure forgings	SA182 F316
Pressure castings	SA351 Grade CF8A
Bolts	SA540 B23 Class 3
Reactor coolant piping	
Tubes	SA351 Grade CF8A
Flanges	SA 508 Class 2

Safety core cooling system

Figures IV-15 and IV-16 show the innovative check valve designed and patented for the MARS SCCS and the SCCS primary heat exchanger.

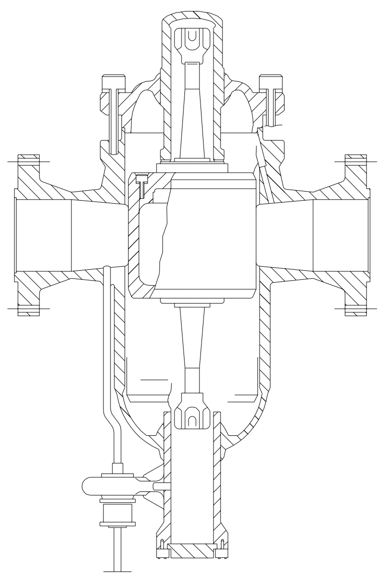


FIG. IV-15. SCCS special check valve.

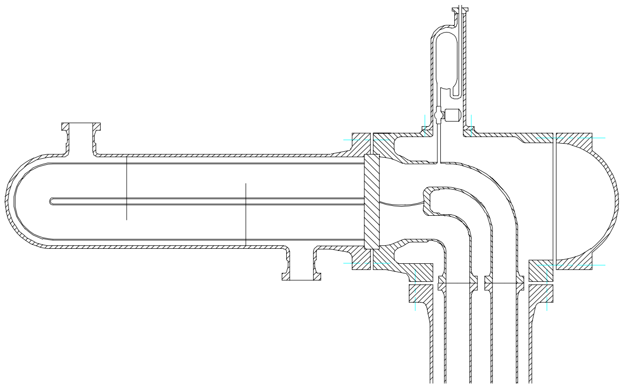


FIG. IV-16. SCCS primary heat exchanger.

Main heat transport system, with specification of heat removal path in normal operation and in accidents

The main heat transport system in normal operation includes the primary coolant system and the secondary coolant system. In emergency conditions, the main heat transport system includes the primary coolant system and the safety core cooling system (SCCS). Figure IV-17 shows the heat path in normal and emergency conditions.

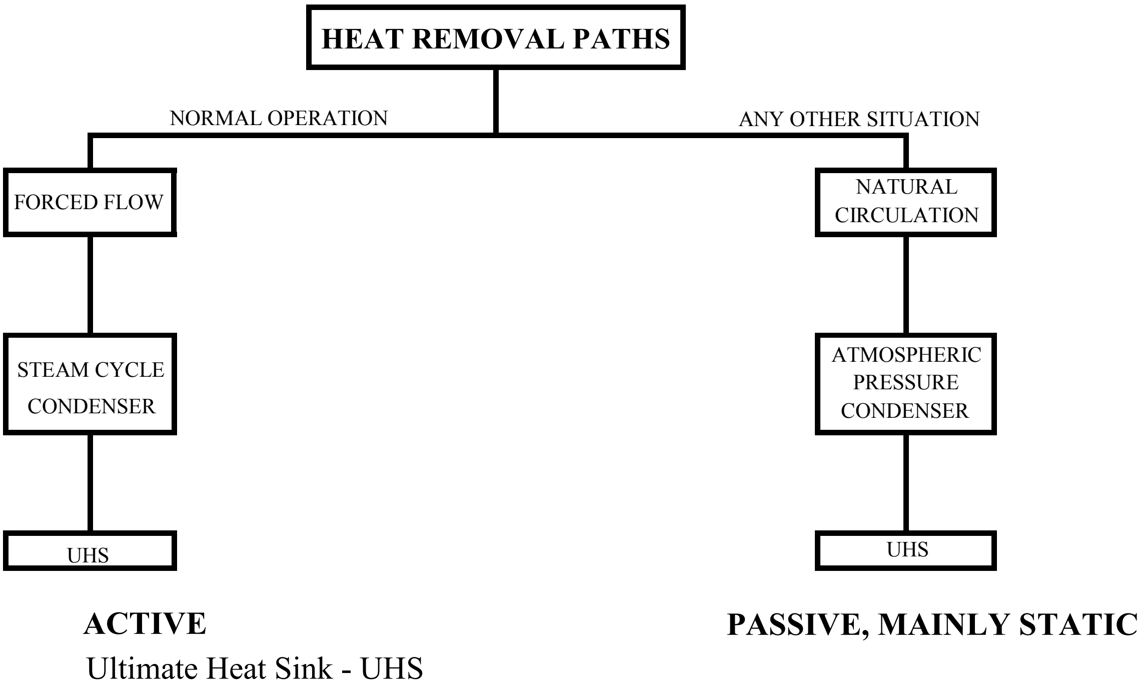


FIG. IV-17. Heat removal paths in the MARS NPP.

Auxiliary systems

The main auxiliaries of the nuclear steam supply system irrelevant to safety are:

- The volumetric control system (VCS), which allows volumetric control of the primary coolant.
- The primary coolant cleaning system (CCS) and the chemical additive control system (ACS), which allow control of the chemical characteristics of the primary coolant.
- The residual heat removal system (RHR), which allows cooling of the primary coolant during shutdown conditions.

These systems are greatly simplified due to the typical parameters of the primary coolant and the core. In fact, the large amount of primary coolant inventory, together with the large volume of the pressurizer, strongly limits the requirement for the VCS, while the RHR operation is required only during refuelling.

In addition, the absence of the possibility of fast nuclear transients, together with the considerable thermal inertia of the reactor coolant system makes thermal transients slow, thus limiting thermal stresses both on structural materials and on oxide coatings.

These features allow a drastic reduction of the chemical and radioactive contamination of primary coolant, leading to a simplification of the CCS and of the ACS.

IV-2.2. Description of the turbine generator plant and systems

The characteristics of steam produced in the MARS steam generator are listed in Table IV-10.

TABLE IV-10. STEAM GENERATOR DATA (SECONDARY SIDE)

Steam pressure	18.5 bar
Steam temperature	209.5°C
Steam moisture	0.25 %
Steam enthalpy	27898.7 kJ/kg
Steam flow	277.6 kg/s

The thermal cycle of the MARS NPP is shown schematically in Figure IV-18. In Figure IV-19, the expansion line of steam in the turbine is shown. Overall cycle data are reported in Table IV-11.

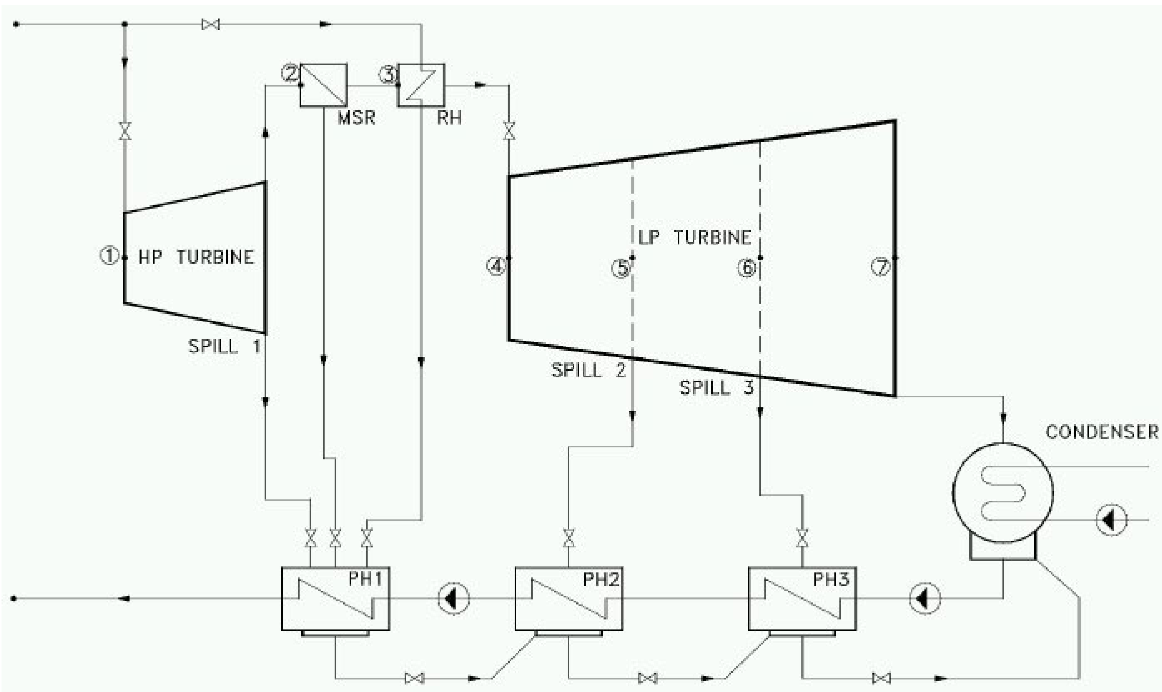


FIG. IV-18. Steam cycle.

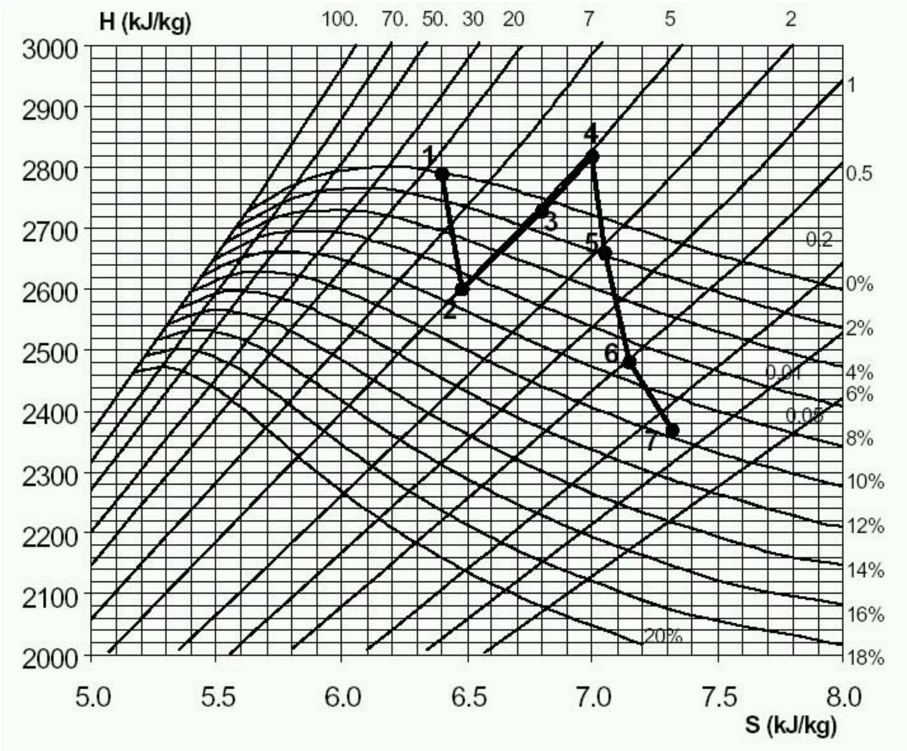


FIG. IV-19. Steam expansion line.

TABLE IV-11. OVERALL CYCLE DATA

Steam generator thermal power	597 181	kW
Condenser thermal power	444 576	kW
Gross power	149 070	kW
Mechanical losses	752	kW
Electric losses	2236	kW
Electric power	146 100	kW
Heat rate	14 718	kJ/kWh
Efficiency	24.5	%

The characteristics of saturated steam produced by the steam generator (in particular the low pressure) require intermediate dehumidification and superheating to obtain an acceptable value of cycle efficiency. The intermediate dehumidification and superheating allow a low steam pressure at the turbine exhaust with humidity within the limit of 9.8%. The preheating line for the regenerative heating of the steam generator feedwater by steam draining includes a high pressure preheater and two low-pressure preheaters.

The turbine is a tandem-compound type, with a high pressure, double-flow multistage drum and a low-pressure double-flow multistage drum. The low-pressure section is equipped with two steam spillages.

Design criteria for electric power system

Safety features of the MARS nuclear reactor do not require the intervention of energized components to perform safety actions; therefore the electric power system has no role in safety as in traditional nuclear power plants. In the MARS plant, very few electric users are considered relevant to safety; they do not require big power loads and include only

instrumentation systems. For these users, the electric supply is also guaranteed in case of accidents, as in the loss of on-off site power accidents.

The electric supply and distribution system will be designed according to the following main criteria:

- To assure operation of all users relevant to general plant safety in all plant conditions.
- To implement protective actions with selective criteria, in case of faulty events; protection shall be designed to be actuated upstream of faulted components.

The electric supply system is divided into separate systems guaranteeing different levels of electrical continuity (see Figure IV-20). In the MARS project, three main user categories are foreseen:

- “A” class plant users (such as the protection system), for which an emergency electrical supply is required.
- “B” class plant users, which represent the whole of the electric users of the nuclear steam supply system (NSSS) and of its auxiliaries, with the exception of the "A" class users.
- Plant service users, which represent plant services and auxiliaries not relevant to nuclear safety, for which electrical supply interruptions do not compromise plant safety management.

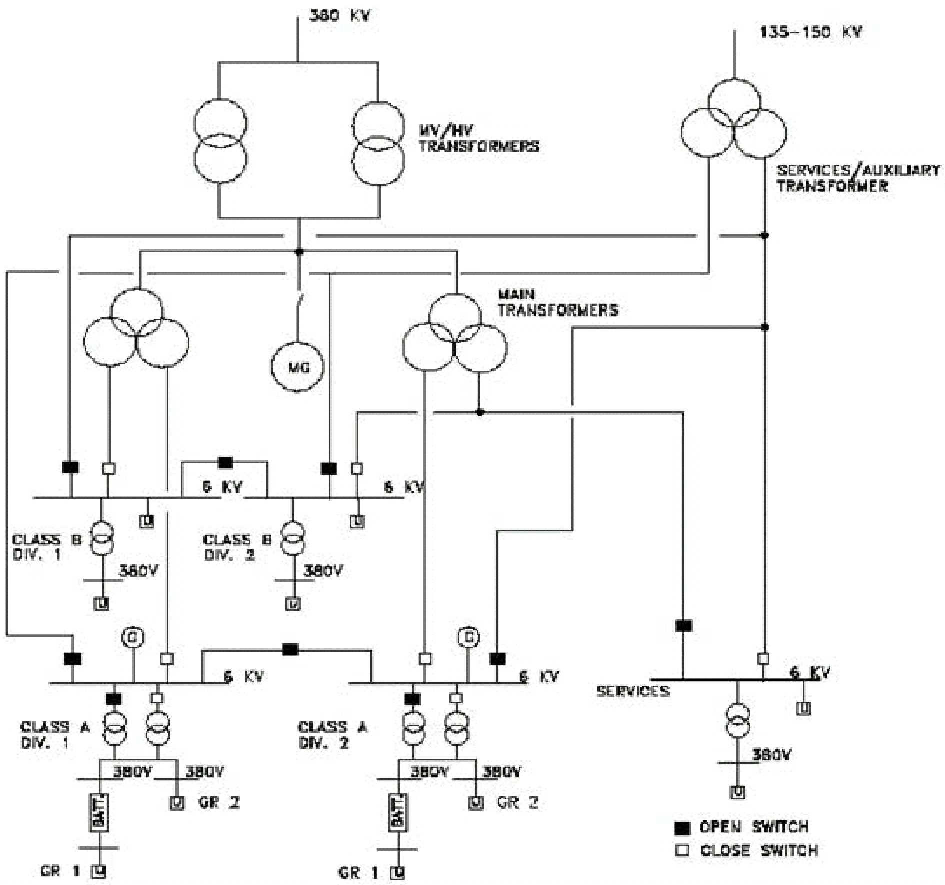


FIG. IV-20. Electric power system scheme.

IV-2.3. Systems for non-electric applications

The cogeneration schemes examined included a steam turbine cycle fed by steam generated in the SG of the MARS NSSS, operating in counter-pressure mode, and a pressurized-condenser

feeding either a multi-effect distillation (MED) desalination system or a district heating system. Since no special safety aspect is involved, traditional plant schemes may be applied.

IV-2.4. Plant layout

The MARS reactor has been designed for plant modularity, using several reactor modules and a unique balance of the plant (see Figure IV-21).

1–2–3	Reactor building
4	Turbine building
5	Fuel way
6	Spent fuel building
7	Laboratories
9–10–11	Auxiliary building
12–13–14	Control room
15	Power station
16–17–18	Motor power center
19–20–21	Wastes building

22–23–24	Dressing and decontamination
25	Backup diesel electric supply system
26	Workshop for activated components
27	Chimney
28	Offices
29	Workshop
30	Medical center
31	Wastes storage

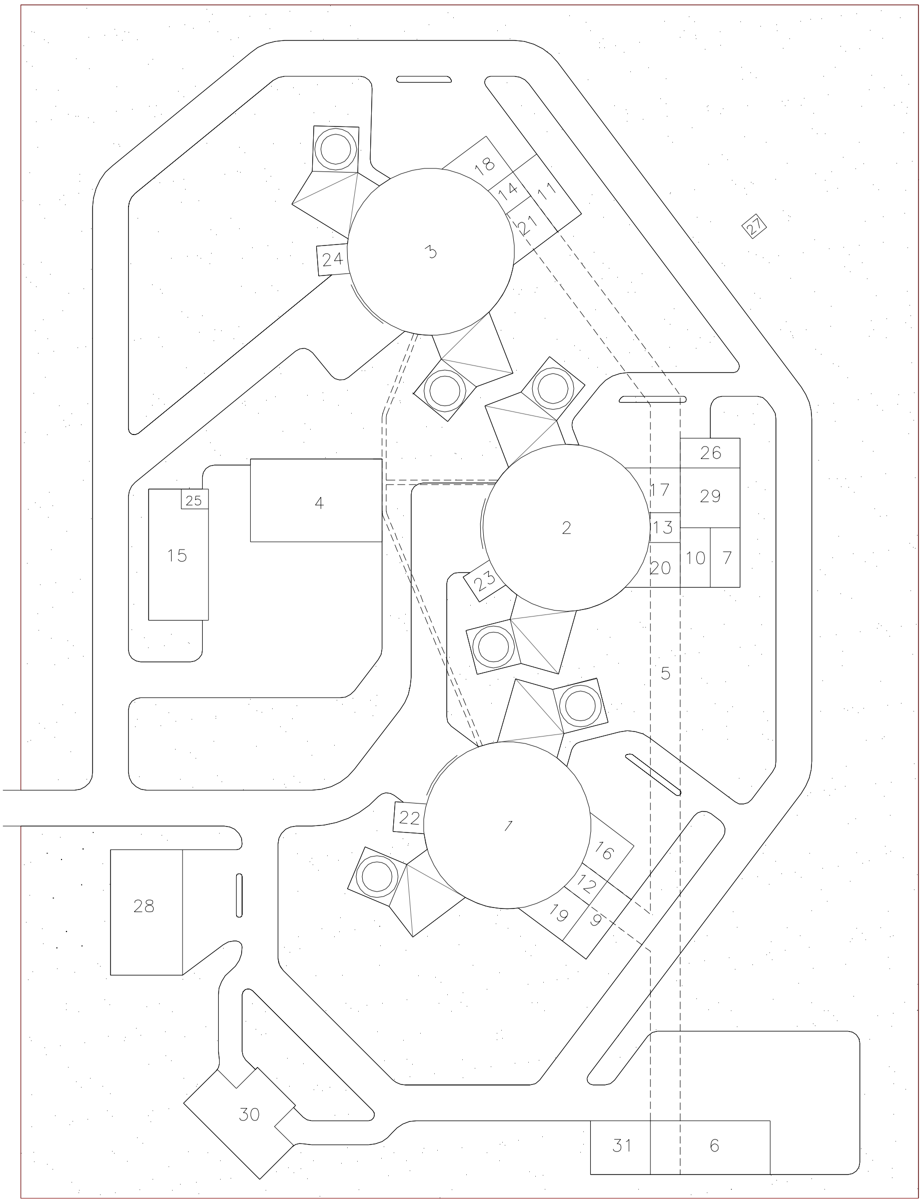


FIG. IV-21. Three reactor-modules plant layout.

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SIMPLE COMPACT REACTOR (SCOR)

CEA, France

V-1. General information, technical features and operating characteristics

V-1.1. Introduction

Since 2000 the Simple Compact Reactor (SCOR) concept has been developed in line with the European utility requirements for enhanced reliability and safety and improved economics for next generation reactors. The advanced SCOR concept relies on proven light water reactor technology.

The SCOR design is based on a compact reactor vessel that contains all the reactor coolant system components, including the pressurizer, reactor coolant pumps, control rod drive mechanism and dedicated heat exchangers of the passive decay heat removal systems. The single steam generator is located above the reactor vessel.

This compact reactor configuration uses a small, inert pressure suppression containment resulting in a high level of safety and economic attractiveness. The SCOR reactor development has applied a “safety by design” approach to reduce or eliminate the consequences of most accident sequences. Fuel assemblies have standard French PWR features. To take advantage of the small power density and improve plant availability, a simplified concept of the safety systems was developed, enabling an extension of the interval between maintenance shutdowns to 24 months or more.

The SCOR is a 2000 MW(th) integrated pressurized light water reactor (PWR); the design for the reactor was developed at the Nuclear Energy Division of Commissariat à l’Energie Atomique at Cadarache, France.

V-1.2. Applications

The SCOR is mainly developed for electricity generation, providing competitive costs compared with large sized reactors through system simplification and compactness in plant layout. However, the SCOR may be used in cogeneration schemes like seawater desalination using low temperature processes, such as thermo-compression or multi-effect distillation.

V-1.3. Special features

The SCOR is an integrated reactor with new features with respect to the design of standard integrated reactors, which usually contain several modular steam generators inside the vessel. This architecture generally leads to the design of a large vessel limiting the size of the reactor to a maximum 1000 MW(th). In the SCOR concept, the steam generator is located above the vessel and acts as the vessel head. This component layout gives space inside the vessel to increase the core size and therefore, with the same safety advantages (elimination of large LOCA) the SCOR power is twice as high as the maximum power of a standard integrated reactor.

Passive safety characteristics allow the SCOR to respond safely to all incidents within the design basis with few operator actions. Except LOCA, where low electric power is needed at

mid-term, no alternative current power is required to manage the other incidents. Most of the design extension conditions are eliminated or passively managed as accidents within the design basis. This simplifies the scope of operator training, equipment qualification and surveillance to meet safety requirements.

V-1.4. Summary of major design and operating characteristics

Installed capacity

Power plant output, net	630 MW(e)
Reactor thermal output	2000 MW(th)

Mode of operation

The plant control scheme will be specifically designed for operation with the single steam generator and will be based on the "reactor follows plant load" strategy. The SCOR design is similar to French standard PWRs and can withstand the following operational occurrences without generation of a reactor trip:

- ±5%/minute ramp load change within 30% and 100% power;
- ±10% step load change within 30% and 100% power;
- 100% generator load rejection;
- 100-50-100% power level daily load follow over 90% of the fuel cycle life;
- Grid frequency changes equivalent to 10% peak-to-peak power changes at 2%/minute rate;
- Loss of a single feedwater pump.

Load factor/ Availability

With this simple plant layout, reductions in the safety systems and the long fuel cycle duration, the expected availability is higher than 91%.

Summary of major design characteristics

Active core height	3.66 m
Equivalent core diameter	3.04 m
Fuel material	UO_2
Fuel inventory	82.2 t UO_2
Average linear heat rate	12.9 kW/m
Average fuel power density	24 kW/kg UO_2
Average core power density (volumetric)	75.3 kW/l
Thermal heat flux	430 kW/m²
Heat transfer surface in the core	4528 m²

Simplified schematic diagram

The SCOR is a pressurized water reactor featuring a compact circuit layout typical of the integrated reactor instead of the standard PWR type loop configuration [V-1]. The main primary system components (core, pressurizer, reactor coolant pumps, control rod drive mechanism (CRDM) and the heat exchangers of the decay heat removal system) are located inside the reactor pressure vessel. The single steam generator acts as the reactor vessel head as shown in Fig. V-1. Water flows upward through the core and the riser region (defined by the extended core barrel) and through the centre of the pressurizer. At the top of the vessel, the fluid flows upward and downward through the U-tubes of the steam generator. At the outlet of

the steam generator, the coolant is collected in an annular plenum where the suction of the reactor coolant pumps is located.

The coolant is directed downward through a venturi and through heat exchangers devoted to decay heat removal. The flow path continues down through the annular downcomer region to the lower plenum and then back to the core.

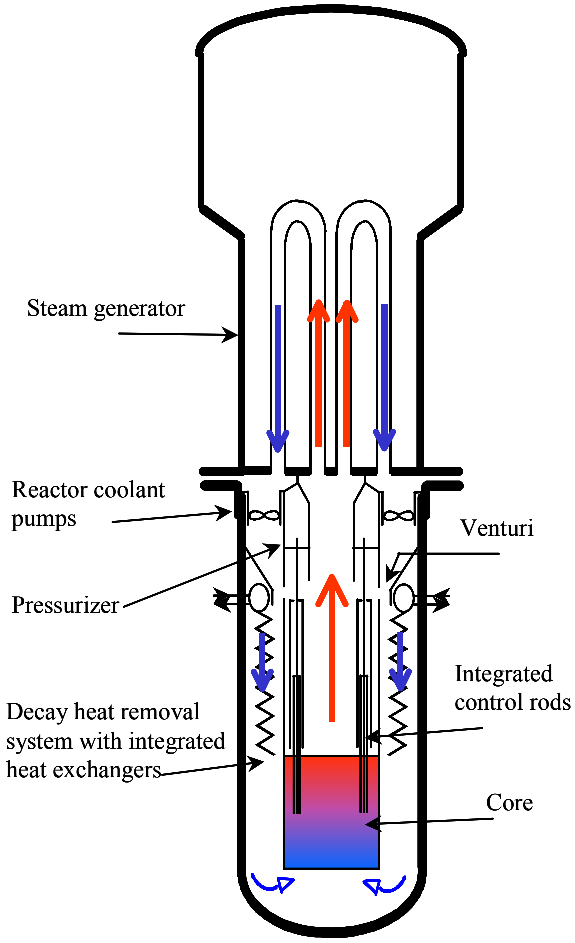


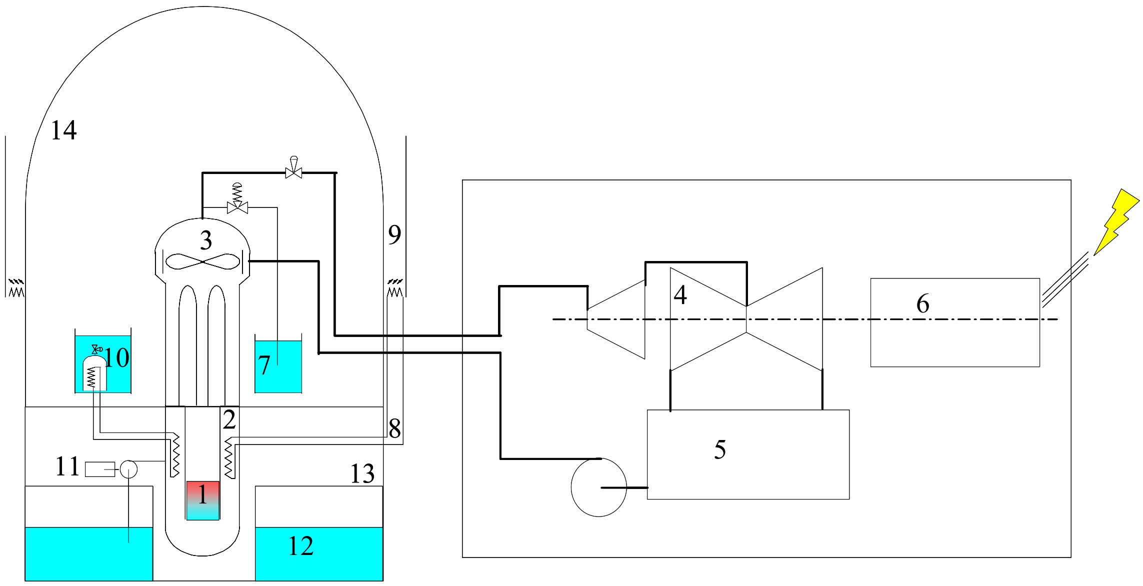
FIG. V-1. Primary coolant system.

The integrated primary circuit layout eliminates large primary vessel penetrations thus excluding the possibility of large breaks and associated large-break loss of coolant events. The number of other pipes is also reduced limiting the probability of occurrence of small breaks and small-break loss of coolant events. The integrated reactor coolant system pressure boundary acts as a barrier against the release of radioactivity generated within the reactor and is designed to provide a high degree of integrity throughout the operation of the plant.

A simplified schematic diagram of the SCOR plant is shown in Fig. V-2.

According to the results from low-pressure PWR studies [V-2], the SCOR operates at a pressure of 8.8 MPa. The main reasons for this selection are:

- A considerable reduction in the pressurized component thickness (reactor vessel, steam generator (SG), etc.);
- A potential increase in fuel burn-up (less cladding corrosion); and
- Simplification of the safety systems.



1	Core	8	Residual heat removal system on primary circuit (RRP)
2	Reactor vessel	9	Air-cooling tower of the RRP
3	Steam generator	10	Heat sink pool of the RRP
4	Turbine	11	Low pressure safety injection system
5	Condenser	12	Pool of the wetwell
6	Generator	13	Primary containment (drywell)
7	Steam dump pool	14	Containment building

FIG. V-2. Schematic diagram of the SCOR plant.

In the secondary circuit, the pressure at the turbine inlet is 3.0 MPa, which is lower than in standard PWRs; the net thermodynamic efficiency is also slightly lower.

The SCOR design is based on well-proven nuclear reactor technologies; major innovations concern safety and auxiliary systems. The innovative solutions of SCOR are the following:

- Elimination of large diameter connections on the reactor pressure vessel.
- Passive and integrated emergency core cooling systems based only on natural circulation and using external air as the ultimate heat sink.
- Operation with a soluble-boron-free core and with control rod mechanisms integrated in the vessel.
- A low power density, enabling a large operating margin (i.e. DNBR) in core power parameters.
- Reduction of the reactor building maximum pressurization.
- Reduction of human factors affecting the safety systems.
- Easy testing and maintenance of all safety systems.

Neutron-physical characteristics

No information was provided.

Reactivity control mechanism

Reactivity control is achieved through the use of control rods with in vessel drives; no soluble boron system is foreseen. To reduce the reactivity at the beginning of the cycle, the loaded portion of fuel contains burnable poison. As in standard PWRs, the rod clusters are moved in guide thimbles but since the steam generator acts as a vessel head, there is no possibility of

using an external mechanism to move the rod clusters. The control rod drive mechanism (CRDM) appears as an integrated hydraulic system. There is around one control rod cluster per two assemblies; this choice is sufficient to control the reactivity from full power to cold shutdown. In accident conditions, redundancy is achieved by another device, called the MP98 system; this system enables movement of a liquid neutron absorber in dedicated tubes in the guide thimbles of the assemblies without rod clusters. Main characteristics of the reactivity control systems are as follows:

Burnable absorbers	Yes
Number of control rods	78
Absorber rods per control assembly	24
Drive mechanism	Hydraulic
Soluble neutron absorber	No
2^{nd} system for accidental conditions	Liquid neutron absorber (MP98)

Cycle type

SCOR is an indirect cycle PWR with the following thermodynamic cycle parameters:

Absolute turbine inlet pressure	3.0 MPa
Absolute pressure at the condenser	5.0 kPa
Power plant efficiency, net	31.5 %

Thermal-hydraulic characteristics

Main thermal-hydraulic characteristics of the SCOR are the following:

Primary coolant flow rate	10465 kg/s
Reactor operating pressure	8.8 MPa
Coolant inlet temperature, at core inlet	246.4°C
Coolant outlet temperature, at riser outlet	285.4°C
Mean temperature rise across core	39.5°C
Primary circuit volume, including pressurizer	278 m^3
Number of coolant loops	0 (Compact reactor coolant system)
Steam flow rate at nominal conditions	987 kg/s
Feedwater flow rate at nominal conditions	987 kg/s
Steam temperature/pressure	237/3.2 °C/MPa
Feedwater temperature	183 °C

Maximum/average discharge burn-up of fuel

The average discharge burn-up of fuel is 45 000 MW·d/t.

Fuel lifetime/period between refuellings

The operating cycle length (fuel cycle length) is 24 effective full power months.

Design basis lifetime

The plant has a design life of at least 60 years with no replacement of the reactor vessel.

Economics

The capital cost is 10% below the specific capital cost of a standard large loop-type PWR; the reduction of the cost of the kW·h is about 13% against the classical PWRs.

V-1.5. Outline of fuel cycle options

The reference fuel characteristics are similar to those of conventional French PWRs. The reduction of the average linear power density (by about 25 percent compared with current PWRs) improves the thermal margin and provides operational flexibility, enabling longer fuel cycles and increases in overall plant capacity. The low core power density may allow use of alternative fuel cycles, e.g. MOX fuel, advanced fuels with increased burn-up, etc.

The SCOR reactor design can also accommodate alternative fuel cycles with an adapted external infrastructure. For example, fuel reprocessing could be similar to that of current French PWRs.

V-1.6. Technical features and technological approaches that are definitive for SCOR performance in particular areas

V-1.6.1. Economics and maintainability

The SCOR design enables improvements in economic features through the following:

- Extensive plant simplification.
- Extensive use of proven technologies.
- Extensive use of pre-manufactured components (including large components of the primary coolant loop).
- Refuelling and maintenance outages significantly less frequent than current outages.
- A plant design life of at least 60 years with no replacement of the reactor vessel.
- A short construction schedule.

It is widely accepted that for a given technology, the investment cost diminishes with reactor power according to the exponent law:

$$C(P_1) = C(P_0) \times (P_1/P_0)^n,$$

where *n*, a design dependent coefficient, is between 0.4 and 0.7. But, for small or medium sized reactors, it may be possible to simplify the design architecture to achieve a threshold effect as shown in Fig. V-3. For the integrated standard reactor, there is a threshold at around one thousand megawatts thermal. At this power, the gain of the change from a loop-type to an integrated reactor is not sufficient to get an overnight cost lower than for a loop-type reactor. The technical point limiting the size of a standard integrated reactor is mass of the reactor vessel, which reaches 1000 tons.

To increase the gain, increased reactor power is needed while maintaining the safety advantages of the integrated reactor. This is achieved in the SCOR concept with a thermal power of two thousand megawatts thermal. Clearing the standard integrated limit is achieved by localization of the steam generator above the vessel. Low pressure and low temperature operating points ease reactor design. The low efficiency is offset by a lower component specific mass, less corrosion and the possibility of integration of the pressurizer. The soluble-boron-free core associated with the integrated design and the low pressure operating point lead to a radical simplification of auxiliary systems. Moreover, the medium sized power, the lower core power density and the large reactor vessel diameter lead to the possibility of in-vessel core retention in case of a hypothetical severe accident. A strategy of retention is another threshold effect of the medium sized reactor.

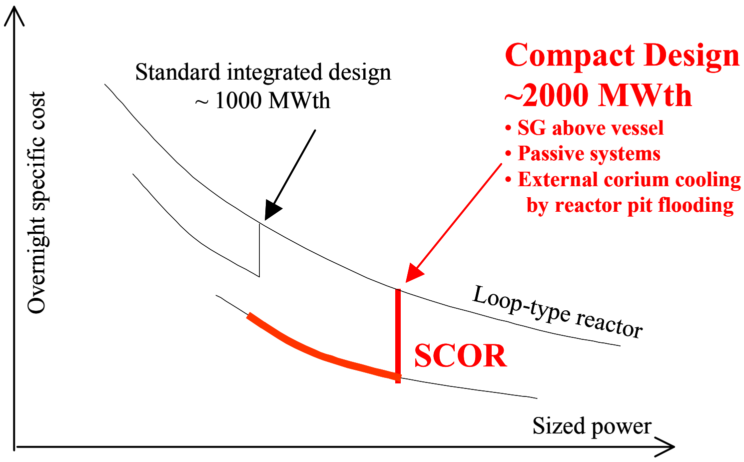


FIG. V-3. Visualization of a threshold effect on the overnight cost.

Table V-1 presents the calculations of costs performed with the CEA SEMER code developed for the economic evaluation of nuclear plants [V-3]. A comparison of calculations is made between two sites with the same electrical power: the first, with large classical PWRs, the second, with SCOR reactors. This cost assessment is performed with nomenclature recommended by the IAEA.

TABLE V-1. ECONOMIC ASSESSMENT – COMPARISON BETWEEN A LARGE LOOP-TYPE REACTOR AND SCOR (RELATIVE UNITS)

Reactor Type	Classical PWR	SCOR	
Thermal power (MW)	4250	2000	
Electrical power (MW)	1450	630	
Number of units on site	2	4.6*	
Plant life time (years)	30	30**	
Interest rate (%)	8	8	
Discount rate (%)	8	8	
Availability (%)	85	91	
Results in per cent of classical PWR (%)			
Direct costs (%)	100	94.1	
Indirect costs and dismantling (%)	16.2	21.7	
Construction lead delay (years)	6	6	3
Interest during construction	30.9	30.8	14.3
Capital costs (investment)	147.1	146.6	130.1
Cost of kW·h(% of classical PWR cost)			
Base investment	63.6	59.1	52.5
Operation & maintenance	15.8	14.8	14.0
Fuel cycle	20.6	20.6	20.6
Total (%)	100	94.5	87.1

*The power of a PWR and a SCOR site is assumed to be equal

** Plant lifetime is 60 years for SCOR, but the economic assessment is performed with 30 years.

The extent of simplification inherent in the SCOR design reduces the direct costs by nearly 6 points compared to a traditional PWR. The main reasons are:

- The specific mass of the primary system, ~2 t/MW(e), is slightly less than in large loop-type PWRs.

- The elimination of soluble boron leads to removal of a number of systems, allows a reduction in the dose rates associated with the auxiliary systems and tritium waste, leading to a reduction in operation and maintenance costs.
- The reactor vessel life is prolonged since neutron fluence onto the vessel is reduced by the increase in width of the annular space.
- An integrated design and low operating pressure reduce the number of safety systems and maintenance costs.
- The use of a compact pressure suppression containment reduces the number of buildings.
- The absence of an external core catcher and easier management of hydrogen risk with simple systems.
- The essential reduction of electrical equipment because of the reduction of auxiliary systems.

In spite of unfavourable indirect costs, the overnight cost is similar; this point constitutes a good performance for a medium sized reactor like SCOR compared to a large sized PWR.

The simplification of the nuclear buildings, reduction in components and the elimination of welding between large components reduces the construction duration to 3 years instead of 6 years, for a classical large loop-type reactor. This leads to a gain of 17 points compared to a reactor (classical PWR or SCOR) with 6-year construction duration.

With a prolonged refuelling interval, design simplification and reactor maintenance provided in power operation, the SCOR is expected to exceed 91% availability. The reduction of the dose rate due to a reduction in the number of radioactive circuits, the smaller amount of radioactive wastes (especially tritium), the essential reduction in the number of systems and the simplification of their maintenance can lead to reduced operation and maintenance costs. These advantages of the compact reactor make it possible to reach a 13% lower kW·h cost than that of a classical PWR.

V-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

Due to an integrated configuration and infrequent reloading/outages, the SCOR design reduces occupational radiation exposure compared with the standard PWR. The soluble-boron-free core lowers radiation exposure by reducing the auxiliary radioactive circuits and at the same time, by reducing the production of tritium.

V-1.6.3. Safety and reliability

Safety concept and design philosophy

The SCOR design philosophy is as follows. The first line in the SCOR defence-in-depth approach is to eliminate initiators that could realistically lead to core damage. This concept is implemented through “safety by design”, which can be simply described as designing the plant in such a way as to eliminate the accidents from occurring, rather than cope with their consequences. If it is not possible to eliminate accidents entirely, then the design should be such that it inherently reduces the consequences and/or decreases the probability of recurrence. The key difference from previous practice is that the compact reactor design is intrinsically conducive to eliminate accidents, to a degree impossible in conventional loop-type reactors.

The "safety by design approach" results in the following major implications:

- Since there are no large primary penetrations of the reactor vessel or large loop piping, large-break LOCAs are essentially eliminated.
- The integrated control rod drive mechanisms eliminate the risk of rapid reactivity insertion through control rod ejection.
- The residual heat removal system on primary circuit (RRP) with heat exchangers located in the vessel very close to the core eliminates the loop with primary water typical of a standard residual heat removal system.

Provisions for simplicity and robustness of the design

In SCOR, the number and complexity of the safety systems and required operator actions are minimized compared to standard loop-type PWRs. The net result is a design with significantly reduced complexity, improved operability and extensive plant simplifications.

The SCOR design is built on the proven technology provided by 40 years of PWR operating experience and on the established use of passive safety systems studied at CEA over the last ten years.

The low primary operating pressure enables a reduction in wall thickness of the pressure-resistant components and reduces the pressurizer volume.

The absence of AC powered safety systems (except for the low pressure safety injection system that operates at low flow rate) reduces the need for complex active systems with sensors, actuators, etc. that must be qualified for reliable operation over the full range of conditions (e.g. fire, seismic events) that might be encountered. Another important implication of the design simplification might be related to improved human reliability, as discussed in more detail below.

Design simplicity and human factors

Most human reliability assessment (HRA) models acknowledge the fact that the human performance in operating a system (especially in performing cognitive, demanding tasks) is largely influenced by complex characteristics. Although this notion of complexity may appear somewhat subjective at a certain level (the perceived complexity of a system is highly dependent on the knowledge and skills the operators have developed), it still exhibits an objective component directly correlated to the intrinsic complexity of features of the system.

Thus, minimizing the intrinsic complexity of the system, even in the early phases of its design, appears to be an attractive way of improving the system operation with an account of human factors.

This is the basis of the approach [V-4] proposed by the CEA to assess the relevance of human factors in advanced nuclear reactor concepts, particularly during the very early phases of the design, that is, when it is still possible to propose alternative solutions at a limited cost.

This method consists of characterizing the design features, especially in the safety system architecture, that are likely to pose problems in the operation, notably during the degraded situations in which the plant safety strongly depends on human reliability. The characterization of the intrinsic physical behaviour of the plant processes (safety functions), of the operating constraints of the safety systems and finally of the interrelations between these entities (most complexity theories consider these interrelations to be the main contributors to the complexity of a system), lead to the definition of an *operational complexity index* and to the identification of the sources of the operational constraints bearing on the operation crews.

Figure V-4 illustrates characterization of the complexity features as defined by the relationships between safety functions and safety systems [V-4].

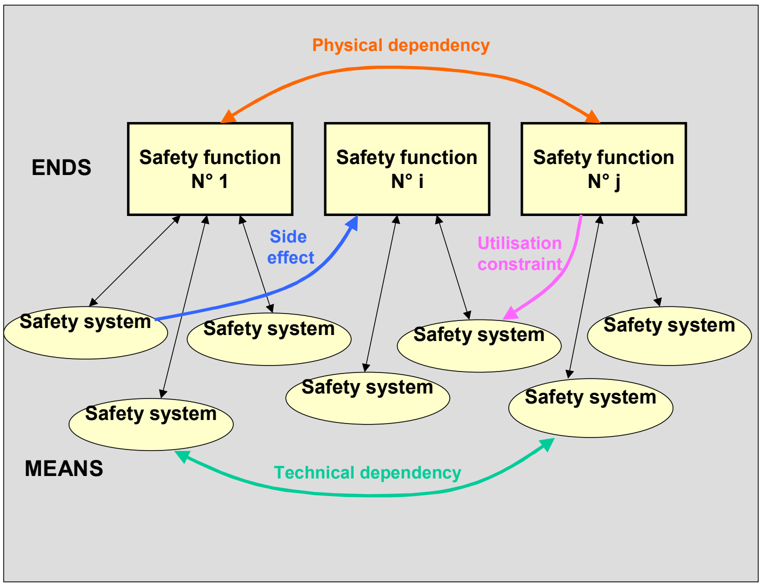


FIG. V-4. Characterization of the complexity features (illustration).

On this basis, comparative studies between various designs are possible, showing a new approach to design optimization which considers human factors at a very early phase in the conceptual design, whereas customary approaches only consider these aspects during instrumentation and control (I&C) and man-machine interface (MMI) design phases.

Even though the design of SCOR is still in an early conceptual phase, the knowledge of safety design options is sufficient for a preliminary assessment of the operational complexity. Figure V-5 hereafter presents the first results of such assessment performed in comparison with a standard loop-type PWR.

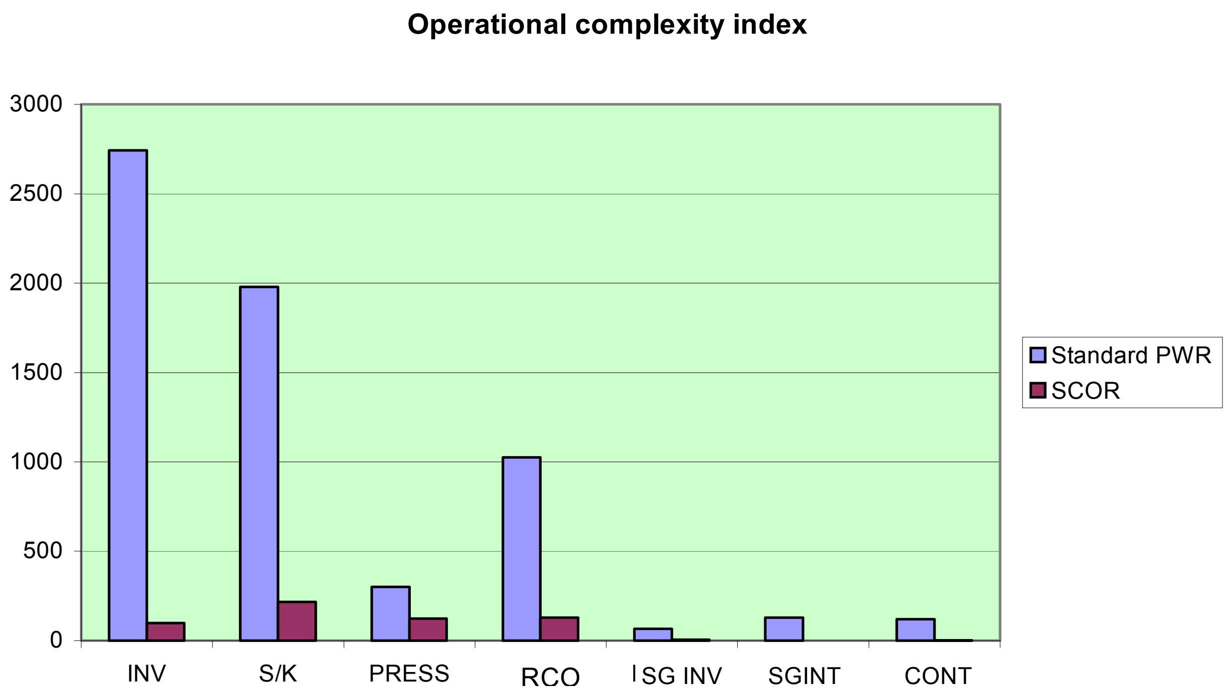


FIG. V-5. Operational complexity vs. safety functions for SCOR and standard PWR.

These results acknowledge a potential considerable decrease in the operational complexity of the SCOR, compared to a standard loop-type PWR. The origins of this simplification are twofold:

- First, it may originate from a *modification of the physical processes of the plant*, as defined by specific selected design options. For example, this is the case for the SCOR coolant inventory function (INV), where the choice of integrated design of the primary system limits the flow rate in possible LOCAs and increases the *grace period* for managing such events. This is also true for steam generator integrity management (SGINT), where the absence of direct evacuation of the steam to the atmosphere in SCOR obviates the need to explicitly manage the steam generator tube rupture, while it appears to be a major source of operational complexity in standard PWRs;
- Second, this simplification may originate from the *performance features of the engineered safety systems*. This is the case for systems dedicated to reactor cooling (RCO) which, in the SCOR – use of passive and closed loop cooling systems instead of active and open loop systems - exhibit much fewer operating constraints than in standard PWRs. This is also true for sub-criticality (S/K) management - elimination of soluble boron in SCOR - and for the coolant inventory control (INV) systems – simplification of the configurations of the low-pressure safety injection.

Even though the assessment of aspects of the human factor in SCOR is preliminary (it focuses on degraded operation but similar analysis is required for normal operation, maintenance and testing), the results confirm that design options for the SCOR concept may lead to a considerable simplification of operation and to a possible improvement in human reliability in operation. This result appears particularly valuable since probabilistic safety assessments (PSA) point to the major contribution of human failures to the global risk in existing nuclear plants.

Active and passive systems and inherent safety features

The consequences of a significant number of accidents are either eliminated outright or reduced by the SCOR concept at the design level. The major safety systems are passive; they require no operator action or off-site assistance for a long period after an accident. Moreover, core and containment cooling is provided during a long time without AC power.

The inherent safety features provided by the SCOR design are:

- No large break in the primary circuit; the maximum is double rupture of the pressurizer line (50 mm).
- Large thermal inertia of the primary circuit.
- A core low power density that results in large thermo-hydraulic margins.
- No reactivity insertion accidents by rod ejection since the CRDMs are integrated.
- No reactivity insertion in case of water dilution since reactivity control is achieved without liquid boron.
- A strong negative temperature coefficient in the duration of the whole cycle.

SCOR design incorporates the following passive safety systems:

- Passive decay heat removal by the RRP system. The passivity of the system is ensured simultaneously in the primary circuit, in the RRP loop and in the final heat sink.
- An RRP system with two types of heat sinks: pool and air-cooling tower.

- A dedicated steam dump pool to prevent radioactivity release into the atmosphere in case of steam generator tube rupture.
- Passive control of the containment pressure by pressure-suppression in case of a LOCA.
- In-vessel core retention with corium cooling by pit flooding in case of an hypothetical severe accident.
- Infinite autonomy with the air-cooling tower heat sink.
- Prevention of hydrogen combustion by an inert atmosphere in the reactor vessel compartment.

The active safety systems are:

- Low pressure safety injection (required at the earliest one hour after the start of the most penalizing transients with a low mass flow rate). Due to the small required operating time, the power may be provided by batteries.

Structure of the defence-in-depth

As it was already mentioned, level 1 of the SCOR defence-in-depth strongly relies on the intrinsic design features that eliminate initiators that could realistically lead to core damage. The "safety by design" features are complemented by the passive safety systems.

The barriers for radioactivity confinement are as follows:

First barrier

The first barrier is the fuel cladding, as in current PWRs.

Second barrier

When the reactor vessel is closed, i.e. under normal operation or accidental conditions, the second barrier is provided by the reactor vessel, the tube bundle of the steam generator and the integrated exchangers of the RRP loop, Fig. V-6.
When the reactor vessel is open, i.e. during refuelling operations, the second barrier is provided by water of the refuelling cavity, as in current PWRs.

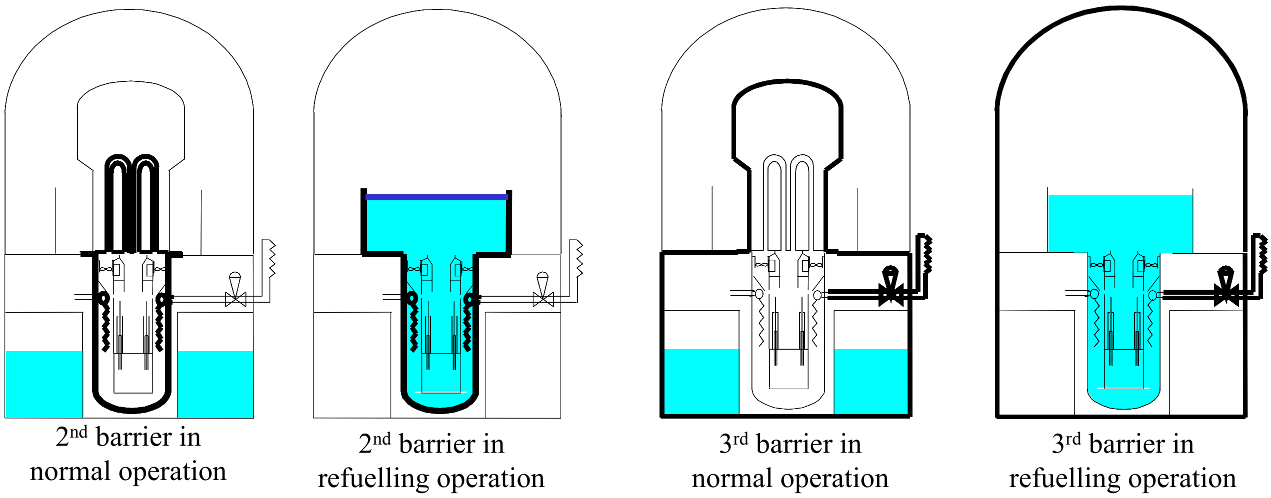


FIG. V-6. Barriers for radioactivity confinement.

Third barrier

When the reactor vessel is closed, the third barrier is provided by the compartment of the primary side, the steam generator, and the external heat exchanger of the RRP loop, Fig. V-6.

When the reactor vessel is open, the compartment of the primary side and the compartment of the secondary side provide the third barrier.
The isolating valves of the RRP loops inside the compartment of the primary side are only used to provide the third barrier in case of an external break in the RRP loop. These valves are not used in start-up of the RRP.

Design basis accidents

The main design basis accidents (NPP blackout, steam line rupture, SG tube rupture, and LOCA) were studied with the CEA's CATHARE code. All calculations were performed with 4 out of 16 available RRP loops. The scenarios of design basis accidents are summarized below.

NPP blackout

For this transient, the power is first removed by the SG and then by the RRP system. The RRP system reaches full operation at about one thousand seconds. After one hour and a half, the power removed by the RRP is enough to correctly cool the reactor.

Steam line rupture

In a steam line rupture, the amplitude of the cold shock is 22°C at the core inlet. The number of control rods is sufficient to stop the reactivity until the cold shutdown. After the trip, residual power is first removed by the released steam of the SG and then stored through the large thermal inertia of the primary circuit. The RRP loops reach their full power in one thousand seconds. One hour after the beginning of the transient, the RRPs remove all residual power. In this transient no water is released through the safety valve of the pressurizer.

Loss of coolant

The largest LOCA considered is a break in the line between the vessel and the boiler of the pressurizer (2×50 mm). At the beginning, power is removed through the break and by the steam generator. As for blackout, the RRPs reach their full power in one thousand seconds. After stabilization at the pressure of the secondary safety valve, the primary pressure reaches the threshold pressure of the safety injection system at 4000 s.

Steam generator tube rupture

During the first thousand seconds, residual power is removed by the SG and by the RRP system. To prevent steam release to the atmosphere, the steam is condensed in a dedicated pool. With 4 RRP loops and 5 SG tubes ruptured, the mass of released steam is about 40 to 50 tons, depending on the RRP heat sink capacity; with 8 RRP loops the mass of released steam is 20 tons. After six thousand seconds, the RRP system is sufficient to adequately cool the reactor and the steam release from the SG is stopped.

Summary of performance in design basis accidents

A comparison of the progress of typical design basis accidents between standard PWRs and SCOR is summarized in Table V-2.

The calculations performed for SCOR show that all transients could be adequately managed in a passive way (in the vessel, in the RRP loop, and in the heat sink) with only 4 out of 16 RRP loops, whatever the heat sink: pool or air-cooling tower. This represents a redundancy of 16 times 25%. The RRP operation is compatible with an active or passive mode whatever the primary pressure or temperature. Since the in-vessel heat exchangers of the RRP loop are located very close to the core, and thanks to the flow bypass of the venturi, the RRPs are operational in a two-phase flow mode in case of a small primary water inventory. Long term cooling may be ensured in a totally passive mode thanks to the RRP with air-cooling tower.

Only a safety injection at 2.0 MPa with a small flow rate is needed one hour after the beginning of the most penalizing LOCA, that is, a double break of the pressurizer line (2×50 mm). In the event of SG tube rupture, the steam released from the safety valves of the secondary circuit is condensed in a dedicated pool. No steam is released into the atmosphere.

TABLE V-2. STANDARD PWR AND SCOR RESPONSE TO ACCIDENTAL CONDITIONS

INITIATING EVENT	TRANSIENT PROGRESS IN STANDARD PWRs	TRANSIENT PROGRESS IN SCOR
NPP blackout	• Natural convection in the primary circuit; • Need for an external electric source (diesel) for systems in support (seal pump, safety injection, etc.); • Heat sink covers few hours.	• Natural convection in the primary circuit; • Very few systems in support (reduced power of the diesels or battery); Infinite autonomy of RRP systems with air heat sink.
Steam line rupture	• Risk of return criticality; • High pressure safety injection (HPSI) with borated water required.	• No return criticality; • Not need for safety injection.
LOCA	• Possible fast core dewatering depending on break size; • Need for three types of safety injection systems: HPSI, accumulators, low pressure safety injection (LPSI); Fast request for safety injection (according to break size); Long term cooling by LPSI (active system).	• No fast core dewatering (at least 1.5 hours after the transient start with no RRP operation); • Only one type of safety injection - LPSI with small flow rate is needed; • No fast request of LPSI, • Long term cooling by the RRP systems in passive mode.
SG tube rupture	• Risk of primary water release through the broken SG; • Request for safety injection disturbing the transient management; • Delicate management of the decreasing pressure to prevent secondary water without boron from flowing into the primary circuit through the steam generator broken tubes.	No steam release to the atmosphere (steam is condensed in a pool); • Cooling by the RRP systems; no need for safety injection; • Primary coolant has no soluble boron; hence, no risk of dilution by secondary coolant.

Beyond design basis accidents

For the SCOR, transients in the design extension conditions are essentially eliminated:

- H1 (total loss of the heat sink): the SCOR concept is based on several independent decay heat removal (RRP) loops ready to operate in a passive mode with a heat sink either in pools with a limited autonomy of several hours or in an air cooling tower in which the autonomy is infinite.

- H2 (total loss of feed water supply to the SG): decay heat is removed by systems of the primary circuit with redundancy (16×25%). There is no need for a safety auxiliary feedwater system.
- H3 (total loss of the power supplies): natural convection is possible in all the decay heat removal systems with the integrated exchangers, from the primary circuit to the heat sink.
- H4 (loss of the containment spray or loss of the low pressure safety injection): the SCOR has no containment spray, because it has a pressure suppression type containment. The low pressure safety injection has a less significant role than in standard PWRs because of the large primary circuit inertia, the elimination of large LOCAs and the effectiveness of the decay heat removal systems.
- ATWS (anticipated transient without SCRAM): the SCOR has two independent shutdown systems so that these transients will be treated individually as for standard PWRs. The management would be eased due to the constantly negative and higher moderator temperature coefficient than in standard PWRs.
- Multiple rupture of steam generator tubes and non-isolable containment: the discharge of the SG is carried out in a dedicated pool.
- Failure of HPSI: no HPSI is foreseen in SCOR.

The hypothetical case of a core meltdown is manageable through the following measures:

- Core meltdown: corium cooling can be ensured by reactor vessel pit flooding because the core power density is small and the large grace period before an hypothetical core meltdown reduces the decay heat when the corium enters the lower plenum.
- Hydrogen risk: the reactor vessel compartment atmosphere is inert to prevent hydrogen combustion (as in BWRs).

V-1.6.4. Proliferation resistance

No features different from those of standard PWRs are foreseen at the moment.

V-1.6.5. Technical features and technological approaches used to facilitate physical protection of SCOR

The inherent and passive safety features provide a strong defence against internal and external threats to the nuclear plant. All safety components and all buildings containing radioactive components (like the spent fuel area) are located inside the containment building. This building is constructed of thick reinforced concrete and can withstand the impact of an aircraft while maintaining the integrity of components within the cavity.

The chimneys of the RRP air-cooling tower are located outside, around and in the upper part of the containment building, Fig. V-16. The segregation of the air-cooling tower around the building makes certain that all chimneys will not fail due to impact. The high performance of the RRP systems (only 4 out of 16 RRP loops are needed to remove the decay heat) makes certain that the core will be adequately cooled. In case of a hypothetical failure of all RRPs with an air-cooling tower, the four RRP loops with immersed heat exchangers inside the building would back up the decay heat removal function. Their autonomy is ten hours before refilling the pools.

The absence of dependence on AC powered active safety systems (omit the LPSI system which needs low energy for a few hours), the inherent safety features of the SCOR, the

passivity of the decay heat removal systems and the selected safety component layout reduce the potential consequences of internal sabotage or external attacks, including aircraft crashes.

V-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of SCOR

No information was provided.

V-1.8. List of enabling technologies relevant to SCOR and status of their development

The list of enabling technologies relevant for the innovative features of SCOR is presented in Table V-3, complete with identification of the needs in R&D.

TABLE V-3. LIST OF ENABLING TECHNOLOGIES FOR SCOR

INNOVATIVE FEATURES OF SCOR	REQUIRED SUBSTANTIAL R&D
Fully integrated primary pumps	Confirmation of the possibility to build such pumps for operation at primary temperature conditions
A single steam generator acting as vessel head	
Low operating temperature	
Low operating pressure	Substantial R&D
Boron free core	Substantial R&D
Dedicated and integrated decay heat removal exchangers	
From a decay heat removal point of view, use of the SG as a thermal buffer	
Integrated and annular pressurizer	
Integrated hydraulic CRDMs	Assessment of the performance of the new drives
Backup of the main control rods by integrated fluidic neutron absorbers (MP98)	Confirmation and assessment of the reliability of MP98 system
Use of a venturi to reduce primary circuits inside the vessel in case of a failure of the SG, pumps, and for low water mass inventory	Assessment of the venture performance and the primary circuit hydraulic behaviour
Spent fuel area inside the containment building	
Containment with a pressure suppression device and inert atmosphere	
In-vessel cooling of corium	Substantial R&D

The SCOR design is built on technology proven by 40 years of PWR operating experience with new components introduced in the architecture. Some of these components are operating but not under the SCOR conditions.

This is the case for the primary pumps: spool type pumps with immerged coils operate in cold conditions and electromagnetic pumps for sodium operate in hot conditions. The SCOR needs R&D to develop spool type pumps with immerged coils that operate in hot conditions.

The proposed hydraulic CRDM for SCOR was derived from hydraulic systems developed for the BWR. The proposed adaptation needs R&D to confirm this option.

A private company is developing the backup liquid neutron absorber (MP98 system).

Many studies were done on the thermal valve; they confirm potential interest in this device to control the heat flux in a decay heat removal system. Nevertheless, R&D is needed to apply this component to the RRP loops.

The water flow path in the vessel with the venturi bypass is different from that used in standard PWRs. A mock-up test will be necessary to validate the hydraulic performance.

The needs of R&D for SCOR are essentially to confirm the performance of the innovative devices, but they do not constitute high technological efforts.

V-1.9. Status of R&D and planned schedule

The SCOR is a conceptual design developed in the framework of the French programme on innovation for light water reactors (LWRs). The SCOR is developed by the Nuclear Energy Division of Commissariat à l'Energie Atomique at Cadarache in France. The work was partially supported by the AREVA / FRAMATOME-ANP.

The current stage is a conceptual design. Due to the low level of R&D required, SCOR could be deployed in the next fifteen years.

V-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

The SCOR concept is based on an integrated design of the primary circuit, which represents a radical change in system configuration against the operating PWRs. SCOR also incorporates innovative design features that make it possible to increase the reactor power up to 2000 MW(th), i.e. nearly twice against the power of other integrated PWRs under development. The SCOR design also includes other innovative features with several of them requiring a substantial amount of R&D, as outlined in Table V-3.

V-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

The list of SMR designs having certain similarities to the SCOR is given in Table V-4.

TABLE V-4. SMR DESIGNS OF RELEVANCE TO SCOR

ITEM	SCOR	IRIS [V-5]	CAREM [V-5]	MARS [V-6]
Primary operating pressure	Low			Low
Number of steam generators	One			One
Fully integrated primary pumps	Yes	Yes		
Hydraulic CRDMs	Yes		Yes	

V-2. Design description and data for SCOR

V-2.1. Description of the nuclear systems

Reactor core and fuel design

The core is similar to the core of French 900 MW(e) PWRs, and consists of 157 assemblies of 264 fuel rods in a 17×17 square array. The central position may be used for in-core instrumentation, while the remaining 24 positions have guide tubes. The active fuel height is 3667 m. The core thermal power is 2000 MW, and the specific power is lowered by 28% compared to a standard French PWR.

The advantages of low power density are:

- Longer cycle duration.
- The core power and power density are *a priori* compatible with an in-vessel corium retention strategy achieved by reactor vessel pit flooding; therefore, an external core catcher is not required.
- The increased thermal margin provides improved operational flexibility, including increased overall plant capacity factors.

The fuel has burnable poisons to compensate reactivity changes during burn-up. The discharge burn-up is the same as in a French standard PWR ~45 000 MWd/t·U. The fuel assembly of SCOR is shown in Fig. V-7.

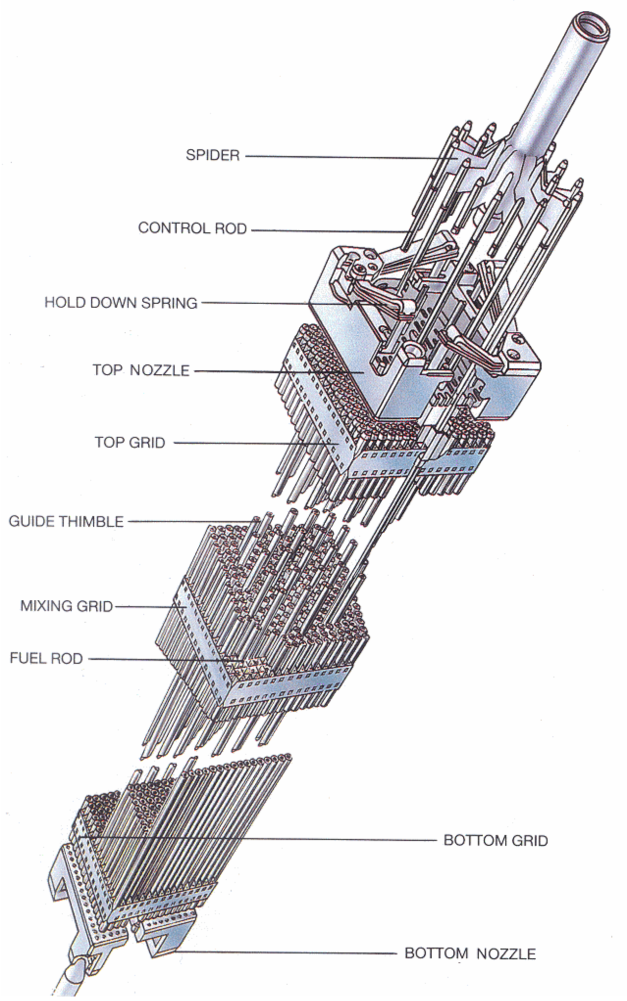


FIG. V-7. SCOR fuel assembly.

Main design characteristics of the fuel assembly are:

Fuel assembly total length	4058 mm
Arrangement of rods	17×17, square
Number of fuel assemblies	157
Number of fuel rods/assembly	264
Number of control rod guide tubes	25
Number of grids per assembly	9
Cladding tube material	Zircaloy
Cladding tube wall thickness	0.57 mm

Outer diameter of fuel rods	9.5 mm
Overall weight of fuel assembly	664 kg
Active length of fuel rods	3660 mm

Reactivity control systems

Control rods

The use of integrated drives is necessary, because standard control rod drives (CRDs) are incompatible with an SG located above the reactor vessel. This system leads to the elimination of the risk of rod ejection (rupture of the CRD nozzle on the reactor vessel head) and removes constraints associated with reactivity insertion accidents for the determination of the maximum discharge burn-up.

The hydraulic drives, which were developed for the BWR [V-7], are adapted to the SCOR design. They consist of a cylinder and a mobile piston as shown in Fig. V-8.

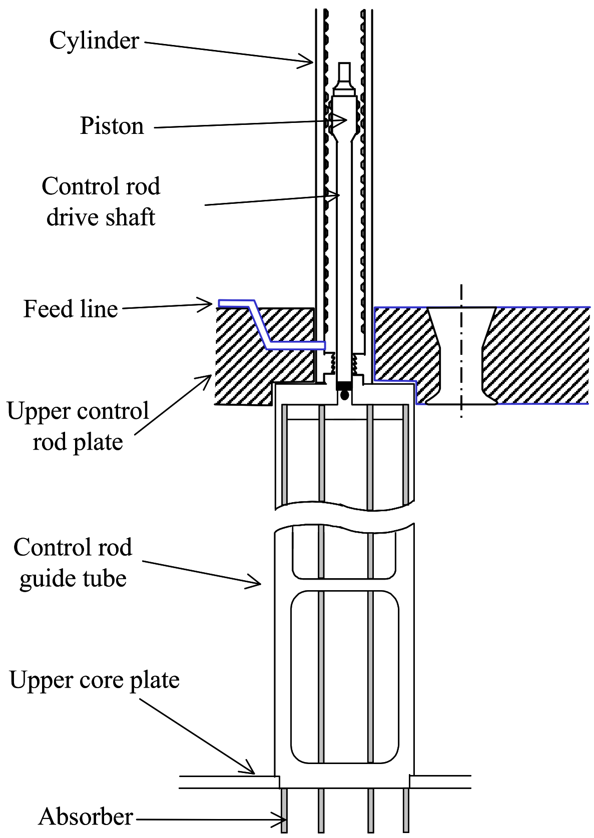


FIG. V-8. Hydraulic drive adjustment to the SCOR design.

The cylinder is attached to the upper control rod plate and the piston is connected to the control rod drive shaft. The entire height of the cylinder is grooved. The piston and cylinder groove geometries are similar. The piston position is maintained by introducing a given primary fluid flow rate into the interior of the cylinder. The piston is designed to rise or descend along a distance equal to the groove pitch by temporarily increasing or decreasing the primary fluid flow rate. The displacement piston acts as the drive rod moving the absorbing rod clusters (similar to those used in standard PWRs). In accident conditions the reactivity control is backed-up by the original MP98 system [V-8].

Soluble-boron-free core

The choice of a soluble-boron-free core is based on studies conducted by the CEA in the 1980s [V-9]. The SCOR features allow adapting the boron-free core due to:

- A 10% increase in the moderation ratio due to an increase in the moderator density.
- The accommodation of local power peaks slightly more intense than in standard PWRs, due to the low power density.
- The possibility of using one control cluster for two assemblies or even one per assembly due to the compact integrated hydraulic drives.
- The elimination of the potential blocking of several CRDs in case of a large primary break, due to the integrated design.
- Elimination of local power excursions by using integrated CRDs and removing the risk of rod ejection accidents.
- Significantly decreased reactivity control requirements due to the low power density (reduced power and Doppler effects), and the low operating point (reduction of the absolute value of the moderator coefficient).

The choice of the soluble-boron-free core leads to simplification of the auxiliary systems related to boron management, resulting in a significant reduction in the investment and maintenance costs. It also simplifies the chemical control system and reduces personnel exposure.

Safety injection

Since large LOCAs are eliminated by design and since the primary system thermal inertia is larger than in loop-type PWRs, the safety injection system requires devices with a smaller flow rate. Given the intrinsic low-pressure option for the reactor, there is only one type of safety injection with a pressure of about 2.0 MPa. The pump power needed for the safety injection is very small, about 35 kW(e).

Decay heat removal systems
Since the reactor has only one steam generator, the decay heat removal systems are diversified in both the primary and secondary circuits, Fig. V-9.

Residual heat removal system on the primary circuit

The decay heat from the primary system is removed by heat exchangers located in the downcomer. Each exchanger has a dedicated heat sink so there are sixteen independent loops, called the RRP system (Residual heat Removal system on the Primary circuit). There are two types of heat sinks:

- Four RRPs are cooled by the heat exchangers immerged in a pool (RRPp).
- The other twelve are cooled by heat exchangers in the air-cooling tower (RRPa).

All RRPs can operate in a natural convection mode provided both in the loop and in the heat sink.

The design of the circuits is very simple. The RRP loops are designed to resist the primary pressure. Isolating valves on the circuit are to avoid the risk of primary water passage outside the containment in the event of heat exchanger tube rupture. A surge tank to compensate for the water expansion from cold shutdown to the full power operating state provides the pressure control of the RRP circuit.

The control valves are placed on the level of the heat sink: thermal valves [V-10] or air leaves function so that the temperature of the RRP loop remains high when the reactor is in power. In this manner, the RRP operates passively, by the opening of the air leaves on the RRP air coolers or by the opening of the thermal valve on the RRP pools.

Forced convection is only required when cooling is needed for core refuelling. The twelve RRPs cool the primary system to a cold shutdown state. They replace the conventional reactor heat removal system.

The maximum power removed by each RRP loop is 5 to 7 MW(th) according to operating conditions. This small quantity of removed power, in whatever the reactor power state, makes it possible to test the heat removal system without significantly disturbing the operation. The procedure of testing constitutes a significant element in securing the reliability of these systems.

The RRPp are safety grade. The RRPa are safety grade, chilled water loop and pumps excepted.

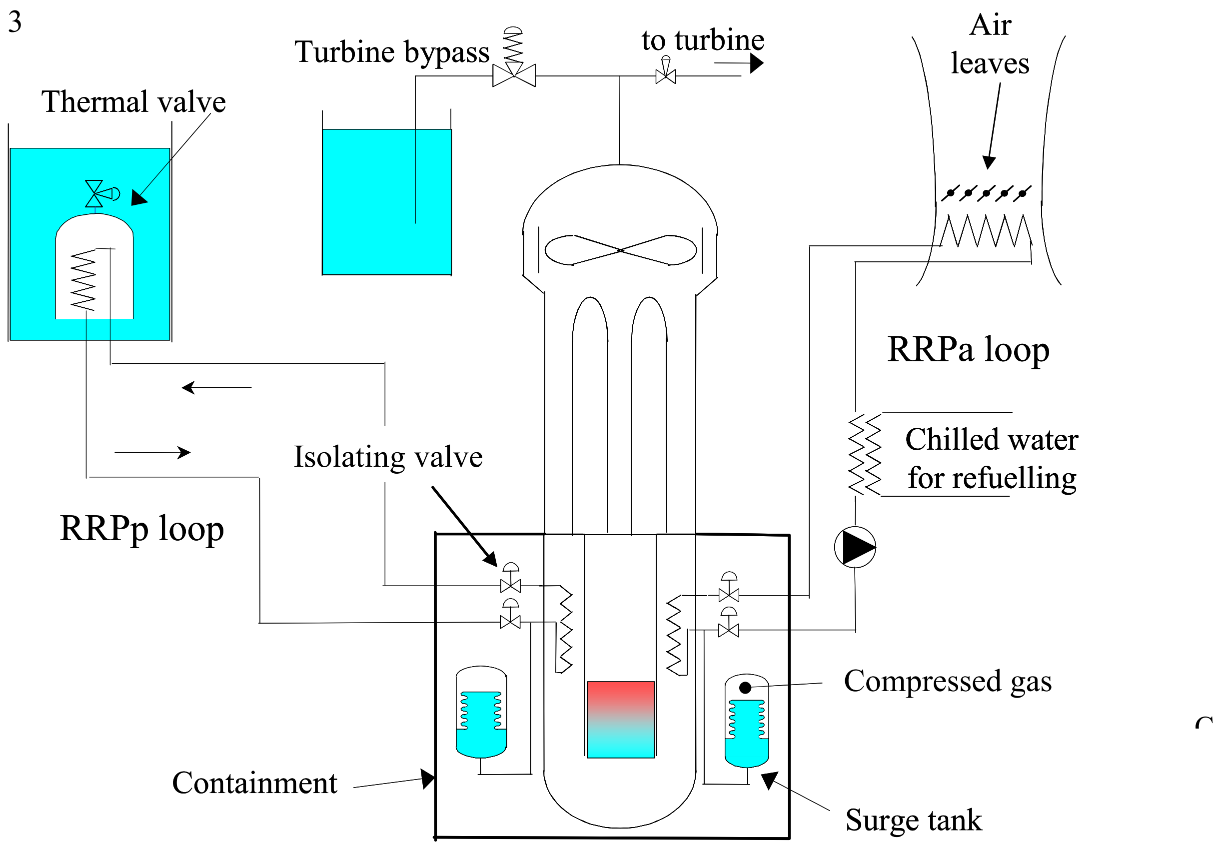


FIG. V-9. Schematic diagram of the decay heat removal systems.

The thermal valve concept can be illustrated in a very simple way (Fig. V-10). It consists of a bell-shaped compartment submerged in the pool surrounding the in-pool heat exchanger and is provided with one or more mechanical valves (depending on the need for the flow area) on the upper part of the bell (called the pilot valve). The compartment is open in its lower part.

With this device, fluid circulation within the in-pool heat exchanger is controlled by actuation of the pilot valve.

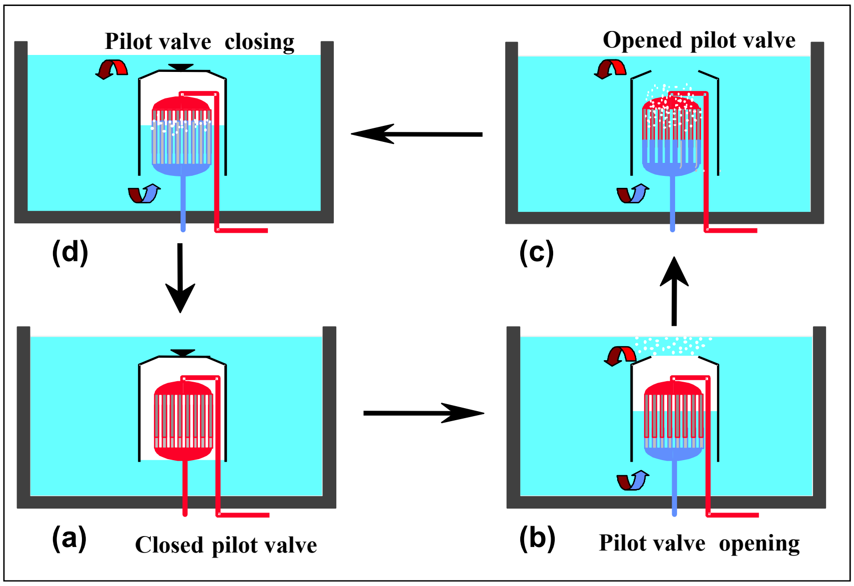


FIG. V-10. Thermal valve diagram.

During normal operation of the reactor the pilot valve is closed, Fig. V-10(a). The steam produced is confined under the compartment and the natural circulation is drastically reduced as is the power exchanged.

When the pilot valve is opened, Fig. V-10(b), steam is released from the compartment and condensed; cold water flows from the opening in the lower part of the compartment and natural circulation is re-established, Fig. V-10(c), allowing passive removal of decay heat.

The pilot valve closing, Fig. V-10(d), leads to the confinement of the steam produced under the compartment until the heat exchanger becomes isolated again.

Residual heat removal in the secondary circuit

The steam generator is not considered the main system for decay heat removal, it acts as a thermal buffer until safety systems on the primary side are fully operational.

Modes of residual heat removal

In hot conditions, residual heat is removed through the steam generator. The steam is discharged to the atmosphere and the SG is fed by the start-up shutdown system (SSS). The system is not safety grade. Then, at low temperatures, the RRP with the air-cooling tower (RRPa) removes the decay heat.

When the vessel is opened, especially during refuelling, decay heat is removed by the twelve RRPa cooled by chilled water to obtain a very low primary temperature compatible with the conditions of maintenance. The primary circuit operates in a natural convection mode and the RRPa loops operate with a forced circulation.

Chilled water is only used during refuelling. In case of a chilled water circuit or an RRPa pump failure, the heat sink is backed up by the air-cooling tower of the RRPa. In these conditions, the maintenance (refuelling) in the primary circuit is stopped and the temperature goes slightly higher, to around 100°C.

The design of the RRP is sufficient to ensure core cooling from the hot to cold temperature, whatever the primary pressure. The RRP system replaces the normal residual heat removal system with an external loop like in standard PWRs.

Reactor pressure vessel

The SCOR primary circuit is integrated. The reactor vessel (RV) houses not only the nuclear fuel and control rods with their mechanisms but all the major reactor coolant system (RCS) components (Fig. V-1). This includes sixteen modules, each one holding a spool type reactor coolant pump (RCP) and a heat exchanger for decay heat removal; a steel reflector surrounds the core in the RV downcomer to improve neutron economy and to reduce neutron fluence on the RV; and a pressurizer located in the RV. This simplified integral arrangement eliminates the individual component pressure vessels and large connecting loop piping, resulting in a compact configuration and in the elimination of the large loss-of-coolant accidents as a design basis event. Because the SCOR vessel contains all the RCS components except the steam generator, it is larger than a traditional RV. Main characteristics of the reactor vessel are the following:

Cylindrical shell inner diameter	4983 mm
Wall thickness of cylindrical shell	141 mm
Total height	14813 mm
RPV head	No (steam generator)
Base material: cylindrical shell	Carbon steel
Liner	Stainless steel
Design pressure/temperature	9.78/309 MPa/°C
Transport weight (lower part), and	280t

The heat exchanger-pump modules

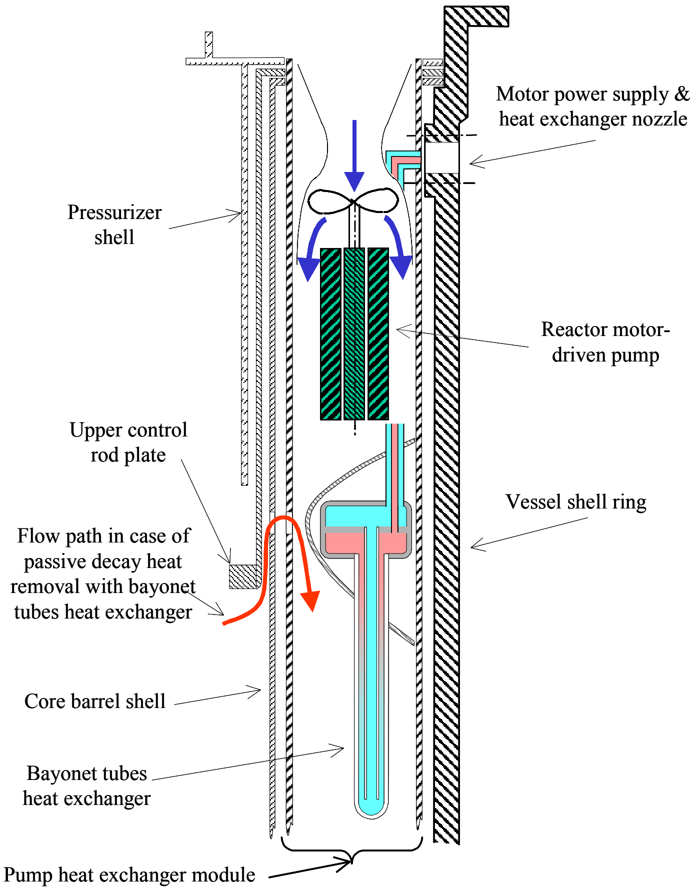


FIG. V-11. Decay heat exchanger-pump module.

The large annular space between the core barrel shell and the reactor vessel contains the heat exchanger-pump modules. Each of the sixteen modules (Fig. V-11) comprises a primary pump and a heat exchanger to remove residual heat.

The spool type pump, located in the upper part, is supplied with water from the steam generator. The submerged coil-type motor is located downstream of the impeller. The motor and pump consist of two concentric cylinders, in which the outer ring is the stationary stator and the inner ring is the rotor carrying high specific speed impellers; only small penetrations are needed for the electrical power cables. The primary water flowing around the outer ring cools the coils and therefore, associated piping penetrations through the RV for cooling water are eliminated. The spool pump needs an additional external motor to provide high inertia during a coastdown to mitigate the consequences of loss-of-flow accidents (LOFAs). This additional inertia is provided by an external motor with a flywheel, electrically linked to the spool type pump.

At the outlet of the pump, water is accelerated by a venturi, passes into a diffuser and then through the decay heat exchanger tube bundles.

The decay heat exchanger of the RRP system consists of bayonet tubes in which the outside surface is wetted by the primary fluid. The secondary water flows first in the internal tube and then upward through the annular space bound by the two tubes. The water box is located in a dead zone, behind the venturi. This type of heat exchanger was selected, since it does not require a water box at the exit of the heat exchanger. This reduces the primary pressure drop and allows free expansion of the tubes. Thermal loads are reduced leading to an increased mechanical resistance and an enhanced reliability.

A flow bypass is installed where the venturi is located, between the core exit and the cold leg. It allows natural convection of the primary fluid during pump shutdown. During normal operation, high flow velocity at the venturi throat leads to a decrease in the local pressure. The cross sectional area of the venturi throat is designed to balance the pressure between the hot leg (core exit) and the cold leg (heat exchanger-pump module) to prevent bypass flow under normal conditions.

The decay heat exchanger-pump module can easily be extracted from the reactor vessel once the steam generator has been removed. The pump power supply and the heat exchanger secondary feed-lines are set in the vessel via a removable opening in the upper part of the reactor vessel. Main characteristics of the reactor coolant pump are as follows:

TYPE	SPOOL TYPE
Number	16
Design pressure/temperature	9.78 /310 MPa/°C
Design flow rate (at operating conditions)	654 kg/s
Pump head	44 m
Power demand at hot coupling	450 kW
Pump speed	3000 rpm

Pressurizer

The SCOR pressurizer is integrated into the upper part of the riser, just below the steam generator. The pressurizer region is designed in an annular shape in the form of an inverted U (Fig.V-12). The coolant flows through the central part of the pressurizer. The bottom portion of the inverted U contains the opening to allow water insurge and outsurge to/from the pressurizer.

Electric heaters are located in a small volume tank outside the reactor vessel and act as a steam source. The cold water supply is tapped off just downstream of the pumps and the two-phase mixture is reinjected at the top of the pressurizer.

Due to the low pressure and low temperature operating point leading to smaller variations of water density versus temperature, the volume of the pressurizer is smaller than those used in plants with a classical, separate pressurizer vessel. The SCOR pressurizer has a total volume of ~21 m^3. This volume is large enough to manage a blackout without steam release through the safety valve of the pressurizer.

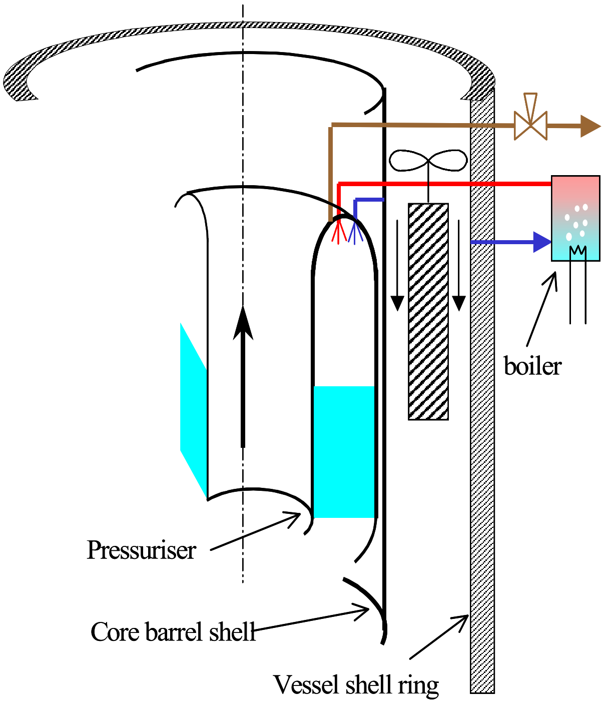


FIG. V-12. Pressurizer diagram.

The pressurizer data is as follows:

Total volume	21.3 m^3
Steam volume: full power	10 m^3
Design pressure/temperature	9.78/309 MPa/°C
Inner/outer diameter	1.66/3.20 m
Total height	4.2 m
Material	Stainless steel

Steam generator

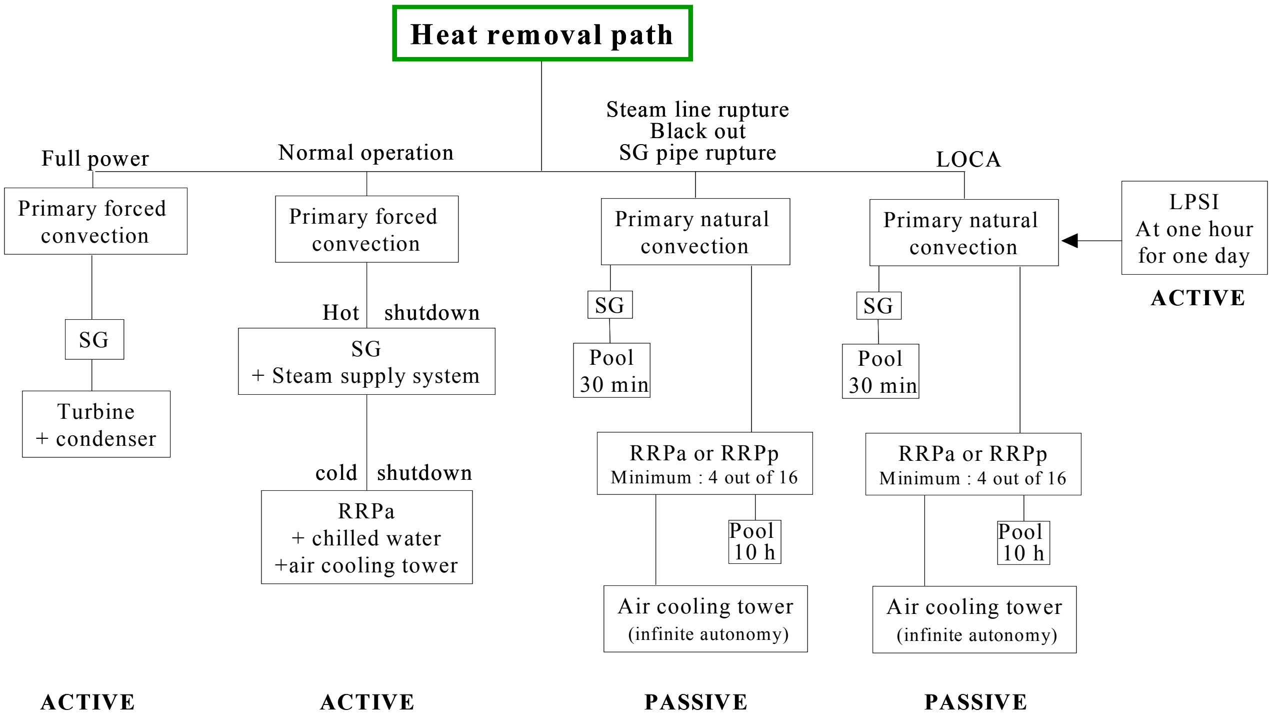
The SG data is as follows:

Type	Axisymmetric, boiling, vertical, U tubes
Number	1
Heat transfer surface	10 707 m^2
Number of heat exchanger tubes	11 000
Tube dimensions	19.5/17.9 mm
Shroud outer diameter	5265 mm
Total height	16 000 mm
Transport weight	568 t
Tube material	Inconel 690

There is only one U-tube boiler type steam generator. Like in propulsion reactors, the SG is placed above the core. In contrast to standard SGs, the present generator has an axial symmetry. The hot leg is located in the centre and the cold leg is located around the centre.

Main heat transport system

A scheme of the SCOR main heat transport system with specification of heat removal paths in normal operation and in accidents is shown in Fig. V-13.



SG : Steam Generator
LOCA : Loss Of Coolant Accident
LPSI : Low Pressure Safety Injection
RRP : Residual heat Removal on Primary Circuit
RRPa : RRP with air cooling tower
RRPp : RRP with in pool heat exchanger

FIG. V-13. Heat removal paths of SCOR.

V-2.2. Description of the turbine generator plant and systems

The turbine generator plant and systems are similar to the conventional islands of standard reactors. In spite of the low inlet turbine pressure, improvements in the EPR turbine design [V-11] lead to a SCOR efficiency of 31.5%.

V-2.3. Systems for non-electric applications

No information was provided.

V-2.4. Plant layout

The SCOR plant layout has been developed with criteria of minimizing radiological risks in case of accidents and minimizing radiological releases to the site. Other relevant design

criteria concern minimizing the impact of the plant on the environment through minimal land use requirements.

To minimize radiological releases in normal operation or in accidents the volume and number of buildings with radioactive fluids are reduced through effectively merging them within a single containment building. A re-enforced containment is designed to protect safety systems against external events.

The turbine plant building is similar to those of standard nuclear power plants. The containment houses the primary circuit, which is located in the lower part of the containment building. All the safety systems (low pressure safety injection system, decay heat removal system, etc.) are located inside the containment building.

The compactness of the primary circuit of SCOR (Fig. V-14) makes it possible to use the design of pressure suppression containments typical of BWRs.

This feature of the SCOR design is used to minimize the volume of the containment.

The containment building consists of two physically separate compartments; the lower is the reactor containment; the upper building is mainly to protect the secondary circuit against external hazards, e.g. missiles (Fig. V-15).

All primary pipe connections are located under the reactor vessel-SG mating surface (Fig. V-16).

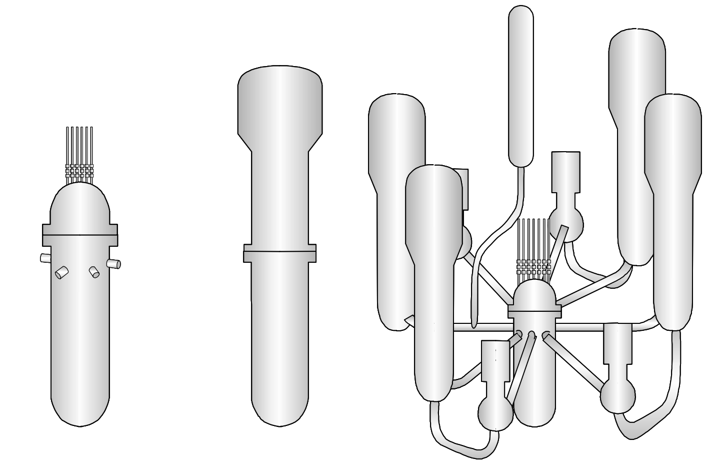


FIG. V-14. Relative size of the SCOR primary system.

The two compartments of the containment building are:

- The compartment of the primary side (or primary containment) is located under the reactor vessel-SG mating surface. It contains the vessel and primary pipe connections. The volume is small. A pressure suppression device, as in BWRs, controls the pressure. This compartment has an inert atmosphere to manage the hydrogen risk.
- The compartment of the secondary side (or secondary containment) houses the steam generator. It has no inert atmosphere since there is no contact with the primary circuit when the vessel is closed.

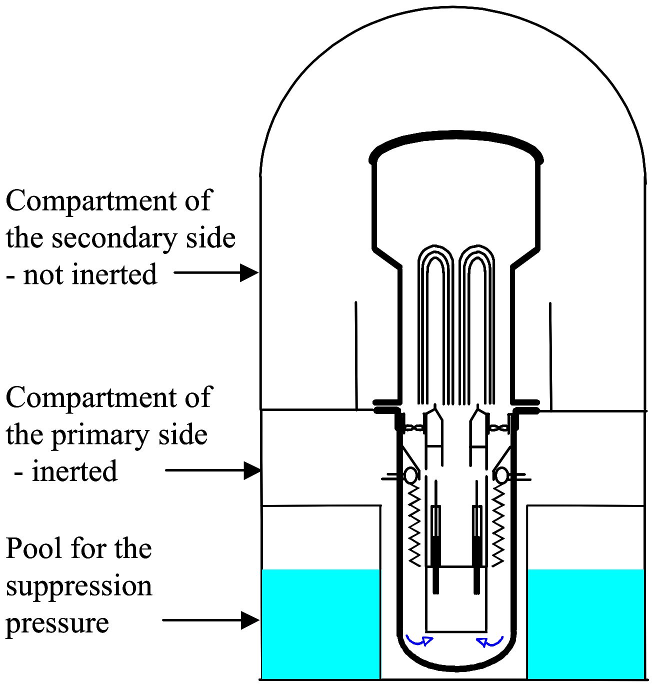


FIG. V-15. Compartments of the containment building.

The primary containment data are as follows:

Type	Pressure suppression
Overall shape	Cylindrical
Dimensions (diameter/height)	30.4/14 m
Drywell: free volume	3500 m^3
Wet well: free volume + water volume	4800 m^3

The containment building layout is shown in Fig. V-16.

The compartment of the primary circuit contains: the reactor vessel, the drywell, the wet well with water pools, the safety injection system and the surge tanks of the RRP loops.

The compartment of the secondary circuit comprises the refuelling cavity with space to store the internal equipment and to store and inspect the SG; a compartment acting as the fuel building, the RRP pools; the steam dump pool for the SG discharge; the zone containing the ventilation devices and the engines with mechanical inertia of the primary pumps.

Around the building containment, there are sixteen areas in the upper part of the building. Twelve areas with air leaves are the chimneys of the RRP air-coolers. Three areas are the chimneys of the passive loops for the reactor containment cooling. The last one is the space for the equipment hatch.

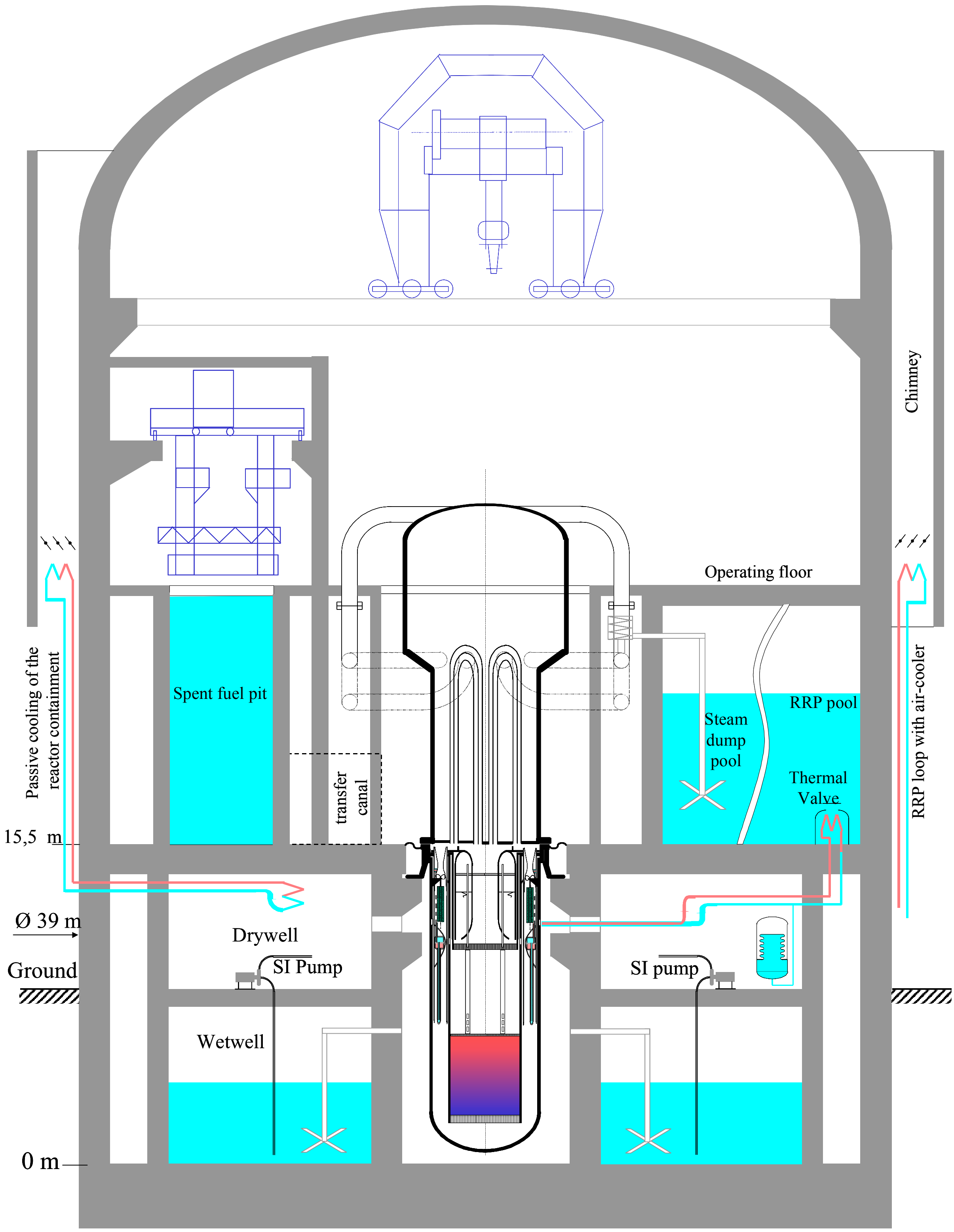


FIG. V-16. Containment building layout.

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ANNEX VI

INTEGRATED MODULAR WATER REACTOR (IMR)

Mitsubishi Heavy Industries, Japan

VI-1. General information, technical features and operating characteristics

VI-1.1. Introduction

The Integrated Modular Water Reactor (IMR) is a type of integrated primary system reactor (IPSR) with a reference output of 1000 MW(th) (350 MW(e)). The design targets of the IMR are to attain electricity generation costs comparable to a large-scale nuclear reactor and a high-level safety by removing the sources causing fuel failures by design. To achieve these targets, IMR employs an integrated design with in-vessel control rod drive mechanisms (CRDM), a hybrid heat transport system (HHTS) employing two-phase natural circulation for primary heat transportation and a stand-alone direct heat removal system (SDHS) for heat removal from the primary system in accidents.

IMR started its conceptual design study in 1999 at Mitsubishi Heavy Industries (MHI), reflecting changes in the business environment such as lower economic growth and electricity demand and deregulation of electricity markets in Japan. An industry-university group led by MHI, including Kyoto University, Central Research Institute of Electric Power Industries (CRIEPI), the Japan Atomic Power Company (JAPC), and MHI is currently developing related key technologies, funded by the Japan Ministry of Economy, Trade and Industry from 2001 to 2004. In this project, the feasibility of HHTS and SDHS concepts has been tested through three series of experiments. They are: (1) air-water scale tests to confirm void distribution and void behaviour in the reactor, (2) high temperature natural circulation tests to study two-phase natural circulation in the reactor with the actual temperature, pressure, and axial dimensions of the IMR and (3) SDHS tests to study passive heat transport with the actual temperature, pressure and axial dimensions of the SDHS. The test facilities were built and operated at the MHI Takasago R&D centre. In-vessel CRDM technology is based on marine reactor (MRX) development by Japan Atomic Energy Research Institute (JAERI) and MHI.

After these conceptual design efforts, basic design and validation tests are required before making an application for IMR licensing. Specific construction plans for the IMR have not yet been fixed.

VI-1.2. Applications

The IMR is primarily designed to generate electricity. Because of its modular characteristics, it is suitable for large-scale power stations consisting of several modules and also suitable for small distributed-power stations, especially when the capacity of grids is small.

The IMR also has the capability for district heating, seawater desalination, process steam production, etc., but design work for these applications has not yet been accomplished.

VI-1.3. Special features

The IMR is a land-based power station module. The capacity of the power station can easily be increased and adjusted to the demand by constructing additional modules.

Each module is constructed by installing integrated sub-modules of buildings containing pre-fabricated equipment. Such construction methods are expected to reduce construction time to two years.

VI-1.4. Summary of major design and operating characteristics [VI-1, VI-2, VI-3, VI-4]

The IMR is a small-to-medium sized power reactor to be built after 2010. The cross-section of the IMR reactor is shown in Figure VI-1 and the plant concept of IMR is in Figure VI-2.

The IMR design goals are as follows:

- Economics competitive with other electric power sources including large-scale reactors.
- A high degree of reliance on intrinsic safety features, unachievable in large-capacity LWRs, i.e. elimination of initiating events that might cause fuel failure, operator-free management of accidents, no need in external water and power during accidents, etc.

To achieve these goals, the IMR employs the following design features:

- An integrated primary system design. The reactor vessel contains the whole primary circuit including steam generators (SGs) and control rod drive mechanisms (CRDMs). The design achieves a compact primary system and containment. It also eliminates initiating events for loss of coolant (LOCA) and control rod ejection accidents, which makes it possible to realize a simple safety design without safety injection and containment spray systems, Fig. VI-1.
- The hybrid heat transport system (HHTS). The IMR employs natural circulation and a self-pressurized primary coolant system, altogether resulting in a simple primary system design without reactor coolant pumps and pressurizer; it also reduces maintenance requirements. In addition, the use of HHTS concept makes it possible to reduce the size of the reactor vessel. The HHTS is a kind of two-phase natural circulation system. The coolant starts boiling in the upper part of the core; two-phase coolant flows up in the riser and is condensed and cooled by SGs. Such design approach increases coolant flow rate and thus, reduces the required height of the reactor vessel (RV) to transport the heat from the core, Fig. VI-2.
- Stand-alone direct heat removal system (SDHS), Fig. VI-2. The SDHS is a passive safety system that directly removes decay heat to the atmosphere through SGs and depressurizes the primary system without opening the primary pressure boundary. The safety design is greatly simplified because the safety injection system (SIS), containment spray system (CSS) and safety grade support systems such as the component cooling water system (CCWS), essential service water system (ESWS) and emergency AC power system are not required. SDHS works continuously from early stages of the accident to long term cooling by automatically changing the cooling mode from water-cooling to air-cooling without external supports such as cooling water, power, etc. [VI-5].

Installed capacity

Thermal power:	1000 MW
Electrical power:	350 MW

Mode of operation

The IMR is capable of both base-load and load-follow operation.

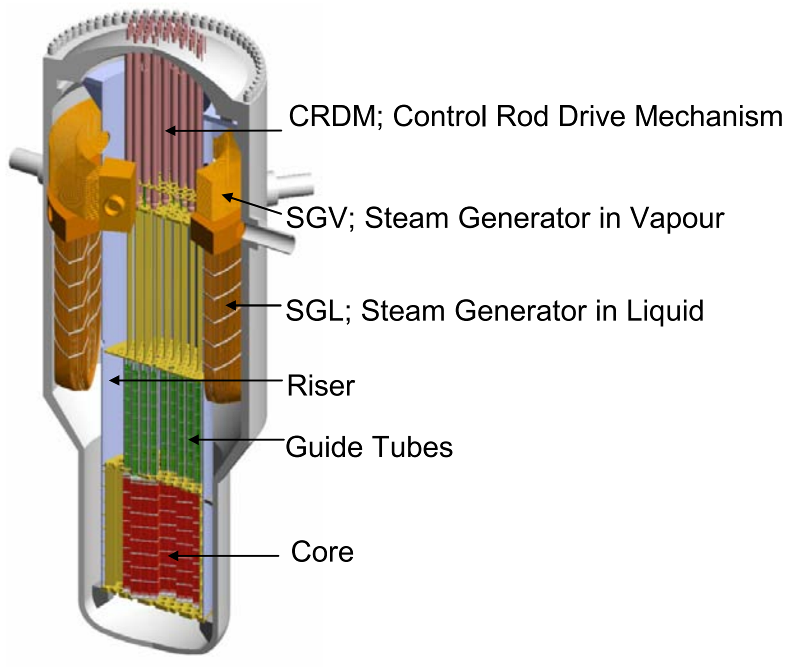

FIG. VI-1 Cross section of IMR reactor.

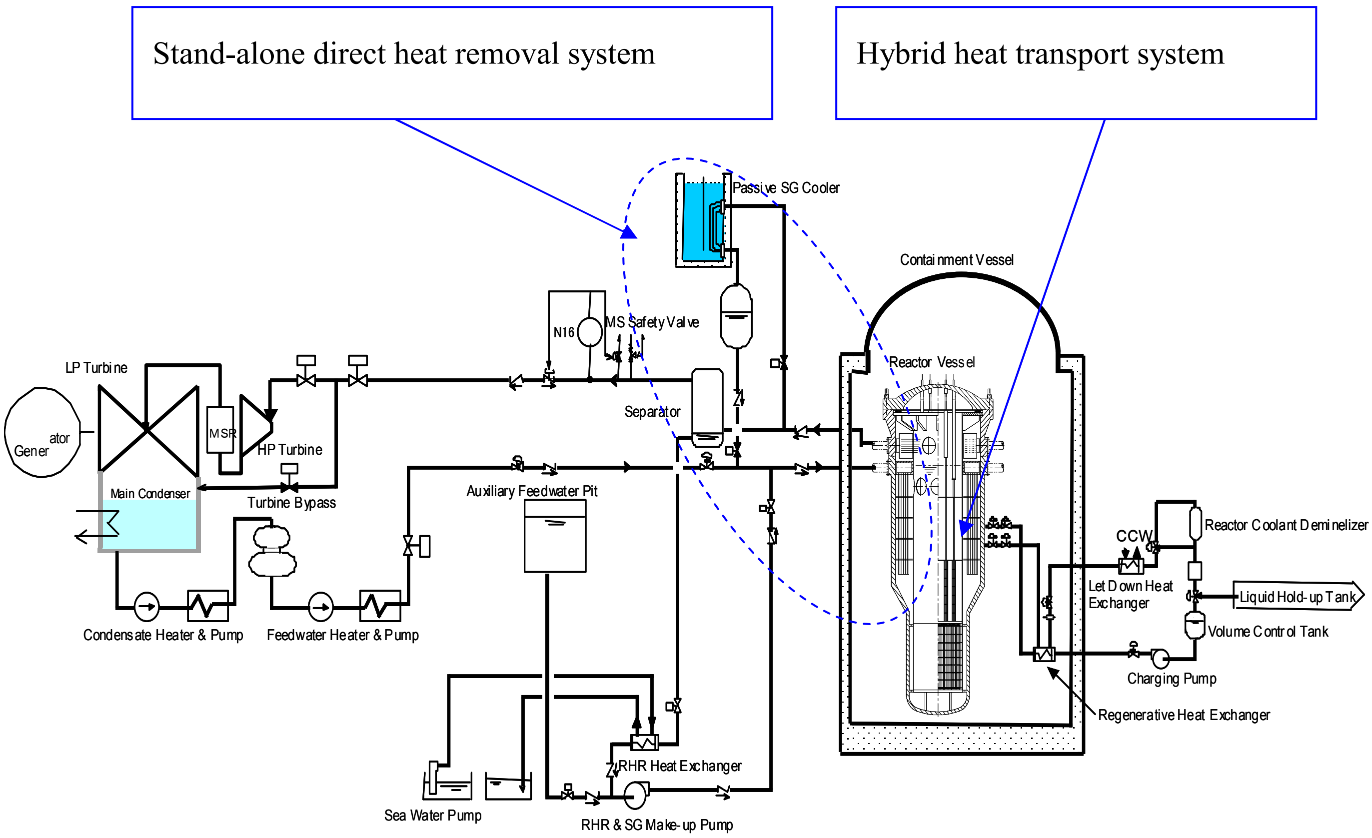

FIG. VI-2 Schematic diagram of IMR.

Load factor/ Availability

Load factor target 95% (in case of base-load operation)
Availability factor target 95%

Major design characteristics of the IMR are summarized in Table VI-1.

TABLE VI-1. MAJOR DESIGN CHARACTERISTICS

TYPE OF FUEL	SINTERED UO_2, UO_2+Gd_2O_3
Fuel enrichment	<5 weight % (4.6 weight % on average in the reference 26 effective full power months (EFPM) design)
Type of coolant/moderator	Light water
Core type Equivalent diameter Active height	Open lattice, square 3.0 m 3.7 m
Reactor vessel type Structural material Total height Maximum inner diameter	Cylindrical Low alloy steel 18 m 6 m
Thermodynamic cycle type	Indirect cycle
SG type Tube material	U type (SGL); C type (SGV) TT 690 alloy
Number of loops	Integrated system

Neutron-physical characteristics [VI-6, VI-7]

The IMR fuel assemblies use square type open lattices, like in conventional PWRs. The major differences are as follows:

- The IMR has no liquid boron system. Therefore, the IMR has a large negative reactivity feedback for coolant temperature and void fraction.
- The hydrogen-to-uranium ratio is set to 5, which are larger than in conventional PWRs, to reduce the pressure drop in the primary circuit. This design feature is facilitated by the absence of liquid boron system.
- The burn-up reactivity swing is around 1% $\Delta\rho$, which is achieved by Gd integrated in fuel and separate burnable absorber rods (Fig. VI-3).
- The coolant boils in the upper part of the core and the core outlet void fraction is around 20%. To reduce axial power peaking caused by coolant boiling, the fuel consists of two parts, the upper part with higher enrichment and the lower part with lower enrichment. Additionally, hollow annular pellets are used in the upper part fuel.

Reactivity control mechanism [VI-5, VI-6, VI-7]

Control rods perform the reactivity control; a soluble boron chemical shim system is not used in the IMR except for the backup shutdown system.

A neutron absorber of the control rods is 90% enriched B_4C, which is the design solution to increase the reactivity worth of control rods and reduce the number of Rod Cluster Controls (RCCs).

The RCCs are separated into two groups: the control group (32 clusters) and the shutdown group (60 clusters). The control RCCs govern reactivity changes with burn-up and power level; the shutdown RCCs govern reactivity changes between the cold zero power state and the hot zero power state. Only 12 control RCCs are inserted in the core during full power operation. Either of these groups can move the reactor from a hot full power to a hot shutdown state.

For redundancy of the shutdown systems, a boric acid injection system is used as the backup to control rods.

Control rod drive mechanisms (CRDMs) are located inside the reactor vessel. Rotating motors and separable ball-nuts, which open to scram, drive the RCCs.

To reduce the burn-up reactivity swing and the number of control rods required, all fuel assemblies contain integrated Gd-fuel rods. Separate burnable absorber rods are also used to reduce the reactivity swing. The RCC data is summarized in Table VI-2.

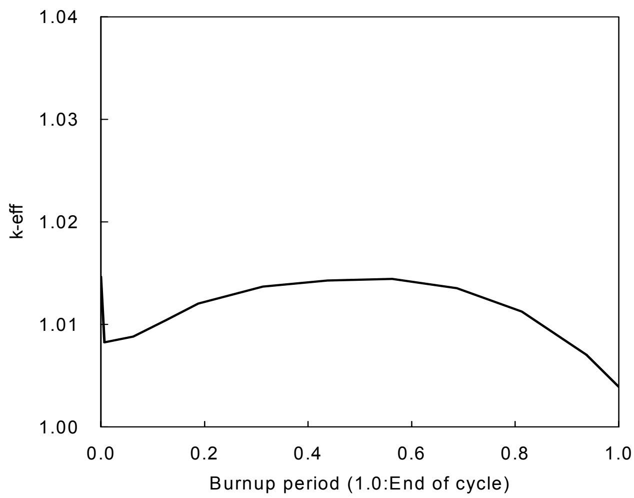

FIG.VI-3. K_{eff} change with burn-up.

TABLE VI-2. REACTIVITY CONTROL SYSTEM

Burnable absorber Fuel Burnable poison rod	 9 weight % Gd_2O_3 Pyrex including natural boron
Control rod Absorber material Number of rods	 90 weight % enriched B_4C 32
Number of RCCs Shutdown margin for scram Shutdown margin for refuelling	92 >1% Δρ, including 1 rod stuck >5% Δρ

Cycle type and thermodynamic efficiency

Cycle type: Indirect cycle

Plant thermal efficiency: 35%

Thermal-hydraulic characteristics

The thermal-hydraulic characteristics are summarized in Table VI-3.

TABLE VI-3. THERMAL-HYDRAULIC CHARACTERISTICS

CIRCULATION TYPE	TWO-PHASE FLOW NATURAL CIRCULATION
Primary circuit volume	Approximately 250m^3
Core inlet temperature	307°C
Core outlet temperature	345°C
Primary coolant flow rate	3000 kg/s
Reactor operating pressure	15.5 MPa
Core outlet average void fraction	20%

Maximum/average discharge burn-up of fuel

In the reference 3 batch core, corresponding to 26 EFPM operation cycle:

Maximum discharge burn-up: 49 000 MW·d/t U
Average discharge burn-up: 45 000MW·d/t U

Fuel lifetime/period between refuellings

In the reference core design:

Fuel lifetime: 6 years (3 batch refuelling)
Period between refuellings 790 EFPD (26EFPM)

Mass balances/flows of fuel and non-fuel materials

The mass flows of materials are summarized in Table VI-4.

TABLE VI-4. MASS FLOWS OF MATERIALS

	TYPE	MATERIAL	WEIGHT
Fuel	Pellet	UO_2, Gd_2O_3	8.6 t/y/MW(th)
Structural materials	Cladding tubes, Thimble Tubes, Grids, Nozzles	Zircaloy, Inconel	2.9 t/y/MW(th)

For the reference 26 EFPM / 3batch core with 95% load factor

Design basis lifetime for reactor core, vessel and structures

The design basis lifetime for main structures is 60 years.

Design and operating characteristics of systems for non-electric applications

The design for such applications has not yet been performed.

Economics

Capital costs:	Equivalent to large scale reactors (target);
Estimated construction period:	2.5 years from 1st concrete pour to commercial operation;
O & M costs:	Reduced by the simplicity of installation but not yet estimated.

VI-1.5. Outline of fuel cycle options

Basically, the fuel cycle option of the IMR is the same as with conventional LWRs. The reference IMR design study has been done assuming the use of low-enriched uranium dioxide fuel (less than 5 weight % U-235); the capability of partially or fully loading MOX fuel will be studied later.

Additionally, the properties of the spent fuel of the IMR are similar to conventional LWRs; it can be reprocessed at existing reprocessing plants. The applicability of MOX fuel reprocessed by methods such as the pyro-metallurgical process, with low decontamination factors, will be studied later.

VI-1.6. Technical features and technological approaches that are definitive for IMR performance in particular areas

VI.6.1. Economics and maintainability

The capital costs per MW(e) are expected to be similar to large-size LWRs. The low total capital cost is due to the small size, reduction of site construction labour, easy normal and post-accident operation and reduction of maintenance to facilitate construction of the IMR in many countries, including developing countries.

The main features of the IMR to achieve low capital and construction costs are:

- Elimination of heavy components such as reactor coolant pumps, pressurizer and main coolant pipelines.
- The small containment vessel and reactor building achieved by the integrated primary system design and simplified systems.
- A simplified chemical and volume control system (CVCS) and waste disposal system (WDS) achieved by the boric-acid free design.
- Elimination of the emergency core cooling system (ECCS) and containment cooling system, such as a containment spray system (CSS).
- Simplification of supporting systems, such as the component cooling water system (CCWS), the essential service water system (ESWS) and the emergency AC power system. These are designed as non-safety grade systems, possible by use of the stand-alone direct heat removal system (SDHS).
- Rationalization of design, manufacture and construction, is achieved by modular and standardized design.
- A short construction period achieved by modular construction and use of pre-fabricated equipment.

The main features of the IMR to reduce O&M costs are:

- Reduced operating manpower through the reactor being capable of self-adjustment to the load and a essentially operator-free performance during accidents.

- Reduced maintenance by simplification and elimination of certain equipment items, such as the reactor coolant pumps and some safety systems, the 2-year long operation cycle.

The low-power density core design enables an efficient 3-batch refuelling strategy even for a two-year long cycle operation and results in lower fuel cycle costs. The fuel type is almost the same as in conventional PWRs and the enrichment is less than 5 weight %.

VI-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

The fuel design and core neutron-physical characteristics are equivalent to conventional LWRs but the power density is lower. This makes it possible to increase the average fuel burn-up especially for long cycle operation and therefore, fuel utilization is similar but a little better than in conventional LWRs.

The amount of solid, liquid and gas waste from the equipment and systems is reduced by a boron-free design, and simplification of the primary system helps reduce waste generated during maintenance.

The radiation exposure of workers is reduced by using less primary system equipment and by the integrated reactor vessel, which limits the boundary of the primary coolant circulation.

The reduction of the amount of spent fuel and waste is advantageous for waste management costs.

VI-1.6.3. Safety and reliability [VI-4, VI-8]

Safety concept and design philosophy

The basic principles of the IMR safety design are as follows:

- By design, eliminate initiating events that might cause fuel failure.
- During accidents, require no operator actions or external support such as water, power, etc.

By adopting the integrated primary system design and the HHTS without reactor coolant pumps and main coolant pipelines, the possibility of accidents that may cause fuel failure, such as a large-break loss of coolant accident (LOCA), rod ejection (R/E), loss-of-flow (LOF) and locked rotor (L/R), is essentially eliminated. During the normal operation, the water level in the reactor vessel is controlled by injection from the charging pumps. However, since the diameter of the pipes connected to the primary system (reactor vessel) is less than 10 mm, water level can be maintained to submerge the top of the core without any injection.

The IMR employs the stand-alone direct heat removal system (SDHS) as a safety system. Figure VI-4 shows the concept of SDHS; it is a closed natural circulation system supplying cooling water to the secondary side of the SGs by gravity. The cooling water is evaporated at the SGs and condensed in the passive steam generator coolers (PSGCs) of the SDHS.

In this system, decay heat is removed directly from inside the reactor vessel to the atmosphere. Therefore, even if water leakage occurred and the charging pumps did not work, water leakage would be terminated automatically when the pressures inside and outside the reactor vessel are equalized and the core is maintained being submerged. In the PSGC, decay heat is removed by water-cooling in early stages of an accident, with air-cooling replacing it at later stages. Therefore, since the SDHS has an infinite grace period, an external support such as water, power and human intervention become unnecessary.

A significant risk reduction is achieved in the IMR as follows:

- Fuel failure does not occur in design basis accidents.
- No external support is required to maintain the plant in a safe condition after accidents;
- A passive safety system with an infinite grace period is adopted.
- The reactor vessel integrity will be retained by core cooling through the reactor vessel wall, even if severe accidents occur.
- The containment vessel integrity will be maintained by submerging the containment vessel head in water for refuelling, even if severe accidents occur.

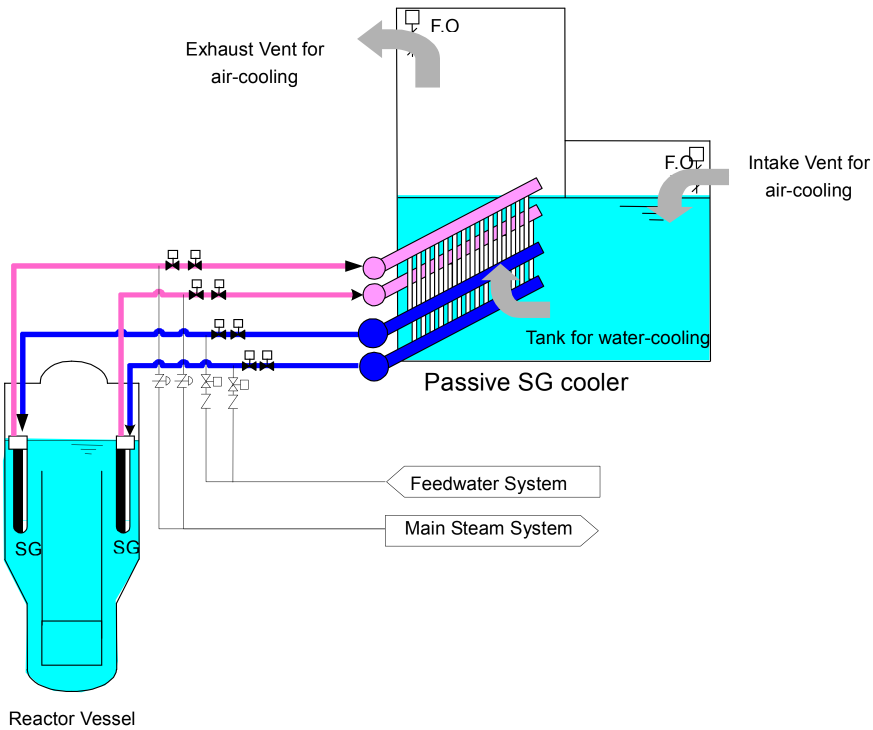

FIG. VI-4. Stand-alone direct heat removal system (SDHS).

Provisions for simplicity and robustness of the design

The ECCS and CSS installed in conventional PWRs are eliminated. The residual heat removal system (RHRS), CCWS, ESWS, emergency AC power system, and heating, ventilating, and air conditioning (HVAC) system of the main control room are designed as non-safety grade, since the SDHS directly removes decay heat from core to the atmosphere through the SGs without operator actions or external supports such as power, water, etc.

Structure of the defence-in-depth

The IMR will be designed, manufactured, constructed, and operated with the same quality, reliability, and safety margins, including negative reactivity feedback characteristics, and based on the same philosophy as conventional LWRs. Additionally, as the design basis, the IMR eliminates causes of initiating events which might result in fuel failure such as, loss of coolant accident, control rod ejection, loss of flow, and locked rotor, by employing integrated primary system design, the hybrid heat transport system (HHTS) and in-vessel control drive mechanisms.

In case of design basis accidents, the IMR detects abnormal condition and trips the control rods. Since the IMR has no soluble boron system as a chemical shim, control rod worth is enough to maintain cold shutdown conditions. Additionally, in case of a trip failure, stand-by shutdown systems inject borated water to shutdown the reactor. Residual heat is removed by a passive stand-alone direct heat removal system (SDHS). The SDHS works without operator action and external supports and keeps core conditions within the safety criteria.

The IMR design could remarkably decrease the possibility of radiological release. In addition, the containment vessel would work as a barrier even if a large radiological release from the reactor vessel were hypothesized.

Active and passive systems and inherent safety features

The active and passive safety systems and inherent safety features are summarized in Table VI-5.

TABLE VI-5. SAFETY FEATURES AND SYSTEMS OF IMR

CATEGORY	SYSTEM	NOTE
Passive system	Primary shutdown system (Control rods)	Rod insertion by gravity Enough reactivity worth to move the reactor from power operation to a cold shut-down
	Backup shutdown system (Boric acid injection system)	Boron injection by accumulated pressure
	SDHS	Closed natural circulation system with infinite grace period for emergency heat removal
	Emergency DC Power System (Batteries)	Batteries are required at the early stage of accidents to operate plant protection system including valves of SDHS
Active system	None	-
Inherent safety features	Integrated primary system	Eliminates large-break LOCA by design
	HHTS	Eliminates LOF, L/R by design
	In-vessel CRDM	Eliminates R/E by design

Design basis accidents and beyond design basis accidents

Design basis accidents of the IMR are supposed to cause no fuel failure with the functions of reactor shutdown and residual heat removal being shouldered by the SDHS.

In case of beyond-design-basis accidents (BDBA) such that the SDHS is not available, normal cooling systems, such as the component cooling water system (CCWS), residual heat removal system (RHRS), etc. are used if possible. In case normal cooling systems are unavailable, the reactor vessel integrity is retained by core cooling through the reactor vessel wall, and submerging the containment vessel head, using the water for refuelling, retains the containment vessel integrity.

Figure VI-5 and Figure VI-8 show the measures for severe accidents. When decay heat removal through the SGs is not applicable, water leaking out of the reactor vessel will fall to the bottom of the reactor vessel cavity. Since decay heat can be removed through the reactor vessel wall, molten core debris could be retained inside the reactor vessel. In addition, decay heat in the containment vessel could be removed through the containment head, which will be immersed in water supplied by the operators.

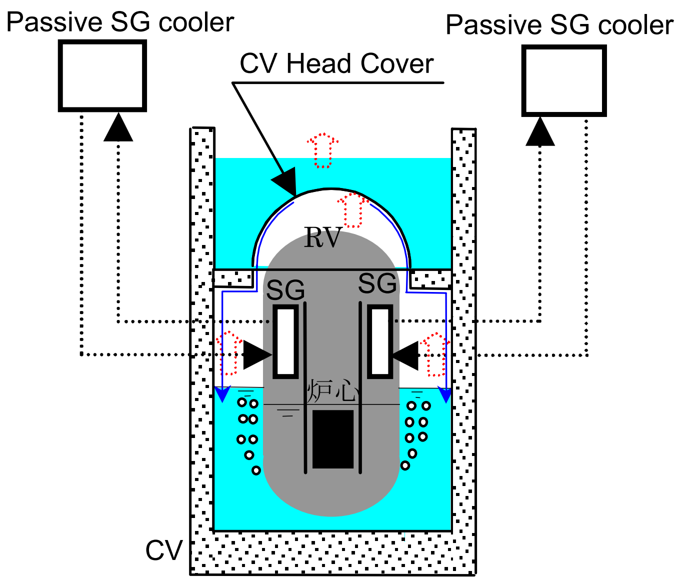

FIG. VI-5. Heat removal in BDBA.

Provisions for safety under seismic conditions

The IMR is designed for seismic conditions of Japan, an earthquake country. Seismic isolation devices could be installed in case of a site with severe seismic conditions.

VI-1.6.4. Proliferation resistance

The IMR is an LWR with moderation ratios similar to conventional LWRs, so that properties of fresh and spent fuel are also similar. Therefore, proliferation resistance is expected to be similar to conventional LWRs, i.e., the initial enrichment required is less than 5 weight % and spent fuel is hard to convert to a weapons-usable material. The low power density core of the IMR enables extending the refuelling interval to two to five years.

VI-1.6.5. Technical features and technological approaches used to facilitate physical protection of IMR

The IMR plant safety is supposed to be maintained by the SDHS, with no or little reliance on external support such as water, power and operators. This enhances protection against external impacts and sabotage.

VI-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of IMR

The IMR is a small to medium sized reactor, and lower initial costs must be favourable to reduce the risk and burden of investment. Simple and essential safety features of the IMR could also help gain public acceptance around the construction sites.

VI-1.8. List of enabling technologies relevant to IMR and status of their development

The major enabling technologies are:

(1) Technologies related to the HHTS and two-phase natural circulation. The water-steam two-phase flow under high temperature and pressure such as in the IMR is a technology with little practical experience, therefore, research, development and validation is needed in the following areas [VI-9 toVI-12, VI-13 to VI-15]:

- Two-phase fluid behaviour. The basic feasibility of the HHTS and the two-phase flow behaviour in the IMR have been tested in one-dimensional high temperature, pressure experiments. Analytical methods have also been validated through simulation analyses of the experiments. However, to understand actual (three-dimensional) behaviour of a two-phase flow, larger scale tests and analysis are required.
- Coupling analysis of nuclear physics and thermal hydraulics. The methods for such analysis have been developed and are applicable to the reactor design.
- Physical and chemical conditions and durability of the materials in the IMR. More data are required to validate the integrity of fuel cladding under boiling conditions. A strategy of water chemistry must be developed. Flow induced vibration (FIV) related issues could be neglected because of low coolant velocity (less than 1m/s in the riser).

(2) Integrated reactor vessel. There are no manufacturing issues but some plant facilities may need to expand their capacity.

(3) In-vessel CRDMs and their rod position indicators (RPIs). The technologies of in-vessel CRDM and RPI in the IMR are based on a design experience of the MRX marine reactor MRX developed by JAERI and MHI. The basic feasibility has been tested but in addition to the development programme, more data are required to confirm the durability of motors, bearings, and ball-nuts under the IMR high temperature operating condition. Specifically, high temperature RPI testing under the IMR design conditions is necessary. Methods of CRDM and RPI maintenance should be improved [VI-16 to VI-18].

(4) In-vessel SG. There are no manufacturing issues for the SGs. Tube inspection is available from the secondary side of the SG, and inspection devices should be improved [VI-6].

(5) Safety system (SDHS). The basic feasibility of the SDHS, such as heat removal capability, natural circulation, and the effect of non-condensable gases has been confirmed through the on-going development programme [VI-8].

VI-1.9. Status of R&D and planned schedule [VI-4]

The IMR is an IPSR categorized under international near term deployment (INTD) in the Generation IV International Forum. The Japan Ministry of Economy, Trade and Industry has been supporting its conceptual design study and feasibility testing of key technologies such as the HHTS and SDHS from 2001 to 2004.

Currently, the development team led by Mitsubishi Heavy Industries (MHI) consists of Kyoto University, Central Research Institute of Electric Power Industries (CRIEPI), the Japan Atomic Power Company (JAPC), and MHI.

The IMR is now at the conceptual design stage, to be completed in 2005. Validation testing, R&D for components and design methods, and basic design development are required before

licensing. The time required for development and deployment of the IMR depends on the financial situation and the extent of construction requirements. The target year to start licensing is 2011 at the earliest.

VI-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

The IMR is an IPSR and employs innovative technologies such as the HHTS, in-vessel CRDM, and SDHS to downsize the system and improve its economy and safety. Each elemental technology requires validation testing but construction of a prototype reactor is not considered to be necessary because the IMR is based on well-developed LWR technologies. Demonstration of the HHTS will require a large-scale thermal hydraulic test because there is little experience with a two-phase flow under high temperatures and pressures such as in the IMR.

VI-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

No information was provided.

VI-2. Design description and data for IMR

VI-2.1. Description of the nuclear systems

Reactor core and fuel design (VI-6, VI-7)

A cross section of the IMR fuel assembly and a core map of the IMR are given in Fig. VI-6 and Fig. VI-7 respectively. Design specifications for the IMR fuel and core are given in Table VI-6.

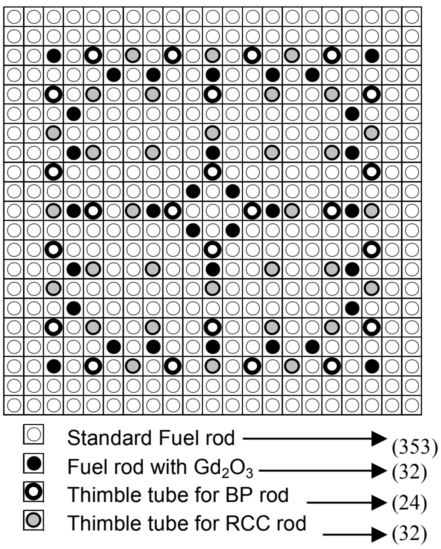

FIG. VI-6. Cross section of fuel assembly.

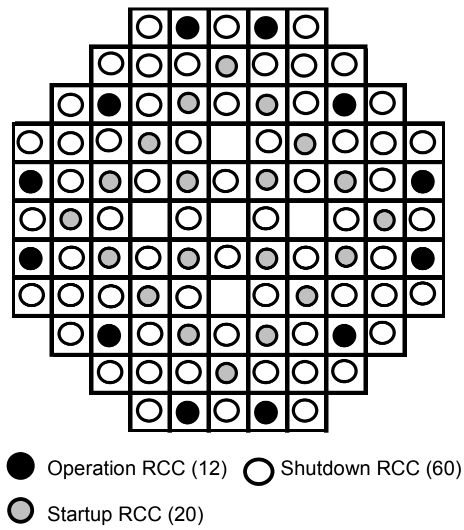

FIG. VI-7. Core map.

TABLE VI-6. FUEL AND CORE DESIGN SPECIFICATIONS

ITEM	SPECIFICATION
Fuel rod Outer diameter (mm) Effective height (mm) Cladding material	 9.0 3650 Zircaloy
U-235 enrichment (weight %)	4.55 and 4.95 for the reference core (26 EFPM / 3 batches)
Fuel assembly Rod arrangement Rod pitch (mm) Number of RCC thimbles Number of burnable poison thimbles	 21×21 12.6 32 24
Core Number of fuel assemblies Average linear heat rate (kW/m) Average power density (kW/l) Number of RCCs	 97 7.2 40 92

Main heat transport system

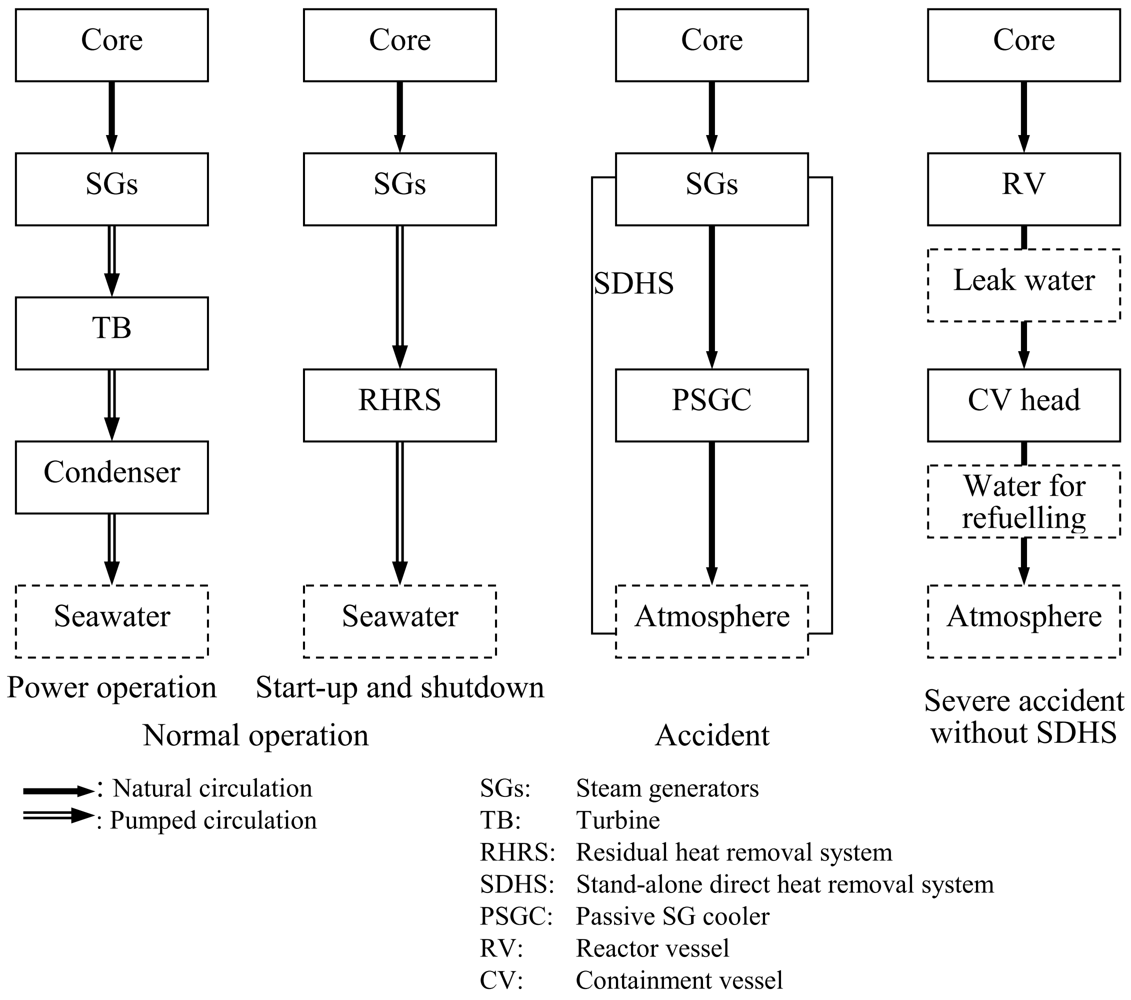

FIG. VI-8. Heat removal paths of IMR.

The scheme of main heat transport system with specification of heat removal path in normal operation and in accidents is presented in Fig. VI-8.

Design options

Several design options are being considered for the IMR, among them:

- Simplified SG design. To simplify the reactor design, the possibility of integrating SGL and SGV into a single SG has been studied. The final SG design will be built based on natural circulation experiments and analyses.

- 100 MW(e) output option. The output of the reference IMR design is 350 MW(e) (1000 MW(th)), which is selected as the maximum capacity under current manufacturing capabilities. To show the adaptability of the IMR for smaller applications, a 100 MW(e) (290 MW(th)) reactor has been designed for trial. Figure VI-9 shows a cross section and major dimensions of such reactor.

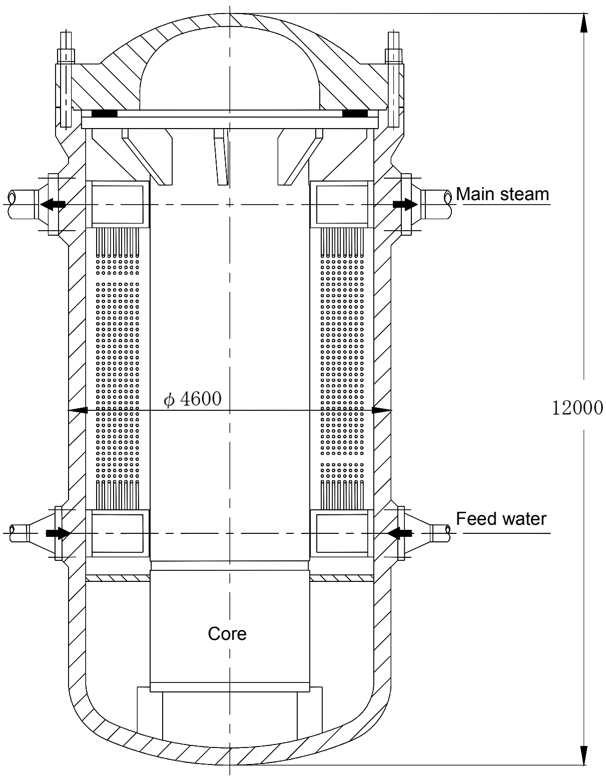

FIG. VI-9. 100MW(e) (290MW(th)) IMR.

VI-2.2. Description of the turbine generator plant and systems

The turbine generator plant and systems of the IMR are basically the same as in conventional PWRs, in which the power rate is similar to the IMR.

VI-2.3. Systems for non-electric applications

No information was provided.

VI-2.4. Plant layout

The plant layout of the IMR is optimized to satisfy various needs, such as safety, radiation, etc, with the same philosophy as applied to conventional PWRs. The important point of the IMR plant layout is the downsizing of building volumes to reduce construction costs. For this, the IMR adopts a small containment vessel and a small reactor building, including a containment vessel and a fuel handling building.

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ANNEX VII

WATER COOLED MODULAR POWER REACTOR (VBER-300)
OKBM, Russian Federation

VII-1. General information, technical features and operating characteristics

VII-1.1. Introduction

VBER-300 is the Russian abbreviation for a nuclear power plant (NPP) with a light water cooled modular power reactor of 300 MW(e).

The VBER-300 is a NPP with a modular pressurized water reactor developed on the basis of shipboard reactor technologies. The Russian Federation has a solid experience in the design and technology development, as well as in the construction and operation of such reactors.

The high quality of shipboard modular design reactors is guaranteed by the long-term accident-free operation of Russian nuclear-powered icebreakers “Arktika”, “Sibir”, “Rossia” and others, under more exacting conditions than those of land-based nuclear power plants (NPPs).

Modular pressurized water reactors are based on the most highly developed reactor technologies, examined and proven by the successful operating experiences of shipboard nuclear power plants. The operating experience of shipboard reactors exceeds 6000 reactor-years. This experience is comparable to that of the nuclear power industries in such industrialized countries as France and Japan.

Long-term experience in the design, construction and operation of shipboard reactors and the results of R&D for their design validation, the technological base and the personnel potential of Russian enterprises are the background to create highly-reliable nuclear power sources for NPPs.

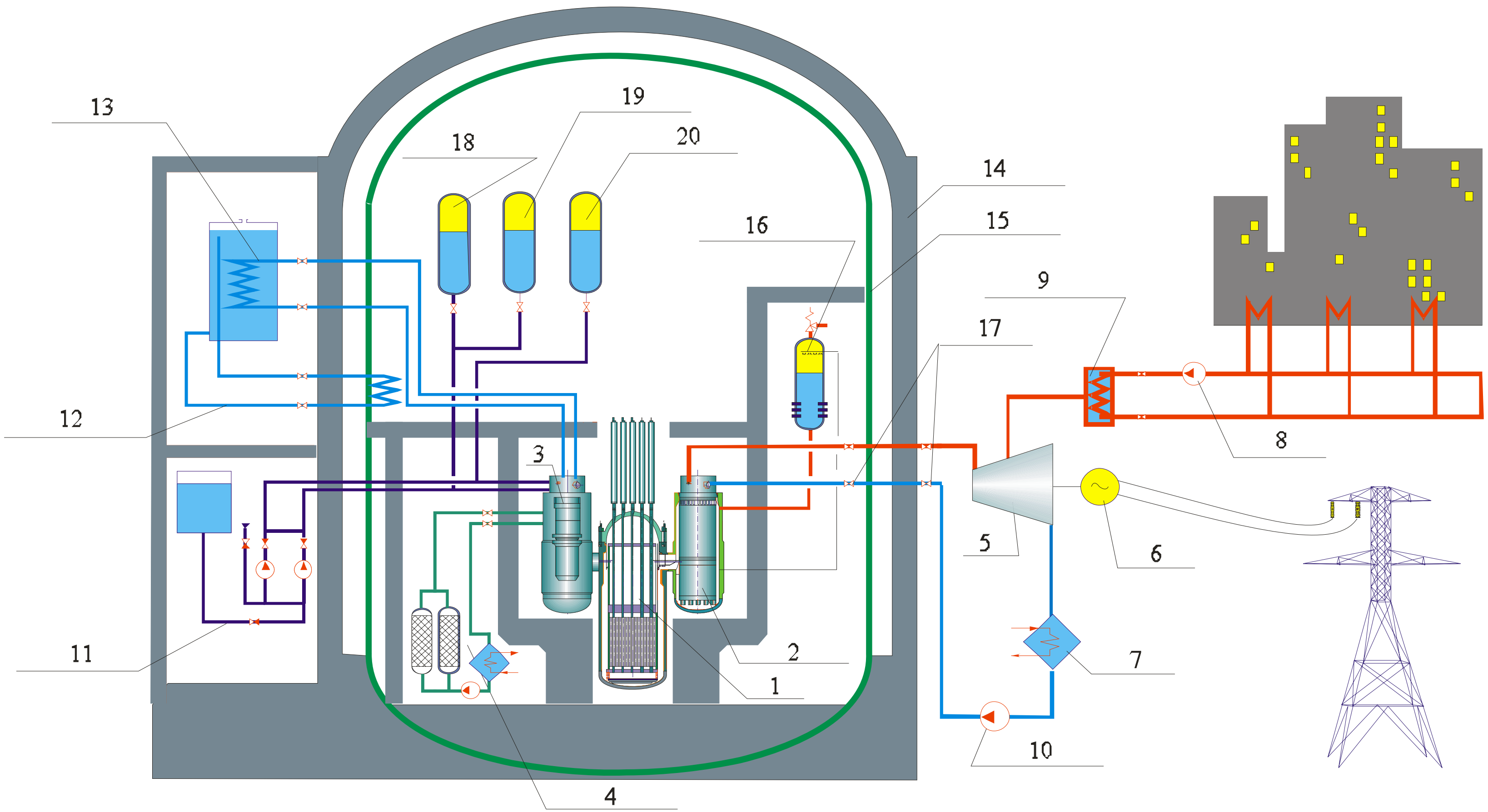
The VBER-300 plant design is a result of the evolution of shipboard modular reactors. The thermal power increase is due to an increase in mass and overall dimensions keeping a reactor plant pattern and the main design solutions as close as possible to those of shipboard reactors. The design was developed using operating experience with the VVER-type reactors and achievements in the field of nuclear power plant safety.

The principal stakeholders are Russian research and design organizations: OKB Mechanical Engineering (OKBM, Nizhny Novgorod), Russian Research Centre “Kurchatov Institute” (RRC “Kurchatov Institute”, Moscow), Scientific-Research and Design Institute “Atomenergoproekt” (NIAEP, Nizhny Novgorod), and Public Company “Lazurit”, Nizhny Novgorod

VII-1.2. Applications

The VBER-300 reactor is a small-to-medium power source for land-based NPPs and cogeneration plants as well as for floating NPPs and desalination complexes. The applications are:

- Electricity generation;
- Cogeneration of electricity and heat for district heating; or
- Seawater desalination.



1-Reactor
2-Steam generator
3-Main circulating pump
4-Primary
5-Turbine
6-Generator
7-Condenser
8-Circuit pump
9-Circuit heat exchanger
10-Feedwater pump
11-Water and boron solution makeup system
12-Protective enclosure pressure drop system
13-Emergency heat removal system
14-Containment
15-Steel protective enclosure
16-Steam pressurizer
17-Stop valves
18-Hydraulic accumulator
19-Secondary stage ECCS tank
20 Boron solution passive supply system

FIG. VII-1. Principal scheme of a nuclear cogeneration plant with VBER-300 reactor.

VII-1.3. Special features

Special features of the VBER-300 are [VII-1 to VII-3]:

- An option of building both land-based and floating NPPs;
- A variety of cogeneration options;
- The possibility to enlarge or reduce the VBER module power using unified equipment only, i.e. the use of the two- or three-loop NPP layout.

The VBER-300 is well suited for autonomous operation and for operation in small power networks in immediate proximity to the customer.

VII-1.4. Summary of major design and operating characteristics

The principle scheme of a land-based cogeneration plant with the VBER-300 is given in Fig. VII-1.

A standard two-circuit system is used for heat removal from the core. Light water acts as a primary coolant and moderator. The hot primary coolant is cooled in a once-through steam generator. A slightly superheated steam is supplied to the turbine in the secondary circuit. Part of the steam is taken off from the turbine and directed to the heat exchanger of a district heating circuit.

TABLE VII-1. SUMMARY TABLE OF MAJOR DESIGN AND OPERATING CHARACTERISTICS

PARAMETER	VALUE
Design characteristics	
Reactor power, MW - Thermal; - Electric	 850 295
Operation mode	Base load operation; load follow modes, e.g., to track daily load changes, or a dispatcher mode with maintaining the frequency are possible
Capacity factor	0.85-0.9
Fuel	
Fuel type	Pellets of sintered uranium dioxide
Fuel element	Rod-type fuel element similar to standard fuel elements of the VVER-1000 reactor
Fuel assembly	Ductless skeleton-structure fuel assemblies of AFA type
Fuel enrichment	Not more than 5%
Coolant	Water (H_2O)
Moderator	Water (H_2O)
Core	
Number of fuel assemblies	85
Circumscribed diameter, mm	2420
Height, mm	2900

TABLE VII-1 (cont.)

Reactor vessel and internals	
Overall height, mm	14 750
Circumscribed diameter, m	10
Operating mass, t	988
Reactor core barrel diameter, inner/outer, mm	3300 / 3700
STRUCTURAL MATERIALS	
Core	
Fuel element cladding	Zirconium alloy
Fuel assembly structural elements	Zirconium alloy
Reactor vessel and internals	
Reactor vessel	Heat-resistant pearlite steel with anticorrosive facing
Steam generator vessel	As above
Vessel of the hydraulic chamber of a circulating pump	As above
Steam generator piping	Titanium alloy
Reactor internals	Stainless steel 08Cr18Ni10Ti
NPP	
Thermodynamic cycle	Steam-turbine cycle on slightly superheated steam
Number of circuits	2
Reactor type	Modular pressurized water reactor on thermal neutrons

NEUTRON-PHYSICAL CHARACTERISTICS	
Reactivity coefficients	*Rated value*
Coolant temperature reactivity coefficient (taking into account coolant density changes), 1/°C	-78.6×10^{-5}
Coolant density reactivity coefficient (without taking into account coolant temperature), $1/(g/cm^3)$	0.33
Fuel temperature reactivity coefficient, 1/°C	-1.7×10^{-5}
Boron reactivity coefficient, % / (g/kg)	-1.38

REACTIVITY BALANCE		
*Parameter**	*BOC*	*EOC*
Samarium-149 poisoning in cold state, %	-0.80	-0.74
Worth of operating-concentration boron acid in cold state, %	-11.3	~0
Temperature reactivity effect, %	-2.60	-7.20
Doppler effect, %	-1.36	-1.38
Effect of irregular distribution of coolant temperature and density over core volume at power increase from zero to $N=N_{rated}$, %	-0.20	-0.30
Stationary xenon-135 poisoning, %	-2.50	-2.60
Reactivity margin for fuel burn-up (between refuellings), %	5.5-9.5	
Reactivity effect at complete drainage of the core, %	-60	

* For the core option of 2700 mm height; at a height increase of 200 mm the presented parameters are practically not changed.

TABLE VII-1 (cont.)

PEAKING FACTORS	
Fuel assembly	1.36
Core volume	1.58
Approaches used to reduce power peaking	Fuel assembly reshuffling scheme; use of boric acid to compensate for reactivity margin
REACTIVITY CONTROL	
Reactivity margin for fuel burn-up	Fuel elements with gadolinium oxide integrated in fuel pellets; boric acid
Compensation of thermal and power effects of reactivity; reactivity margin for core poisoning by xenon-135 and samarium-149; operating margin for reactivity changes under reactor power changes and for maintaining core subcriticality in the cold unpoisoned state	Control rod clusters: bundles of 18 absorber rods joined by a common traverse and travelling inside the fuel assembly guide tubes. Separate actuator for each cluster.
Control and protection system (CPS)	All CPS rods (72 clusters/control rod drives) enter the core driven by gravity at de-energization of their drives according to an emergency signal. System of boric acid emergency injection using makeup pumps.
THERMAL-HYDRAULIC CHARACTERISTICS	
Primary circuit parameters	
Circulation type	Forced circulation using canned MCPs with fly-wheels
Coolant flow rate, t/h	13 610
Coolant temperature at core outlet, °C	332
Coolant temperature at core inlet, °C	294
Coolant velocity in the core, m/s	2.5
Primary circuit coolant pressure, MPa	15.7
Maximum fuel temperature, °C	1000
Average fuel temperature in the core, °C	600
Maximum temperature of fuel element cladding, °C	352
Average temperature of fuel element cladding, °C	340
Maximum acceptable fuel temperature, °C	2500
Maximum acceptable temperature of fuel element cladding, °C	700
Minimum margin to heat exchange crisis (DNBR)	1.56
Secondary circuit parameters	
Steam pressure after steam generator, MPa	6.38
Steam output, t/h	4×365
Steam temperature at steam generator outlet, °C	305
Feedwater temperature, °C	185

TABLE VII-1 (cont.)

BURN-UP CYCLE AND MATERIAL BALANCES		
Cycle length	Flexible, with operating period between refuellings from one to two years	
Cycle characteristics (for operation mode with one partial refuelling per a year and half *)		
Number of fuel assemblies in a batch	22	
Partial refuelling repetition factor	3.68	
Uranium inventory in a batch, t	7.94	
Cycle length, effective days	485	
Fuel burn-up	64.2 MW·d/kg U in the mode with 15 fuel assemblies in a refuelling batch	
Specific consumption of natural uranium, g / (MW·d)	213	
* Cycle characteristics for other operation modes (refuelling once in one or two years) are presented in Table VII-2.		
DESIGN SERVICE LIFE		
Reactor vessel and internals, yr.	60	
Steam generator piping, yr.	25-30	
Main circulating pump, yr.	25-30	
Economic characteristics	Nuclear cogeneration plant	Floating NPP, PAES-600
Plant construction cost*, million US $	640	484
Capital investments, US $/kW(e)	1084**	820
Costs for operation and maintenance, thousand US $/year	~ 91	-
Fuel cost (initial fuel inventory), thousand US $	~ 31	~ 31
Projected initial cost of generated electric power (condensation mode), cent/kW·h	2.2	1.8
Pay-back period, yr.	12-14	7

* For the central part of European territory of Russia

** For VVER-1000 (V-392 design) this value is ~US $920 /kW as per the paper "Technical and Economical Analysis of NPP Designs with VVER-1000 (V-428), VVER-1000 (392), BN-800, MKER-1000, VVER-640, VVER-1500, UVR-1500 reactors" – paper of a working group of the Ministry for Atomic Energy of the Russian Federation, Moscow 1999

VII-1.5. Outline of fuel cycle options

The basic fuel cycle option for the VBER-300 is a once-through fuel cycle with enriched uranium dioxide fuel, similar to that used in the standard Russian VVER reactors. The major characteristics of the considered refuelling modes of the VBER-300 are given in Table VII-2.

Like standard VVER-1000 reactors, the VBER-300 could also operate with MOX fuel in a closed fuel cycle.

TABLE VII-2. VBER-300 REFUELLING OPTIONS

PARAMETER	VALUE			
Length of cycle between refuellings, effective full power days	362	485	543	629
Number of fuel assemblies in a batch - With 2.4% uranium enrichment; - With 5.0% uranium enrichment.	 - 15	 1 21	 1 24	 - 30
Partial refuelling repetition factor	5.66	3.86	3.40	2.83
Uranium enrichment in a refuelling batch, %	5.41	7.94	9.02	10.82
Uranium-235 load in a refuelling batch, kg	270	386	440	539
Specific consumption of natural uranium, g/(MW·d)	199	213	216	229
Discharged fuel burn-up, (MW·d)/kg U: - Average for fuel assemblies; - Maximum for fuel assemblies; - Maximum for fuel elements in fuel assemblies	 56.7 59.4 64.2	 53.6 58.5 63.2	 52.6 54.1 58.4	 49.4 57.2 61.8

The fuel cycle using thorium based on A. Radkowsky's RTR (Radkowsky Thorium Reactor) concept [VII-4], which has been studied in the RRC "Kurchatov Institute" for some years, is considered the prospective development of the VBER-300 to increase cost efficiency, enhance safety and protection of the environment. As a result of investigations performed by the RRC "Kurchatov Institute", the concept of the VVER-T reactor has been developed, making maximum use of the existing VVER-1000 technology, equipment and engineering solutions [VII-5]. This concept is also supposed to be usable in the VBER-300.

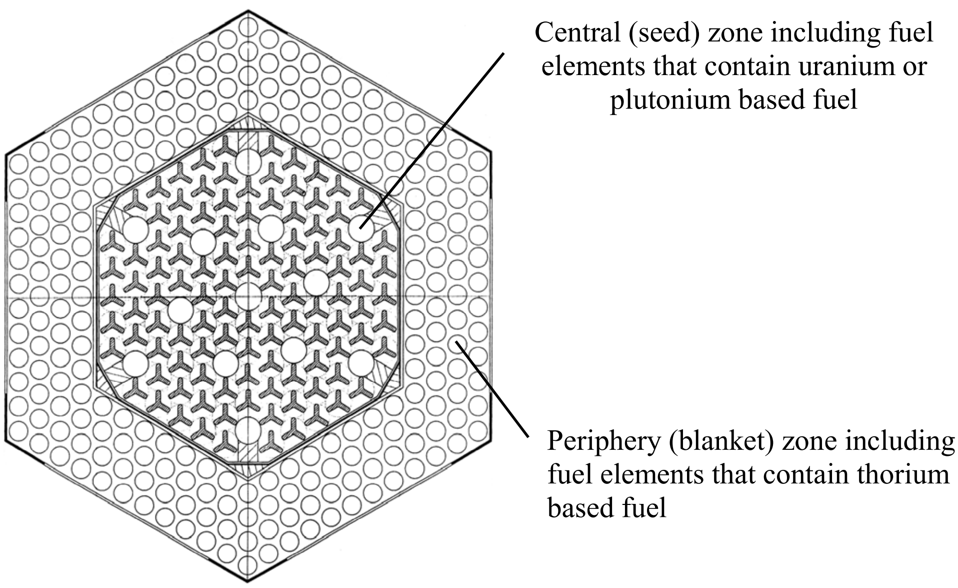


FIG. VII-2. The diagram of fuel arrangement in fuel assembly with thorium fuel.

As per the concept, the fuel assembly with thorium fuel consists of two demountable parts (Fig. VII-2):

- Fuel assembly's central zone (the seed zone) arranged of the fuel elements containing fuel based on uranium or plutonium;
- Fuel assembly's peripheral zone (the blanket zone) that consists of the fuel elements containing fuel based on thorium.

The application of the RTR fuel concept provides the following benefits:

- A solution to the proliferation issue: neither fuel loaded nor that discharged from the reactor can be used for nuclear weapons production;
- Enlargement of the fuel base as a result of the involvement of thorium, with the generation of a large part of the power due to U^{233} burn-up without recycle;
- Savings in natural uranium consumption and the associated fuel cost reduction;
- Decreases of highly active and highly toxic wastes including plutonium and minor actinides, compared to reactors under operation; and
- An option to dispose of weapon grade plutonium.

Based on engineering requirements developed by the RRC "Kurchatov Institute", OKBM is performing design studies of the RTR type fuel assemblies for the VVER-T reactor using the AFA design. These studies are applicable in full to the VBER-300.

As provided by the RTR concept, in the central (seed) part of the fuel assembly, the fuel elements based on shipboard reactor technologies are used. These include bimetallic (U-Zr) fuel elements or cermet fuel (UO_2 in metal matrix). Fuel elements of the peripheral (blanket) zone are similar in design to those of the VVER-1000, with a pellet fuel based on the mixture of thorium dioxide (~90%) and uranium dioxide (~10%).

There are 108 fuel elements in the seed zone at the fuel element pitch in a regular triangular lattice is 12.75 mm; the fuel element pitch in the blanket zone is 11.73 mm. Uranium enrichment in the seed zone and blanket zone does not exceed 20%. Maximum fuel burn-up in the blanket zone is 90 MW·d/ (kg heavy atoms), in the seed zone ≥100 MW·d/kg U. The design parameters of the seed and blanket zones ensure that about 30% of the generated power comes from fission of ^{233}U, produced and incinerated *in-situ*. Therefore, the natural uranium consumption is decreased by ~20%.

The VBER-300 design based on the RTR type fuel assemblies developed by the RRC «Kurchatov Institute», decreases the generation of actinides and plutonium, which determine the long-term radiation waste hazard. The specific annual plutonium production is about 4 times less than that in the reactor with purely uranium fuel [VII-6].

Considerable advantages are also provided in the volume of fuel enrichment, fuel element manufacturing and in the size of spent fuel storage. The plutonium-thorium fuel cycle (with plutonium fuel arranged in the seed zone) is also being considered as a prospective trend of the RTR concept application to the VBER-300, with a specific goal of the excess weapon grade and reactor grade plutonium utilization.

VII-1.6. Technical features and technological approaches that are definitive for VBER-300 performance in particular areas

VII-1.6.1. Economics and maintainability

To enhance the technical and economic characteristics of an NPP with the VBER-300, the following solutions and approaches are accomplished in the design:

- A compact modular layout of the reactor unit, including primary circuit and the main primary equipment, providing a reduction of the metal intensity of the reactor unit and accordingly, reduction in the construction scope of a reactor compartment;
- Increase of the reactor plant service life to 60 years;
- Increases in plant efficiency due to the combined use of installed capacity for cogeneration;
- NPP location in immediate proximity to cities results in a corresponding minimization of the expenses for heat transport to the customers and in reduction of the energy losses to the environment;
- The simplified requirements to safety systems (as comes to their scope, capability, operation speed, power supply, and control and monitoring requirements) due to a strong reliance on inherent safety features and the principles of passive actuation and systems operation;
- Improvements in fuel cycle characteristics, decreasing the annual requirements for fresh fuel and for spent fuel management;
- The adoption of proven designs for fuel assemblies and structures, based on the established technologies of nuclear reactors for icebreakers (including the experiences of serial production), as well as the VVER-1000, the AST-500, and the KLT-40S reactors;
- The optimized parameters of the thermodynamic cycle, resulting in improvements of the plant efficiency (generation of steam with economically optimal parameters);
- A reduction of waste management costs through the reduction of liquid and solid radioactive wastes by the use of leak-tight equipment and systems and by increases in the service life of the main replaceable equipment (steam generators, pipe systems, removable parts of the main circulating pumps, etc.); and
- Reduction of the refuelling period and an option to perform scheduled maintenance and repair works during a refuelling.

The use of a well known and mastered design scheme with a compact modular layout of the primary circuit in which short nozzles connect the main plant equipment (reactor, steam generator, MCP), is the main factor contributing to improvement of the technical and economic characteristics of the VBER-300. Such a design provides minimal mass and overall dimensions of the primary circuit, reduces the primary piping, minimizes the construction scope of the reactor compartment and consequently, reduces the capital construction investment per unit.

The corresponding values specific for a nuclear cogeneration plant with the VBER-300 are given in Table VII-3.

TABLE VII-3. SPECIFIC INTENSITY, CONSUMPTION AND CONSTRUCTION VOLUME DATA FOR A NUCLEAR COGENERATION PLANT WITH VBER-300

PARAMETER	VALUE
Specific metal intensity of primary circuit equipment (t/MW)	3.17
Specific construction volume for main reactor vessel (m^3/MW)	633
Specific consumption of concrete and reinforced concrete (m^3/MW)	110.1
Specific consumption of metal structures taking into account the protective enclosure made of steel (t/MW)	9.3

The PAES-300 floating plant with the VBER-300 possesses practically all the characteristics of a land-based plant and has the following additional advantages:

- Floating plants can be serially manufactured under shipyard conditions, and then delivered to a customer fully assembled, tested and ready for operation.
- The minimum scope and cost of capital construction is required for a floating plant location in a water area (4.5 ha), compared to large areas of alienable territory (up to 30 ha) required for land-based NPPs.
- There is no need to create transportation links and energy communications or preparatory infrastructure to realize the PAES-300 project.
- A high degree of freedom in selecting the location of a floating NPP; the possibility of mooring in many coastal regions of the world independent of their seismicity.
- Considerable reduction in the construction period (down to 4–5 years) and consequently, reduction in the repayment period of construction credit.
- The infrastructure for nuclear ship maintenance available in Russia would make it possible to minimize maintenance costs in operation of the plants as well as reduce the requirements for local labour skills, which could be especially important when exporting such plants to developing countries.
- The adoption of the available technologies for the utilization of ships with nuclear reactors, which would make it possible to realize the “green lawn” concept at the site of a floating NPP operation or, if necessary, to replace the amortized floating plant with a new one.

VII-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

The improved fuel utilization efficiency and the reduced natural uranium consumption are secured by the following conceptual and engineering solutions of the VBER-300:

- All improvements of the nuclear fuel and fuel cycles of the VVER-1000 reactors are directly applicable to the VBER-300, including the transition to a closed fuel cycle.
- Specifically, there is a possibility to enlarge the fuel base by involving thorium for generation of a large part of the power by ^{233}U production and fission without recycle.
- There is also a possibility to use MOX fuel.
- Increases in the fuel burn-up are provided by geometrical stability and operational reliability of the skeleton-design ductless fuel assemblies.

The following design features of the VBER-300 contribute to minimization of the radioactive wastes:

- The leak-tight primary circuit, which is standard for shipboard reactors.
- A closed-loop system of primary coolant purification and boron removal.
- The use of waste less technologies in coolant management.
- The reprocessing of radioactive wastes using the state-of-the-art low waste technologies.

The use of the design features and technologies proven by multi-year operation experience of icebreaker reactors secures that the quantity of radioactive waste from the VBER-300 plant will not exceed: for solidified liquids ~ 40 m^3, and for solid wastes ~ 20 m^3.

The radiation safety of a NPP with the VBER-300 reactor is achieved by meeting the requirements for limiting the irradiation impacts on personnel, population and the environment during all operation regimes, including abnormal operation occurrences and accidents up to those with severe fuel damage.

The design provides for a set of technical features and organizational measures to minimize the possible level of personnel and population irradiation, among them:

- Effective biological shielding.
- A modular layout of the main equipment, including the reactor, steam generator, leak-tight circulating pumps, and the connections of vessels and equipment with short nozzles to replace lengthy pipelines that permit depressurization accidents with the rupture in a large section.
- A closed-loop system of primary coolant purification and boron removal that excludes the leakage of the radioactive medium in the primary circuit and prevents the corresponding activity from entering the atmosphere during plant operation.
- Use of the intermediate water loops, including those between the primary and district heating circuits.
- A containment sharing the tasks of protection against natural and human-induced external impacts and resistance to the internal emergency impacts.
- Radiation control measures.
- A division of the plant production area into two zones: the controlled access zone and the free access zone.
- Establishment of a control area and a radiation-control area near the NPP.

VII-1.6.3. Safety and reliability

Safety concept and design philosophy

The principal decisions on safety of the power unit with the VBER-300 reactor are made based on a systems approach integrating the experience and achievement of safety in nuclear stations and shipboard nuclear plants and the requirements defined by locating a nuclear source near large populated areas and providing resistance against human actions of malevolent character.

The engineering solutions incorporated in the design correspond to worldwide trends followed by all state-of-the-art advanced NPPs:

- The priority is given to measures preventing accidents from occurrence.
- The plant design is simplified.
- The design incorporates inherent safety features and passive safety systems.
- The defence in depth approach is used to assure a high level of safety.
- An effort is made to increase protection against the impacts of the external events.
- Measures are foreseen to limit the consequences of severe accidents.

Active and passive systems and inherent safety features

Self-protection properties

Principal attention in the VBER-300 design is paid to the provision of self-protection properties intended to activate energy release self-limitations and reactor self-shutdown, the limitations of pressure and temperature, coolant heating rate, as well as to limit the scope of the primary circuit depressurization and outflow rate; all to maintain the reactor vessel integrity in severe accidents.

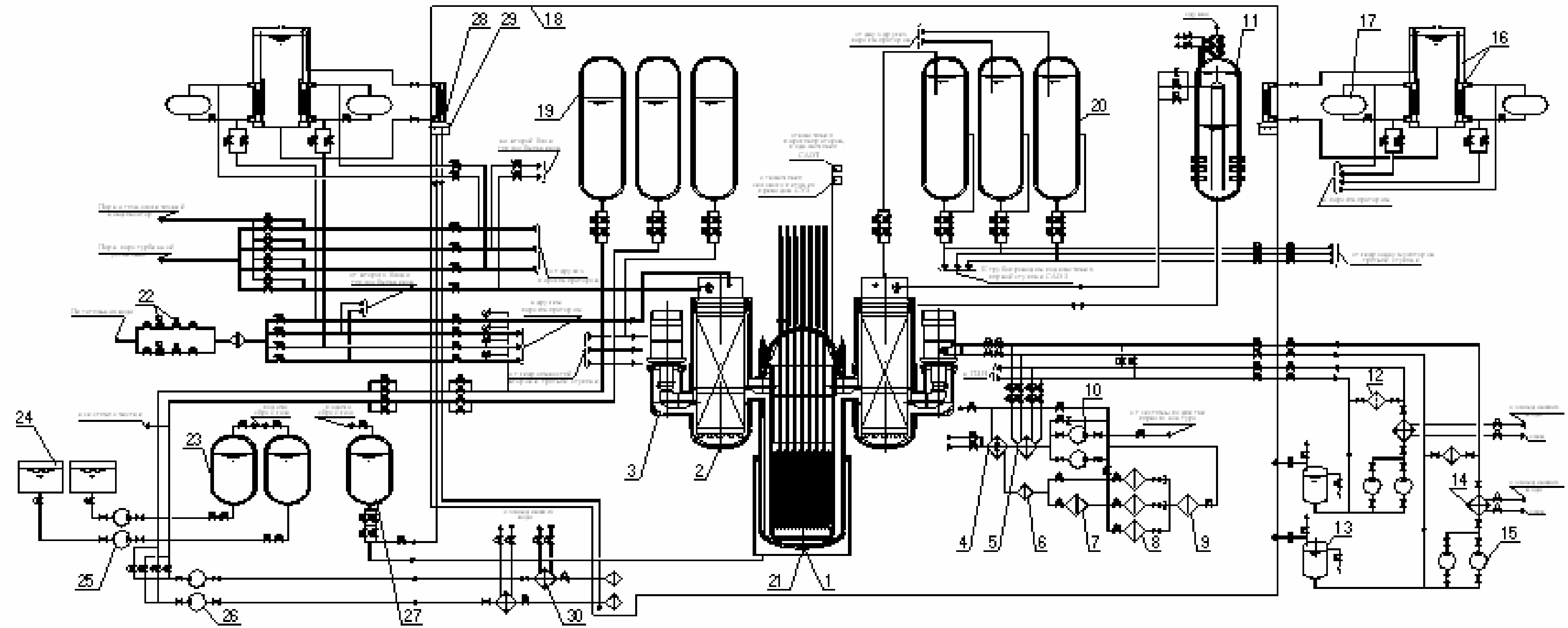
The self-protection properties are stipulated by the inherent and passive safety features of the reactor, ensuring stability to all possible disturbances including mistakes of the personnel and human actions of malevolent character. The inherent and passive safety features of the VBER-300 are:

- Negative fuel and coolant temperature reactivity coefficients, negative reactivity coefficient on coolant specific volume, as well as negative steam and integral power reactivity coefficients.
- Decreased core power density, compared with shipboard reactors and the VVER-1000 type reactors (less than 72 kW/l).
- A stable natural circulation in all heat transfer circuits, providing heat removal from the shut down reactor.
- Connecting most parts of the primary circuit pipelines to "hot" parts of the circuit with the location of nozzles on the reactor vessel above the core, that ensures a steam outflow to take place during transients and decreases requirements for the emergency core cooling system (ECCS) flow rate characteristics.
- The use of a reactor unit with short load-bearing nozzles between the main equipment units with no lengthy large-diameter pipelines in the primary circuit.
- The use of narrowing devices of small diameters in nozzles of the primary circuit auxiliary systems that, in combination with modular layout of the main equipment, excludes accidents with large and medium leaks from the primary circuit.
- Provision of such material properties and stress-strained state of the vessel structures that, in combination with the high requirements for production quality and diagnostic systems, practically exclude the loss of operability.
- The use of leak-tight main circulating pumps.
- The use of once through steam generators, limiting the increase of heat removal rate from the secondary circuit (the overcooling of the primary circuit coolant) at a steam line rupture.

Safety systems

The following main safety systems are incorporated in the VBER-300 design (Fig. VII-3):

- A reactor emergency shutdown system.
- An emergency heat removal system.
- An emergency core cooling system (ECCS).
- Emergency localization systems, including a double protective enclosure and locking valves on the primary circuit auxiliary systems and systems adjoining them.
- The system of reactor vessel cooling.



1-Reactor
2-Steam generator
3-Main circulating pump
4-Recuperator
5-Heat exchanger for cooling down the reactor
6-Filter trap
7-Filter based on a combined principle of operation
8-Anion exchange filter
9-Cation exchange filter
10-Purification and cooldown system pump
11-Pressurizer
12-Intermediate circuit filter
13-Intermediate circuit tank
14-Intermediate heat exchanger
15-Intermediate circuit pump
16-Unit of heat exchangers
17-Water storage tank
18-Containment
19-Hydraulic accumulator
20-Secondary stage ECCS tank
21-Reactor vessel cooling tank
22-Feedwater complex
23-Soluble poison tank
24-Water supply tank
25-Primary circuit make-up pump
26-Recircurlating pump
27-Reactor caisson filling tank
28-Heat exchanger of containment pressure suppression system
29-Condensate tank
30-Heat exchanger of recirculation system

FIG. VII-3. Safety systems of VBER-300.

The following design features and engineering solutions ensure high reliability of the safety systems:

- Passive functioning of the systems without exceeding the prescribed design limits over the entire range of design basis accidents, including losses of coolant (LOCAs) and loss of all alternate current (AC) sources, during not less than 72 hours.
- Redundancy and diversity of the reactor shutdown, core cooling and residual heat removal systems.
- Radioactive products release localization using a double protective enclosure and passive systems and redundant fast-acting valves.
- Separation of the safety systems channels to exclude common cause failures, the use of elements meeting the principle of a safe failure.
- Redundancy and diversity of control systems achieved through the use of self-actuated devices.
- The use of diagnostic means and periodic inspections to exclude failures in the safety system elements not revealed during operation.

Using a two-channel scheme of the safety systems, the regulatory requirements for safety are met by both deterministic and probabilistic characteristics due to redundancy of the elements inside channels and redundancy of the safety systems themselves.

The control safety systems provide an automated and remote control of the safety systems equipment from independent control panels (located in the main control room and in a standby control room).

Protective enclosure

To provide personnel and population protection against the consequences of the design basis and beyond design basis accidents, the following engineering solutions for the protective enclosure are used in the design of the land-based and floating VBER-300 units:

- The system of passive heat removal from the protective enclosure, designed to limit pressure in the protective enclosure in LOCAs.
- The system of fuel retention in the reactor vessel in accidents with severe core damage.
- Separation of the protection functions against external natural and human-induced impacts and internal emergency impacts.
- The system of iodine and aerosol purification of the air of the inter-enclosure space (the space between the protective enclosure and the containment) from radioactive leaks coming from the protective enclosure in accidents involving a pressure increase in it.

The protective enclosure of the reactor compartment of a land-based nuclear cogeneration plant is double, consisting of an internal steel protective enclosure and a containment made of reinforced concrete without pre-stressing (Fig. VII-4).

The steel protective enclosure is cylindrical with a 28.0 m diameter and a height of 34 m. The containment is monolithic reinforced concrete with a 34 m outer diameter and a height of 42.2 m, without pre-stressing.

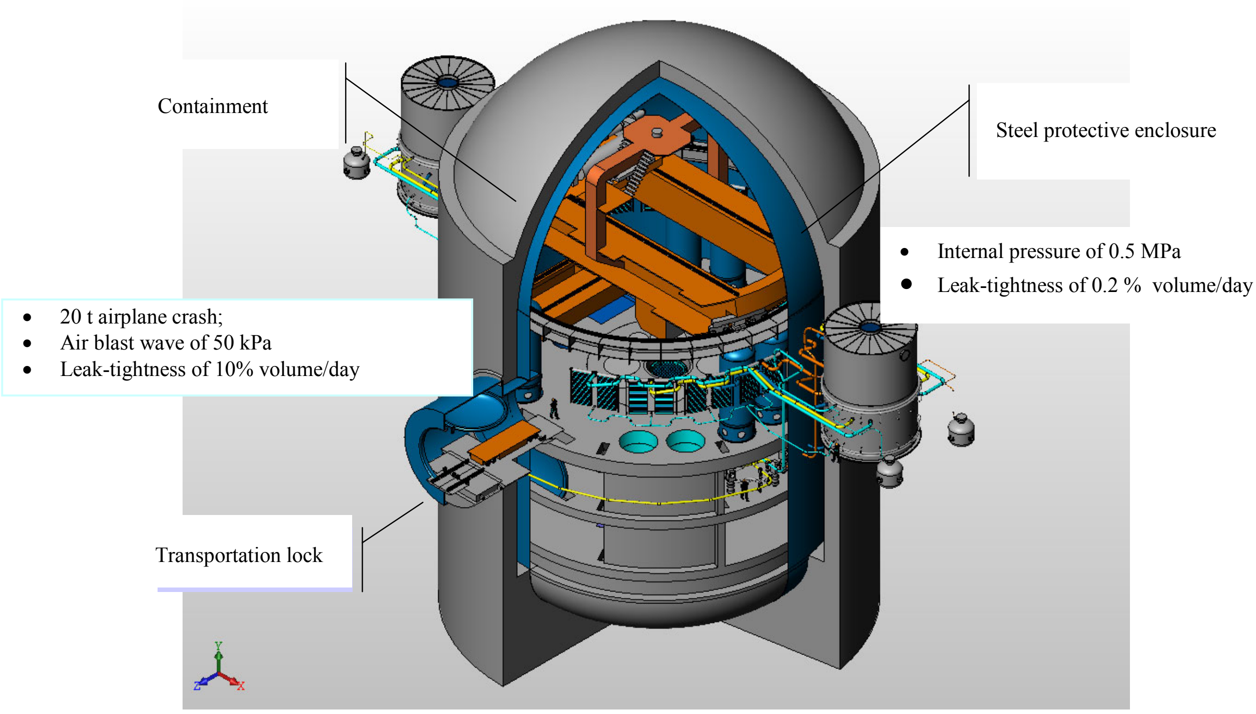


FIG.VII- 4. Containment of a nuclear cogeneration plant with VBER-300.

Design basis accidents and beyond design basis accidents

A systematic approach is realized in the design to analyze and validate safety based on both deterministic and probabilistic methods.

The deterministic safety analysis is performed using a set of calculation codes developed in OKBM and proven in calculations of stationary and non-stationary modes of NPP operation. The codes take into account specific features of the plant design, circulation circuits, steam generators, cooldown systems, control systems etc., and are based on the experimentally proven methods of calculation and correlations, with long-term experience in application.

The codes were verified on the results of studies of several separate effects and phenomena and on the results of the integral experiments performed at thermo-physical test facilities-mock-ups of modular reactors, as well as on the experimental data available from the validation and testing experiences of full-scale nuclear installations.

The codes used for safety analysis underwent the procedure of examination by the Council on Software Certification of the Gosatomnadzor of Russia (Russian regulatory authority) and have been certified.

Along with design basis events, a wide range of beyond design basis accidents, including failures of safety systems coupled with various initiating events and/or mistakes of personnel, were analyzed.

Specifically, the list of beyond design basis accidents includes:

- Overpower transients with control safety system failure (Fig. VII-5).

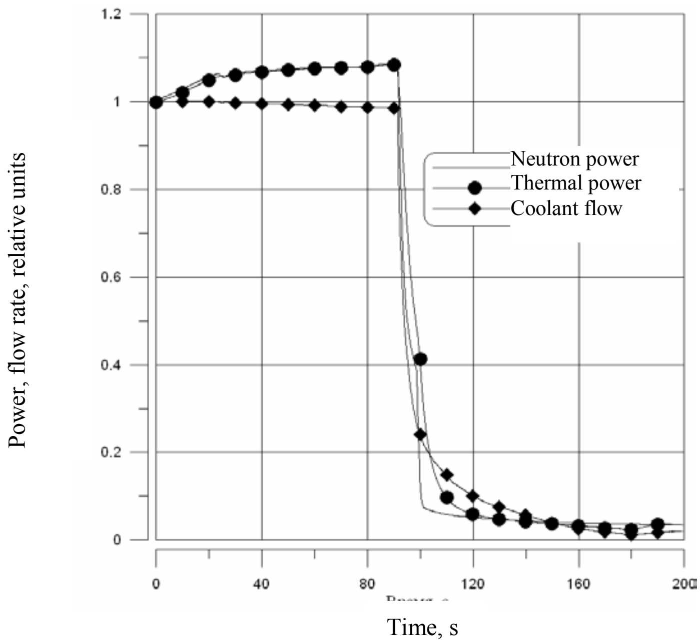


FIG. VII-5. Uncontrolled withdrawal of a control rod group with control safety system failure.

- Primary circuit pipeline rupture with a total NPP blackout (see Fig. VII-6) or a failure of core cooling systems. and

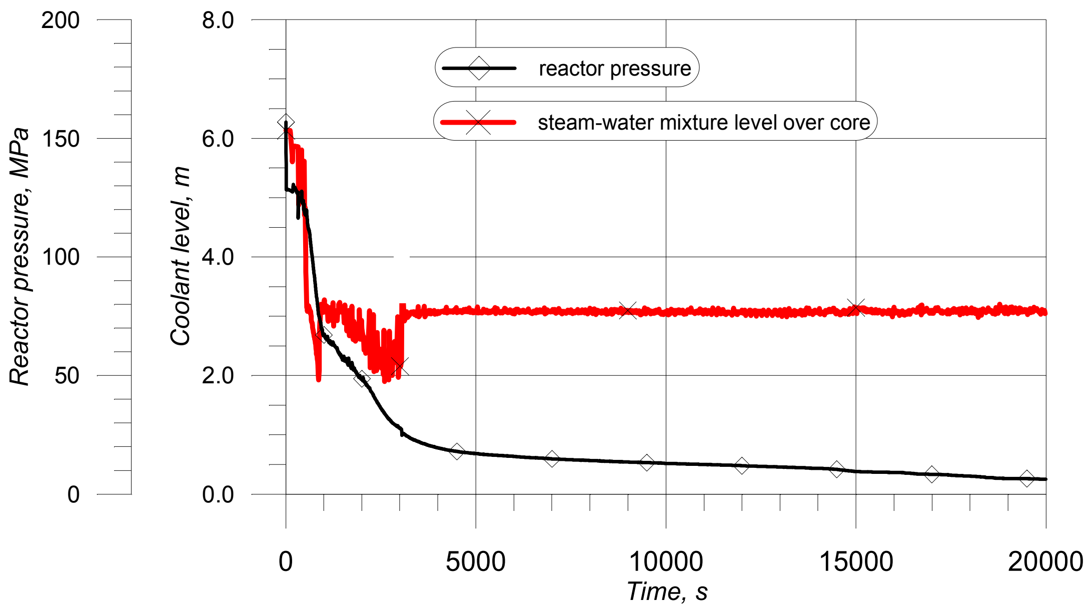


FIG.VII-6. Primary circuit pipeline rupture with total NPP blackout.

- Total NPP blackout with a failure of control safety system (see Fig. VII-7), or a failure of channels of the emergency heat removal system.

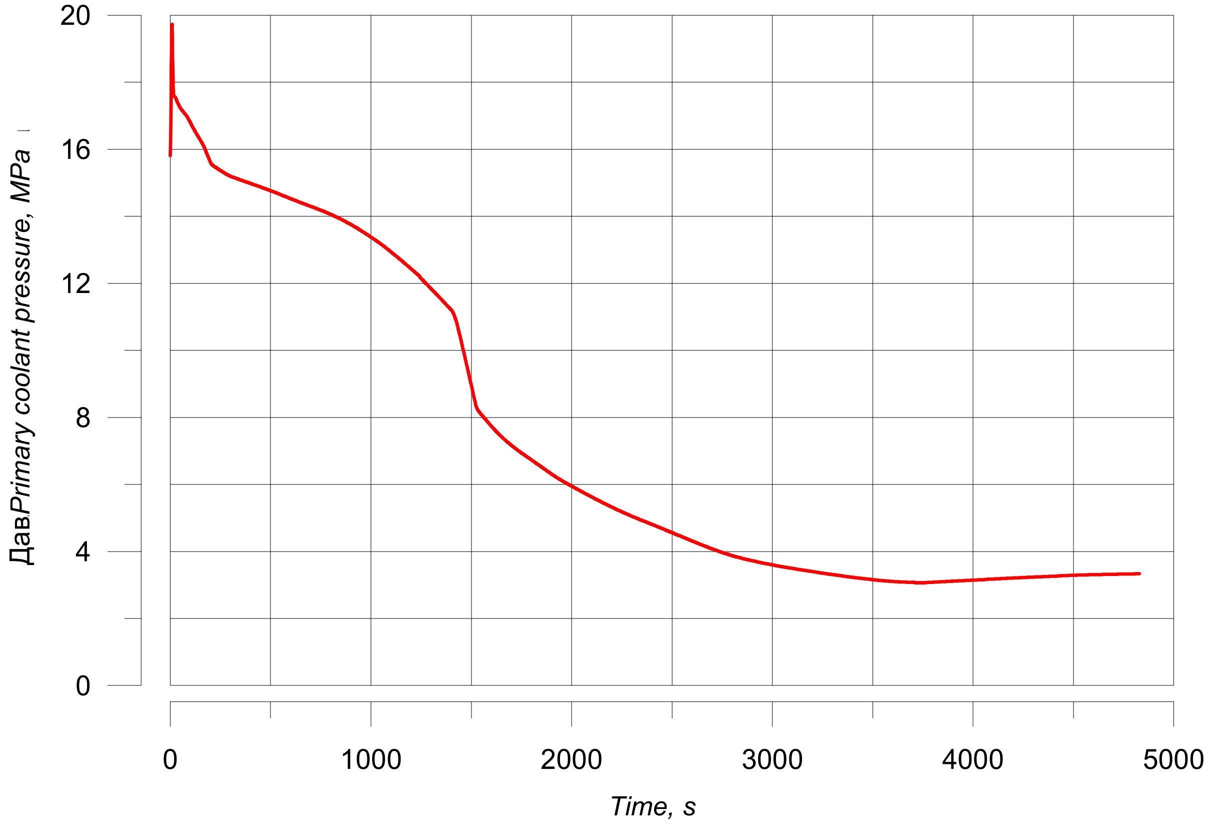


FIG. VII-7. Total NPP blackout with failure of control safety system.

Probabilistic safety analysis was used to support the deterministic analysis with regard to the elimination of weak points of the design and an effectiveness assessment of decisions on the perfection of safety features and measures, i.e., to ensure a more balanced defence-in-depth approach.

Mitigation of severe accident consequences

The evaluated core meltdown probability for the VBER-300 is low. Nevertheless, in accordance with the regulations and considering the design experience of similar domestic and foreign new-generation reactors, the problems of safety in postulated severe accidents were considered in the VBER-300 design.

In accordance with regulations, not exceeding the allowable emergency doses to population should be ensured in accidents with severe core damage [VII-7, VII-8], and the necessity of population evacuation should be excluded. These requirements conform to the current internationally established norms and the IAEA recommendations [VII-9, VII-10].

The standard approach to a severe accident is based on the combination of design solutions and accident management of the following two types:

- Directed at the prevention of core damages (decrease of core damage probability). and

- Directed at the limitation of severe accident consequences.

Retaining the melted core in the reactor vessel is considered a priority in limiting the consequences of severe accidents in the VBER-300, since the consequences are determined to

a great extent by the reactor vessel failure and relevant initiation of the additional loads on the protective enclosure when core melt exits the vessel.

The VBER-300 features that facilitate the retention of melted core in the reactor vessel are:

- A decreased core power density compared with large power reactors (VVER, PWR).
- A relatively low level of residual heat at the stage of core degradation and melt displacement to the bottom.
- No penetrations in the reactor vessel bottom, which are potential 'weak spots' under core melt impact on the bottom. and
- Smooth outer surfaces of the reactor vessel bottom creating more favourable conditions for steam evacuation under core cooling by boiling water.

The VBER-300 design provides a special system of emergency vessel cooling to solve the problem of retaining the melt inside the reactor vessel in severe accidents. This system functions in a passive mode: the reactor vessel is cooled by boiling water, the generated steam is condensed in the protective enclosure and the generated condensate is again involved in the reactor vessel cooling through the system of condensate gathering tanks.

The performed design evaluations show that the task of melt retention in the VBER-300 reactor vessel can be successfully solved.

Calculations also show that in a severe accident the limits of the allowable emergency doses to population are met and, therefore, measures on obligatory population evacuation are not needed. The boundary of the area of protection is not more than 1 km distant from the NPP. These results meet in full the safety requirements for new-generation reactors that are set by the NRC, the US industry, and the NPI consortium (EPR reactor) and also match the IAEA recommendations on safety of advanced reactors [VII-9].

Plant protection against the impacts of natural and human-induced external events

The structures, systems and equipment of a nuclear cogeneration plant with the VBER-300 are developed considering natural and human-induced external impacts and provide for a variety of siting options in line with the current regulations on NPP siting.

The external events considered in the VBER-300 design include: earthquakes, extreme wind loads, low and high temperatures, the fall of an aircraft or its parts, a shock wave, as well as other impacts.

For a floating NPP with the VBER-300, the following additional protection measures / features are provided:

- Water area protection against unauthorized access by floating objects.
- The design of vessel structures and seawater systems meets the floodability requirements of the sea shipping register of Russia.

VII-1.6.4. Proliferation resistance

The following technical features and conditions facilitate proliferation resistance of the VBER-300:

- The enrichment of uranium dioxide fuel by ^{235}U is less than 5%, which meets the IAEA recommendations.
- A standard once-through fuel cycle of the present-day VVER reactors is used, for which a well-established infrastructure and proliferation resistance measures are in place.

- Regarding a floating NPP, the infrastructure of nuclear ship service and maintenance is available in Russia. and
- The use of thorium fuel based on the RTR concept can ensure that neither fuel loaded in the reactor nor fuel discharged from the reactor could be used for nuclear weapons production; the plutonium recovery could be decreased by 4 to 5 times.

VII-1.6.5. Technical features and technological approaches used to facilitate physical protection of VBER-300

Technical features and technological approaches used to support physical protection of a NPP with the VBER-300 are similar to conventional approaches used for NPPs with the VVER- and PWR-type reactors.

The system of physical protection (SPP) of the plant includes a complex of the following engineering measures:

- Alarm systems, monitoring systems, and operative communication systems.
- The system of access control.
- The system of engineering safeguards.
- A subsystem of organizational measures.

The zoned principle of SPP arrangement is applied. For floating NPPs, a water area zone (limited by protective breakwaters and protective dams) and a coastal zone are provided along with a coastal technological site and a floating power unit (FPU) zone, which is a zone of increased control.

The following technical features contribute to protection of the VBER-300 against external impacts:

- A double protective enclosure providing separation of the functions of protection against external natural and human-induced impacts and internal emergency impacts. and
- Protection of the water area against unauthorized access by floating structures or floating objects (for a floating NPP).

VII-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of VBER-300

Non-technical factors and measures to facilitate effective development and deployment of the VBER-300 are better clarified for the floating NPP. They include:

- A decision on the construction of a pilot floating NPP with the reactor of the same type in Severodvinsk has been adopted, to demonstrate the advantages of this technology.
- Floating NPPs can be configured as power sources for desalination complexes, that considerably enlarges the number of potential customers-states as the shortage of potable water becomes more acute in many regions of the world. and
- Floating NPPs can be a powerful emergency cogeneration source in regions of natural disasters.
- Floating NPPs can be located in any coastal region of the world irrespective of its seismicity and proximity to cities; alternatively, they also can be located in remote areas.

- Floating NPPs can be leased under ‘build-operate-transfer’ conditions, considerably decreasing the present political and economical restrictions on the use of nuclear technologies in developing countries. and
- Russian plants-manufacturers of shipboard reactors have a proven capability of fabricating the main equipment of reactor units using the available production technology; these plants have appreciable equipment and highly skilled personnel specially trained to manufacture the equipment of such a type.

The possibility of involving a wide range of countries with shipbuilding facilities sufficient to build large-capacity barges and with industrial capacities capable of power machinery production is considered an important advantage of the proposed technology.

In this, namely the nuclear technology including the supply of NPPs and fuel, the operation and maintenance, and spent fuel management could be controlled by the Russian Federation.

The proposed NPPs could be of interest for foreign investors and customers, including a wide range of developing countries.

VII-1.8. List of enabling technologies relevant to VBER-300 and status of their development

The list of basic enabling technologies with status of their development for the VBER-300 is presented in Table VII-4.

TABLE VII-4. LIST OF BASIC ENABLING TECHNOLOGIES FOR VBER-300

ENABLING TECHNOLOGY	DEVELOPMENT STATUS
Technologies of pressurized water reactors with modular equipment layout used in shipboard reactors of the Russian nuclear fleet.	A well established reactor technology; the operation experience with shipboard reactors of different destination exceeds 6000 reactor-years.
Technologies of VVER-1000 power reactors (reactor core, control rod drives)	A well established reactor technology (20 reactors are under operation)
Technologies of reactors of the AST-500 nuclear cogeneration plant (safety design)	Under construction; safety review by the IAEA has been performed.
Technologies of the KLT-40S icebreaker-type reactor for a pilot floating NPP (location of a floating NPP)	The reactor and FPU designs have been developed, a regulatory body (Gosatomnadzor of Russia) license for the FPU construction in Severodvisnk (Russia) has been awarded

The specific enabling technologies behind the main engineering solutions of the VBER-300 are as follows [VII-1-VII-3]:

- A modular layout of the main equipment: reactor, steam generator, and main circulating pumps. Welding, using short nozzles without lengthy pipelines directly join the equipment vessels. The primary coolant circulation through connection nozzles of the main coolant path is performed according to a “co-axial” piping scheme.
- The use of a vessel-type pressurized water reactor, the one that is most proven through worldwide experience.

- The leak-tight design of the primary circuit using welded joints, leak-tight packless pumps, and leak-tight bellows valves.
- A four-loop path with forced and natural coolant circulation in the steam generating unit providing a high degree of redundancy and reliability of heat removal from the core in normal operation and emergency modes.
- The use of once-through coil type steam generators.
- A leak-tight MCP of the type used in shipboard reactors, with the required increase of its run-out.
- A cassette type core with VVER-type fuel with a decreased power density, meeting the requirements of the existing VVER nuclear fuel cycle.
- Electromechanical control rod drives are exactly the same as used in the VVER-1000 reactor.
- The use of passive safety systems for the emergency reactor shutdown, core cooling and reactor after-cooling.
- The vessel system service life is 60 years; proven technologies of the metallurgical, press-forging and machine-assembly production available at the shipboard reactor fabrication plants are used.
- Highly reliable systems of the nuclear propulsion reactors and of the state-of-the-art NPPs are used.
- The use of proven technologies for mounting, repair and replacement of the equipment; the use of diagnostics and monitoring systems and devices to control the equipment status.

The basic technologies for equipment fabrication, already mastered in the commercial production, are:

- Welding technologies of the unit of vessels.
- Production processes for the titanium alloy piping system of the steam generator.
- The technology of manufacturing and mounting of the ‘coaxial-type’ internals providing coolant circulation.
- The design and fabrication technology of canned MCPs.
- Production processes for the skeleton AFA-type fuel assemblies of the VVER-1000.
- The manufacturing technologies for elements of the systems of normal operation and of the safety systems (self-actuated devices, pressurizer, tanks, heat exchangers, pumps, filters).

VII-1.9. Status of R&D and planned schedule

The development of NPP designs with the VBER-300 reactor is performed at the initiative of several Nizhny Novgorod region companies having the unique experience of design, construction and operation of shipboard nuclear reactors. The list of stakeholders identifying their responsibilities within the project is given in Table VII-5.

TABLE VII-5. RUSSIAN ENTERPRISES INVOLVED IN VBER-300 PROJECT

ENTERPRISE	AREA OF RESPONSIBILITY
OKB Mechanical Engineering (OKBM), Nizhny Novgorod	Chief designer of VBER-300 reactor
RRC «Kurchatov Institute», Moscow	Scientific adviser of the reactor project
Scientific-Research and Design Institute «Atomenergoproekt» (NIAEP), Nizhny Novgorod	General designer of nuclear cogeneration plant
Public Company «Lazurit», Nizhny Novgorod	General designer of floating nuclear power plant

The design development of nuclear power sources with the VBER-300 is performed at the expense of the financial assets of Russian enterprises and is partially supported by the Federal Agency of Russia on Atomic Energy (the Rosatom of Russia) within the framework of the national programme “The use of nuclear power sources for district heating and cogeneration purposes”.

The VBER-300 design incorporates the results of R&D on reliability, safety and production processes previously achieved during the design development of similar reactors and power units, including the results of inter-industry programmes on the development of new-generation nuclear power plants.

Specifically, the VBER-300 design development:

- Makes a full use of the calculation methods, codes and databases validated and certified during the previous design development activities.
- Eliminates the necessity for development and certification of new materials.
- Incorporates proven designs of fuel assemblies and structures, as well as production processes already proven in serial production.

Based on the above and considering the factor of increased power compared with operating modular shipboard reactors, the research, design and demonstration (production and testing of prototypes) is required only for separate equipment items of the VBER-300.

The preliminary design of the VBER-300 reactor was completed in 2002. “The technical and commercial proposal” on the nuclear cogeneration plant and the floating NPP with the VBER-300 (a brief version of the technical and economic assessment) has been produced.

The VBER-300 preliminary design and the technical and commercial proposal on the NPPs were reviewed by the RRC “Kurchatov Institute”, which recommended the VBER-300 as a basic project to support export potential increase and conversion in Russia.

At present, the design is undergoing an expertise of the Rosatom of Russia to be submitted for consideration at its Scientific-Technical Council. The phase of optimization of individual design features and schemes is near completion, and development of the detailed design documentation for the VBER-300 reactor is underway.

The timetable for development and deployment of the NPPs with the VBER-300 is given in Table VII-6.

TABLE VII-6. TIMETABLE FOR DEVELOPMENT AND DEPLOYMENT OF NPPs WITH VBER-300

STAGE	DURATION
Detailed design development, including licensing	3 years
Plant construction (including licensing and major demonstrations)	
Nuclear cogeneration plant	5 years
Floating NPP	4 years

VII-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

The VBER-300 is a scaled analogue of the icebreaker-type KLT-40 reactor, providing for thermal power increase from 180 to 850 MW. The scaling factors are not expected to influence the practicality of the engineering solutions for the reactors of this type.

There is no necessity to perform wide-scale R&D; experimental validation of individual equipment is necessary, such as: aerodynamic testing of the reactor setting; validation of the production processes for main units of the steam generator; development, fabrication and testing of a pilot model of the MCP; design, fabrication and testing of a refuelling machine.

It is necessary to construct a floating NPP prototype to solve general problems associated with floating NPPs intended to supply power and energy products for coastal areas. The use of floating NPPs for seawater desalination and electricity cogeneration is an innovation. The use of a unified reactor unit design for both land-based and floating NPPs appears to be an innovation too.

VII-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

No information was provided.

VII-2. Design description and data for floating nuclear power plant with VBER-300

VII-2.1. Description of the nuclear systems

Reactor core and fuel design

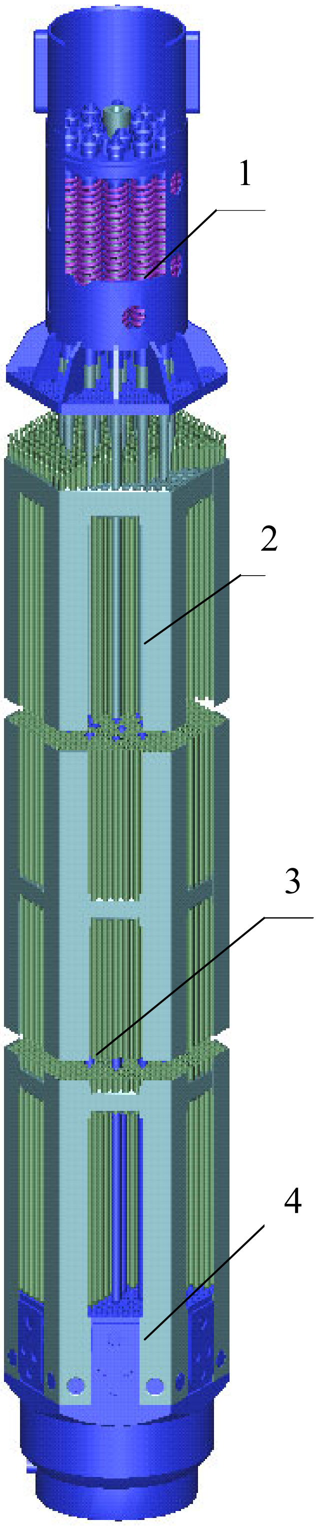
In the VBER-300 design, priorities were directed toward ensuring the reliability and safety of the core and NPP as a whole, while improving economic indices for the fuel cycle.

In line with this, the design incorporates the core of a cassette structure, for which the reliability is validated by the long-term successful operating experience of PWR-type reactor cores.

The fuel element design is similar to that used in the VVER reactors. The fuel element cladding has a 9.1 mm outer diameter and a 7.73 mm inner diameter. The fuel cladding material is E-110 or E-635 zirconium alloy.

The uranium dioxide pellets of 7.6 mm diameter are used as fuel; the uranium enrichment is up to 5% (maximum licensed enrichment).

The fuel assembly design is that based on casing-free skeleton-structure fuel assemblies of the AFA-type, originally developed by OKBM for the VVER-1000 reactor [VII-11], Fig. VII-8.



1: Head 2: Stiffening angle 3: Spacer grid 4: Stem

FIG. VII-8. VBER-300 fuel assembly.

The results of testing performed within the VVER-1000 reactor core of the Kalinin NPP Unit 1 have confirmed serviceability, high load-carrying capacity and high resistance to deformation of the AFA type fuel assemblies [VII-12].

The AFA fuel assembly design used in the VBER-300 provides the following:

- A flexible fuel cycle.
- The possibility of load-follow operating modes.
- Vibration strength and geometrical stability of the assembly ensured by a load-carrying skeleton with the enhanced stiffness.
- Core demountability and an option to repair fuel assemblies at the site.

Main characteristics of the AFA design fuel assemblies and the VBER-300 reactor core are given in Table VII-7.

Some fuel elements within the core contain gadolinium in the uranium dioxide fuel pellets and are used as burnable poisons. The geometric characteristics of such gadolinium fuel elements are similar to those of the core fuel elements. The gadolinium content in the fuel elements is identical with that in the VVER reactors.

TABLE VII-7. MAIN CHARACTERISTICS OF VBER-300 REACTOR CORE

PARAMETERS	VALUE
Rated thermal capacity, MW	850
Number of fuel assemblies	85
Circumscribed diameter, mm	2420
Equivalent diameter, mm	2285
Height, mm	2900
Volume, m^3	11.89
Power density, MW/m^3	71.5
Number of control rod drives	72
Pitch of fuel assemblies, mm	236
Fuel assembly dimensions on a turnkey basis, mm	234
Pitch of fuel elements in a fuel assembly, mm	12.75
Number of fuel elements and gadolinium fuel elements	26 520
Area of core heat transfer surface, m^2	2198.7
Average heat flow from the surface of fuel elements and gadolinium fuel elements, MW/m^2	0.375
Average linear heat rate of fuel elements and gadolinium fuel elements, W/cm	107.2

The core reactivity margin for fuel burn-up is compensated by a boric acid solution together with the gadolinium fuel elements. For that purpose, the boric acid is gradually discharged from the coolant by special filters during the reactor operation.

The cluster system of reactivity compensation is used to compensate for temperature and power reactivity effects, reactivity margins for core poisoning by xenon-135 and samarium-149, operating margins to change reactivity during reactor power changes, and to provide core sub criticality under reactor shutdown.

A diagram of the fuel assembly arrangement is shown in Fig. VII-9.

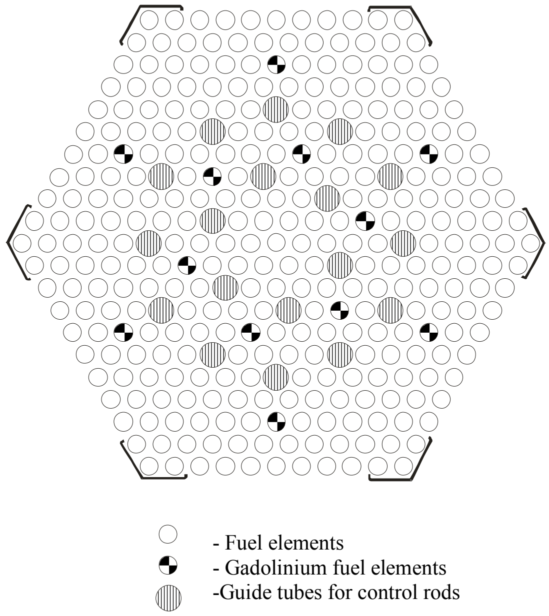


FIG. VII-9. Fuel assembly map.

Clusters are in the form of a bundle of 18 absorber rods joined by a common traverse, moving inside the E-635 zirconium-alloy guide tubes of a 12.6 mm outer diameter and a 0.6 mm wall thickness. Each cluster has its own drive. On the total, there are 72 control rod clusters of the reactor control and protection system (CPS). The core map is shown in Fig. VII-10.

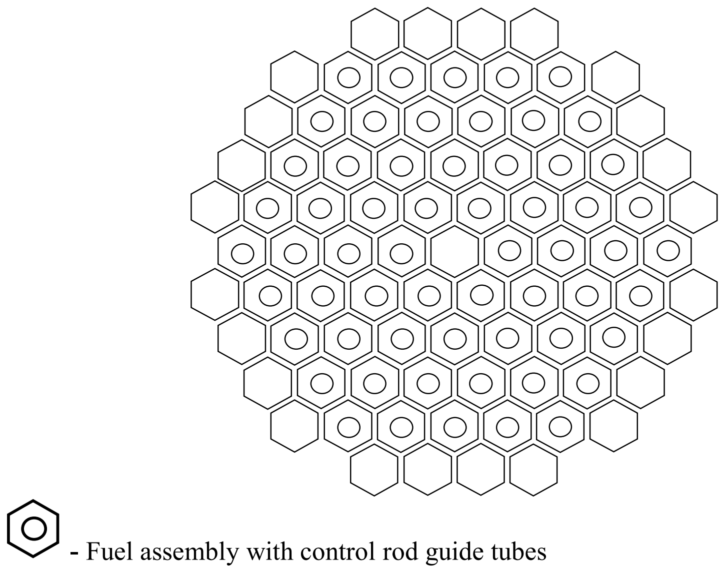


FIG. VII-10. Core map indicating the positions of control rods.

All CPS control rods simultaneously shoulder the functions of reactivity compensation and emergency protection. In this, the emergency protection function is executed by passive sinking of the CPS control rods through gravity, in case of a control rod drive de-energization according to a signal of the control system.

The adopted CPS scheme can provide core subcriticality under all operating conditions including reactor cooldown and de-poisoning, with account of a 'one-rod-stuck' event.

Primary circuit
The primary circuit is a leak-tight closed system intended for heat removal from the reactor core and heat transfer to the secondary water-steam circuit through the steam generator.

The primary circuit includes:

- A reactor unit.
- A pressurizer system.
- A purification and shutdown cooling system.

Water with a content of 6.0-7.0 g H_3BO_3/kg H_2O at the beginning of the cycle and not more than 0.1 H_3BO_3/kg H_2O at the end of the cycle is used as a coolant.

Reactor unit
The reactor unit is intended to generate steam of required parameters. The following are parts of the reactor (Fig. VII-11 and Fig. VII-12):

- A vessel system.
- A reactor core.
- Once-through steam generators.
- The main circulating pumps.
- CPS control rod drives.

The main technical characteristics of the reactor unit are given in Table VII-1 in the beginning of this Annex.

The vessel system consists of a reactor vessel and four "steam generator and main circulating pump" units connected to the reactor vessel by powered nozzles, designed according to a "co-axial" scheme.

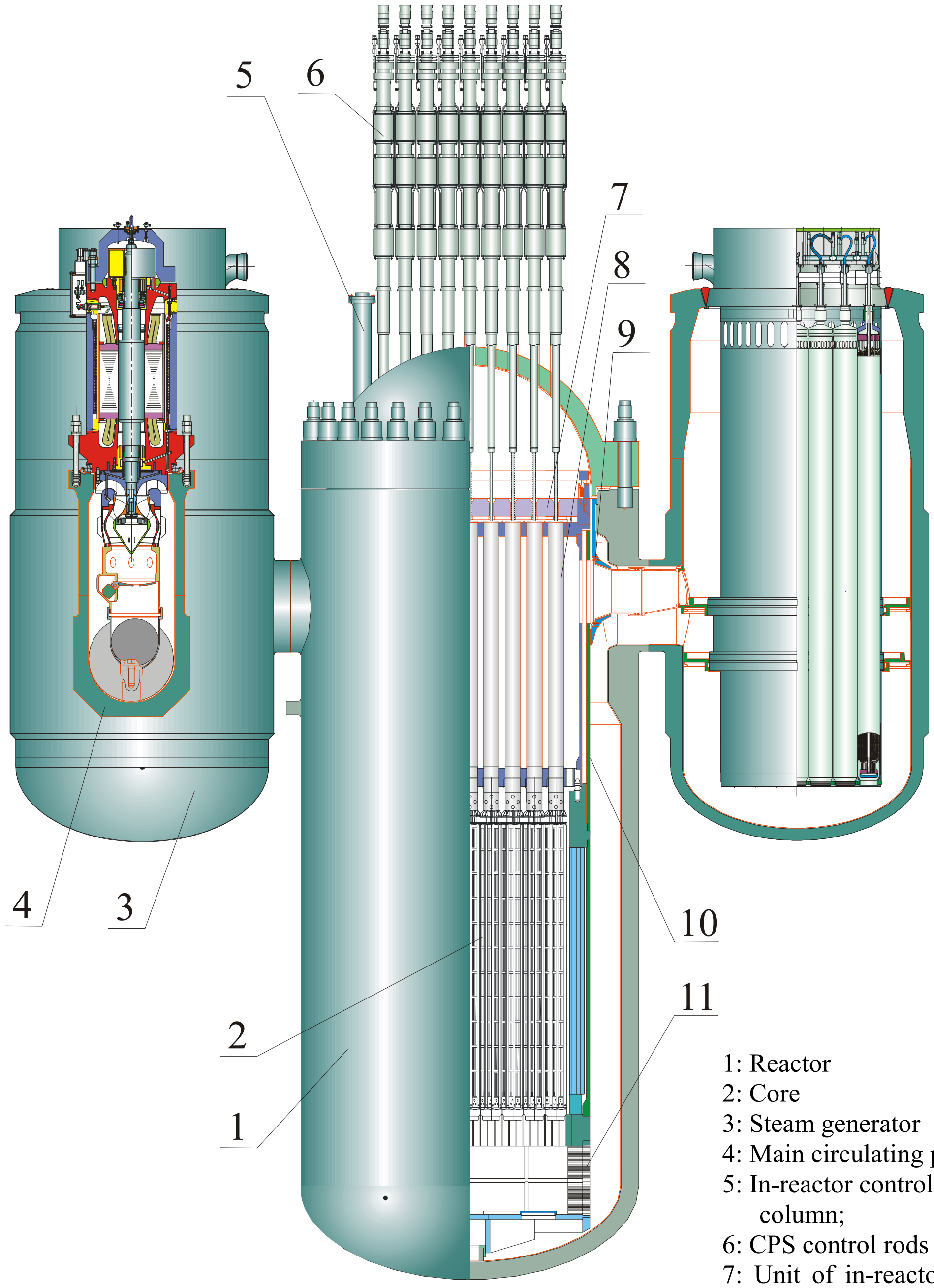
The reactor vessel is a welded cylindrical vessel with an elliptical bottom, four main nozzles and a flanged part. The reactor vessel accommodates an in-vessel unit.

The "steam generator and main circulating pump" unit consists of a steam generator vessel, connected with the hydraulic chamber of a main circulating pump by a powered nozzle.

A modular design minimizes mass and overall dimensions of the reactor unit and the scope of construction of the reactor, consequently decreasing capital outlay per unit. It also excludes main circulation pipelines and the associated large and medium break LOCAs. The maximum scope of the primary pipeline depressurization with account of narrowing devices does not exceed DN 42 mm.

The main mass and overall dimension characteristics of the reactor unit are as follows:

Overall height of the steam generator unit, mm	14 750
Total mass of the steam generator (without remote pressurizer)	988
Reactor core barrel diameter, inner/ outer, mm	3300/3700
Reactor core barrel diameter, inner/ outer, mm	988
Circumscribed diameter of the unit, mm	≈10 000

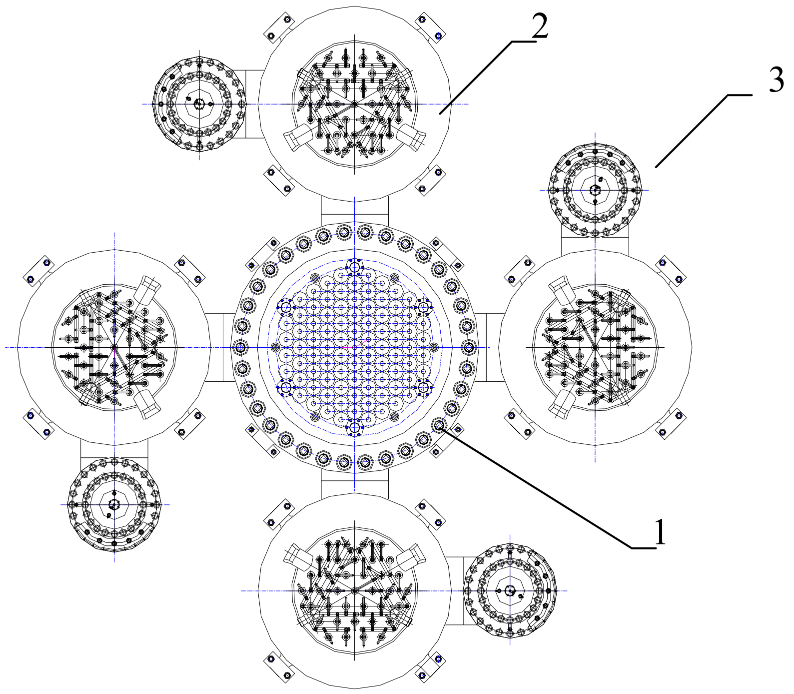


1: Reactor
2: Core
3: Steam generator
4: Main circulating pump
5: In-reactor control probes column;
6: CPS control rods
7: Unit of in-reactor control system probes
8: Unit of protective tubes
9: Shell
10: In-vessel cavity
11: Edge filter

FIG. VII-11. Reactor unit.

Steam generator

The pipe system of the steam generator together with the casing is a modular coil-type vertical-cylindrical surface-type heat exchanger in which the steam exchange between the primary coolant circulating in tube space and the secondary working medium circulating in the inter-tube space is achieved.



1: Reactor 2: Steam generator 3: Main circulating pump

FIG. VII-12. VBER-300 reactor unit (view from above).

The pipe system heat exchange surface consists of 37 unified coil-type steam generating modules (Fig. VII-13) covered by a box-like casing on their periphery. By feedwater and steam, the steam generating modules are consequently integrated into two independent sections of 18 and 19 modules.

The module heat exchange surface consists of seven rows of cylindrical multiple-thread coils wound on the central tube of the module. Coils are spaced in longitudinal and lateral directions using spacing combs and supporting strips.

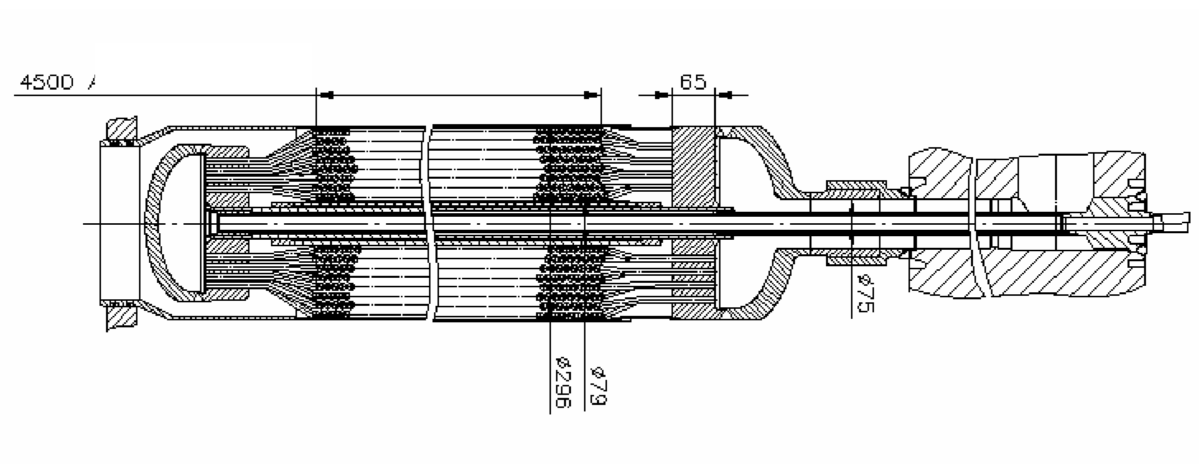


FIG. VII-13. Steam generator unit.

Main circulating pump

The main circulating pump is a unit consisting of a diagonal flow pump and a canned electric motor both in a singular module.

The pump setting consists of a guide flange, a diagonal-type console wheel and a guide device. The guide flange, along with the guide device, is intended to form the flow at the wheel inlet and to discharge a pumped coolant from the wheel to a pressure cavity.

The electric motor is asynchronous consisting of a stator, a rotor, bearings and a cover. A magnetic conductor and stator windings are separated from a rotor cavity by a thin-wall partition, welded to a stator casing.

The electric motor rotor is rotated in two slide bearings. A thrust bearing and a journal mounting at the rotor top take the thrust load forcing on the rotor. A flywheel is located on the top end of the rotor.

Pressurizer system

An external steam pressurizer system with a two-zone steam pressurizer is used in the VBER-300. Electric heaters are used as a steam source.

In the steam volume of the pressurizer, two zones are structurally organized: the steam zone 9.75 m^3 in volume and the steam-gas zone 0.25 m^3 in volume.

The pressurizer steam zone is intended to compensate pressure; the steam-gas zone is intended to keep a stable concentration of gases dissolved in the primary circuit coolant.

The design of the two-zone steam pressurizer provides a self-sustaining water-gas mode in the primary circuit coolant under operating regimes of a NPP with a leak-tight primary circuit, without relief of the steam-gas medium from the pressurizer.

Purification and shutdown cooling system

The purification and shutdown cooling system is intended to maintain the primary circuit coolant of a required quality, to decrease the boric acid concentration in the primary coolant, to provide normal and emergency reactor cooldown and to inject chemical reagents compensating for chemical conditions into the primary circuit.

In addition, primary circuit filling, make-up, drainage, and sampling, as well as the injection of chemical agents to correct the water chemistry are carried out through this system.

The system consists of the following components (see Fig. VII-3):

- A recuperative heat exchanger.
- Two cooling heat exchangers.
- Two circulating pumps (operational and standby).
- An ion-exchange filter with a combined principle of operation, three anion-exchange filters and cation-exchange filters.
- A filter-trap.
- Piping and valves.
- Transducers.

To limit coolant loss in case of a pipeline or system depressurization the nozzles connecting the system with the reactor unit are equipped with restriction inserts.

Main heat transport system

The scheme of the VBER-300 main heat transport system with specification of heat removal paths in normal operation and in accidents is shown in Fig. VII-14.

VII-2.2. Description of the turbine generator plant and systems

Secondary circuit

The secondary circuit is the steam and feedwater circuit, which is intended:

- To supply feedwater to the steam generators and to discharge steam.
- To generate steam with the required parameters in the steam generators.

The secondary circuit system consists of steam generators, feedwater and steam piping with valves.

Feedwater is supplied from a condensate-feed system of the steam turbine plant to the steam generators through feed piping. Each pipeline has double pneumatically actuated valves. The first valves by medium flow are the stop valves.

Steam is removed from the steam generators by four pipelines equipped by double pneumatically actuated stop valves and then enters the steam generator unit.

The emergency heat removal system is connected to steam and feedwater pipelines.

Turbine-generator unit

The floating nuclear power plant design provides for a turbine-generator unit based on the "LMZ" T-275/200-60/50 design modified in accordance with barge-mounting requirements.

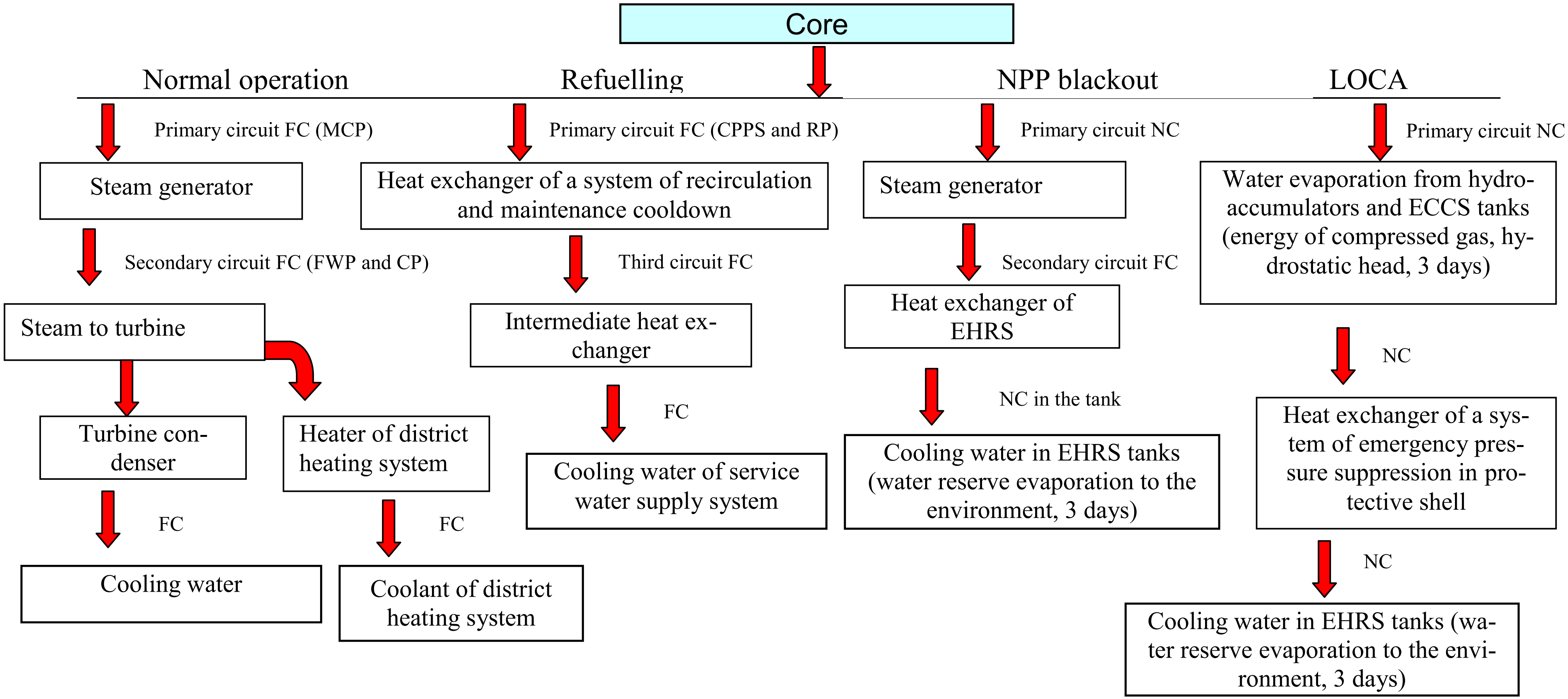
The turbine will have the following approximate characteristics:

- The turbine-generator unit is structurally arranged as a double cylinder (high pressure and low pressure cylinder [double flow cylinder]).
- The turbine length without the generator is about 20 m.
- The maximum width is about 11 m.
- The height measured from turbine island floor is 6 m.
- The condenser is cellar-cross type; an overall height of the condenser group is about 15 m.
- The total area occupied by the turbine plant equipment is about 1450 m^2.

Main characteristics of the turbine-generator unit are given in Table VII-8.

TABLE VII-8. MAIN CHARACTERISTICS OF THE TURBINE-GENERATOR PLANT

CHARACTERISTIC	VALUE
Live steam pressure before high-pressure cylinder valves, MPa	6.03
Live steam temperature before high-pressure cylinder valves, ºC	300
Feedwater temperature, ºC	185
Rated electric power, MW	295
Speed of rotation, revolutions per minute	3000
Installed capacity utilization per year, not less than, hr.	8000
Lifetime, yr.	60



EHRS	Emergency heat removal system
ECCS	Emergency core cooling system
FC	Forced circulation
NC	Natural circulation
FWP	Feedwater pump
CP	Condensate pump
CPPS	Circulation pump of purification system
RP	Recirculation pump

FIG. VII-14. Main heat transport system of VBER-300.

The condensation unit consists of a surface condenser, condensate pumps of the modular desalinating plant, air-ejectors, special valves, fittings and piping.

The feedwater system includes three electric feedwater pumps (one operative and two standby pumps). Each pump has a recirculation line to the deaerator providing pump testing and operability in transients.

Power output system

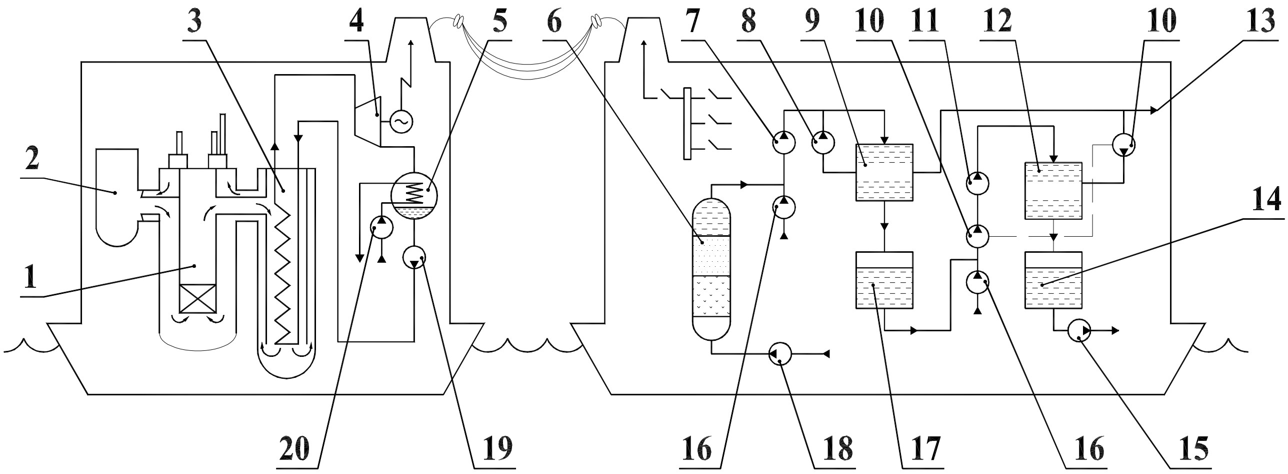
Each power unit includes a reactor unit, a steam turbine, a generator and a modular step-up transformer. It is proposed to use a 300 MW power generator of the TZV-320-2UZ type with complete water cooling developed by the "Elektrosila" (Sankt-Peterburg).

A power output scheme is used in which electric power of the floating power unit is supplied to the distribution station bus lines through the high voltage power lines; the voltage is 220 kV. The units are connected to existing supply lines by the "generator-transformer unit-line" scheme. Besides two power units, standby transformers are connected to the existing supply lines.

To supply power to consumers, while securing safety of expensive major equipment under the conditions of a NPP blackout, there are two redundant diesel generators (one operative and one standby). These generators are designed to compensate operating loads of consumers under a failure of one diesel generator.

DC loads are distributed through the boards such that, the power unit will still be operable in case of a failure of one accumulator or a DC board.

VII-2.3. Systems for non-electric applications



1 – Reactor
2 – Main circulating pump
3 – Steam generator
4 – Turbine generator
5 – Condenser
6 – Preliminary filter
7 – Medium-pressure pump
8 – Recirculation pump
9 – Ultra-filtration membranes
10 – Energy recovery system
11 – High pressure pump
12 – Reverse osmosis membranes
13 – Brine discharge
14 – Potable water storage tank
15 – Potable water discharge pump
16 – Chemical additions injection
17 – Ultra-filtration tank
18 – Seawater supply pump
19 – Condensate pump
20 – Circulating pump

FIG. VII-15. Principal flow diagram of the power-desalination complex with a condensing turbine and a reverse osmosis desalination plant.

It is possible to use floating power units with the VBER-300 as a power source for seawater desalination coupled with electricity cogeneration.

A power-desalination complex includes the reactor, the condensing turbine, the condenser and a reverse osmosis plant (Fig. VII-15). Electric coupling is used between the reactor and the reverse osmosis plant. Electric power generated at the complex is partially supplied to the reverse osmosis plant and the excess power is supplied to coastal consumers.

VII-2.4. Plant layout

General philosophy governing plant layout

A two-reactor floating power unit (FPU; Fig. VII-16) is a non-self-propelled autonomous floating structure classified as a harbour ship, per classification by the Sea Shipping Register of Russia. The FPU is on a platform supported by three pontoons (one central and two sides).

The two-reactor FPU has been functionally divided into three parts (central part, bow and stern). Reactor units are mounted in the central part and the turbine generators (TGs) on the bow and stern.

The main power equipment (reactor unit and TG) is installed in the longitudinal center plane of the central vessel (Fig. VII-16). Compartments for the auxiliary systems and mechanisms, power distribution systems, ventilation plant, ship systems, control systems, etc. are located in the hull compartments and in the superstructure.

The central pontoon carries two independent reactor units. Each consists of a reactor compartment, power unit control panel compartments and an electric equipment compartment. There is also a refuelling and repair compartment.

The main reactor unit equipment with auxiliary systems is located inside the steel containment. Fuel assembly storage is located on the central pontoon between the reactor plants.

The turbine generators together with the turbine generator auxiliary equipment and systems are mounted on the bow and in the stern of the central pontoon.

The port side pontoon carries the equipment intended for conversion, distribution and supply of electricity with a voltage of up to 220 kV to coastal objects and for the own needs of a floating NPP.

The starboard side pontoon carries the auxiliary equipment such as standby and emergency electric power sources and pumps.

The philosophy governing the FPU layout is as follows. The starboard side (“clean side”) facing the coast has been equipped with facilities for coupling to a floating dock, providing continuous plant communication and high-voltage terminals for electric power transmission to the coast.

The port side (“dirty side”) is intended for the logistics of ship mooring and providing transfer facilities for fresh and spent fuel casks and radioactive waste casks.

Each pontoon has the following dimensions:

Length, m	170
Width, m	19
Board depth, m	12

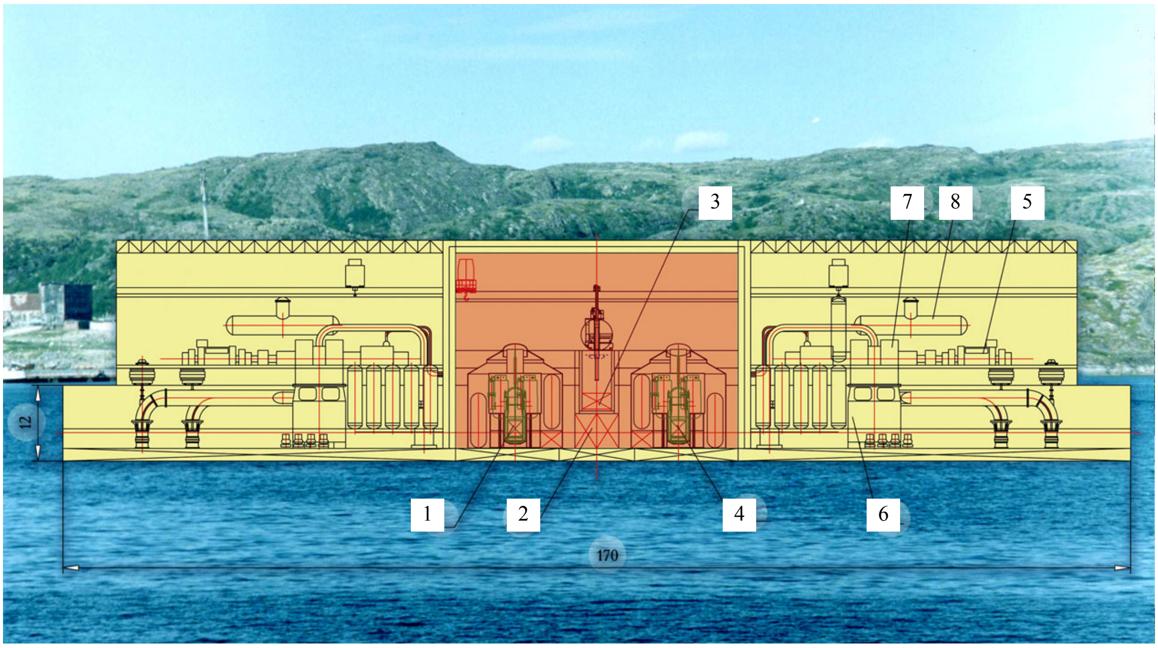
The total displacement of a two-reactor FPU is 49 000 tons. Due to the power unit composite vessel using both steel and reinforced concrete, there is no need for scheduled maintenance docking during the entire floating NPP lifetime. The service life of the floating NPP is 60 years.
The possibility of FPU construction at Russian shipbuilding facilities, first of all at Far East region facilities which are located not far from probable plant operating regions and have sufficient experience and infrastructure to create nuclear power vessels, is a foreground task of FPU designing. Based on the above information it was determined to construct the FPU vessel in the form of a trimaran, consisting of three right-angled steel vessels joined at an outfitting yard, using special coupling facilities afloat.

A priority task in the design of the floating NPP was to ensure the possibility of its construction at Russian shipbuilding enterprises with the sufficient experience and infrastructure, first of all those located in the Far East region of Russia, not far from the targeted regions of plant operation. A trimaran layout of the ship has been adopted, consisting of three rectangular steel vessels joined at an outfitting yard, using special coupling facilities afloat.

The dimensions of a floating NPP are given in Table VII-9.

TABLE VII-9. BASIC DIMENSIONS OF A FLOATING NPP WITH TWO VBER-300 REACTORS

Length, m	170
Width, m	62
Board depth, m	10
Draught, m	5.5
Overall height, m	35
Displacement, t	49 000



1: Reactor unit 1 2: Radioactive waste storage 3: Storage of fresh and spent fuel assemblies
4: Reactor unit 2 5: Electric generator 6: Condenser 7: Steam turbine 8: Deaerator

FIG. VII-16. Floating power unit with two VBER-300 reactors (PAES-600).

Reactor building and containment layout

Reactor compartment:
In the central part of the floating NPP there is a reactor compartment, in which two stand-alone VBER-300 reactors are placed.

Each reactor has its own steel, leak-tight containment. The reactor compartment is closed by a protective enclosure consisting of the multi-layered ceilings of a superstructure roof, walls of the stern and bow machine rooms and the superstructure boardrooms.

This construction constitutes the external protection of a reactor compartment and is capable of withstanding the extreme external impacts including aircraft falling on the floating NPP.

Turbine Island:
The floating NPP has two autonomous machine rooms intended for mounting turbine-generator units and auxiliary systems. The machine rooms are located to the bow and stern of the reactor compartment and separated by the cross walls of the reactor compartment protective enclosure.

The machine room of each power unit has dimensions of 54×36 m and a height (from the double bottom) of 33 m. This height is due to the installation of a bridge crane for turbine-generator unit maintenance.

One turbine-generator unit is mounted in each machine room in a longitudinal arrangement.

To provide the floating NPP construction according to an accepted construction layout, the main equipment of the turbine-generator unit is mounted in the central vessel of a floating NPP.

Plant plot

A specific arrangement of water space and creation of coastal infrastructure are needed for normal operation of the floating NPP.

The coastal infrastructure includes the following objects (Fig. VII-17):

- Protective waterworks (jetties, beacons, boom barriers).
- A waterfront structure (sea-walls, piers, etc.).
- Anchor links.
- Power line supports intended for the transmission of generated electricity to consumers.
- Facilities to provide security of the external plant perimeter from the land and from the sea (fence, watch-houses, supervision and control system, etc.). and
- Communications and connections (roads, telephone and secure communication lines).

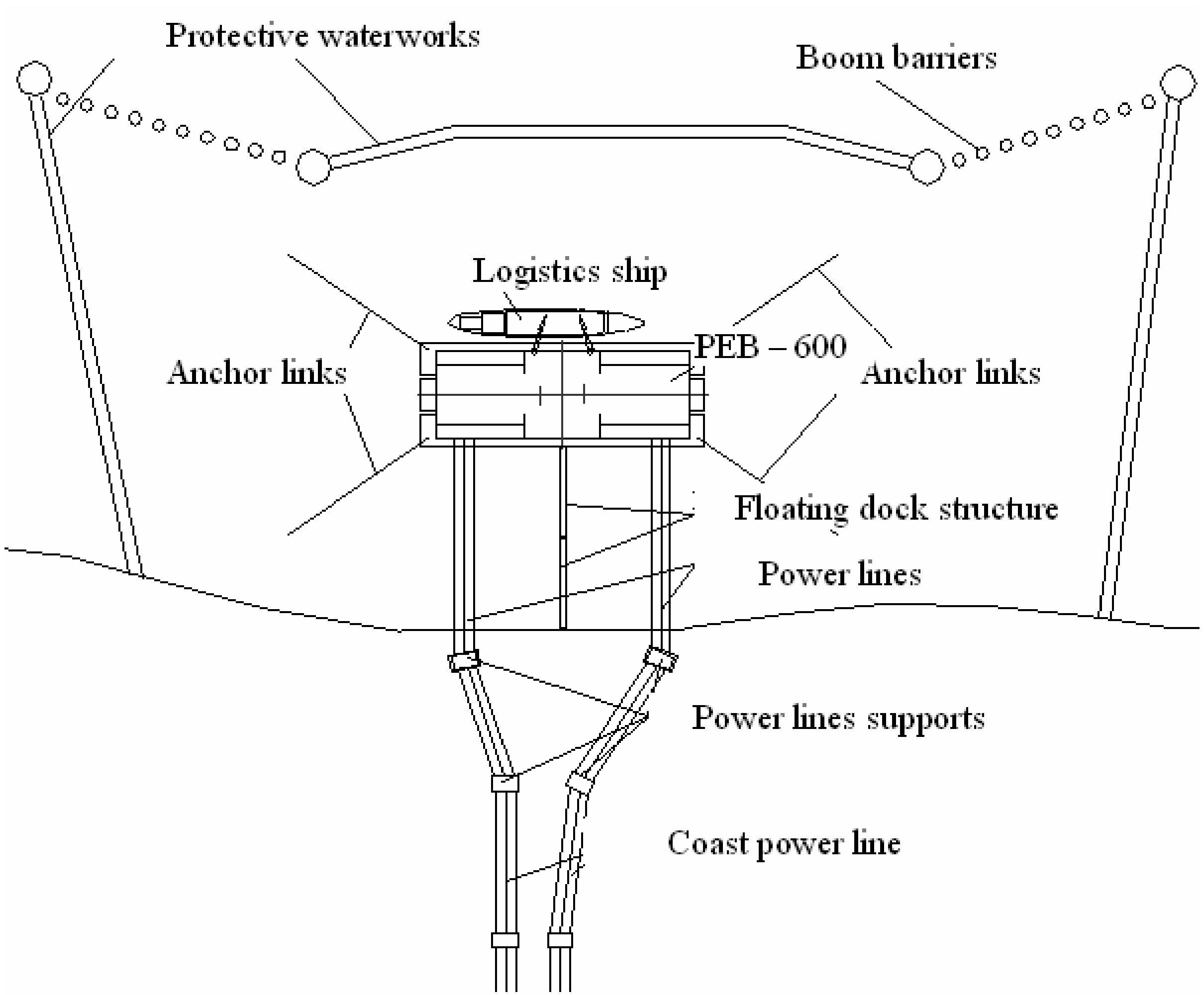


FIG. VII-17. PAES-600 plant plot.

VII-3. Design description and data for nuclear cogeneration plant with VBER-300

The design of a nuclear cogeneration plant is developed based on the unified VBER-300 reactor design. Design unification (ensuring that the VBER-300 can be used both in land-based and floating NPPs) reduces costs and increases quality of the design development.

In this section, descriptions of the turbine-generator unit and systems as well as design and architectural solutions for a land-based nuclear cogeneration plant with the VBER-300 reactors are presented, with a focus on elements that are different from those used in a floating NPP design.

VII-3.1. Description of the nuclear systems

Description of the VBER-300 nuclear systems is given in the previous section.

VII-3.2. Description of the turbine generator plant and systems

The design of a nuclear cogeneration power plant with the VBER-300 includes a turbine-generator unit of the "LMZ" T-275/200-60/50 type. The turbine-generator unit includes a heating unit of 420 GCal/h output (two units generate 920 GCal/h). The structural arrangement of the turbine-generator unit is one high-pressure and one low-pressure cylinder. Such turbine design solutions have been proven in present-day NPPs with the VVER reactors.

Main characteristics of the turbine-generator unit are given in Table VII-10.

TABLE VII-10. MAIN CHARACTERISTICS OF TURBINE-GENERATOR UNIT

PARAMETER	VALUE
Live steam pressure before high-pressure cylinder valves, MPa	6.2
Live steam temperature before high-pressure cylinder valves, ºC	300
Feedwater temperature, ºC	185
Feedwater pressure, MPa	9.5
Condensation mode: - Rated electric power, MW	295
Heat-extraction mode: - Electric power, MW, not less than - Heat output, GCal/h	200 460
Speed of rotation, revolutions per minute	3000
Installed capacity utilization per year, not less than, h	8000
Lifetime, yr.	60

VII-3.3. Systems for non-electrical applications

A nuclear cogeneration plant with the VBER-300 reactor(s) is considered to generate electric and heat power with maximum heat output.

The heat supply system from a nuclear cogeneration plant includes:

- Delivery water heaters.
- Supply-line pumps.
- Transit pipelines of delivery water at the nuclear cogeneration plant site.
- Stop valves.

The heat supply is realized through a double-circuit, double-stage scheme using a temperature chart of 150/70°C.

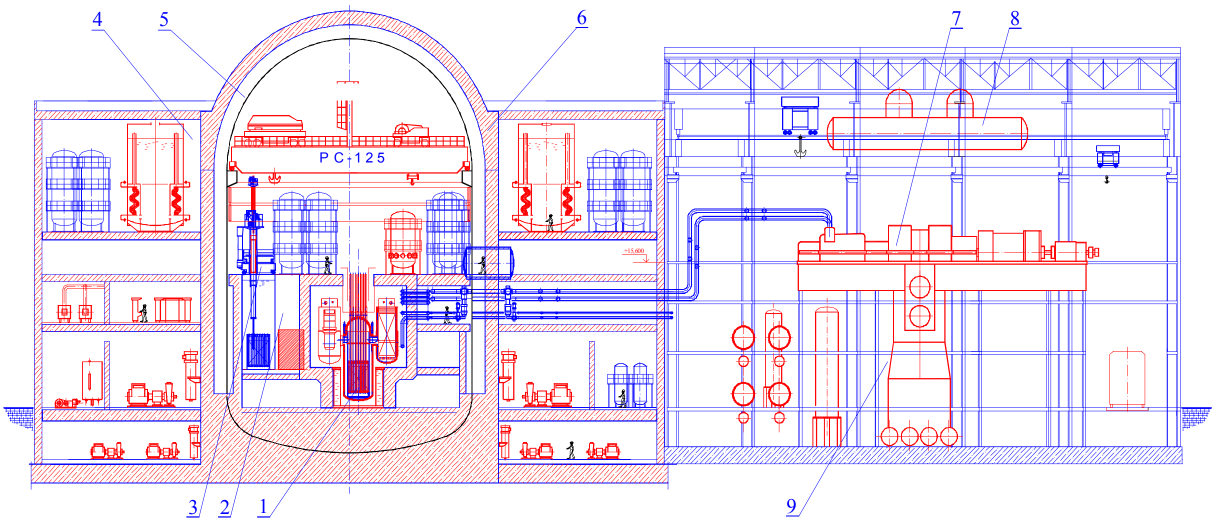
Two supply-line pumps of 50% capacity each supply the delivery water, separately mounted on each unit.

Hot delivery water is supplied from the locking gate valves to the transit heat network from the nuclear cogeneration plant to the city. The delivery water is returned through a return pipeline.

VII-3.4. Plant Layout

General philosophy governing plant layout

The basic engineering solution for nuclear cogeneration plants is to provide independent main buildings for the two power units, each of which consists of a reactor compartment, main control room building and a turbine island, which includes a turbine compartment, auxiliary systems compartment, deaerator compartment, and electric equipment compartment, Fig. VII-18.



1: Reactor	4: Safety systems rooms	7: Turbine generator
2: Storage pool	5: Leak-tight enclosure	8: Deaerator
3: Refuelling machine	6: Containment	9: Condenser

FIG. 18. Main building of VBER-300 power unit.

Reactor building and containment layout

From the side of the turbine islands, buildings of the “free access” zone are arranged as follows: turbine islands of units 1 and 2 with compartments for electrical equipment; deaerator compartments; buildings and structures for outdoor switchgear; the integrated auxiliary building with a makeup water demineralizer unit and sewage disposal structures; and a heating system makeup unit.

From the side of the reactor compartments, “strict control” zones are arranged as follows: the reactor compartments (RC 1 and RC 2); a special building for two units, which is placed symmetrically between RC 1 and RC 2; the integrated auxiliary building, which includes fresh fuel storage and a unit of reprocessing and storage of solid radioactive wastes.

Reactor compartment

The reactor compartment consists of a cylindrical protective shell with a spherical dome and two independent blocks attached to the protective shell from opposite sides, see Fig. VII-19.

A protective shell houses the reactor unit with auxiliary systems, spent fuel storage pool, and transport and process equipment.

The attached blocks of the reactor compartment contain the equipment of the normal operation and safety systems, electrical and control systems, and the equipment of ventilation systems of the protective shell and the attached blocks.

The attached blocks of the reactor compartment are box-type structures of reinforced monolithic concrete 0.9 m in thickness in the outer walls and a cover inflexibly joined to the reinforced concrete containment from the two opposing sides, containing channels of the process safety systems, control safety systems, and normal operation systems.

A separation of the safety system channels from opposite sides of the containment is provided to avoid simultaneous damage of safety systems under an airplane crash.

Turbine island

The turbine island adjoins by the face to the reactor compartment.

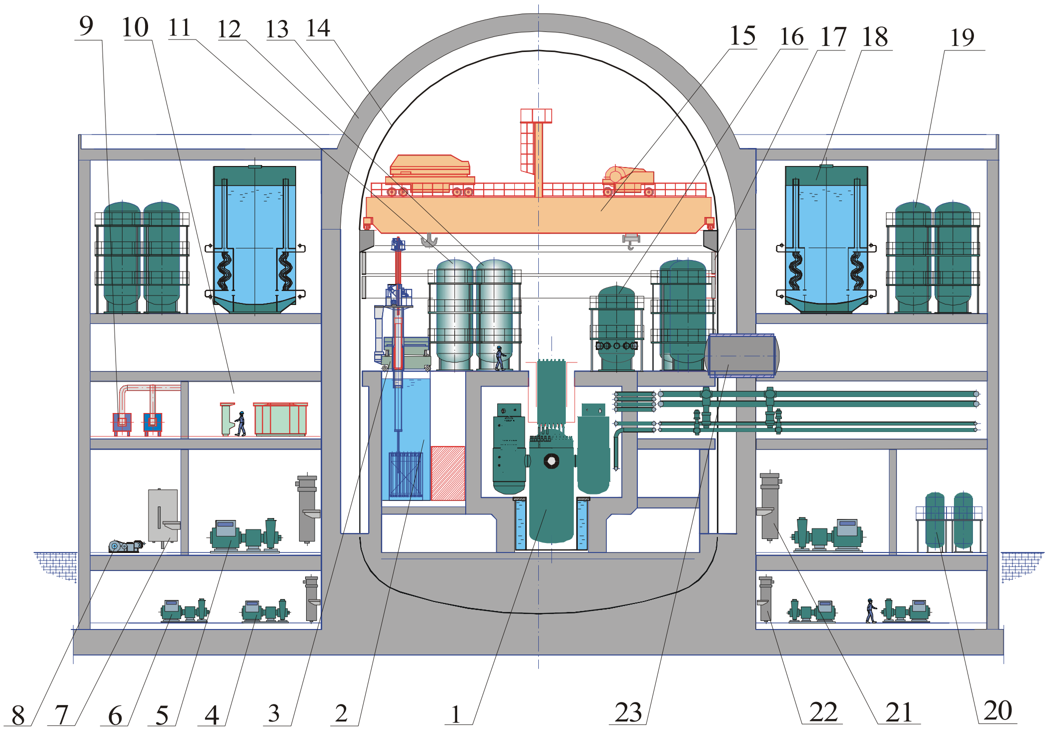
The skeleton the turbine island with a deaerator compartment is metal in the form of a two-span framing of 36 m and 12 m spans. The spacing of the cross framings is 12 m. Building stability in the transverse direction is formed by rigid framing joints of the skeleton of the deaerator compartment and rigid coupling of the columns with basements.

The standard-type turbine island is provided. The diameter of the turbine island is 36.0 m; the height to the bottom part of the framework is 33.6 m.

The turbine island is a two-span construction with an annex from the “A” line side:

- The main span – turbine compartment - with dimensions of 36×54 m.
- The auxiliary span – deaerator compartment - with dimensions of 12×54 m.
- The annex from the “A” line side, auxiliary systems compartment with dimensions of 12×24 m.

The turbine is installed in the turbine compartment longitudinally.



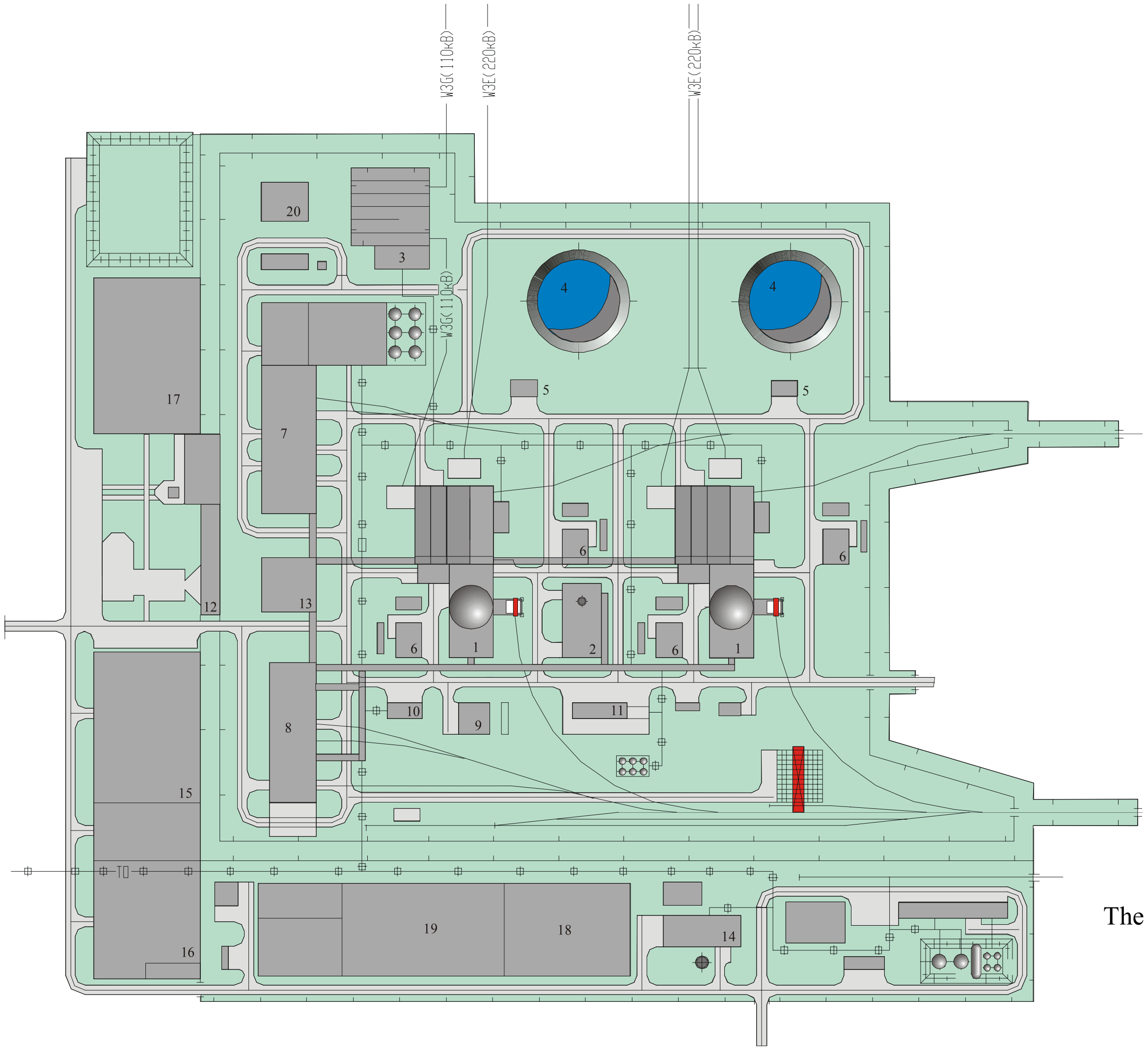
1: Reactor unit
2: Storage pool
3: Refuelling machine
4: Recirculation pump
5: Intermediate circuit pump
6: Storage pool cooling pump
7: Water supply tank
8: Make up pump
9: Ventilation installation room
10: control safety system room
11: Hydraulic accumulator of core emergency cooling system
12: Core emergency cooling system tank
13: Containment
14: leak-tight protective enclosure
15: Polar crane
16: Pressurizer
17: Heat exchanger of pressure drop system in leak-tight enclosure
18: Emergency heat removal system tank
19: Emergency core cooling system tank
20: Boron solution tank
21: Intermediate circuit heat exchanger
22: Recirculation system heat exchanger
23: Lock

FIG. VII-19. Reactor compartment of VBER-600 nuclear cogeneration plant.

Plant plot

The scheme of the general layout (Fig. VII-20) provides for the placement of buildings and structures in accordance with process requirements and maintains sanitary and fire-prevention guidelines.

In the center of the site, there are main buildings of the first and the second power units. From the side of the turbine compartments there are pumping plants with cooling towers of the turbine compartment cooling system. Between power units there is a special building for both units with a vent stack. This special building is connected with the main buildings by a pedestrian and transport overpass. On different sides of the reactor compartments, there are cooling water buildings with a standby diesel power station. For each unit there are two buildings.



1 Main building;
2 Special building;
3 Building with main control panel;
4 Cooling tower;
5 Pump plant of the turbine compartment;
6 Cooling water building with standby diesel power station;
7 Auxiliary building;
8 Special auxiliary building;
9 Modular standby diesel power station;
10 Gas building;
11 Nitrogen-oxygen station;
12 Office building;
13 Sanitary-social building;
14 Start-up and backup boiler house;
15 Fire-protection building;
16 Parking;
17 Training center;
18 Building and structures of water-supply;
19 Building and structures of drainage;
20 Emergency response control room

The nuclear cogeneration plant area including fence is 30 ha

FIG. VII-20. General layout of nuclear cogeneration plant with twoVBER-300 reactors.

The power line output from the power unit is made toward the cooling towers. For the electrical supply of the standby transformer of unit 1 and the construction substation, provision is made to construct a 110 kV outdoor switchgear and a main control panel with the unit of auxiliary structures, which is also placed from the side of the cooling towers.

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ANNEX VIII

WATER COOLED AND MODERATED NATURAL CIRCULATION BOILING WATER REACTOR (VK-300)

RDIPE, Russian Federation

VIII-1. General information, technical features and operating characteristics

VIII-1.1. Introduction

The VK-300 is a 250 MW(e) simplified water cooled and water moderated boiling water reactor with natural circulation of coolant and passive systems.

The major technological principles of the concept are: the simplicity in design resulting from a single circuit scheme and natural circulation of coolant in the core; the use of water as a coolant and moderator the production of steam of required parameters directly in the reactor due to an integral arrangement; and strong reliance on passive features and systems to achieve a high level of safety.

The VK-300 is a direct successor of the VK-50, — one of the simplified boiling water reactors developed in Russia The VK-50 has been successfully operated for decades at the Research Institute for Nuclear Reactors (RIAR) in Dimitrovgrad, the Russian Federation.

In 2001, the design of the VK-300 of 750 MW(th), 250 MW(e) was developed to supply electricity and heat of up to 400 GCal/h within a nuclear co-generation plant to be built at the Krasnoyarsk Mining and Chemical Combine. This design was developed by Russian research and design organizations: the Research and Development Institute of Power Engineering (RDIPE) named after N.A. Dollezhal and also known as NIKIET, the Russian Research Centre “Kurchatov Institute”, and the Institute of Physics and Power Engineering (Obninsk), also involving the RIAR, VNIINM, VNIIPIET, VNIIPIPT and others.

During project development, maximum use was made of the experience in design, production and operation of the following equipment used in the following operating reactors:

- VVER-1000 — reactor vessel, cap, internals, fuel assemblies, fuel elements, steam separators, structural materials [VIII-1].
- RBMK — in core power detectors, pulse current fission chambers, analogue level gauges and level indicators.
- VK-50 — core cooling by natural circulation of boiling water coolant.
- SM.3, IVV.10 research reactors — control and protection system (CPS) actuator drives.

Several options were considered in the core arrangement and reactor design at the developmental stage of the VK-300 project to enhance safety and economic efficiency. Specifically, of all options considered the principle of passive operation of main safety systems was retained to achieve an optimum balance between the enhanced safety and improved economic characteristics.

As of 2004, the project documentation on the VK-300 was developed for a four-unit nuclear cogeneration plant to be built in the Arkhangelsk region of the Russian Federation.

VIII-1.2. Applications

The VK-300 reactor is designed to operate within a cogeneration plant to produce electricity and heat for district heating. The idea of using the VK-300 reactor for fresh water production in countries experiencing a deficit of potable water and having decentralized and relatively

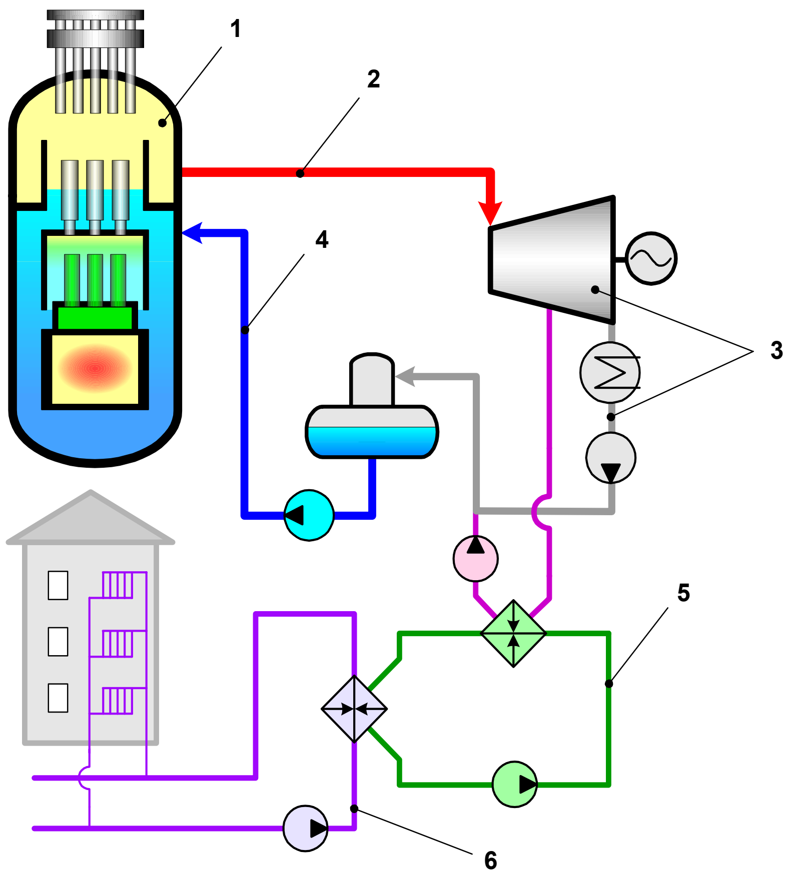
small electricity grids could also be exploited. There is also a potential for the VK-300 use within a floating nuclear cogeneration plant, possibly, under leasing arrangements.

VIII-1.3. Special features

The VK-300 is a land based cogeneration plant.

VIII-1.4. Summary of major design and operating characteristics

A schematic diagram of a power-and-heat supply unit with the VK-300 reactor is shown in Fig. VIII-1. This unit features direct steam feed from the reactor to the turbine. After passing a number of stages, a portion of the steam is removed from the turbine and fed to the primary circuit of the heating plant. Heat is supplied to the consumers from the second circuit of the district heating plant; pressure in this circuit is chosen to exclude radioactivity ingress from the primary circuit.



1 – VK-300 reactor
2 – Steam feed to the turbine
3 – Turbine unit
4 – Feedwater supply to the reactor
5 – District heating plant
6 – Heat consumers

FIG. VIII-1. Schematic diagram of a cogeneration unit with VK-300.

Major design and neutron-physical characteristics of the VK-300 are summarized in Tables VIII-1 and VIII-2 respectively.

TABLE VIII-1. SUMMARY OF MAJOR DESIGN CHARACTERISTICS

CHARACTERISTIC	VALUE/ DESCRIPTION
Installed capacity, Thermal Electric	750 MW(th) 250 MW (e) - in condensation mode 150 MW (e) - in heating mode
Heating unit capacity	400 GCal/h
Fuel type	Cylindrical fuel elements with fuel pellets of sintered uranium dioxide, a density of 10.4–10.7 g/cm^3 and cladding of the E110 zirconium alloy
Fuel enrichment	^{235}U content - 4%
Coolant	Boiling water
Moderator	Boiling water
Structural materials	Fuel element cladding - E110 zirconium alloy Reactor vessel- steel
Core	Cylindrical, made up of 313 hexagonal fuel assemblies, equivalent diameter - 3.16 m, height of active part of the fuel assemblies - 2.42 m
Core height, m	2.42
Core volume, m^3	8.79
Number of fuel assemblies	313
Number of fuel elements per fuel assembly	107
“Turn-key” size of a fuel assembly, m	0.160
Pitch of fuel assemblies in the core, m	0.170
Reactor vessel	The VVER-1000 vessel with necessary back fitting. Outer diameter - 4535 mm; Wall thickness - 200 mm; Height of the vessel (cap inclusive) - about 13100 mm.
Number of circuits, thermal cycle type	Single circuit scheme with saturated steam at the reactor outlet
Nuclear steam supply facility	Once-through steam supply facility with integral arrangement of the primary circuit within a water cooled and water moderated boiling water reactor
Mode of operation	Basic, in the mode of electric power and heat production, permitting a load follow operations schedule
Thermodynamic efficiency	33%
Design capacity factor	0.913
Design service life, years	60

TABLE VIII-2. NEUTRON-PHYSICAL CHARACTERISTICS

CHARACTERISTIC	VALUE
Reactivity effects, % ΔK/K:	
Heating from 20°C to saturation line t_s, $\Delta\rho_{ts}$	- (0.3 - 0.0)
Steam fraction, $\Delta\rho_{\varphi}$	- (6.2 - 4.4)
Doppler effect from t_s to the nominal temperature, $\Delta\rho_U$	- 0.7
Full temperature reactivity effect, $\Delta\rho_t$	- (7.0 - 5.1)
Xenon poisoning, $\Delta\rho_{Xe}$	- (2.6 - 2.4)

CHARACTERISTIC	VALUE
Reactivity margins, %ΔK/K:	
In cold state, $\Delta\rho_{cold}$	9.7 – 11.8
For fuel burn-up, $\Delta\rho_{work}$	0.15 – 3.5
Reactivity coefficients*, ΔK/K:	
α_U, 10^{-5}/°C	- (2.1 - 1.9)
α_φ, 10^{-4}/%φ	- (11.0 -10.4)
α_N, 10^{-5}/%N_{nom}	- (23.0 - 22.4)
Effective fraction of delayed neutrons β_{eff}, %	0.56 - 0.61
Peaking factors**:	
Axial (cumulative), K_z	1.65 - 1.18
Radial, K_r	1.54 - 1.34
Axial (fuel burn-up), K_{zb}	1.37 - 1.32

* α_U - reactivity coefficient in on fuel temperature, α_φ - reactivity coefficient on steam quality, related to change in the average volumetric steam content per 1 absolute %, α_N - power reactivity coefficient related to power change by 1% of the nominal value.

**Average factors for all fuel assemblies per the irradiation cycle.

The reactor has two independent mechanically driven control and protection systems (CPSs), each consisting of 135 absorber assemblies, three in a control member of the reactor CPS. The reactor also has a liquid boron shutdown system, based on injection of the sodium pentaborate water solution (NaB_5O_8, 3 g/kg). Each of the systems can scram the reactor and maintain it in a sub-critical state, as illustrated by the data of Table VIII-3.

TABLE VIII-3. REACTIVITY CONTROL AND PROTECTION SYSTEM

REACTIVITY IN VARIOUS STATES AFTER ACTUATION OF THE MECHANICAL PROTECTION SYSTEM OR LIQUID BORON SHUTDOWN SYSTEM, %ΔK/K		
REACTOR STATE/ PARAMETERS	FOR 90/ 89[A] ACTUATED MEMBERS OF THE CPS	OPERATION OF LIQUID BORON SHUTDOWN SYSTEM (NaB_5O_8)[B]
Nominal Xe_{steady}; P=7 MPa; $\gamma^{(c)}$=530 kg/m^3; T_U=540°C	- 31 / -24	
Residual heat release, 5% N_{nom} ; Xe_{steady}; γ=710 kg/m^3 ; T_U=310°C	- 22.5 / -19	
Residual heat release up to 1% N_{nom} ; Xe_{steady}; $\varphi^{(d)}$=0; γ=760 kg/m^3; T_U=284°C	- 21.5 / -15	- 4.9
Cooled down to 200°C; Xe=0; φ=0; γ=870 kg/m^3; T_U=200°C	- 14.6 / -8.4	- 4.4
Cooled down to 20°C; Xe=0; γ=1000 kg/m^3; T_U=20°C	- 8.3 / -3.6	- 5.2

(a) Allowance is made for failure of the actuator of a most effective member the protection system.
(b) With all members of the control and protection system withdrawn from the core.
(c) γ - core averaged coolant density
(d) φ - volumetric steam quality averaged over the core fuel element-to-fuel element space

Table VIII-4 presents main thermal-hydraulic characteristics of the VK300; the design limits for the VK-300 are given in Table VIII-5.

TABLE VIII-4. THERMAL-HYDRAULIC CHARACTERISTICS

CHARACTERISTIC	VALUE/ DESCRIPTION
Circulation type	Natural circulation
Feedwater temperature, °C	190
Steam parameters at the reactor outlet: - Pressure, MPa - Temperature, °C - Maximum moisture content, % by weight	 6.86 284.5 0.1
Steam output of the reactor, t/h	1360
Hydraulic diameter of a fuel assembly, m	0.0121
Heat exchange surface for a fuel assembly, m^2	7.4025
Coolant velocity in a fuel assembly, m/s	0.8–1.1
Average linear heat rate, W/cm	92.5
Maximum operating linear heat rate, W/cm	284
Outlet steam quality for average power fuel assembly, %	15.5
Outlet steam quality for maximum power fuel assembly, %	26.9
Outlet volume fraction of steam for average power fuel assembly, %	68.4
Outlet volume fraction of steam for maximum power fuel assembly, %	76.9
Minimal dry out margin	1.27

TABLE VIII-5. DESIGN LIMITS FOR VK-300

CATEGORY OF LIMITS	DESCRIPTION	VALUE
Operation limits	Fuel element damage limit in terms of the number and size of defects: Gas leaks due to micro cracking, % Direct fuel - coolant contact, % Fuel cladding temperature: External surface, °C Internal surface, °C	 0.02 0.002 400 500
Safe operation limits	Fuel element damage limit in terms of the number and size of defects: Gas leaks due to micro-cracking, % Direct fuel - coolant contact, %	 0.1 0.01
Accident limits	Fuel cladding temperature, °C	1200
	Local depth of fuel cladding oxidation, % of the initial cladding thickness	18
	Portion of reacted zirconium, % of its weight within fuel claddings	1
	Maximum fuel temperature, °C	2800

The lifetime maximum fuel and fuel cladding temperatures for the VK-300 reactor are shown in Fig. VIII-2 and VIII-3.

During the entire service life of a fuel element, the maximum temperature of the cladding external surface does not exceed 295°C; the internal surface temperature does not exceed 343°C.

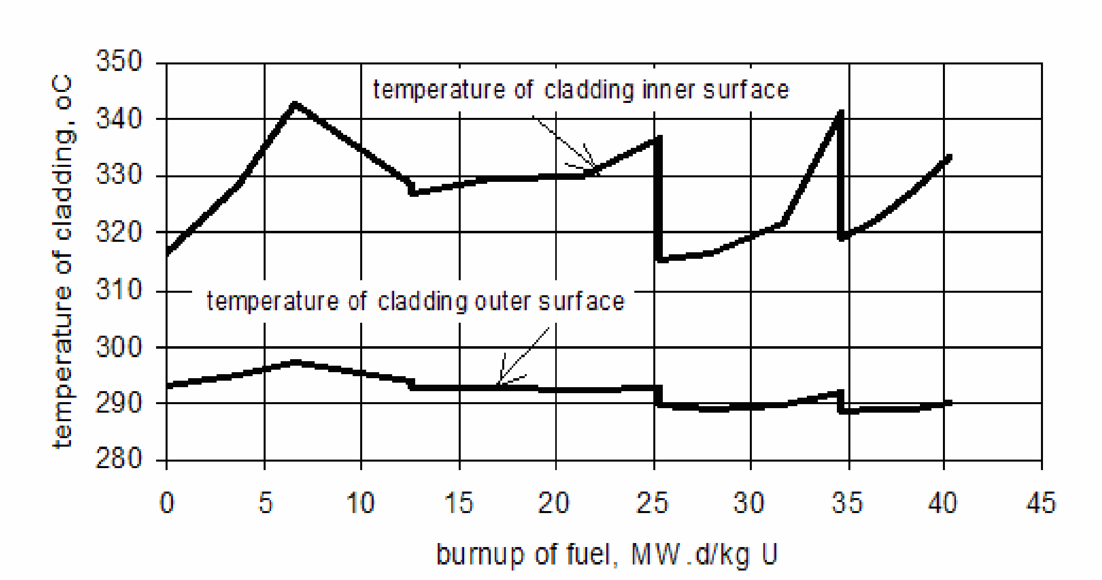


FIG. VIII-2. Lifetime variation of the maximum temperature of outer and inner cladding surface for a maximum power fuel element.

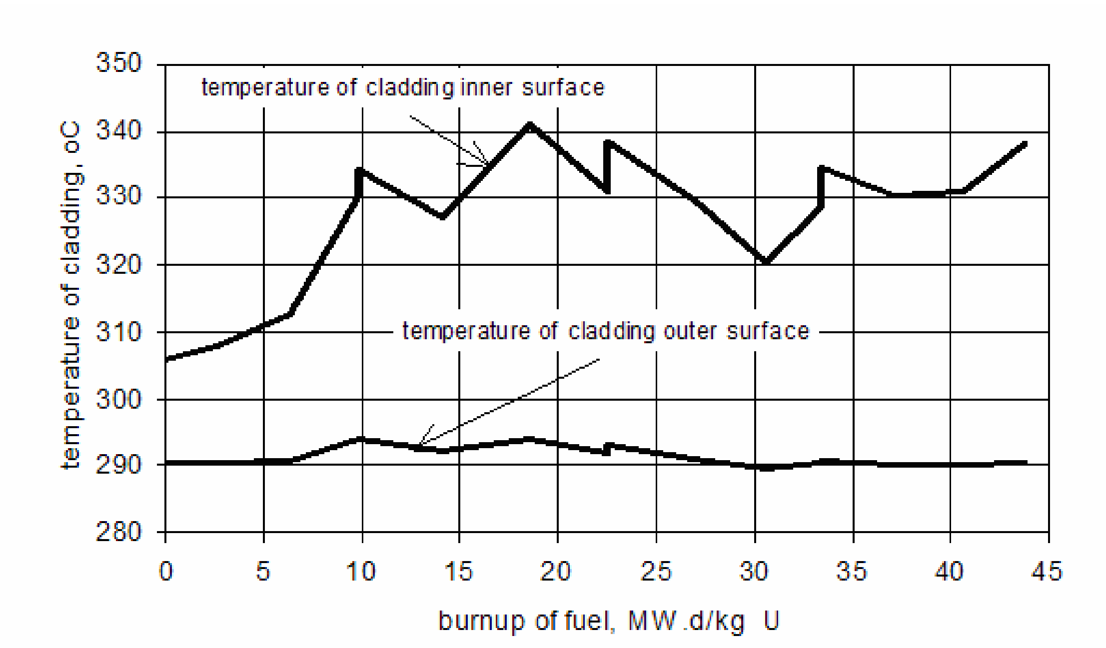


FIG.VIII-3. Lifetime variation of the maximum temperature of outer and inner cladding surface for a maximum burn-up fuel element.

The maximum fuel temperature does not exceed 1230°C during the entire lifetime of a maximum power fuel element, and 1050°C for a maximum burn-up fuel element.

The burn-up cycle and material balance data are given in Table VIII-9.

TABLE VIII-9. BURN-UP CYCLE AND MATERIAL BALANCE DATA

CHARACTERISTIC	VALUE
Full lifetime, effective days (years)	1748 (6)
Irradiation cycle duration, effective days (calendar months)	437 (18)
Fraction of reloaded assemblies	¼
Number of reloaded assemblies	78
Uranium loading, t	31.8
Fuel enrichment, % by weight	4
Average burn-up of discharged fuel, MW·d/kg	41.4
Contents of selected isotopes in discharged fuel, kg/t:	
Uranium-235	6.68
Uranium-236	5.23
Uranium-238	936.0
Plutonium-239	3.95
Plutonium-240	2.19
Plutonium-241	1.06

Economic estimates for the VK-300 were performed in conformity with the current rules, regulations and methodical recommendations on the definition of construction costs for NPPs on the territory of the Russian Federation. The assessments were performed for a project of the 500 MW(e) (2×250 MW) heat and power plant linked to a specific site in the North of Russia. The results of these assessments are summarized in Table VIII-10.

TABLE VIII-10. RESULTS OF ECONOMIC ASSESSMENTS

ITEM	VALUE/ DESCRIPTION	REMARKS
Specific cost of equipment	450 US $/ kW(e)	
Specific capital costs for construction	US $330/ kW(e)	This assessment made an allowance for the site being partially developed. The scope of the available development was estimated at US $0.5 million
Other specific capital investments	US$ 85/ kW(e)	The operation assets are defined by the cost of 313 fuel assemblies and fuel reserve (10%)
Operation assets	US$ 10 million	
Design net cost of: - Electric power - Heat	 US$ 0.01/kW-h US$ 3.33 /GCal	

VIII-1.5. Outline of fuel cycle options

Since VVER-1000 type fuel elements are used in the VK-300, the fuel cycle for the VK-300 reactor is assumed to be similar to that of other water cooled and water moderated reactors.

VIII-1.6. Technical features and technological approaches that are definitive for VK-300 performance in particular areas

VIII-1.6.1. Economics and maintainability

For the VK-300, improvement of the economic characteristics is attained through the following design philosophy and features:

- Some expensive primary circuit equipment items (main circulation pumps, steam generators, remote separators, etc.) are eliminated through use of the vessel-type boiling reactor with an integral arrangement of the primary circuit inside the vessel and natural circulation of the coolant;
- Maximum simplicity of the plant design based on a single circuit scheme of the power unit to reduce construction and operating costs; in this, an additional barrier to radioactive substance propagation is provided by application of a special scheme for the separation of phases inside the reactor;
- Elimination of some expensive safety-related equipment through the application of passive principles of systems operation and employment of a primary containment;
- Maximum possible use of time-proven equipment, structural materials and processes; and
- Provision of an option for basic equipment manufacture and testing under factory conditions.

VIII-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

The VK-300 is a nuclear cogeneration plant designed to shoulder certain functions in the structure of integrated large-scale nuclear power, where each element would play its own role. Specifically, the VK-300 could replace fossil-fuelled heat and power plants located nearby or within cities or settlements. The factors contributing to this are as follows.

An insignificant dose burden on the personnel and population is provided for in the VK-300. For normal operating conditions, the dose burden for populations on the control area boundary will make up pro mille versus the sanitary norm.

In design-basis accidents with the most severe consequences, the effective dose of population radiation exposure at a distance of 1 km or more will not be in excess of 5% of the regulated value.

Analyses of radiation impacts in beyond design basis accidents define the control area around the nuclear cogeneration plant as coinciding with the site territory, while the planning area for post-accident measures was defined as 3 km. No post-accident measures planning the mandatory evacuation of population is required.

The values of heat emissions per unit of power for fossil-fuelled heat and power plants and nuclear cogeneration plants are similar so that heat pollution induced by nuclear cogeneration plants can be considered inconsequential.

Waste produced during the operation of a nuclear cogeneration plant, liquid, solid and gaseous, both radioactive and non-radioactive, is subject to processing in a way similar to that applied in present-day LWRs. Liquid radioactive waste is processed by specialized water treatment facilities with subsequent solidification and compaction. Solid radioactive waste is treated and compacted. The processed waste in containers is removed to centralized storage; gaseous radioactive waste is trapped by filters.

Spent fuel management for the VK-300 is similar to that used in standard PWRs and BWRs. Any advancements in spent fuel management for LWRs would also be applicable to the VK-300.

VIII-1.6.3. Safety and reliability

Safety concept and design philosophy

The objective of the VK-300 safety design is to assure radio nuclide confinement in the fuel under normal operating and emergency conditions so that radiation exposure of personnel and radiation in the area of the nuclear power plant site are maintained within the limits prescribed by regulations.

The safety design principles of the VK-300 are:

- Consistent implementation of the defence-in-depth concept;
- Enhancement of the reactor inherent and passive safety features (self-protection features);
- Implementation of approaches to assure reliable functioning of safety systems, such as:
  - Redundancy;
  - Spatial and functional independence;
  - Diversification;
  - Maximum reliance on passive principles of operation.

Provisions for simplicity and robustness of the design

The VK-300 relies on natural circulation of light water coolant for core heat removal in normal operation and in accidents. The VK-300 design was optimized to find an effective and efficient combination between inherent and passive safety features and engineered (active and passive) systems.

Structure of the defence-in-depth

The basic principle of the VK-300 safety design is consistent implementation of the defence-in-depth concept based on a system of physical barriers to ionizing radiation and radioactive substance propagation to the environment. The system of barriers incorporates:

- A fuel matrix;
- Fuel element claddings;
- The reactor vessel and adjoining pipelines;
- Pipelines of residual heat removal system;
- A primary containment and emergency cooling water tanks;
- Leak-tight reactor compartment rooms; and
- Secondary containment.

Different levels of the defence-in-depth incorporate the following features and measures of power level control:

- Automatic power level control;
- Preventive power reduction;
- Emergency protection; and
- Power level self-control via stabilizing negative feedbacks.

Active and passive systems and inherent safety features

Inherent safety features and passive systems

Table VIII-11 summarizes self-protection properties (inherent and passive safety features) of the VK-300.

TABLE VIII-11. SELF-PROTECTION PROPERTIES OF VK-300 IN RELATION TO SAFETY FUNCTIONS

POWER LEVEL CONTROL	HEAT REMOVAL FROM NUCLEAR FUEL	PRESSURE MAINTENANCE IN PRIMARY CIRCUIT	CONFINEMENT OF RADIOACTIVITY
Power level self-restriction achieved via stabilizing negative feedbacks.	Fuel element cooling by the all-mode natural circulation based system. Low linear power of fuel elements. Water supply in the reactor vessel assuring the core flooding. Natural circulation based residual heat removal system. Emergency cooling of the reactor provided by water supply tanks passively operated due to differential head. Water supply in the emergency cooling water tanks, providing passive absorption of the reactor heat and the core flooding maintained for a long period.	Self-restriction of pressure change rate in the primary circuit due to damping features of the steam blanket in the reactor vessel, supported by properties of a boiling coolant.	Low temperature of nuclear fuel facilitating confinement of fission products within fuel matrix. Restricted entrainment of radioactive substances from the primary circuit by steam, owing to their different concentration in water and steam, and also due to the separator unit being located within the reactor vessel. Leak-tight compartments of the nuclear cogeneration plant.

Active systems

Two independent mechanical systems of reactivity control and a liquid boron shutdown system are the active safety systems of the VK-300.

More details on safety systems are given in the second part of this design description.

Design basis accidents and beyond design basis accidents

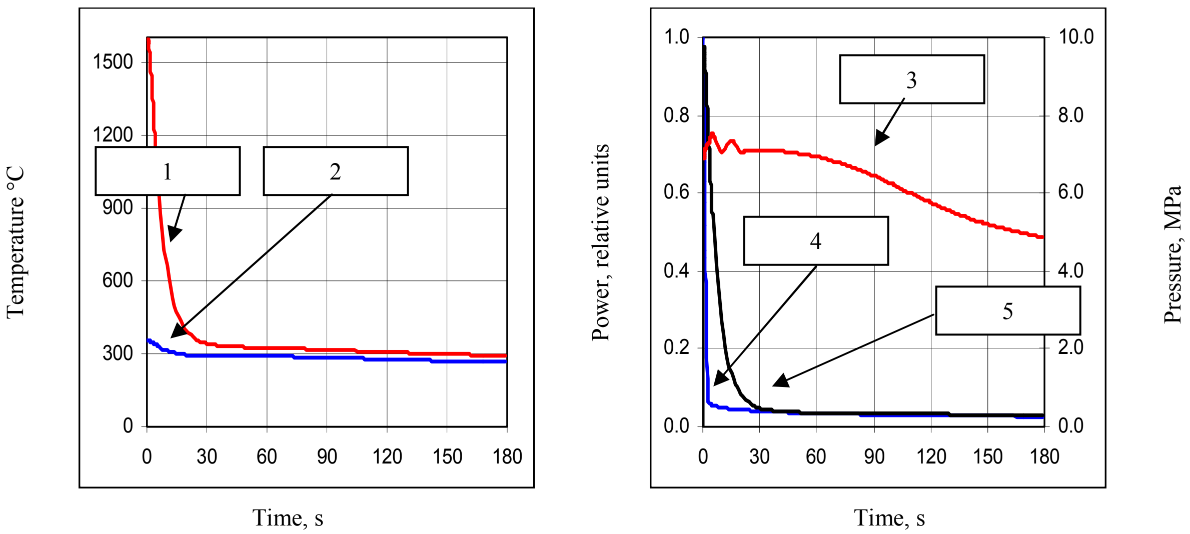
At the detailed design phase, a preliminary safety analysis report for the nuclear cogeneration plant with the VK-300 reactor was developed, providing a list of accidents considered. The list of accidents was developed on the basis of failure analysis of the nuclear cogeneration plant elements, taking into account possible human errors and external impacts.

The experience gained in design and operation of existing nuclear cogeneration plants was also taken into consideration when compiling the list of accidents.

The following groups (classes) of initiating events were considered:

- Changes in reactivity and/or power distribution due to erroneous operation of active reactivity control systems;
- Reduction or termination of normal heat removal from the reactor as a result of equipment failures, including loss of in-house power;
- Termination of normal heat removal from the reactor as a result of a loss of tightness of the steam line or feedwater pipeline beyond the primary containment shell;
- Termination of normal heat removal from the reactor as a result of a loss of tightness beyond the primary containment shell of the actuating mechanisms of cooling system of the CPS, system of primary circuit coolant purification and system of maintenance cooling;
- Loss of the primary circuit tightness inside the primary containment;
- Increase in heat removal from the reactor due to failures or spurious actuation of systems; and
- Errors in nuclear fuel handling.

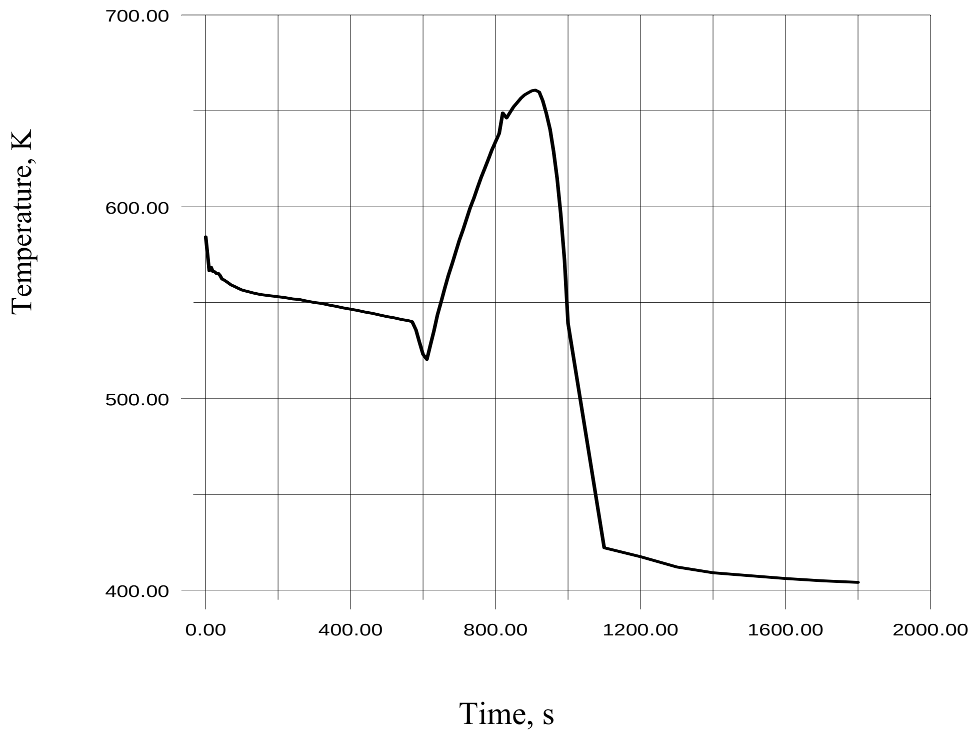
An NPP blackout was rated as the most probable accident. The results of analysis of an NPP blackout accompanied by the concurrent failure of the emergency protection actuation signal are shown in Fig. VIII-4 (the emergency protection system is actuated by a signal of the reduction of steam flow to the turbine unit). No core damage hazard is observed.



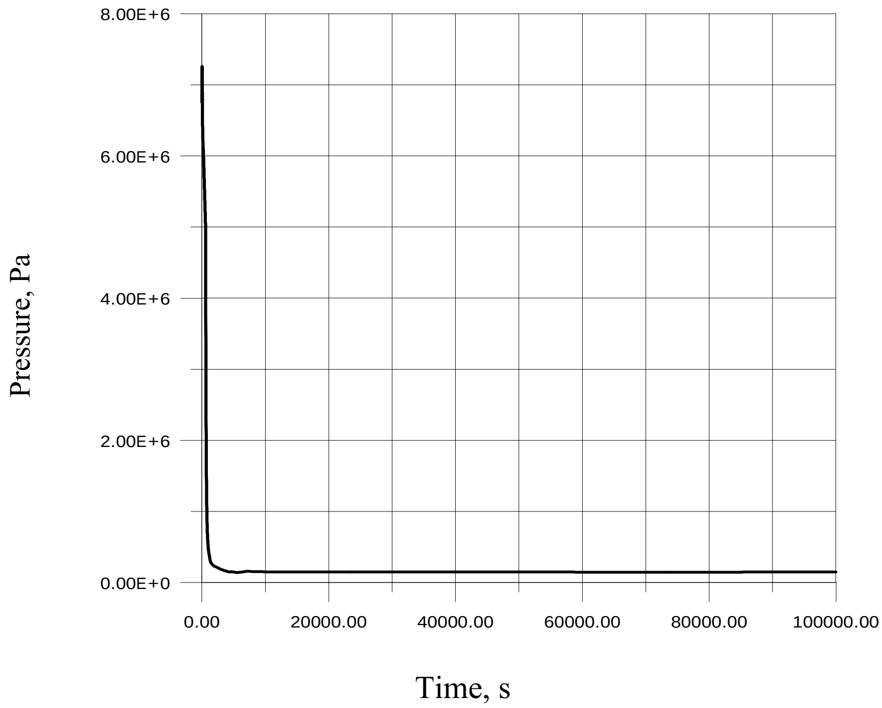
1- Maximum fuel temperature, °C 2 - Maximum cladding temperature, °C 3 - Pressure in the reactor, MPa
4 - Relative power density 5 - Relative power supplied to the coolant

Fig. VIII-4. NPP blackout with failure of emergency protection system.

An accident involving a pipeline rupture in the primary circuit coolant cleaning system, where the fuel cladding temperature is up to 660°C (Fig. VIII-5), is likely to pose the highest risk of a loss of cladding integrity, although the temperature value reached is below the safe operation limit for a fuel element.



(a)



(b)

Fig. VIII-5. Accident with rupture of a pipeline in the primary coolant cleaning system: (a) Temperature, K; (b) Pressure, Pa.

Analysis of the consequences of abnormal operation occurrences and accidents for the VK-300 indicates the following:

(1) For all design basis accidents considered, no exceedance of the fuel cladding limit temperature adopted for normal operating conditions is observed; likewise, no loss of fuel element tightness takes place;

(2) For design basis accidents involving ruptures of the primary circuit pipelines beyond the primary containment shell, the use of isolating devices in the pipelines restricts ingress of the activity to the compartments and the downstream cleaning systems, which ensures compliance with the prescribed regulations on radioactivity releases with a considerable margin;

(3) For design basis accidents involving ruptures of the primary circuit pipelines inside the primary containment, the localization of releases by bubbling air into the emergency cooling water tank essentially reduces the activity, while the downstream cleaning of gas blow offs of the emergency cooling water tank will ensure that the requirements specified by regulatory documents are met[1];

(4) According to the estimates, as a result of design basis accidents involving the most unfavourable radiation consequences, the effective radiation exposure dose at a distance of 1 km or more from the plant will not be in excess of 5% of the maximum permissible value.

Provisions for safety under seismic conditions

It is assumed that safety under seismic impacts will be assured in conformity with practices set forth in the regulatory documents. The strength analysis of the basic project of the VK-300 confirmed the seismic resistance of the plant up to an earthquake magnitude of 7 on the MSK scale.

Probability of unacceptable radioactivity release beyond the plant boundaries

Probabilistic safety analysis was performed at the stage of preparation of a preliminary safety report for the underground nuclear cogeneration plant with the VK-300. The estimated value of core severe damage probability does not exceed 2.0×10^{-8} per reactor-year, with the corresponding radioactivity release set at 1.3×10^{8} Bq. Such low probability value is attained due to broad implementation of inherent safety features and the extensive use of passive principles of safety system actuation.

Measures planned in response to severe accidents

The analysis of radiation consequences of beyond design basis accidents confirms that there will be no need for post-accident off-site emergency measures.

VIII-1.6.4. Proliferation resistance

No major difference with the traditional water cooled and water moderated reactors is envisaged.

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[1] The project of an underground NPP with the VK-300 envisages the appropriate systems. Prior to dumping to the atmosphere, gas from the emergency cooling water tank (blow-offs) is fed (via a cooling gas-holder) to an activity suppression facility, where it undergoes cleaning in aerosol and iodine filters; the cleaning factor is no worse than 1000, in terms of volatile iodine species – no worse than 10.

VIII-1.6.5. Technical features and technological approaches used to facilitate physical protection of VK-300

Strong reliance on inherent safety features and passive principles of safety system operation essentially reduces the potential for premeditated disabling of the VK-300 safety functions.

The VK-300 reactor and associated systems are accommodated within the primary containment, localizing the safety systems and physically protecting the reactor equipment. The equipment in the reactor compartment is arranged within leak-tight boxes.

The power unit is arranged within the containment to withstand natural and human-induced external impacts.

VIII-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of VK-300

The analysis of market demands for cogeneration plants with district heating in certain regions of the Russian Federation was performed prior to designing the VK-300.

VIII-1.8. List of enabling technologies relevant to VK-300 and status of their development

The VK-300 design makes an extensive use of the components and engineering approaches proven in operation of many LWRs therefore, the scope of necessary R&D is expected to be rather limited. Further R&D are needed to:

- Validate and test characteristics of the passive systems;
- Validate and test characteristics of the control and protection system actuator;
- Study fuel assembly characteristics at higher fuel burn-ups; and
- Validate and test thermal-hydraulic characteristics of the in-vessel natural circulation circuit under normal and emergency operating conditions.

VIII-1.9. Status of R&D and planned schedule

As of 2005, the following design documentation has been prepared:

- A detailed design of the VK-300 reactor for a nuclear cogeneration plant;
- Basic regulations for a typical nuclear cogeneration plant using the VK-300 reactors; and
- Substantiation of investments into construction of a nuclear cogeneration plant in the Arkhangelsk region of the Russian Federation.

Research and development activities are currently underway for further validation of the design approaches adopted in the VK-300 design. It is estimated to take 2–3 years to implement the abovementioned activities.

The plan is that construction of the first phase (two power units) of the nuclear cogeneration plant with the VK-300 reactor will be accomplished within five years from the start-up date of construction, including:

- Building of the foundation - 8 months;
- Erection of the superstructure - 24 months;
- Mounting of the equipment - 41 months;
- Construction of other major buildings and structures - 52 months.

It is assumed that the first power unit will be realized at the end of the fourth year starting from the moment of construction and mounting; the second power unit - by the end of the fifth year.

VIII-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

Experience in the design, manufacture and operation of different reactors, with the VK-50 as a direct predecessor, was used to the maximum extent in the project of the VK-300.

In contrast to the VK-50, a number of innovative features discussed above are implemented in the VK-300 design to improve safety, reduce costs and create a potential for the power unit use in a large power sector, i.e., for the cogeneration of heat and power.

These innovative approaches, to be confirmed in a prototype plant, are as follows:

- Application of the original primary circuit scheme with in-pile natural circulation of coolant and multi-stage steam separation;
- Use of axial centrifugal separators built into the reactor vessel; and
- Safety assurance by passive systems.

The prototype plant is also necessary to validate operational modes and algorithms.

Modernization of the operating prototype, the VK-50 reactor, is viewed as the optimal variant for pilot facility construction.

VIII-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

No information was provided.

VIII-2. Design description and data for VK-300

This section provides a summary of the design data for the VK-300, with more details being presented in the International Atomic Energy Agency [VIII-2].

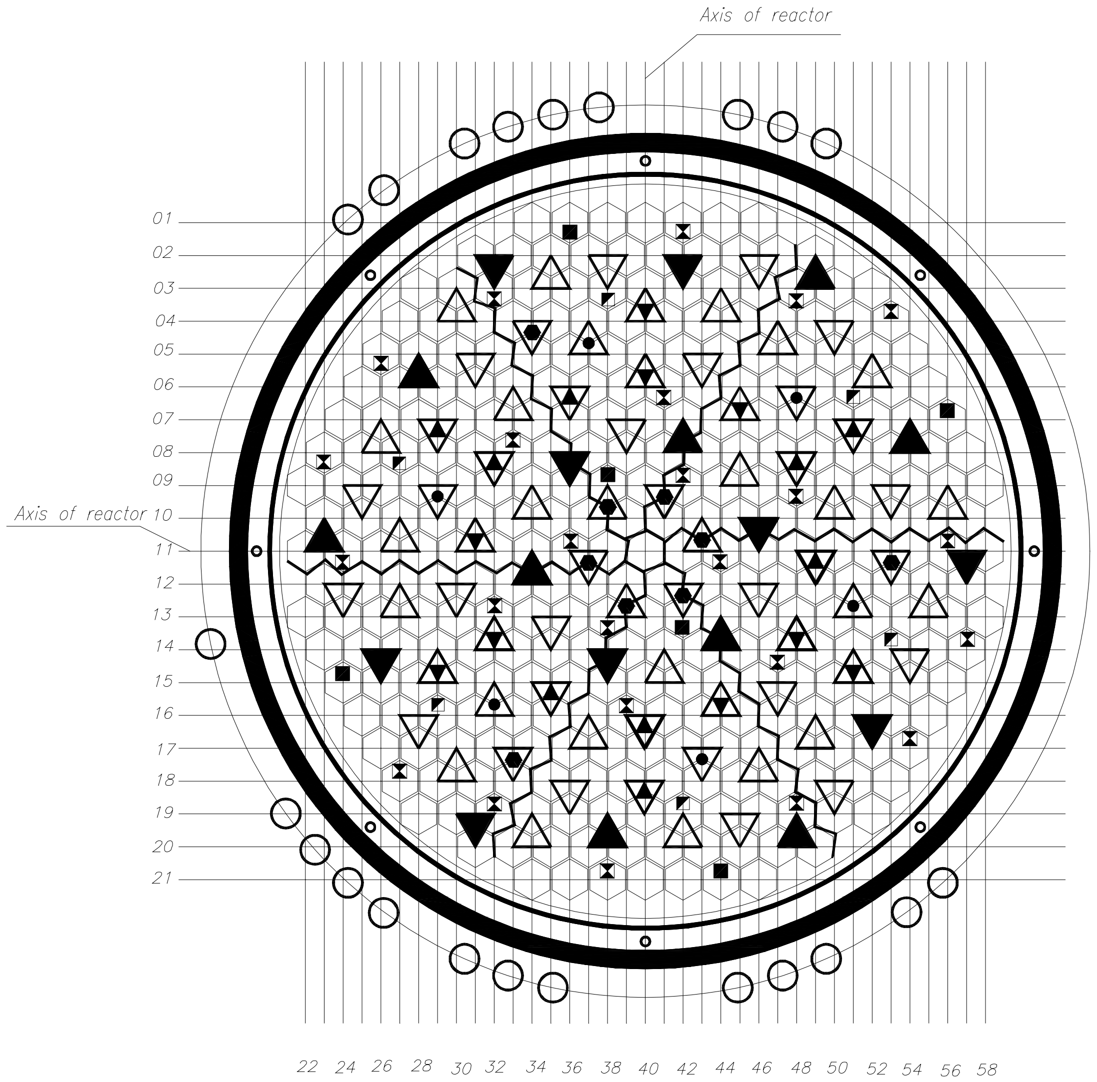
VIII-2.1. Description of the nuclear systems

Reactor core and fuel design

The following specific core design features characterize the VK-300:

- The use of uranium-gadolinium fuel with Gd_2O_3 as burnable absorber; due to the use of the burnable absorber and partial refuelling of the reactor, the reactivity margin could be kept as low as possible;
- The relatively low temperature and specific power of fuel provide for the reduction of energy accumulated in the fuel and the restriction of gas release from the uranium dioxide.

The reactor core of the VK-300 consists of 313 hexagonal fuel assemblies. The fuel assemblies of the flat-to-flat size of 160 mm are arranged in a regular triangular lattice with the pitch of 170 mm. They form a symmetric configuration characterized by the rotational angle of symmetry of 60°. The active height of the reactor core is 2420 mm. The equivalent diameter of the reactor core is 3160 mm. The VK-300 core map is given in Fig. VIII-6.



Legend	Number
- fuel assembly(FA)	313
- control rod of the control system	39
- control rod of the emergency protection system	9
- emergency protection rod	18
- shim rod of the emergency protection system	18
- automatic regulator	6
Total number of the CPS rods	90
- power density sensor (PDS) equipped with the sensor for core inlet temperature monitoring	24
- PDS equipped with the sensor for temperature monitoring in the mixing chamber	6
- PDS equipped with the sensors for core unlet and outlet temperature monitoring and core outlet power density	6
Total number of in-core sensors	36
- channels for the sectional in-core sensors(ICS) based on the pulse current fission chamber	
- for EP including 3 ICS for simultaneous monitoring of subcriticality and power (SPMS)	6
- redundant channels	1
- for the emergency control room	1
Total number of ICS channels	8
-channels for the side ionization chambers(SIC)	
- for emergency protection	12
- for automatic regulation	8
- for emergency control room	1
- redundant channels	1
Total number of SIC channels	22

FIG. VIII-6. VK-300 core map.

The VK-300 fuel assembly map is shown in Fig. VIII-7; the fuel assembly data is in Table VIII-12.

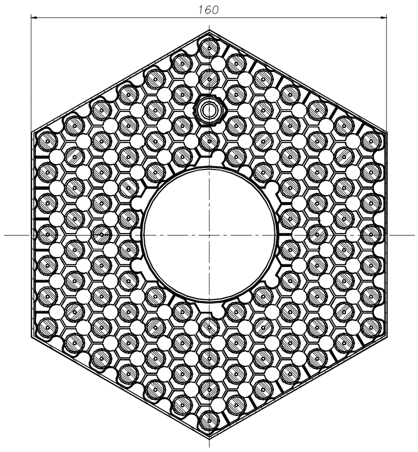


FIG. VIII-7. VK-300 fuel assembly map.

TABLE VIII-12. FUEL ASSEMBLY DESIGN DATA

CHARACTERISTIC	VALUE
Thickness of the outer casing, mm	1
Overall length, mm	3040
Height of fuel column in fuel rods, mm	2420
Height of fuel column in rods with Gd_2O_3, mm	2420
Lattice pitch, mm	14
Sizes of the inner tube, mm	61×1
Sizes of the tube for ICS, mm	12.6×0.85

Main design characteristics of the VK-300 fuel elements are summarized in Table VIII-13.

The design of the rods with Gd_2O_3 (mixed with fuel) is similar to the design of fuel elements. Fuel is manufactured of sintered uranium dioxide in pellet form with the addition of gadolinium oxide at the upper 2150 mm-long section. At the lower 170 mm-long section of these rods, the uranium enrichment is 1.6% and gadolinium is not added.

TABLE VIII-13. FUEL ROD DESIGN DATA

CHARACTERISTIC	VALUE
Material for fuel cladding and end parts	E110 alloy
Filling gas	Helium
Length of fuel element, mm	2621±2
Length of fuel column in cold state, mm	2420±6
Outer diameter of fuel element, mm	$9.12^{+0,08}_{-0,05}$

CHARACTERISTIC	VALUE
Inner diameter of fuel cladding, mm	$7.73^{+0,06}$
Minimum thickness of fuel cladding, mm	0.63
Diameter of a fuel pellet, mm	$7.57_{-0.03}$
Diameter of a hole in the fuel pellet, mm	$1.5^{+0.2}$
Height of the fuel pellet, mm	9÷11
Nominal conventional mass fraction of ^{235}U, % by weight*	4.0
Fuel density, g/cm^3	10.4÷10.7

* At the lower 170 mm-long section, the uranium enrichment is 1.6%.

Safety systems

The safety systems are designed as multi-train systems in keeping with the principles of redundancy, independence, diversity, single failure and the preferred passive principle of action. In compliance with the principle of single failure, safety systems must be capable of performing the assigned functions under any initiating event that causes actuation, in case of failure of active or passive components with mechanical moving parts, acting independent of the initiating event and assuming an additional coincidental failure of any other component which can not be controlled during operation of the reactor.

The following safety systems are envisaged:

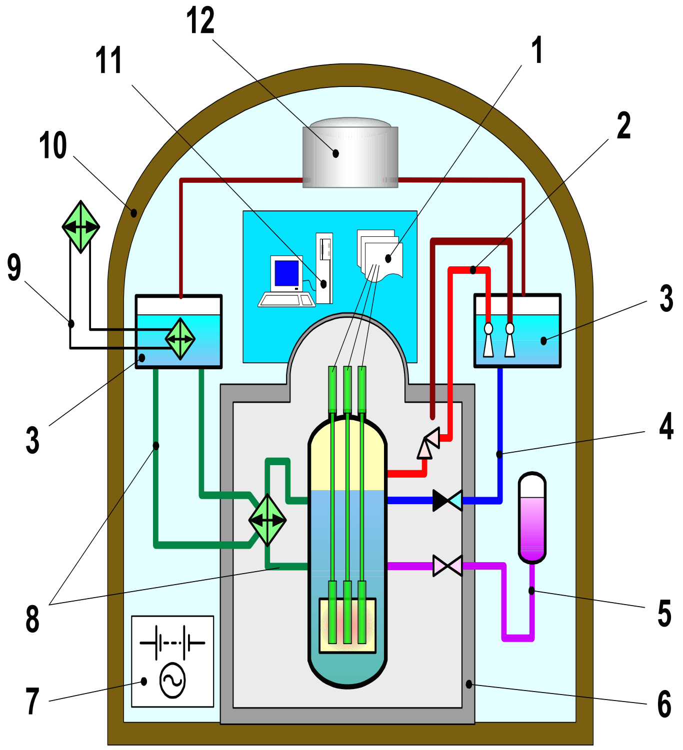
(1) Protection systems:

- Emergency protection system;
- Residual heat removal system;
- Emergency core cooling system;
- Overpressure protection system;
- Depressurization system; and
- Liquid shutdown system.

(2) Localizing safety systems:

- Primary containment including isolation devices and pressure suppression equipment located therein;
- Gas holder, activity suppression system, etc.;
- Support safety systems:
- System for residual heat removal to the ultimate heat sink;
- Emergency power supply system; and
- Control safety systems.

Safety system components are located inside the containment, Fig. VIII-8.



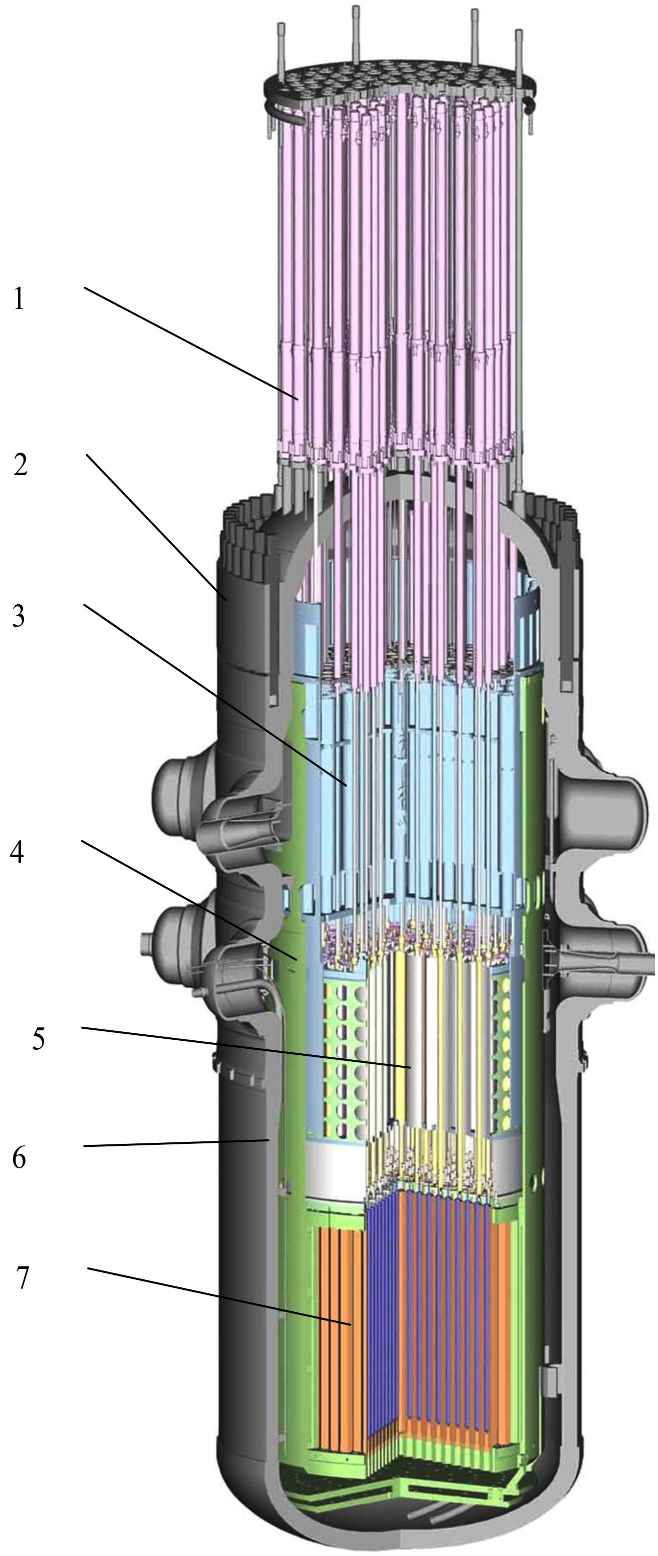
1 - Control and protection system
2 - Overpressure protection system and depressurization system
3 - Emergency cool down tanks
4 - Emergency core cooling system
5 - Liquid boron shutdown system
6 - Primary containment
7 - Emergency power supply system
8 - Residual heat removal system
9 - System for residual heat removal to the ultimate heat sink;
10 – Containment
11 - Control safety systems
12- Gas holder

FIG. VIII-8. Safety systems of VK-300.

Control and protection system

In the reactor core there are 90 control members of the control and protection system. Each control member includes 3 absorber assemblies. The absorber assemblies are located in the central tubes of 270 fuel assemblies. The absorber assemblies consist of 16 absorber elements placed in the circle at regular intervals. The absorber material is vibro-packed powder of boron carbide (B_4C). By using a special cross-bar, the absorber assemblies in three adjacent fuel assemblies are united in the control member of the CPS. Each control member of the CPS can be moved inside the reactor by an individual drive, which is independent in terms of the control action. The travel distance of the CPS control member is 2600 mm.

The absorber assembly consists of 16 absorber elements of 8.2 mm in diameter placed in the circle of 46.2 mm diameter at regular intervals and fixed at the support grid in the central carrier rod, Fig. VIII-9. The control members are inserted to the core from the top, as in PWR (VVER) type reactors.

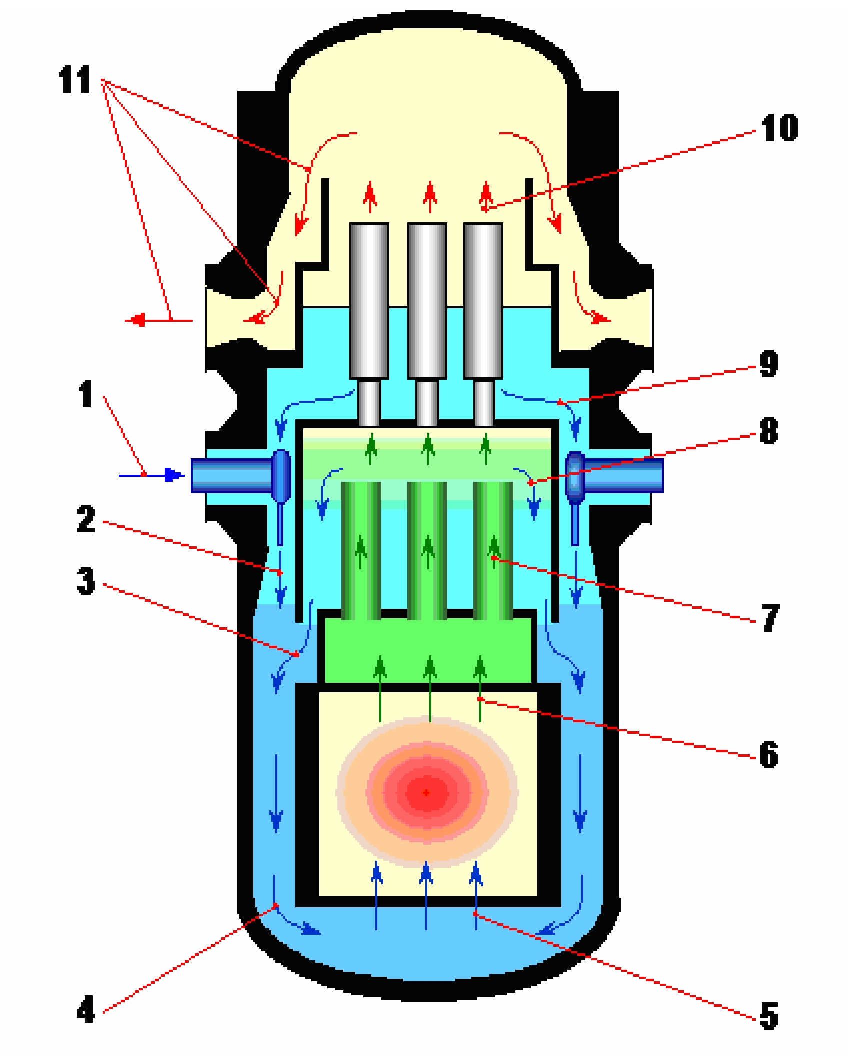


1 – Drive mechanisms of CPS members
2 – Reactor vessel cover
3 – The unit of axial centrifugal separators
4 – Reactor vault
5 – The unit of riser tubes
6 – Reactor pressure vessel
7 – Reactor core

FIG. VIII-9. VK-300 reactor.

Main heat transport system

The scheme of in-vessel natural circulation is shown in Fig. VIII-10. Figure VIII-11 presents a schematic diagram of the VK-300 main heat transport system with specification of heat removal path in normal operation and in accidents.



1 – Feedwater supply
2 – Mixing of feedwater and the major portion of the coolant flow
3 – Mixing of separated water and the downcomer flow
4 – Return of the flow to the core
5 – Water entering the core
6 – Steam-water mixture leaving the core
7 – Steam-water mixture flow in the riser tubes
8 – Preliminary separation at the outlet from the riser tubes
9 – Water leaving the axial centrifugal separators
10 – Steam leaving the axial centrifugal separators
11 – Steam outlet from the reactor

FIG. VIII-10. In-vessel natural circulation scheme.

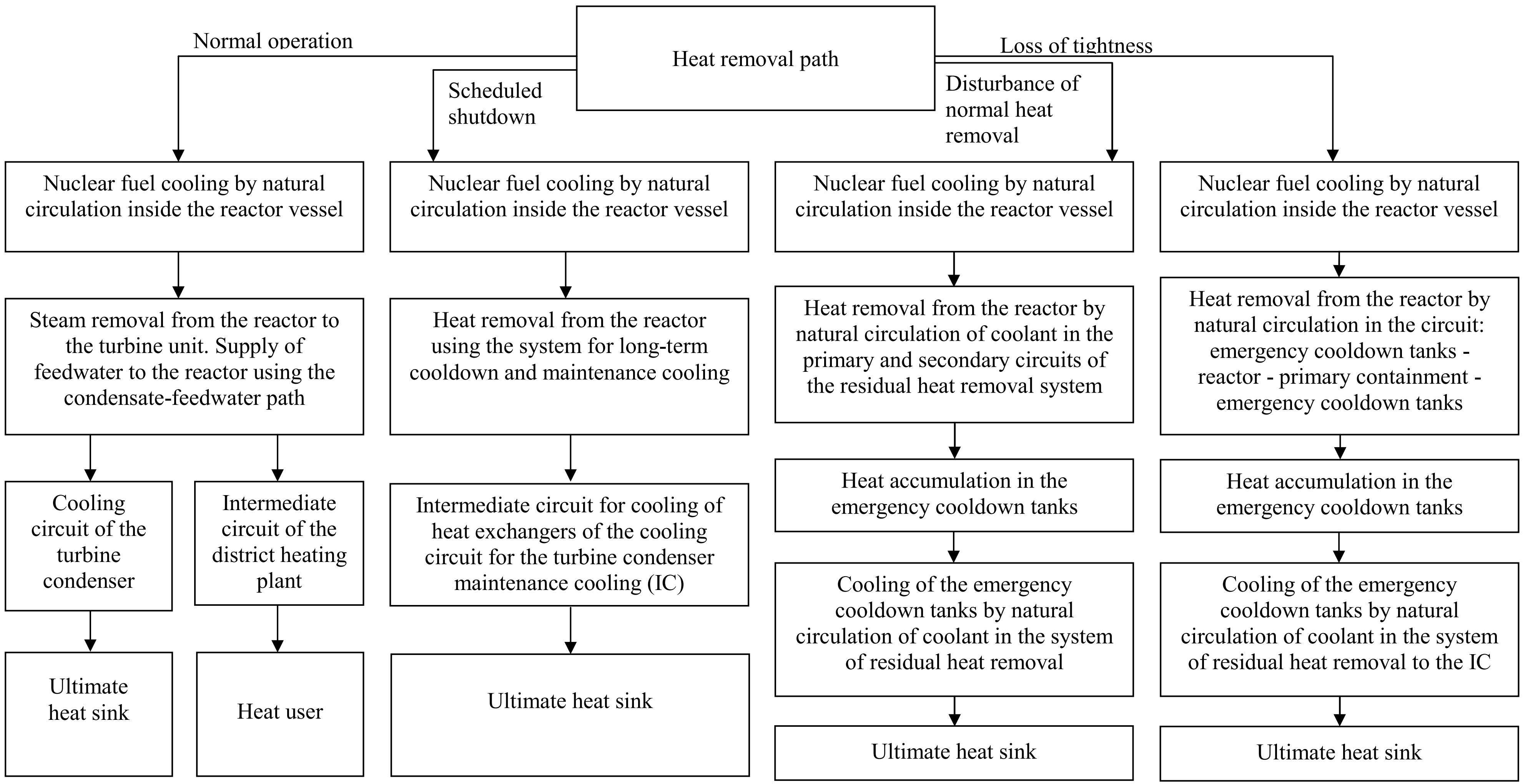


FIG. VIII-11. Main heat transport system of VK-300.

VIII-2.2. Description of the turbine generator plant and systems

The power unit of the VK-300 reactor facility includes the reactor, the turbine and district heating systems, as shown in Fig. VIII-1.

The turbine is designed for operation in a single unit with the VK-300 reactor, in accordance with the direct cycle scheme and for the simultaneous generation of electricity and heat for district heating and hot water supply to municipal and industrial users. The turbine is a single-shaft device consisting of a high and intermediate-pressure cylinder (HIPC) and a low-pressure cylinder (LPC). Nominal power of the turbine in the cogeneration mode is 150 MW(e); maximum power in the condensation mode is 250 MW(e).

VIII-2.3. Systems for non-electric applications

Water is heated step-wise in the intermediate circuit of the heating network in the double-stage district heating facility, designed in accordance with the two-line scheme, as shown in Fig. VIII-1. The LP boiler is supplied with steam extracted at the steam receiver between the HIPC and the LPC, downstream of the LP separator. The LP boilers are supplied with steam extracted downstream of the high-pressure turbine part.

Depending on the load for district heating, either both stages or only the lower stage of the boiler unit can be operated. Steam extraction to the LP boiler is controlled.

Water flow rate in the intermediate circuit varies from 2500 to 5000 t/h depending on the mode of operation.

The nominal heating load is 400 GCal/h.

VIII-2.4. Plant layout

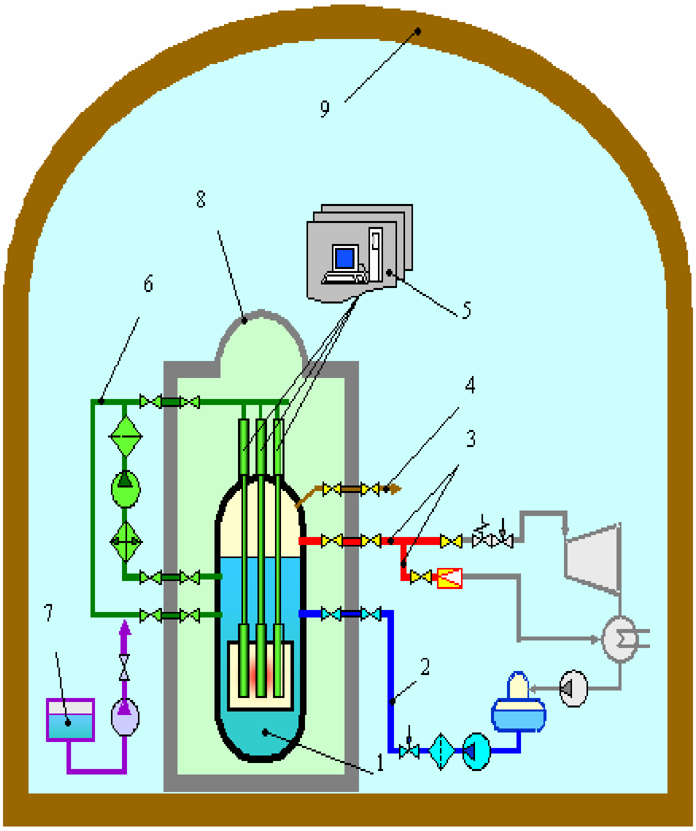
General philosophy governing plant layout

A schematic diagram of the general layout has been developed for siting of the two power units (the figure was not provided). The power units are placed at a distance of 130 m. The buildings and structures were located in the scheme with the following considerations:

- A general layout based on the independent operation of each power unit;
- Separation of the buildings based on the seismic resistance classification;
- Separation of the buildings and structures based on the safety classes;
- Separation of the buildings based on radiation monitoring;
- The reduction of costs and facilitation of construction;
- Requirements for space saving and easy operation; and
- The capability of adding more power units on the same site.

Reactor building and containment layout

The reactor and system components connected directly to the reactor are in the primary containment that acts as a leak-tight enclosing structure, Fig. VIII-12. The reactor system and the turbine unit are inside the secondary containment that protects against the external impacts.



1 - Reactor 2 - Feedwater supply system 3 - Steam removal system
4 - Gas supply and gas removal systems 5 - Computer aided IC system including the CPS
6 - System for cooling of the drive mechanisms of CPS, coolant purification and maintenance cooling
7 - Reactor filling and makeup system 8 - Primary containment 9 - Containment

FIG. VIII-12. Containment arrangement of VK-300.

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[VIII-1] OVCHINNIKOV, F.YA., VOZNESENSKY, V.A., SEMYONOV, V.V. Operation modes of NPP with the VVER-1000. M., Energoatomizdat, 1992 (in Russian).

[VIII-2] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of advanced light water reactor designs, IAEA-TECDOC-1391 (May 2004), Vienna.

COMPACT CONTAINMENT BOILING WATER REACTOR (CCR)

Toshiba Corporation, Japan

IX-1. General information, technical features and operating characteristics

IX-1.1. Introduction

The compact containment boiling water reactor (CCR) is a modular boiling water reactor (BWR) designed by the Toshiba Corporation with the support of the Japan Atomic Power Company (JAPC). The current CCR design falls into the category of innovative small and medium size reactors, featuring 300MW electrical output per module.

In Japan, increases in nuclear plant unit capacity have been promoted to take advantage of the economies of scale while further enhancing safety and reliability. As a result, more than 50 nuclear units are playing an important role in the domestic electric power generation. The next generation reactor with a 1700 MW(e) capacity is currently under development [IX-1, IX-2]. However, the future of nuclear power generation looks uncertain because of increasing competition with other power sources [IX-3] in the deregulated market, in spite of the general recognition that nuclear power is attractive from the viewpoint of energy security and environmental protection. Furthermore, factors such as stagnant growth in recent electricity demand, limitations in grid capacity and limited initial investment to avoid risk, will not favour large plant outputs. Nuclear plants are required that can easily be adopted in any country to globalize nuclear power generation for the mitigation of greenhouse effects.

In the 1980s, the Toshiba Corporation has carried out R&D for BWRs with natural circulation and passive safety features. These R&D included tests and analysis of passive containment cooling systems (PCCS), isolation condensers (IC) and gravity driven cooling systems (GDCS). The results obtained through these tests have been used in the design of a simplified boiling water reactor (SBWR). Based on these activities, the design of a simplified BWR with a long operating cycle (LSBWR) design has been under development since the mid 1990s [IX-4, IX-5, IX-6]. The concept of the LSBWR is to provide flexibility to meet site conditions and electricity demands, to mitigate investment risks and to facilitate public acceptance. To meet the output targets, the power range chosen for the LSBWR is from 100 to 300 MW(e). To overcome the disadvantages of economies of scale for small sized reactors, the following design approaches have been used for the LSBWR:

- Simplification of systems by combining direct cycle and natural circulation and by relying on passive safety means.
- Simplification of structure by integration of the reactor building and the turbine building.
- Elimination of the fuel pool and refueling machine (for a 15-year operation cycle).
- Using a modular design to ensure a short construction period.
- Adoption of a seismic isolation and a ship hull structure (marine type protective enclosure structure).
- Improvement in availability by implementing longer operating cycles (3 to 5 years).

The reactor concept presented herein (the CCR) takes a follow-up on the LSBWR. It has a small power output, the capability of long operating cycles and a simplified and compact BWR type configuration with comprehensive safety features. To be economically competitive, the CCR design includes simplification of systems and compact structure,

modular structures for short construction periods and improved availability. For comprehensive safety, the CCR targets to eliminate off-site emergency planning by using highly reliable equipment and systems, such as a reactor pressure vessel (RPV) with large inventory and the core being located at its bottom, the capability of in-vessel retention of a core melt (IVR) and a compact pressurized containment vessel (PCV) with high-integrity, incorporating several passive features.

The CCR concept is to provide economic flexibility for a variety of site conditions and electricity demands, to mitigate investment risks, and to facilitate public acceptance.

The principal stakeholders are:

- Toshiba Corporation.
- The Japan Atomic Power Company (JAPC).

IX-1.2. Applications

The CCR is designed for electricity generation but is also applicable to district heating and seawater desalination.

IX-1.3. Special features

The CCR is a land-based, modular nuclear power generating unit. When the market requires, more modules can be constructed in a series to form a larger nuclear power plant with the appropriate power output. The plant design is aimed at standardization and modularization.

IX-1.4. Summary of major design and operating characteristics

Some major design characteristics of the CCR are given in Table IX-1.

TABLE IX-1. MAJOR DESIGN CHARACTERISTICS OF CCR.

ATTRIBUTES	DESIGN PARTICULARS
Type of fuel	$^{235}UO_2$
Fuel enrichment	Below 5%
Types of coolant/moderator	Light water
Types of structural materials	Stainless steel for reactor internals
Core type/characteristics, dimensions Equivalent diameter × Active fuel length	Cylindrical, 3.5 m × 2.2 m
Vessel type/characteristics	Carbon steel reactor vessel, 5.6 m of inner diameter and 20.5 m of overall height
Cycle type	Direct cycle
Number of circuits	One circuit, nuclear boiler combined with steam turbine circuit.
Steam pressure	7 MPa
Mode of core heat removal	Natural circulation

Simplified schematic diagram

A vertical cross-section of the CCR reactor and a simplified schematic diagram of the CCR based nuclear power plant are shown in Figures IX-1 and IX- 2 respectively.

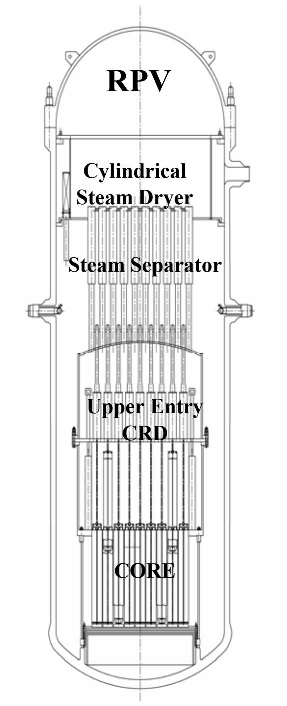

FIG. IX-1. Vertical cross-section of CCR (CRD is for control rod drive).

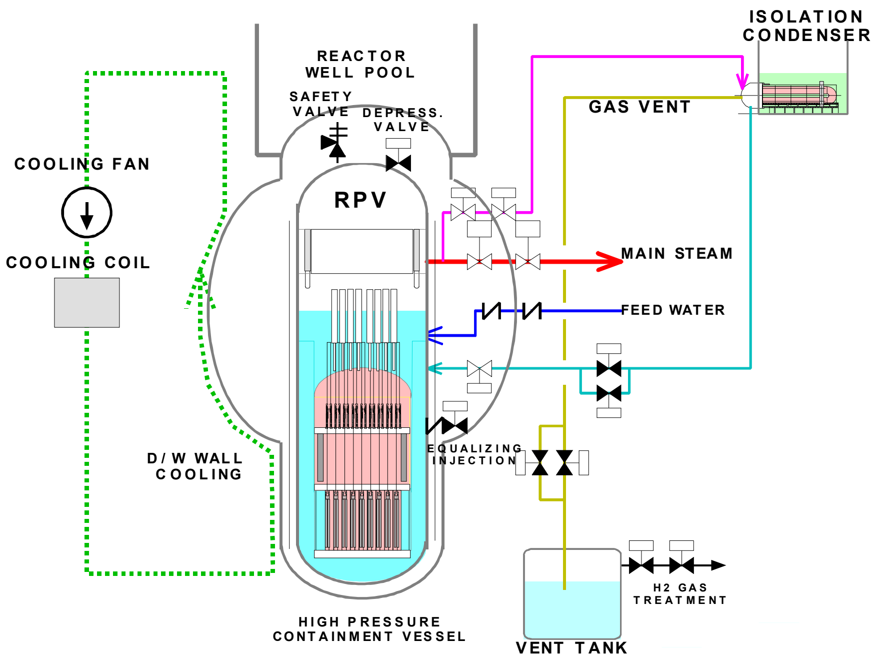

FIG. IX-2. Simplified schematic diagram of CCR (D/W is for dry well).

Installed capacity

- Reactor thermal power: 900 MW
- Generator power: 300 MW (Gross)

Mode of operation

The CCR plant is designed primarily for base load operation, but can also be operated in load follow-mode.

Load factor/ Availability

- Availability factor target: over 90%
- Load factor target: over 80%

For achieve design simplicity and a high level of safety, the CCR reactor system is designed with highly reliable equipment and systems such as:

- A large RPV inventory.
- A bottom core configuration.
- Passive core and primary containment vessel (PCV) cooling.

Core cooling through natural circulation results in a highly reliable operation and eliminates re-circulation pumps and loss of forced flow events, such as through pump seizure. To attain core cooling by natural circulation, the fuel length is shortened from the conventional 3.7 m to 2.2 m, resulting in a decreased pressure drop.

An innovative control rod drive (CRD) is being developed in the configuration of an internally mounted upper entry, above the reactor core, as shown in Fig. IX-1. Since the guide chimney, which has the function of a CR guide and a two-phase flow path above the core, separates the path of the control rod (CR) from the core flow, the CR operates without flow-induced vibration (FIV). The guide chimney also promotes natural circulation.

The internal upper entry CRD is motor-driven and uses electromagnetic coupling technologies; it consists of a CR driving motor, a latch mechanism for gravity driven scram, a position indicator and electromagnetic coupling. The electromagnetic coupling transfers signals and electric power between the outside and inside of the RPV without contact. The electric coils of the magnets are shielded by ceramics that can withstand high temperatures. The innovative internal upper entry CRD is being developed under a government sponsored programme [IX-7].

A low-pressure-loss type separator [IX-8] is mounted above the internal upper entry CRD as shown in Fig. IX-1. This new type separator, under development in a joint study by the Japanese BWR utilities and makers could decrease pressure losses by approximately 20% compared with the separator of a conventional type and effectively promotes natural circulation.

As shown in Fig. IX-1, the cylindrical dryer is located at the top of the RPV; it is designed to simplify the internals and ensure an easy fuel handling, i.e., without removal of the dryer.

An innovative design of the mentioned components provides the following features of the CCR:

- Natural core circulation and an innovative configuration of reactor internals results in a large water inventory above the core and a large safety margin against the loss of inventory;

- The RPV and PCV height is shortened by the internal upper entry CRD and the guide chimney above the reactor core;
- The CRD does not penetrate the steam dryer since the cylindrical type dryer and the CR cap are located under the normal reactor water level.

Neutron-physical characteristics

The CCR adopts lower power density and shorter core height to enhance natural circulation. The low power density results not only in design simplification but also in an increased plant availability by allowing longer refueling intervals. For example, a 2-year operation is achievable with the average discharge burn-up of 48GW·d/t under the limitation of a fuel enrichment of 5% (by weight).

To suppress initial reactivity of the core, gadolinium (Gd_2O_3) is used in the fuel pellets as a burnable poison. The enrichment distribution in the CCR bundle is optimized to flatten the local power. The CCR core has several control cells where the control rods are inserted during operation. To minimize neutron leakage from the core, fuel assemblies with high burn-up fuel are shuffled to the periphery of the core. These design aspects are similar to the approach used in current BWRs.

The void coefficient of the core has a negative value; the absolute value of the void coefficient of a 2-year cycle core is reduced by 30% compared to the current BWR cores.

Reactivity control mechanism

The CCR core has 69 cruciform control rods. The control rods contain boron carbide or hafnium as a neutron absorber.

The gadolinium for a 2–3 year operation is contained in the fuel pellets as a burnable poison. If a longer cycle is desired, isotope-enriched gadolinium is considered an option. For example, the applicability of isotope-enriched gadolinium to the 15-year operation cycle was studied for a long cycle variant of the simplified BWR (the LSBWR) [IX-9]. In a 15-year operation cycle, it is required to increase the enrichment by ^{235}U to about 18% (by weight) and to increase the control rod worth so that large initial reactivity could be suppressed. Control rods combined with smaller bundles (for example, a 30% reduced bundle) is an effective measure because control rod worth increases greatly in the bundle of a smaller design. For this reason, smaller bundles are considered as an option for the CCR with longer operation cycle, such as a 15-year cycle.

In conventional BWRs, the core reactivity is adjusted by the core flow and control rods. In the CCR, the core flow cannot be used for the adjustment of core reactivity because natural circulation is used. Therefore, only the control rods inserted from the top of the core adjust the core reactivity.

Cycle type and thermodynamic efficiency

The CCR is designed to operate in a direct cycle, i.e., to supply nearly dry saturated steam at a nominal pressure of 7 MPa directly from core to turbine, yielding a thermodynamic efficiency of 33%.

Thermal-hydraulic characteristics

The main thermal-hydraulic characteristics of the CCR are given in Table IX-2.

TABLE IX-2. THERMAL-HYDRAULIC CHARACTERISTICS OF CCR

CHARACTERISTIC	VALUE OR DESCRIPTION
Circulation type	Natural for normal operation as well as hot shutdown conditions
Coolant parameters	Core inlet: 551 K, 2722 kg/s; Core outlet: 560 K, average exit quality 17%
Steam and feedwater conditions	Steam at the outlet of steam drum: 7 MPa, 560 K, 486 kg/s Feedwater at the RPV inlet: 488 K
Fuel rod temperatures during normal operation	For maximum rated channel: fuel centre line: 1213 K; Clad surface: 572 K The maximum permissible clad temperature is 673 K.

Fuel lifetime/period between refuelings

The CCR core has 284 fuel assemblies. In the reference CCR core design, on average, 92 fuel assemblies will be replaced per cycle (assuming 24 months of effective full power operation), with an average residence time of a fuel assembly inside the core of approximately 6.2 years. The average discharge burn-up of fuel for the CCR is 48 GW d/t, with a maximum of 50 GW d/t.

If a longer operation cycle is desired, the 15-year operation without refueling is available within the 20% (weight) enrichment by ^{235}U-235 with smaller bundles and isotope-enriched gadolinium [IX-9].

Mass balances/flows of fuel materials

Assuming a 24-month operation cycle, after attaining equilibrium core conditions, the annual material requirements for fuel are given in Table IX-3.

TABLE IX-3. ESTIMATED ANNUAL FUEL REQUIREMENTS FOR CCR

MATERIAL	REQUIREMENT/MW(e) INSTALLED
Natural U	220 kg/yr
UO_2	26 kg/yr
Gd_2O_3	0.4kg/yr

Design basis lifetime for reactor core, vessel and structures

The design life of all non-replaceable structures (including RPV, pressure containment vessel, major piping and concrete structures) is 60 years. All components and equipment with a shorter design life will be easily replaceable during routine shutdowns.

Design and operating characteristics of systems for non-electric applications

The CCR coupled with a reverse osmosis (RO) plant could operate in a co-generation mode, producing 182 MW(e) of electric power and 102 000 m^3/day of potable water.

Economics

For the CCR, preliminary estimates indicate that its overnight capital cost per kW(e) would be equivalent to that of a large sized BWR, such as the ABWR. The estimated construction period from bedrock inspection to commercial operation is within 24 months.

The combination of a high pressure resistant containment and the upper entry CRD can facilitate the compact PCV without a suppression pool and the simplified safety systems with only an isolation condenser (IC) and an equalizing injection system. The adoption of simplified safety systems contributes to the reduction of O&M costs.

IX-1.5. Outline of fuel cycle options

A once-through fuel cycle without reprocessing is assumed as the standard fuel cycle.

A closed fuel cycle is the alternative fuel cycle option. Aqueous reprocessing could be adopted to reprocess spent fuel from the CCR. Dry reprocessing, which is considered in application to the reduced moderation core [IX-10], is also a candidate reprocessing option. An innovative fuel cycle system named BARS (BWR with an advanced recycle system) is proposed as a future fuel cycle option aimed at the enhanced utilization of uranium resources and reduction of radioactive wastes. In the BARS, the spent fuel from LWRs is recycled as a MOX fuel in BWR cores with the fast neutron spectrum, using dry reprocessing and vibro-packing for fabrication of the fuel.

IX-1.6. Technical features and technological approaches that are definitive for CCR performance in particular areas

IX-1.6.1. Economics and maintainability

The simple and easily understandable safety features and low capital cost of the CCR could facilitate its deployment in developing countries with limited financial and technological resources.

For the CCR, several preliminary evaluations of economic potential have been conducted.

For the reactor, the number of CRDs is decreased by approximately 30% by using a core with the bundle lattice 1.2 times broader than in conventional BWRs (the 1.2C lattice core). Applying an internal top-entry CRD eliminates CRD housings from the bottom of the RPV. A cylindrical dryer simplifies fuel handling during refueling. The numbers of components such as pumps, valves and tanks are decreased from those in a conventional BWR plant. Figure IX-3 presents a comparison between the number of major components in an ABWR and CCR. The numbers of major pumps, valves and heat exchangers (HEX) in the CCR are decreased to 50% from those of an ABWR. These reductions are attained by system simplification.

The reactor building volume is reduced remarkably by the compact PCV. The reactor building volume of a 300 MW(e) CCR is approximately 27% of that of a 1350 MW(e) ABWR. Assuming building volume proportional to power output, the building volume of a 300 MW(e) reactor becomes 22% of that in a 1350 MW(e) power plant, which is well matched by the CCR. As for the reactor building volume, economy of scale is almost overcome in the CCR.

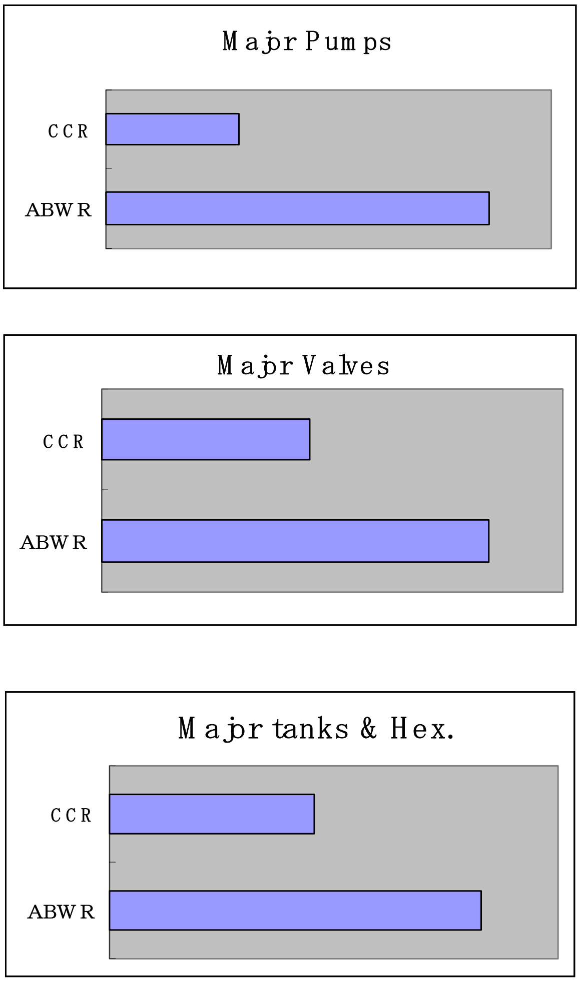

FIG. IX-3. Numbers of major components in CCR and ABWR.

Design simplification resulting from the combined use of a high pressure resistant containment and an upper entry CRD can facilitate the compact PCV without suppression pool and make it possible to reduce the safety systems to an isolation condenser (IC) and an equalizing injection system. Such simplified safety systems contribute to the reduction of O&M costs.

IX-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

If a reduced moderation core concept is applied to the CCR, the conversion ratio can be improved to about 1 [IX-9]. Coupled with an innovative BARS fuel cycle system (outlined before), the CCR could therefore enhance the utilization of uranium resources and reduce the specific amount of radioactive wastes. To achieve a fast neutron spectrum, a tight lattice fuel assembly could be adopted with a square channel box.

The reduced number of components in system design, Fig. IX-3, contributes to a decreased volume of wastes from operation and decommissioning.

IX-1.6.3. Safety and reliability

Safety concept and design philosophy

The CCR safety design philosophy is to increase the reliability of systems and to avoid design complexity, both with the use of passive safety systems, and through it to eliminate the need of emergency planning in the public domain.

Provisions for simplicity and robustness of the design

The incorporation of several passive safety features, natural convection core cooling, low power density in the core, compact PCV and large coolant inventory are some of the important provisions for the simplicity and robustness of the CCR design.

Active and passive systems and inherent safety features

The safety features of the CCR are summarized in Table IX-4, in comparison with those of the ABWR. The basic concept of the CCR safety systems is that the post-accident core cooling and decay heat removal functions are achieved by the passive safety systems. In the CCR, the top-mounted CRD and natural circulation core cooling are adopted and therefore, the coolant inventory per rated reactor power above the top of active fuel is much larger than that in an ABWR. In addition, the reactor pressure vessel (RPV) is surrounded by the high pressure compact containment vessel (PCV), which can prevent a large coolant inventory loss from the RPV in case of a pipe break accident: as pressure in the compact PCV rises up to several mega-Pascal due to the break flow, it is equalized with that in the RPV in the blow-down phase of a LOCA. These features of the CCR design, the large coolant inventory in the RPV and the high pressure compact PCV can eliminate the need for a coolant injection system, such as the HPCF (high pressure core flooder system) or LPFL (low pressure flooder system). The latter are used in the ABWR to replenish the coolant inventory of the RPV.

In the CCR, post-accidental core cooling and decay heat removal are achieved by the isolation condenser (IC), see Fig. IX-2. The IC is a passive system and consists of steam lines, steam condensers in the coolant pool located outside the PCV, drain lines and gas vent lines. The steam line routes the steam produced by decay heat in the RPV to the steam condensers of the IC. The steam is condensed in the steam condenser consisting of horizontal U-type tubes and the condensed water is returned to the RPV through the drain lines, driven by gravity. Non-condensable gases in the steam are vented, if necessary, to the gas vent tank through vent lines to mitigate the heat transfer deterioration effect of the non-condensable gases produced by the water radiolysis, or the metal-water reaction, and contained in the PCV. During normal plant operation, PCV wall cooling is achieved with the forced airflow over the outer surface of the PCV steel wall. Lost inventory from the RPV is stored in the compact PCV and can be returned by gravity to the RPV through flooder lines, which connect the RPV with the PCV when the water level in the PCV is high above the flooder line. The valve in the flooder line is opened by a LOCA signal.

Figure IX-4 shows the scheme of core cooling in LOCA conditions [IX-11]. After a feedwater line break (a), the pressure in the reactor and the PCV is equalized by releasing steam from the reactor to the PCV (b). At the same time, steam from the reactor is introduced to the isolation condenser (IC), and water is returned to the reactor after being condensed by cooling in the IC. On the other hand, the released steam is condensed and accumulated at the bottom of the PCV, the surface of which is being cooled. Therefore, the reactor water level is gradually decreased by heat release from the PCV surface. Before the reactor water level reaches the top of the active fuel, an equalizing valve is opened equalizing the water level between the RPV and the PCV (c). Finally, a long-term cooling gets settled in LOCA and continues until the system is returned to a normal condition

TABLE IX-4. SAFETY FEATURES AND SYSTEMS OF CCR*

EVENT		FUNCTION	SYSTEMS		RELATED EVENT
			CCR	ABWR	
Design Basis Accident		Core Cooling (maintain reactor coolant)	PCIV (PCV Boundary) IC DPV Flooder Line Valves	RCIC HPCF LPFL ADS	LOCA
		Heat Removal	IC	RHR	LOCA Steam System Piping Break Outside PCV
		Reactor Isolation	MSIV	MSIV	Steam System Piping Break Outside PCV
Severe Accident	Before Core Damage	Core Cooling at High Pressure (maintain reactor coolant)	PCIV (PCV Boundary) IC High Pressure Make-up Water Pump	RCIC HPCF ADS	High Pressure Sequence
		Core Cooling at Low Pressure (maintain reactor coolant)	PCIV (PCV Boundary)	HPCF RHR	Low Pressure Sequence
		Heat Removal	IC RHR	RHR	Loss of Heat Removal Sequence
		Emergency Power	AC (2 divisions) DC (4 divisions)	AC (3 divisions) DC (4 divisions)	Station blackout
	After Core Damage	In-Vessel Strategy	Flooder Line Valves	RPV Depressurization Alternative Coolant Injection	-
		Out-Vessel Strategy	IC	D/W Cooler Alternative Coolant Spray	-
ATWS		Reactor Shutdown	Diversity Scram Signal System	ARI	ATWS Sequence

* The acronyms used in this table are:

PCIV - Primary Containment Isolation Valve
IC - Isolation Condenser
DPV - Depressurization Valve
RCIC - Reactor Core Isolation Cooling System
MSIV - Main Steam Line Isolation Valve
ATWS - Anticipated Transient Without Scram
HPCF - High Pressure Core Flooder
LPFL - Low Pressure Flooder
ADS - Automatic Depressurization System
RHR - Residual Heat Removal (System)
D/W - Drywell
ARI - Alternate Rod Insertion

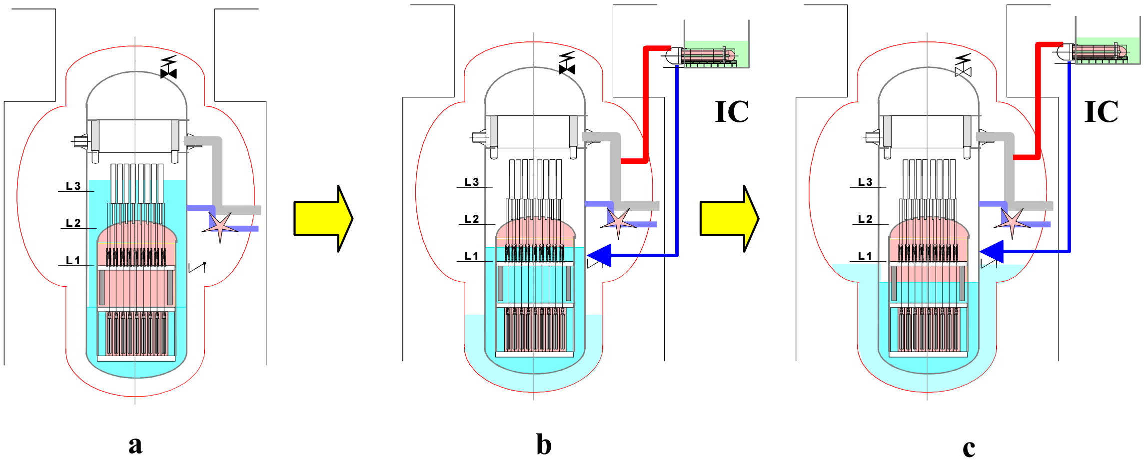

FIG. IX-4. Scheme of CCR core cooling in LOCA.

Structure of the defence-in-depth

Major highlights of the CCR safety design, structured in accordance with the various levels of defence in depth, are summarized below.

Level 1. Prevention of abnormal operation and failure

Since heat removal from the core under normal full power operation is performed by natural convection of the coolant, the hazard of a loss of coolant flow is essentially eliminated.

The extent of overpower and overpressure transients is reduced substantially. Features of the CCR design which help to achieve this objective are the following:

- Low core power density;
- Use of an interlock for control rod operation; and
- Large steam dome volume.

Level 2. Control of abnormal operation and detection of failure

Features of the CCR design which help to achieve this objective are the following:

- The increased reliability of control systems;
- Simplified requirements to operating personnel;
- Large coolant inventory in the RPV.

Level 3. Control of accidents within the design basis

The following features contribute towards achievement of this objective:

- The increased reliability of the ECCS, achieved through passive core cooling and decay heat removal by the IC;
- The elimination of a large coolant inventory loss from the RPV during a LOCA because of the application of the high design pressure compact PCV;
- A large coolant inventory in the RPV;
- The retention of the reactor water level above the top of the active core during a LOCA by actuation of the equalizing valve; and

- The retention of the reactor water level above the top of the active core during a LOCA by actuation of the equalizing valve; and
- The elimination of a reactivity initiated accident (RIA) induced by a CR drop or ejection because of application of the internal upper entry CRD.

Level 4. Control of severe plant conditions, including prevention of accident progression and mitigation of consequences of severe accidents

The following features contribute towards achievement of this objective:

- The multiple ICs;
- The retention of the reactor water level above the top of the active core during a LOCA; and
- The flooding of the reactor cavity by the discharged coolant following a LOCA.

Level 5. Mitigation of radiological consequences of significant release of radioactive materials

The following features help in passively decreasing the containment pressure and minimizing the releases from the containment following a large break LOCA:

- Passive core cooling and decay heat removal;
- High design pressure compact PCV.

Design basis accidents and beyond design basis accidents

The performance of the CCR safety system following a LOCA accident has been evaluated, and the feasibility of the concept was examined.

Major analytic conditions, which have been determined based on the preliminary design of the CCR, are listed in Table IX-5. The safety system performance has been evaluated for a feedwater line break accident without active residual heat removal (RHR). The IC actuates with a low-water level (Level-2) signal that indicates the collapsed downcomer water level is 5.9 m above the top of the active fuel elevation.

TABLE IX-5. MAJOR ANALYTIC CONDITIONS FOR LOCA ANALYSIS

ITEM	CONDITION
Rated thermal power	900 MW (300MW(e))
PCV volume	793 m^3
IC heat transfer area	100 m^2
Accident	Feedwater line guillotine break (break area = 0.02 m^2)

The analysis has been conducted using a transient reactor analysis code (TRAC) coupled with the heat transfer models for steam condensation with the presence of a non-condensable gas in the IC condenser tube. The non-condensable gas, which initially fills the PCV space, will be absorbed into the IC condenser tubes through the broken feedwater pipe.

The calculation results for the reactor and containment pressure responses to the feedwater line break are shown in Fig. IX-5. Figure IX-6 shows the transient of the water levels in both the RPV and PCV. Following the feedwater line break, the containment pressure starts to

increase, being boosted by the break flow from the RPV. After about 20 seconds, the collapsed water level in the RPV reaches the Level-2 and consequently the IC is actuated with the low level signal. The heat removal with the IC accelerates the depressurization of the RPV since the heat removal rate exceeds the heat generation rate in the reactor core, as shown in Fig. IX-7. The results obtained indicate that the IC sizing can be reduced. The pressure in the RPV and the containment space match at approximately 4 MPa in about 130 seconds since the initiating event and then keep decreasing due to the heat removal with the broken feedwater line. The break flow becomes almost zero after 130 seconds (as shown in Fig. IX-8) since the pressures are equalized.

The two-phase level in the RPV shows a temporary increase after about 20 seconds due to the flashing caused by the depressurization in the vessel, as illustrated in Fig. IX-6. After the pressures are equalized at around 130 seconds, the two-phase level transient starts gradually decreasing due to the steam collapse with the sub-cooled condensate flow drained from the IC. However, it is kept at the elevation of the broken line since the steam generated in the reactor vessel and the containment now becomes identical. The two-phase level in the RPV is, therefore, kept well above the elevation of the top of active fuel and there is no heat-up in the reactor core.

In summary, the isolation condenser has the heat removal capability sufficient to reduce the pressure in the containment, and the reactor core remains covered with the coolant all the time following a feedwater line break.

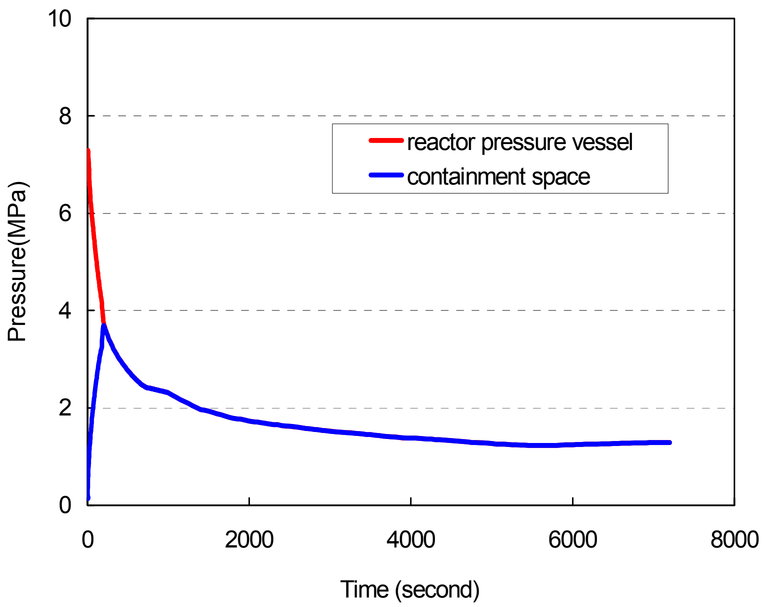

FIG. IX-5. Pressure response to a feedwater line break.

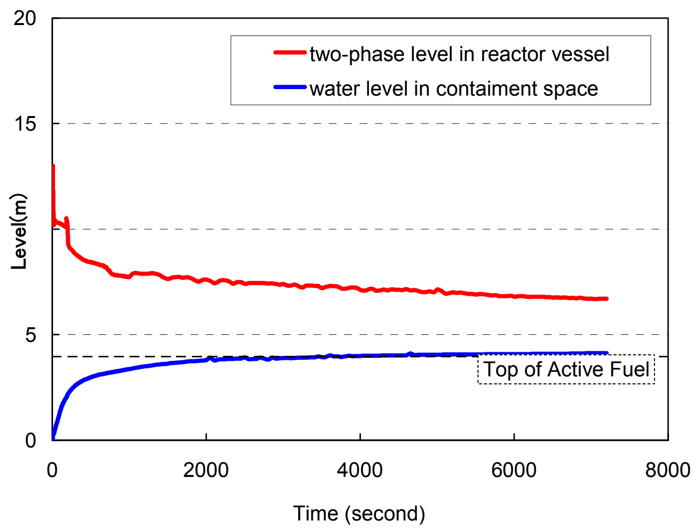

FIG. IX-6. Water level transient.

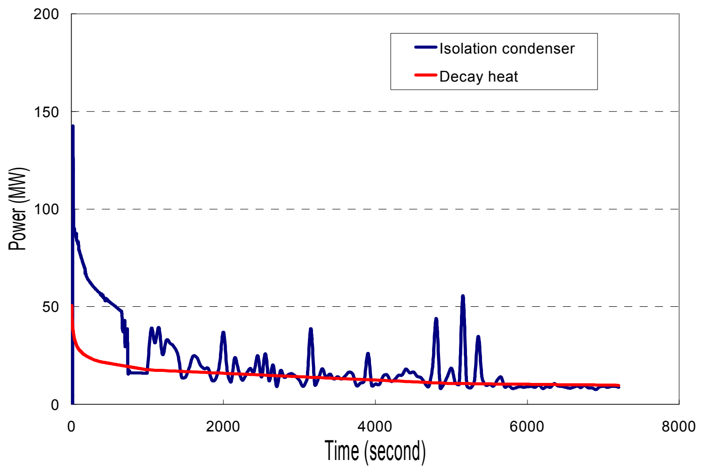

FIG. IX-7. Heat removal rate of isolation condenser.

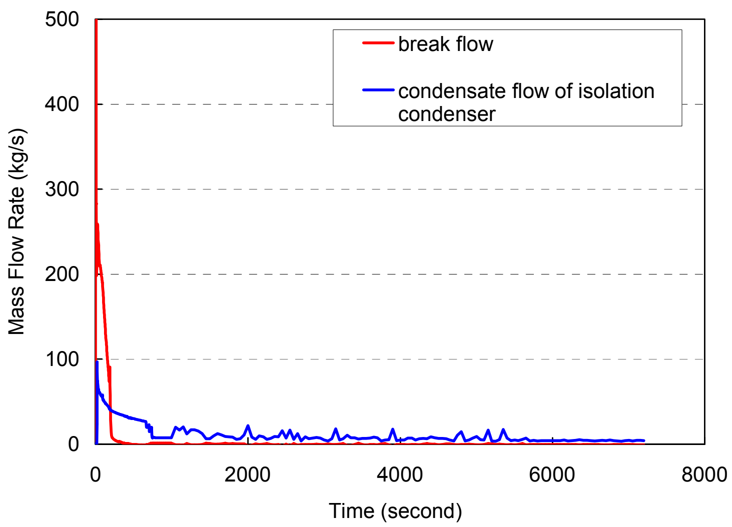

FIG. IX-8. Condensate flow of IC.

Provisions for safety under seismic conditions

The CCR structures, systems and equipment will be designed for high level and low probability seismic events such as an operating basis earthquake and a safe shutdown earthquake, also known as S_1 and S_2 level earthquakes respectively. Seismic instrumentation will be also provided in accordance with the national and international standards.

Probability of unacceptable radioactivity release beyond the plant boundaries

It is expected that the probability of an unacceptable radioactivity release beyond the site boundary for the CCR will be less than that for conventional BWRs.

Measures planned in response to severe accidents

For severe accident conditions, the CCR safety system will be operated according to procedures similar to those used in case of a LOCA.

Hydrogen produced by the metal-water reaction during core melting is separated from steam at the vent tank connected to the IC. Hydrogen will be converted to water (H_2O) and ammonia (NH_3) by the advanced passive autocatalytic recombiner (A-PAR) [IX-12], using special catalysts. Melted core in the RPV will be cooled from outside the RPV by water of the PCV bottom, which is condensed on the PCV inner wall. The target of the CCR design is to attain In-vessel retention (IVR) capability with the use of only a simple passive safety system. Should such capability be confirmed in further design and validations, the CCR would target reduced or eliminated off-site emergency planning.

IX-1.6.4. Proliferation resistance

No analysis has been performed so far.

IX-1.6.5. Technical features and technological approaches used to facilitate physical protection of CCR

No analysis has been performed so far.

IX-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of CCR

The CCR design takes into consideration market demands for smaller power output of 300 MW(e) or about 100 MW(e) with an increased refueling interval; the analysis of corresponding demands is being performed also.

IX-1.8. List of enabling technologies relevant to CCR and status of their development

A list of enabling technologies relevant to the CCR is given in Table IX-6.

TABLE IX-6. ENABLING TECHNOLOGIES RELEVANT TO CCR AND THEIR DEVELOPMENT STATUS

SPECIFIC OBJECTIVE	ENABLING TECHNOLOGY	STATUS OF DEVELOPMENT
Reduction of the number of fuel assemblies	Wide lattice pitch	Design study in progress.
Elimination of re-circulation pumps by using natural convection for core cooling	Short length core	Design study in progress.
	Low pressure drop steam separator	R&D in progress.
	Divided chimney	Design study in progress.
Compact primary containment vessel (PCV)	Upper entry internal CRD	R&D in progress under a Government sponsored programme.
	High pressure resistant PCV	Basic design study has been completed.
	Electric cable penetration	R&D in progress.
Simplified passive safety systems	Compact high pressure resistant PCV	As mentioned above.
	Isolation condensers	Design study in progress.
	In-Vessel Retention (IVR) by passive means	Evaluation in progress.
Short construction period	Module construction methods	Design study in progress [IX-13].
	Steel-concrete structure	Design and evaluation study has been completed.
	Ship hull structure	Design and R&D have been completed [IX-14].

IX-1.9. Status of R&D and planned schedule

The Toshiba Corporation primarily performs design development for the CCR.

The Japan Atomic Power Company (JAPC) supports the R&D for the CCR. The internal upper entry CRD development is supported under the “Innovative and Viable Nuclear Energy Technology Development Project” founded by the Agency of Natural Resources and Energy and the Ministry of Economy, Trade and Industry (METI).

The conceptual design of the reactor and major nuclear systems has been completed. It is expected that by mid-2010 the design will be sufficiently complete to enable initiation of construction related actions, subject to the availability of funds and regulatory and other statutory clearances.

IX-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

The CCR can be rated as an innovative reactor by the following features:

- It has a simplified and compact safety system with a high pressure resistant PCV that makes it possible to eliminate emergency core cooling systems.
- It has an internal upper entry CRD, to be used in a BWR for the first time.

IX-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

No similar SMRs are under development elsewhere.

IX-2. Design description and data for CCR

IX-2.1. Description of the nuclear systems

Reactor core and fuel design

The overriding design target for the CCR is realization of a comprehensive economic performance through system simplification. A reduction in the number of bundles (fuel assemblies) and control rods will be the key issue in core design, because the CCR adopts lower power density and a shorter core height to enhance natural circulation. The simplest way of decreasing the number of bundles is to increase the power density. However, increased power density requires greater vessel height. Therefore, the bundle pitch is expanded to be 1.2 times wider than in the conventional BWR; this wider lattice is called a 1.2C lattice. The letter C means that the control rod is allocated to the “Centre” of four bundles and the width of the bundle is also expanded. The core map and a candidate fuel design based on the 1.2C lattice bundle are shown in Fig. IX-9. The use of 1.2C lattice decreases the number of fuel bundles and control rods by 30%.

The core has 284 fuel bundles and 69 control rods; the fuel rod array of the 1.2C lattice bundle is 12×12.

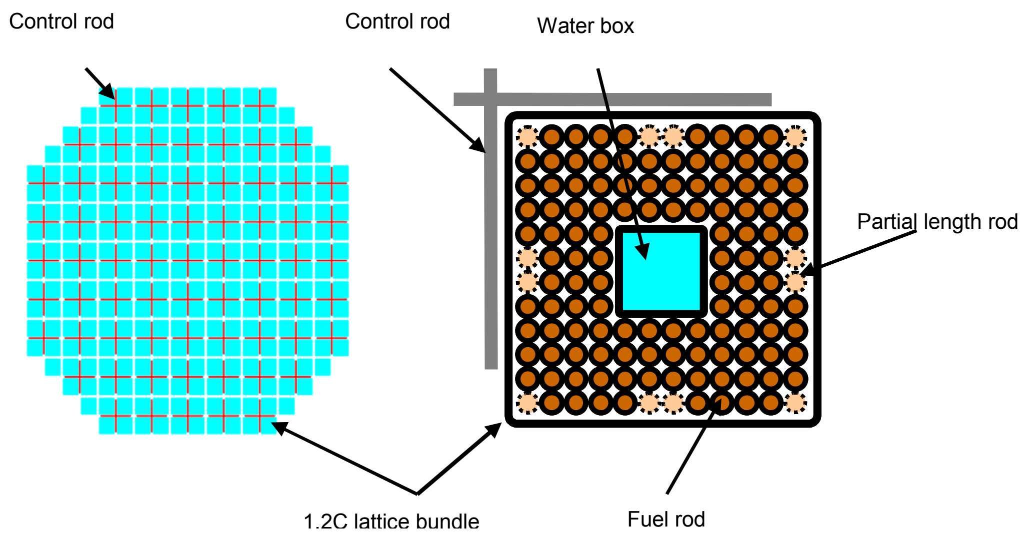

FIG. IX-9. Core map of CCR (left) *and 1.2C lattice bundle (right).*

Main heat transport system

The CCR is a direct cycle reactor, with its primary coolant system being also the main heat ransport system (MHT). The function of the MHT is to remove nuclear heat from the reactor core through natural circulation in both operating and shut down conditions. The design objectives are to achieve a Minimum Critical Heat Flux Ratio (MCHFR) of at least 1.5 at 120% full power and to ensure thermal-hydraulic stability during all operating conditions.

The MHT system transports heat from the fuel rods to the light-water coolant. The two-phase steam-water mixture produced in the reactor core, enters the steam dryer through the steam separator at the upper plenum, see Fig. IX-2. Steam separated from the steam-water mixture in the steam separator and dryer enters the steam dome. Two pipes collect steam from the steam dome and transfer it to the steam chest of the turbine.

During normal operation or in the bypass mode of operation, steam from the turbine exhaust is condensed in a condenser, which rejects the heat to seawater. The condensate is heated in heat exchangers by a steam extraction system. The feedwater temperature is finally raised to 488 K through LP (low pressure) and HP (high pressure) heaters, and the feedwater pumps direct the feedwater into the RPV where it mixes with water separated from the steam separator and dryer. The heat removal paths of the CCR under various operating states and in LOCA are schematically shown in Fig. IX-10.

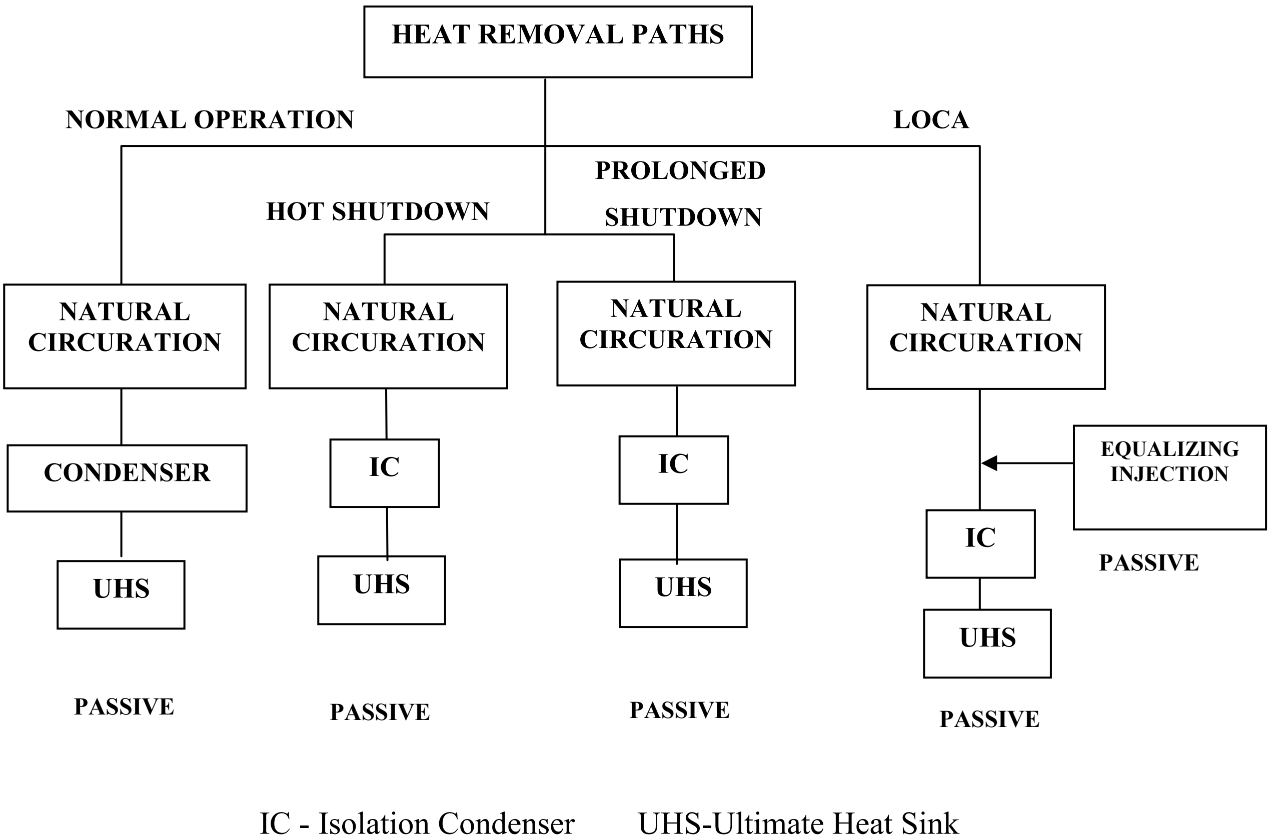

FIG. IX-10. Heat removal paths of CCR.

IX-2.2. Description of the turbine generator plant and systems

Conventional equipment for the BWR will be used in the turbine generator system.

IX-2.3. Systems for non-electric applications

A concept of coupling the CCR with a process for potable water production has been analyzed. In this concept, the balance of plant (BOP) consists of a turbine system generating 182 MW(e) and a seawater desalination system producing about $100\times10^3 m^3/d$ of potable water, Fig. IX-11. The design of the desalination system is based on existing technologies.

The turbine system employs a regenerative steam cycle consisting of the two stages of high pressure feedwater heating and three stages of low pressure feedwater heating. Steam generated in the reactor works in the high and low-pressure stages of the turbine to produce 182 MW of electricity. Steam condenses in the condenser and, to improve the thermal efficiency, the resultant water is heated by the feedwater heaters before it reaches the core.

Part of the steam (30 kg/s) is taken off after the high pressure stage of the turbine and used to drive two pumps, called turbine-driven (TD) pumps. The TD pumps compress seawater up to the osmotic pressure (about 7 MPa) required for the reverse osmosis (RO) process. The RO units produce about $100\times10^3 m^3/d$ of potable water assuming the energy recovery ratio of 40%, see Fig. IX-12.

Instead of backup boilers often accommodated in the distillation type seawater desalination systems, the RO system of the CCR is equipped with a motor-driven pump (MD pump) for backup. Because the MD pump is powered by external sources, backup boilers together with the associated systems become unnecessary.

The RO system, which includes TD pumps as an interface, has advantages in the efficiency, economics and safety over conventional distillation systems for seawater desalination. This RO system produces about 100×10^3 m^3/d of potable water, while the distillation system could produce only up to 80×10^3 m^3/d if the same amount of steam is used in the process. Only the MD pump is added for backup in a RO based seawater desalination system, while backup boilers, together with associated systems including fuel tanks, are necessary for a distillation system. Because the possibility of radioactive contamination of seawater in the RO system of the CCR is eliminated by design, no extra barrier would be necessary for this RO system. Different from this, the distillation system needs extra barriers (e.g. an extra heat exchanger) to reduce the possibility of contamination because only a thin wall in the heat exchangers separates a BWR steam and seawater. Therefore, the RO system was favoured as a nuclear seawater desalination system for the CCR over the distillation system.

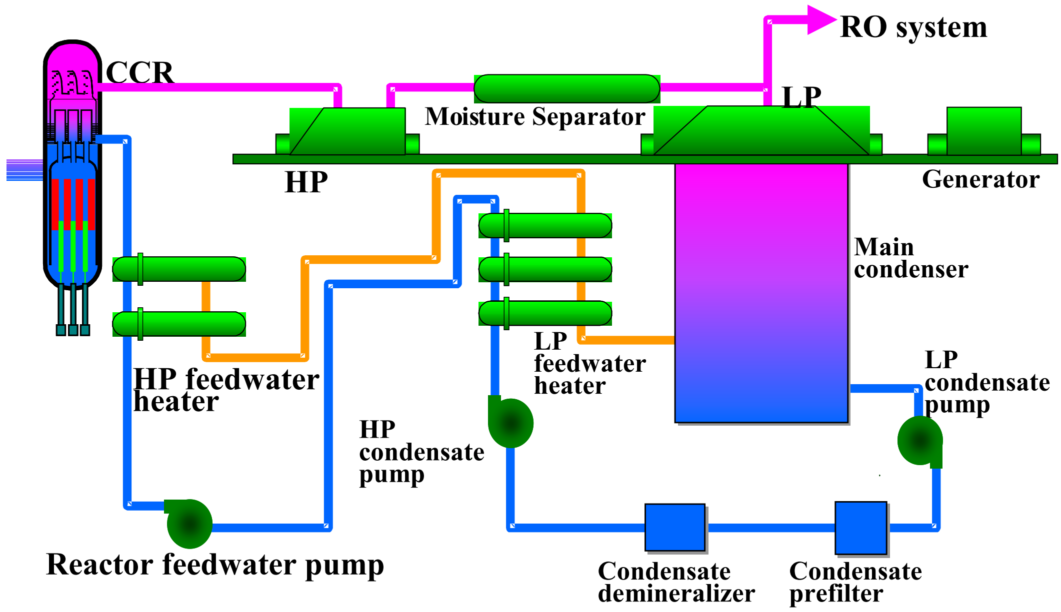

FIG. X-11. Balance of plant for CCR operation in a co-generation mode with potable water production.

Electric power
182 MW

Water production
102×10^3 m^3/d
(MSF*: 80×10^3 m^3/d)

RO high pressure (HP) pump
TD×2 (50%×2)
17 MW/unit

Preheating option (dashed line)

*MSF is for multi-stage flash desalination

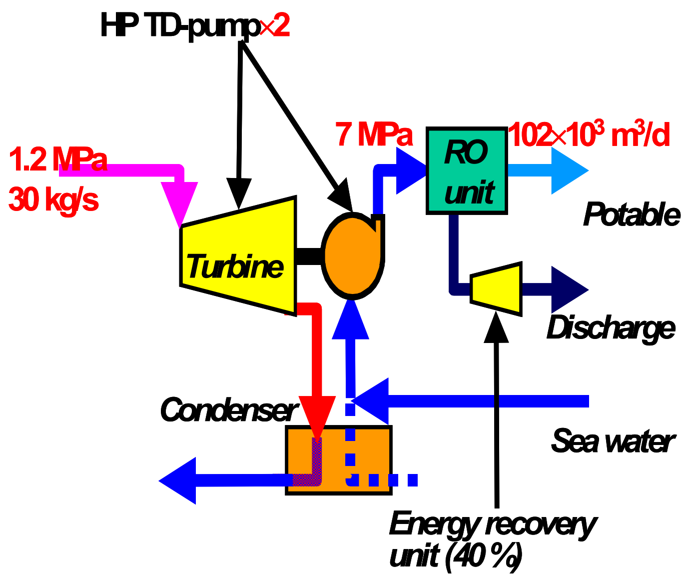

FIG. IX-12. Reverse osmosis (RO) desalination system for CCR.

The characteristics of the CCR based desalination plant are summarized in Table IX-7.

TABLE IX-7. SUMMARY CHARACTERISTICS OF CCR BASED SEAWATER DESALINATION PLANT

CHARACTERISTIC	VALUE
Rated thermal power	589 MW
Rated electric power	182 MW
Water production rate	102×10^3 m^3/d
Steam cycle	
Turbine type	TC4F23"
Stages of feedwater heating	HP×2, LP×3
Desalination	
Process	RO
HP pumps	TD×2 (50%×2)
Capacity	17 MW/unit
Backup	MD×1 (50%)
Pre-heating by condenser	Option

IX-2.4. Plant layout

General philosophy governing plant layout

The basic philosophy for the CCR plant layout is similar to that of the current Japanese BWR plants, including the provisions for maintainability and accessibility, etc.

Additionally, the reactor building volume is reduced remarkably by the use of the compact PCV and simplified reactor auxiliary systems.

The main control room is integrated with the reactor building because it is in the same seismic category as the reactor systems.

Reactor building and containment layout

The pressure suppression system is one of the most effective systems through which both the design pressure and volume of the primary containment vessel (PCV) could be decreased. In the current BWR designs, the use of this system facilitates the PCV achieving compact size and low design pressure. For conventional BWRs, the design pressure of a PCV is approximately 0.3 to 0.4 MPa.

For a dry containment, the PCV design pressure is defined by PCV volume. It is possible to decrease PCV volume in cases where the design pressure of the PCV is increased but hard to design a high pressure resistant PCV because of the dimensions, especially for large reactors.

For small reactors, it is possible to increase PCV design pressure and make it the same as reactor pressure, without exceeding the requirements of practical fabricating size and integrity [IX-15].

The high pressure resistant PCV combined with a rational cooling system are basic features of the CCR design. One of the functions of the PCV is control of the coolant outflow from the RPV, achieved by the equalization of pressure between the PCV and RPV during a LOCA. Therefore, the PCV design pressure is set to be about 7 MPa, which is equivalent to the RPV

design pressure, and both are about 20 times the PCV design pressure of current BWR plants. Moreover, to prevent uncovering the reactor core in accidents, it is required that the cooling water which blows out from the RPV into the gap between the RPV and PCV is returned to the RPV through an equalization irrigation line. For this reason, the PCV volume below the core should be kept as small as possible. On the other hand, the upper portion of the PCV should be planned to provide a reasonable space to accommodate piping arrangement connected to the RPV. And, as a matter of course, it is necessary to consider compact dimensions from the viewpoint of economy.

From these considerations, the PCV configuration of the CCR is designed to have a cylindrical shape for the bottom and a spherical shape for the upper part. The PCV configuration is shown in Fig. IX-13. Executing the preliminary structural design with carbon steel as PCV material, the thickness of the PCV shell was defined as about 230 mm for the cylindrical part and 200 mm for the spherical part.

The arrangement concept for the main steam line (MS) and a feedwater line (FDW) is shown in Fig. IX-14, with these two lines representing a typical piping inside the PCV of a boiling water reactor. The MS header, which is the largest piping in the PCV, is arranged at the centred height of a spherical shell and connected to the RPV nozzles by two piping risers to mitigate the thermal stress of the MS piping. The FDW piping is arranged below the MS header so that the loop for thermal stress relief of the FDW can be secured. The preliminary design study [IX-16] has confirmed rationality of the selected approach to provide a compact design of the PCV for the CCR.

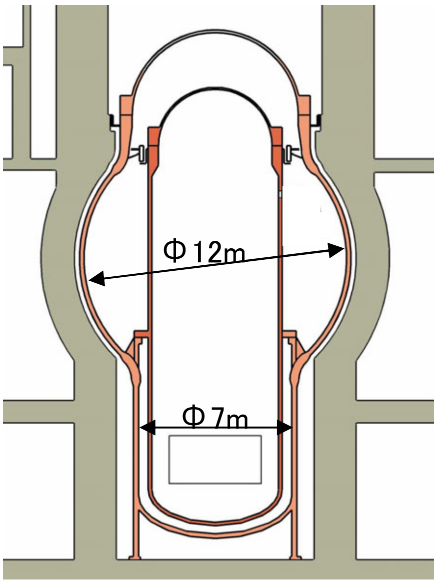

FIG. IX-13. PCV configuration of CCR.

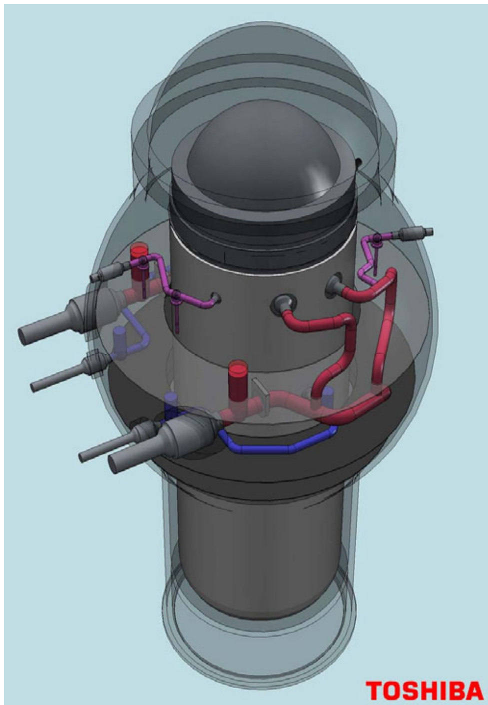

FIG. IX-14. Concept of piping arrangement inside PCV.

Plant layout

Figure IX-15 shows a cross-section view of the reactor building arrangement.

Major considerations governing the reactor building and containment layout are the following:

- Minimization of the primary containment volume;
- Effective utilization of space in the annulus between the primary and secondary containments;
- Unrestricted entry to the reactor top for on-power fuel handling and transfer operations;
- Adequate shielding against radiation and preventing the release of radioactive contamination during normal and accidental conditions;
- Provision of a large water inventory at a suitable height, capable of supporting a number of passive systems;
- Submergence of the reactor core under water before exhaustion of the inventory of the emergency core cooling system;
- Easy access to the maximum of the equipment for operation and maintenance during normal and accidental situations; and
- Fire prevention and control.

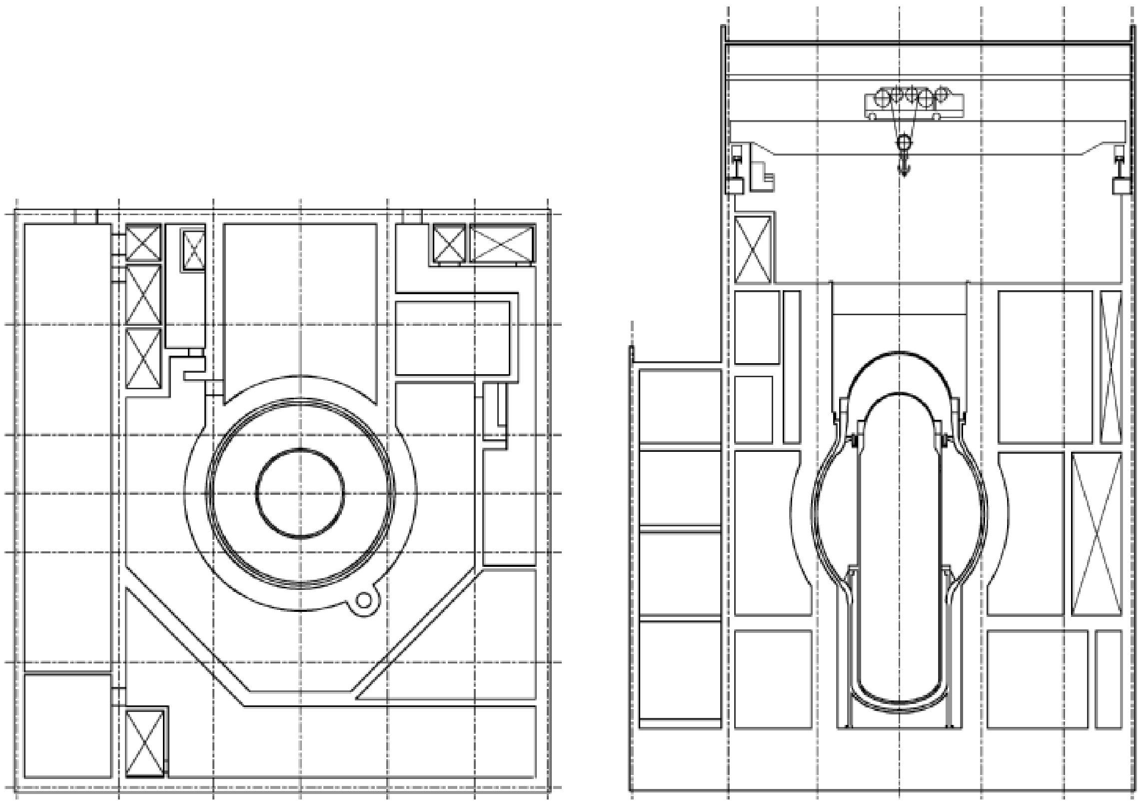

FIG. IX-15. Arrangement plan of CCR reactor building.

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ANNEX X

REDUCED MODERATION WATER REACTOR OF 300 MW(e) (RMWR)

HITACHI LTD, JAERI, JAPC, TITech, Japan

X-1. General information, technical features and operating characteristics

X-1.1. Introduction

An advanced water-cooled reactor, namely the Reduced Moderation Water Reactor (RMWR) [X-1, X-2], aims to achieve a high conversion ratio, over 1.0, with mixed oxide (MOX) fuel, based on proven boiling water reactor (BWR) technology. High conversion ratio is attained by reducing the moderation of neutrons, i.e. reducing the water fraction in the core.

The RMWR presented herein is a 300 MW(e) small sized reactor concept with passive safety features. For small reactors, what is called the scale demerit should be overcome. In this concept, simplifying the plant systems and introducing passive safety features overcome the disadvantages of small scale. One of the major passive safety features is natural circulation of the primary coolant but several passive safety components are also intended for utilization to reduce costs.

The previous designs of 1000 MW(e) class RMWRs have been developed at the Japan Atomic Energy Research Institute (JAERI) in cooperation with the Japan Atomic Power Company (JAPC) and with technical support from the Japanese light water reactor (LWR) vendors. They are described in detail in [X-3].

To develop a 300 MW(e) small reactor concept, there has been collaboration among JAERI, JAPC, Hitachi Ltd. and Tokyo Institute of Technology (TITech) with government funding from the innovative and viable nuclear energy technology (IVNET) development project.

X-1.2. Applications

The RMWR is designed to generate 330 MW (gross) of electric power.

X-1.3. Special features

The RMWR is a land-based nuclear power station. The reactor could achieve a high conversion ratio, over 1.0, using plutonium mixed oxide (MOX) fuel.

X-1.4. Summary of major design and operating characteristics

Major characteristics for a conceptual design of the RMWR are given in Table X-1.

Simplified schematic diagram of RMWR

The schematic overview of the plant system concept is shown in Fig. X-1. In RMWR, natural circulation core cooling is introduced and hence, circulation pumps and the related power supply can be eliminated. This results in a simplified and economic core cooling system. In this system, gravitational steam/water separation is expected due to the low steam velocity from the core and, hence, the steam/water separator and the steam dryer can also be eliminated.

TABLE X-1. MAJOR DESIGN CHARACTERISTICS OF RMWR

ITEMS	SPECIFICATIONS
Reactor type	Integral tank type; modular
Electricity output	330 MW(e)
Thermal output	955 MW(th)
Mode of operation	Base load / load-follow
Plant efficiency, gross	34.5%
Availability factor	96% (target)
Fuel type	(Pu–U)O_2 MOX
Average fissile Pu enrichment	18 weight %
Total Pu enrichment	Approximately 30 weight %
Coolant, moderator	Boiling light water
Structural materials	Zircaloy 4, stainless steel
Core configuration	BWR-type double flat core
Core diameter (circumscribed circle)	4.14 m
Active core height	0.88 m plus upper and lower blankets of 0.20 m and 0.24 m, respectively
RPV outer diameter	6.3 m
RPV height	17.4 m
Cycle type	Direct, similar to Advanced Boiling Water Reactor (ABWR)
Decay heat removal system	1 Accumulator injection (passive); and 2 Flooding systems (active) long-term
Emergency core cooling system	Passive system (accumulators) and flooding system
Containment system	Steel containment vessel

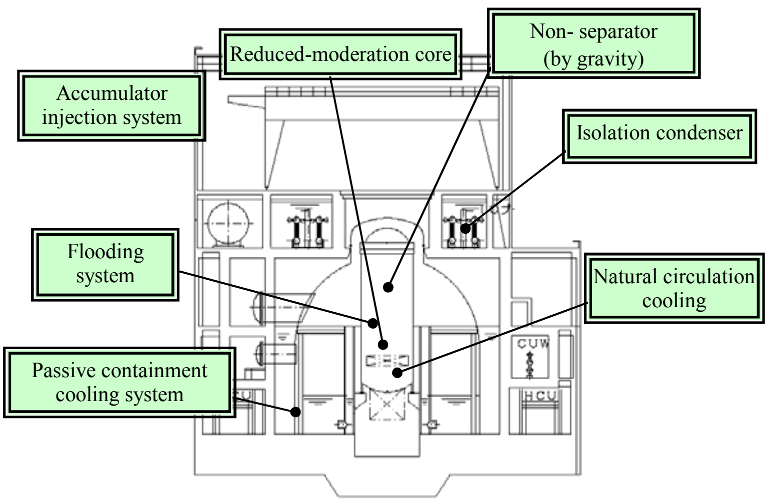


FIG. X-1. RMWR plant system concept.

A more detailed scheme of the nuclear steam supply system (NSSS) including all circuits and turbine plant is shown in Fig. X-2.

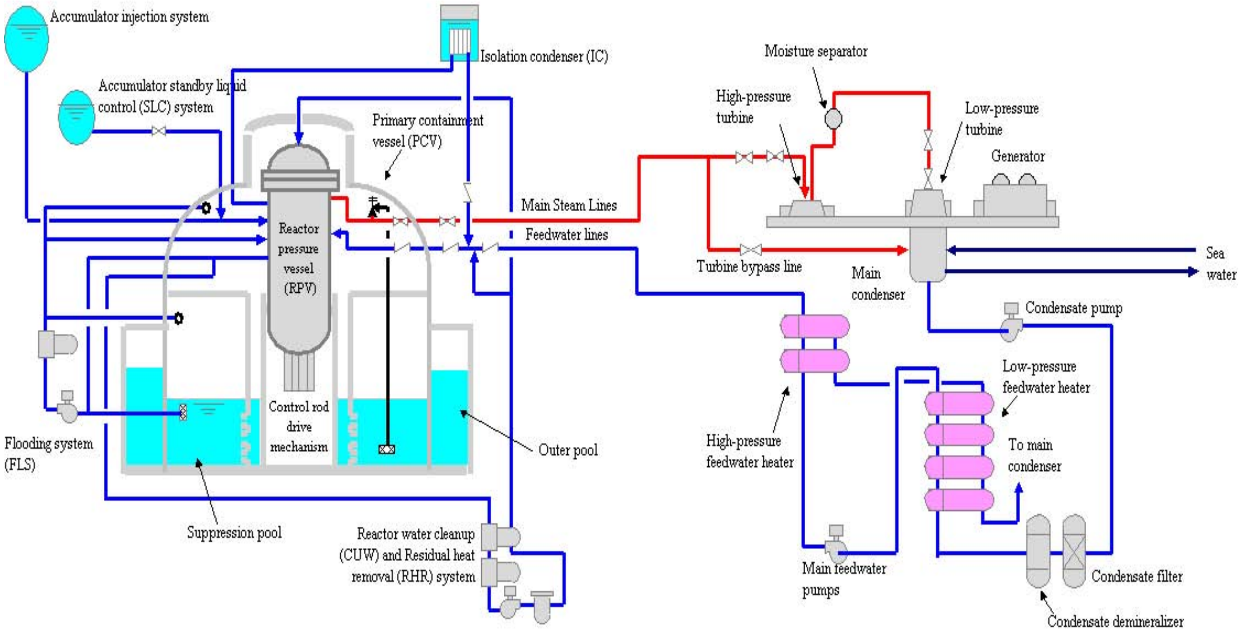


FIG. X-2. Simplified schematic diagram of RMWR plant.

Neutron-physical characteristics

The design targets for the RMWR core were set as follows:

Conversion ratio:	More than 1.0
Void reactivity coefficient:	Negative value
Core average burn-up	65 GW d/t (50 GW d/t including upper and lower blankets)
Operation cycle length	25 months

The first two targets are basics required for the RMWR, the last two are added in consideration of economic factors.

Main neutron-physical characteristics of the RMWR are given in Table X-2.

TABLE X-2. NEUTRON-PHYSICAL CHARACTERISTICS OF RMWR

CHARACTERISTIC	VALUE
Fissile Pu conversion ratio	1.03
Average / maximum fuel burn-up	65 000 / 100 000 MW·d/t U
Void reactivity coefficient	–0.5 × (10^{-4} Δk/k)/(% void)
Burn-up reactivity swing	~1% Δk/k

Reactivity control mechanism

The control rods of the Y-shaped design contain B_4C with highly enriched ^{10}B; the graphite follower structure is intended to remove the water (i.e. the moderator) to attain a high conversion ratio. The control rods with the follower are inserted from the bottom of the core as in current BWRs. The total number of control rods is 85 and they are distributed at a ratio of approximately one for every three fuel assemblies. The cold shutdown margin during the cycle is more than 1% as in current BWRs. Detailed analyses of the control rod operation have been performed and 19 control rods out of 85 are assigned to the operation during the cycle.

The second shutdown system is a standby liquid control (SLC) system.

Thermal-hydraulic characteristics

The core average void fraction is designed to be very high, 70% to achieve the high conversion ratio of more than 1.0. To attain the high core void fraction, the core water flow rate is reduced to 4300 t/h, which is about one third of the corresponding scaled-down value of the ABWR. Due to this lower core water flow rate, the evaluated pressure drop across the fuel assembly is very small, 0.04 MPa, in comparison with the value of 0.18 MPa for current ABWRs. Therefore, cooling through natural circulation is possible for RMWR.

The main thermal-hydraulic characteristics of RMWR are given in Table X-3.

TABLE X-3. THERMAL-HYDRAULIC CHARACTERISTICS OF RMWR

PARAMETER	VALUE
Circulation type	Natural circulation for normal operating as well as hot shut-down conditions
Coolant conditions	Core inlet: 556 K, 1200 kg/s, 7.2 MPa; Core outlet: 561 K, average exit quality 53%; saturated water, self-pressurization
Core average power density	54 W/cm^3
Maximum linear power density	39 W/cm
Core average void fraction	70%
Core pressure drop	0.04 MPa
Minimum critical power ratio (MCPR)	1.3
Steam and feedwater conditions	Steam at RPV outlet: 7.1 MPa, 560 K, 520 kg/s; Feedwater at RPV inlet: 549 K; 2 steam lines.
Fuel temperatures during normal operation	For maximum rated channel: - Fuel centre line: 2403 K, - Clad surface: 582 K (with Baron’s thermal conductivity model for high burn-up MOX) The melting point of MOX fuel is 2790 K.

The minimum critical power ratio (MCPR) at full power operation of the reactor is estimated as 1.3 by the modified CISE correlation [X-4].

Fuel lifetime/period between refuellings

The fuel lifetime is about 3800 effective full power days (EFPD) and the period between refuellings in is about 780 EFPD (about 25 months).

Mass balances/flows of fuel materials

With a closed fuel cycle and the breeding ratio slightly exceeding 1.0, depleted or natural uranium becomes the essential fuel. The estimated mass flow of depleted uranium is 5.4 t/GW(th)·year under an 85% capacity factor, based on detailed core calculation.

Design basis lifetime for reactor core, vessel and structures

The design life of all non-replaceable structures (including the RPV and major piping and concrete structures) is 60 years. All components and equipment with a lower design life will be easily replaceable during routine shutdowns.

Economics

For the system design, the basic approach to reduce the plant cost is simplification by introducing the passive safety features, in addition to the natural circulation core-cooling concept. For example, the high-pressure coolant injection system using pumps is replaced by the passive accumulator system and the emergency diesel generators can be removed, resulting in an effective cost reduction. The cost evaluation of the nuclear steam supply system produces about a 20% cost reduction against a prototype with no passive safety features mentioned above.

X-1.5. Outline of fuel cycle options

The fuel cycle concept of the RMWR is basically a closed cycle and is the same as for FBRs (sodium cooled fast breeder reactors). It has been confirmed that the high conversion ratio, more than 1.0 and the negative void reactivity coefficient can be achieved in the RMWR core under the multiple recycling of Pu including advanced fuel reprocessing schemes.

The simplified PUREX method, in which purification processes for Pu and U are eliminated, could be considered. Centralized reprocessing is assumed. The spent nuclear fuel (SNF) will be reprocessed and only HLW will be discarded; minor actinides (MAs) could be recycled, if MA recovery and MA fuel fabrication processes are established.

X-1.6. Technical features and technological approaches that are definitive for RMWR performance in particular areas

X-1.6.1. Economics and maintainability

It is widely known that the construction cost per power output of the plant becomes higher for a scaled-down design with a smaller power output. That is, the small power reactor design tends to have economic disadvantages although it has other attractions, such as requiring low initial capital cost and flexibility in plant installation to correspond to power demands. In RMWR passive safety features are intended to improve the economy of the plant through system simplification. They are, of course, also favorable in eliminating human factors from the safety systems as much as possible and realizing transparent safety. Based on this direction, natural circulation cooling of the core was adopted in the present design and a decision was

made to introduce the appropriate passive safety components. Although a few different design concepts were investigated, a hybrid design with a combination of passive and active components seems more promising.

In the reactor system design, a natural circulation core cooling system and a gravitational steam/water separation without mechanical separation devices are adopted, taking into account the characteristics of the RMWR core and the low thermal power. The absence of large capacity pumps and valves used in conventional large capacity BWRs reduces: the number of systems on-line, construction costs, and periodic inspection loads.

In the safety system design, some passive features such as the accumulator tank partially shouldering the function of a high-pressure injection, and an isolation condenser and passive containment cooling system partially shouldering the functions of the residual heat removal (RHR) system, are introduced to simplify safety systems and to reduce the electric load on the emergency diesel generator. This permits use of a simple low-priced air-cooled diesel generator. By introducing passive components into the system, simplification is expected and this results in improvements in plant economy. However, passive components tend to be less effective and larger in size. Therefore, in the present hybrid system design, passive components are limited, to effectively improve plant economy.

In the turbine system design, simplification is paramount to thermal efficiency. Hence, re-heaters are eliminated because their economic benefit is relatively lower in the small power plant. Furthermore, redundant feedwater pumps are eliminated because the risk of electric supply loss is lower in the small power plant.

Cost evaluations for the safety related systems of the NSSS are presented in Fig. X-3. Based on the assumption that specific cost increases as $(1/W)^{0.7}$, where W is thermal power, the construction cost per power output for 300 MW(e) NSSS becomes about 1.6 times that of the ABWR. However, by introducing the passive high pressure emergency core cooling system (ECCS) in the present design, about a 20% reduction in the construction cost is estimated. This comes mainly from a significant reduction in the emergency diesel generator (DG) capacity. In the present design, only a small capacity emergency DG is required and hence, the air-cooled system can be adopted instead of the expensive water-cooled system intended for a larger capacity.

The main design features of RMWR leading to reduced capital costs per MW(e) are:

- Elimination of the main coolant pump and associated equipment.
- Reduction of the main steam lines to 2 lines.
- An air cooled type DG.
- Reduced volumes of the reactor and turbine buildings.
- A 60-year design life of the reactor.
- Reduction of active safety-related equipment by using a passive safety system.

The simplification and elimination of equipment and a long operational cycle of 25 months reduces the maintenance load; the high discharge burn-up of 65 GW d/t U can reduce the number of reloaded fuel bundles in the batches and thus, the amount of the spent fuel per unit of power produced.

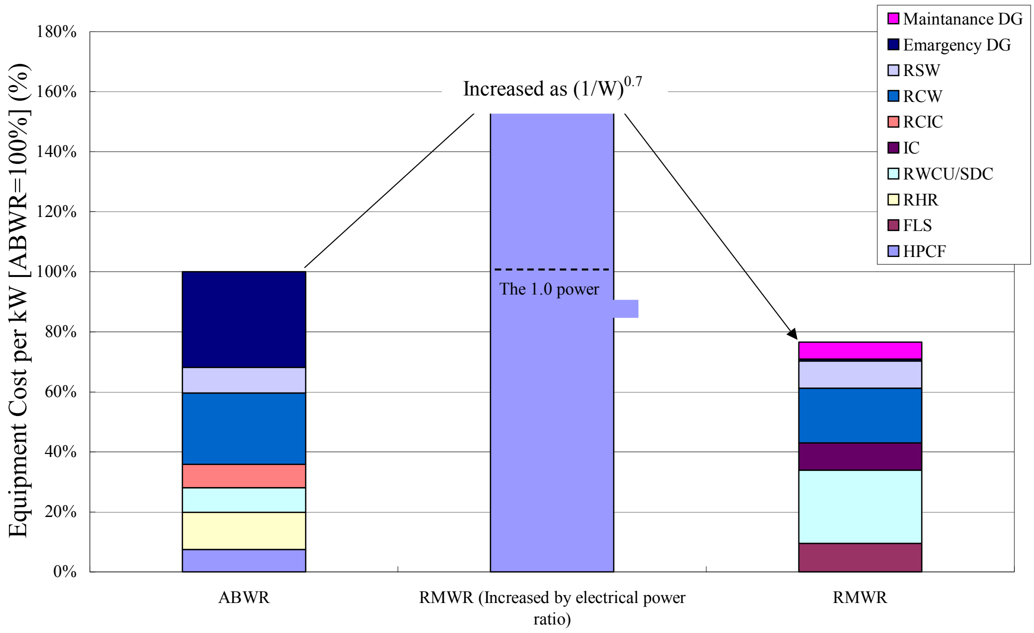


FIG. X-3. Cost evaluation of the NSSS safety-related systems.

X-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

RMWR is designed to have a breeding ratio of more than 1.0 and to operate in a closed fuel cycle, which could ensure a self-sustainable regime on fissile materials, with only depleted or natural uranium being added to fuel during reprocessing. In this way, essential savings of fissile ^{235}U are secured.

The dose limits of the RMWR reactor system are not expected to exceed the current acceptable LWR limits.

The relatively high core burn-up of 65 GW d/t and SNF recycle would result in low SNF and waste volumes. An option to recycle minor actinides (MAs) could be considered to reduce long-term radiotoxicity of waste.

X-1.6.3. Safety and Reliability

Safety concept and design philosophy

The safety concept and design philosophy of RMWR is to provide a high level of safety by implementing a reasonable combination of the active and passive safety features and systems, and to reduce costs by relying on natural circulation for core cooling and by the simplification of safety systems.

Table X-4 summarizes the RMWR safety system components and their features, as compared to the ABWR.

Provisions for simplicity and robustness of the design

The natural circulation system eliminates the coolant circulation pumps, which require reliable electric power.

TABLE X-4. SAFETY SYSTEM COMPONENTS AND FEATURES

SAFETY FUNCTION	SAFETY SYSTEM COMPONENTS FOR:		FEATURES OF RMWR
	ABWR	RMWR	
Electric output	1,356 MW(e)	330 MW(e)	Small power reactor
Core cooling	・High pressure core flooder (HPCF): 2 ・Reactor core isolation cooling (RCIC): 1 ・Low pressure core flooding (LPFL): 3	・Accumulator injection system: 1 ・Flooding system (FLS): 2	Using the active system can keep core covered for a long term. Adopting passive Accumulator Injection System can reduce the required emergency power resource capacity.
Primary containment vessel (PCV) cooling	Residual heat removal (RHR): 3	Water wall (WW)	Primary containment vessel (PCV) cooling can be continued for more than 7 days by passive systems.
Shutdown cooling	Residual heat remover (RHR): 3	・RHR/Clean up water system (CUW): 1 ・Isolation condenser (IC)	Adopting passive systems can reduce the redundancy of active systems.
Emergency power resource	Emergency (DG): 3 (5,000 kW by water cooling)	・Emergency DG: 2 (450kW by air cooling) ・Maintenance DG: 4	Adopting passive systems can reduce the required emergency power resource capacity.

Active and passive systems and inherent safety features

The main features and components in these categories are listed below.

Active safety systems:

- Active decay heat removal provided by residual heat removal (RHR) and clean up water (CUW) systems.
- Flooding system.

Passive safety systems:

- Heat removal by natural circulation during normal operation and hot shutdown conditions.
- The passive injection of accumulators.
- Passive containment cooling through the containment wall in accidents.

Inherent safety features:

- A negative void coefficient of reactivity.
- A double containment system.

The RMWR safety systems are illustrated in Fig. X-4.

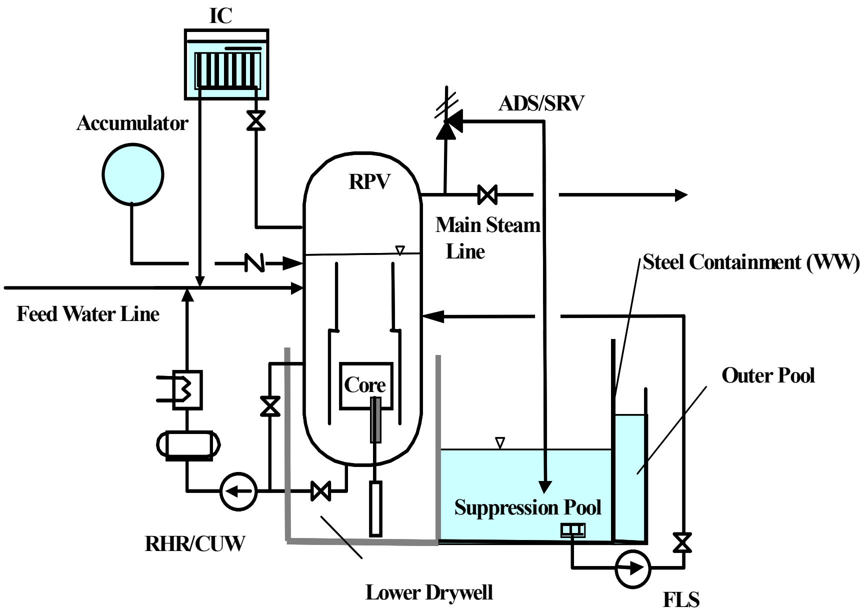


FIG. X-4. Concept of RMWR safety system.

For core cooling, the passive accumulator system (ACC) is foreseen as the high pressure emergency core cooling system (ECCS). This has sufficient capacity to prevent core uncovering for all LOCA events and to maintain core flooding for one day after the accident. The capacity is also designed to be larger than the volume of the dry well and hence, "in vessel retention" (IVR) is possible in severe accidents. After one-day core cooling by the ACC, the active core flooding system (FLS) continues to cool the core for a long period by injecting the water from the suppression pool into the core. The ACC system can eliminate active high pressure ECCS components as well as the related emergency diesel generator (DG) capacity.

Structure of the defence-in-depth

Some major highlights of the RMWR design, structured in accordance with the various levels of defence in depth are brought out below.

Level 1: Prevention of abnormal operation and failure

INITIATING EVENT	PREVENTION LEVEL FEATURES
LOCA	•Lines are designed to meet high quality engineering codes and standards, as well as seismic and environmental requirements; •Stringent water quality and material specifications to limit corrosion;

Level 1 (cont.)

LOCA	•Elimination of longitudinal welds, reduction in the number of welds; •Elimination of all large nozzles below the top of active fuel makes it unnecessary to postulate large LOCAs at low locations; •Reduction of welds to reduce the initiation frequency of leaks and ruptures on pressure boundary.
Transients	•Reduction of transient initiation by increased design margins; •Elimination of primary pump trip event by using natural circulation.
Loss of preferred power	•Standby AC power supply (two diesel generators) and DC power supply (battery).
Rod withdrawal error at low power	•Reactor control and instrumentation system (RCIS) and automatic power regulator (APR).
Rod withdrawal error at power	•Automated thermal limit monitor system (ATLM).
Station blackout	•Capability for safe shutdown and maintaining shutdown cooling under a postulated loss of all AC power for more than 72 hours.
ATWS	•Alternate rod insertion (ARI) system that utilizes sensors and logic, independent of Reactor Protection System.

Level 2: Control of abnormal operation and detection of failure

INITIATING EVENT	PREVENTION LEVEL FEATURES
LOCA	•Automatic initiation of containment isolation and actuation of active and passive systems.
Transients	•Not important due to increased design margins and intrinsic self-protection by negative void reactivity coefficient.
Loss of preferred power	•Standby AC power supply (two diesel generators) and DC power supply (battery).
Rod withdrawal error at low power	•Reactor control and instrumentation system (RCIS) and automatic power regulator (APR).
Rod withdrawal error at power	•Automated thermal limit monitor system (ATLM).
Station blackout	•Capability for safe shutdown and maintaining shutdown cooling under a postulated loss of all AC power for more than 72 hours.
ATWS	•Alternate rod insertion (ARI) system that utilizes sensors and logic, independent of Reactor Protection System.

Level 3: Control of accidents within the design basis

SAFETY FUNCTIONS	MITIGATION LEVEL FEATURES
Fission product confinement	•Pressure suppression type containment vessel.
Coolant inventory (high pressure)	•Control rod drive system; •Feedwater system; •Isolation condenser system.
Coolant inventory (low pressure)	•Accumulator; •LPFL; •Automatic depressurization system.
Decay heat removal	•RHR/CUW; •Water Wall; •IC pool.
Reactivity control	•Fine motion control rod drive (FMCRD); •SLC.
Reactor coolant pressure control	•Auto-depressurization system (ADS).

Level 4: Control of severe plant conditions, including prevention of accident progression and mitigation of consequences of severe accidents

SAFETY FUNCTIONS	MITIGATION LEVEL FEATURES
Containment temperature/pressure	•Water wall; •Containment vent.
Heat removal	•Water wall; •Isolation condenser system.
Tightness control	•Containment.
Inflammable control	•Passive auto-catalytic recombiners •N2 gas inert
Fission product confinement	•Containment.
Corium management	•Core catcher.

Level 5: Mitigation of radiological consequences of significant release of radioactive materialsThe containment vent system mitigates the radiological consequences of radioactive release from the suppression chamber airspace through the suppression pool.

Design basis accidents and beyond design basis accidents

The following accidents are considered in the RMWR, Table X-5.

TABLE X-5. LIST OF ACCIDENTS CONSIDERED FOR RMWR

CATEGORY	ACCIDENT
Design basis accident	•LOCA; •Fuel handling accident; •Main steam line break outside containment; •Control rod drop accident.
Beyond design basis accident	•Station blackout; •ATWS; •Multi-failure in transients.

Figure X-5 shows the analyzed variations of the reactor water level during a drain piping break LOCA.

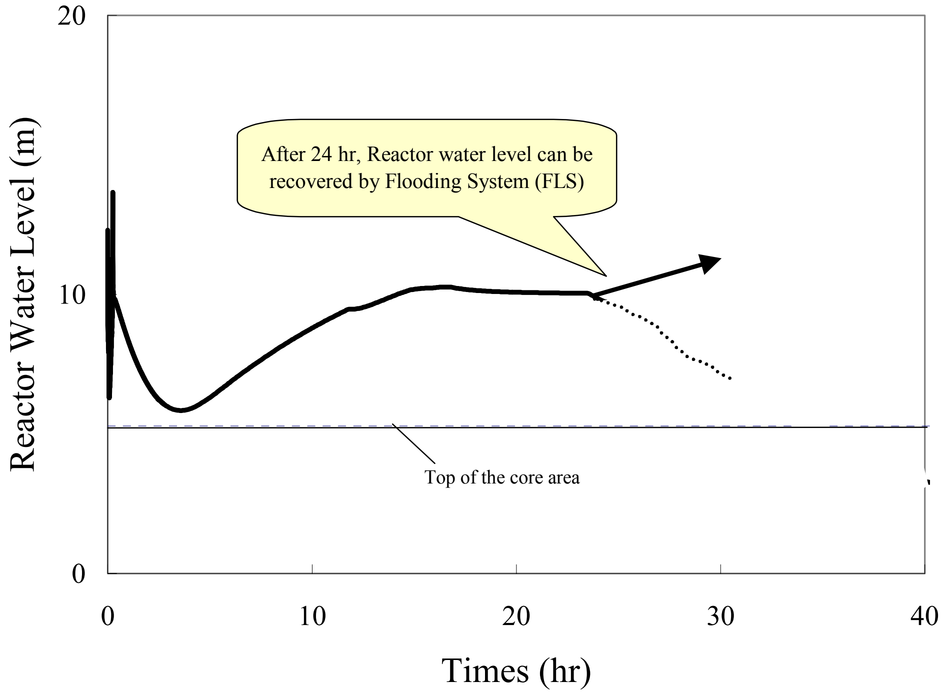


FIG. X-5. Water level during a drain line break LOCA.

Since the drain piping is located at the bottom of the pressure vessel, this accident is the most severe one with respect to core flooding. The results show that the accumulator injection system can keep the water level above the top of the core for one day after initiation of the LOCA. Although the active component of the FLS is not activated in the present analysis to show the performance of the ACC, the FLS is supposed to be activated to maintain long-term core cooling by injecting water from the suppression pool, even one day after initiation of the LOCA. For isolation of the reactor, the passive isolation condenser (IC) is introduced for core cooling whereas the RCIC system driven by the steam turbine is eliminated.

The heat released in the containment vessel during the accident is absorbed in the suppression pool for one day; further heat removal is possible over a longer period by evaporation of the water in the outside pool of the containment. This system gives sufficient time margins to cool the core for severe accidents as well as design base accidents.

In the presented RMWR design, pipes of large diameter, such as the main steam line and the feed water line, are intentionally located above the top water level of the core. Therefore, a small capacity of ECCS is sufficient to keep the core covered with the water during a LOCA.

Provisions for safety under seismic conditions

The RMWR is designed to meet Japanese seismic standards.

Probability of unacceptable radioactivity release beyond the plant boundaries

The probability is estimated to be less than 1×10^{-7}/reactor·year.

Measures planned in response to severe accidents

These consist of containment vent actuation and water makeup for the outer pool.

X-1.6.4. Proliferation resistance

Simplified PUREX reprocessing with a relatively low decontamination factor is introduced and hence, MOX fuel would have higher concentration of radioactive materials than the current LWR MOX fuel. Even fresh MOX fuel for the reactor has a higher radioactivity than UO_2 fuel or the current LWR MOX fresh fuel and this facilitates nuclear material accounting and verification.

The RMWR can be operated with more than 35% of the ^{240}Pu contents in the equilibrium condition of the multiple recycling of Pu [X-5], which would reduce the attractiveness of the RMWR fissile materials for a weapon programme.

Pure Pu is not separated in the whole reactor fuel cycle and Pu always exists with U, Am and FPs.

X-1.6.5. Technical features and technological approaches used to facilitate physical protection of RMWR

No information provided.

X-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of RMWR

A modular construction is envisaged to reduce on-site work in developing counties. Provisions for leasing of fuel and / or full-scope fuel cycle service agreements and options for nuclear power plant (NPP) leasing would be considered.

X-1.8. List of enabling technologies relevant to RMWR and status of their development

A list of enabling technologies relevant to the RMWR based power plant is given in Table X-6.

TABLE X-6. ENABLING TECHNOLOGIES RELEVANT TO RMWR BASED NPP

MAIN OBJECTIVE	ENABLING TECHNOLOGY	STATUS OF DEVELOPMENT
Adequate cooling conditions.	Feasibility of tight lattice core cooling.	Feasibility demonstrated by critical heat flux experiments and sub-channel calculations.
Negative void coefficient and high conversion ratio in tight-lattice core	Double-flat-core design	Feasibility demonstrated by critical experiments and calculations.
Optimum use of passive systems for core heat removal	Natural circulation driven core cooling	No information provided.
	Isolation condensers	
Increased burn-up of fuel	Provision for reliable cladding performance at high burn-up	Increased burn-up data for zirconium alloy cladding in LWRs. A new stainless steel cladding is under investigation.
Reduced O&M costs	60 year design life.	No information provided.

MAIN OBJECTIVE	ENABLING TECHNOLOGY	STATUS OF DEVELOPMENT
Enhanced safety following LOCA.	Passive ECCS (emergency core cooling system) of enhanced effectiveness.	No information provided
Reliable performance of high-content Pu MOX fuel at high burn-ups	Technology for reliable performance of high-content Pu MOX fuel rods at high burn-ups	In-depth analytical studies and irradiation experiments foreseen. The topics for a study are: •Thermal conductivity degradation on the burn-up extension of the pellets; •Swelling of the FP gas pores generated around Pu-rich spots; and •Fuel rod deformation behaviour.

X-1.9. Status of R&D and planned schedule

The design and development of RMWR is supported mainly by JAERI, Hitachi Ltd. and JAPC. R&D for the RMWR is partly supported by the Japanese Government and utilities' foundations.

The current design stage is conceptual design. Once adequate funding is available, the basic design of a 300 MW(e) RMWR could be developed in about one year, and the detailed design in about 2 years after it.

The R&D costs needed to deploy the prototype are estimated at about ~US $10 million. The R&D costs needed to deploy final version of the NPP with an RMWR are estimated as ~US $500 million.

X-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

A light water breeder reactor with MOX fuel has never operated before. Therefore, a prototype plant will be needed before licensing the RMWR in a series

X-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

No other similar SMRs are under development elsewhere.

X-2. Design description and data for RMWR

X-2.1. Description of the nuclear systems

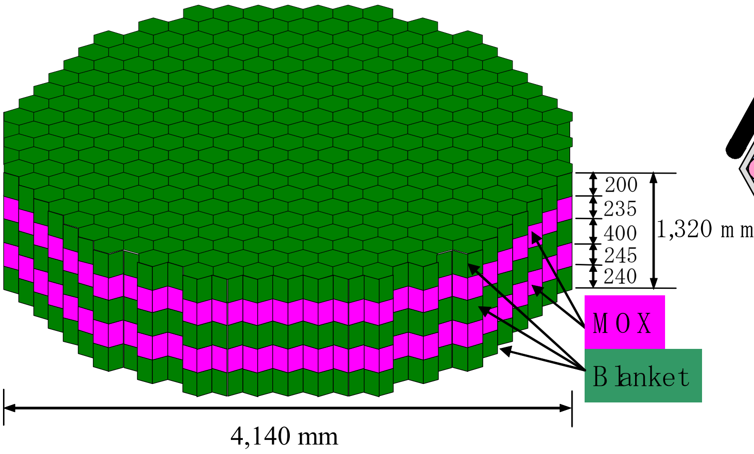
Reactor core and fuel design

It is known from previous experiences on the RMWR core design [X-6] that the pressure loss across the core tends to be large. This is basically due to the characteristic tight-lattice core configuration, with a rod gap width of around 1.3 mm, required for the high conversion ratio. The only exception is the double-flat-core design for the BWR-type core. In this design, the core water mass flow rate is significantly reduced to realize a high core average void fraction and the fuel rod is very short as well. For these reasons, the pressure drop along the fuel

assembly is calculated to be around 40 kPa, which is about one fifth of that in the current ABWR design and hence, core cooling by natural circulation is expected to be possible in this design. Therefore, the double-flat-core design was adopted.

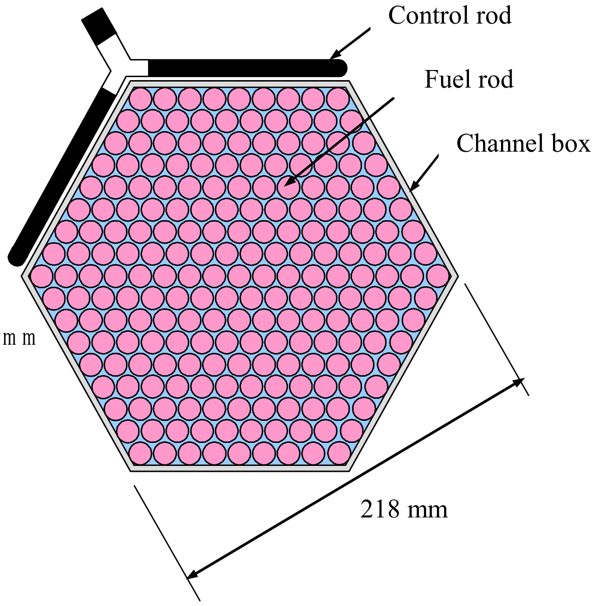
The major specifications and dimensions of the small RMWR core satisfying the four design targets listed in the previous study [X-5], are summarized in Table X-7 and schematically shown in Fig. X-6. A cross section of an assembly with the channel box along with the Y-shaped control rod is shown in Fig. X-7. The core consists of 282 hexagonal fuel assemblies, each of which consists of 217 fuel rods with an outer diameter of 13.0 mm arranged in a triangular lattice with gap widths of 1.3 mm. The core part is very short, only 0.88 m high, and consists of two MOX regions with an internal blanket region between them. Adding the upper and lower blanket regions with 0.20 and 0.24 m heights, the total axial length is 1.32 m. This very short double-flat-core design causes the void reactivity coefficient to be negative. The average fissile plutonium content in the MOX regions is 18 weight %.

The plant system and structures are appropriately designed based on this core specification.



No. of fuel assembly 282
No. of control rod 85

FIG. X-6. Overview of whole core.



No. of fuel rod 217
Fuel rod diameter 13.0 mm
Fuel rod gap 1.3 mm

FIG. X-7. Cross sectional view of fuel assembly.

TABLE X-7. MAJOR SPECIFICATIONS OF THE CORE

Items	Specifications
Core type	2 MOX regions with 1 internal and 2 axial blanket regions
Number of fuel assemblies	282
Number of control rods	85
Number of fuel rods	217 per hexagonal assembly, Zircaloy-4 cladding
Clad diameter / Fuel rod gap	13.0 mm / 1.3 mm
Lattice pitch	218 mm (226.4 mm: the side with the control rod)

The MOX fuel for the RMWR has a high Pu content, about 30% of Pu (total) by weight, and is irradiated to a high burn-up of about 100 GW d/t in the MOX region. Such high irradiation conditions make an evaluation of the thermal and mechanical feasibility of the fuel rods essential. Therefore, the safety evaluation analysis of the fuel behaviour has been conducted using a fuel performance code FEMAXI-RM. The code FEMAXI-RM, an advanced version of FEMAXI-V [X-7, X-8], has been developed to analyze the RMWR fuel rods with features such as the combined structure of MOX and blanket parts.

The analyses were conducted for a singe rod, which is assumed to have the highest burn-up in the RMWR core. The models or material properties applied to the analysis, such as fuel thermal conductivity, fission product (FP) gas diffusion and release and creep rate are derived or extrapolated from those in the usual analysis of LWR fuel rods. For the first analysis, particular focus was on the thermal behaviour, such as FP gas release and internal pressure increase induced by the fuel temperature rise.

Figure X-8 shows one example from the calculated results for the internal pressure rise vs. burn-up, which is essentially caused by the FP gas released from the fuel pellets.

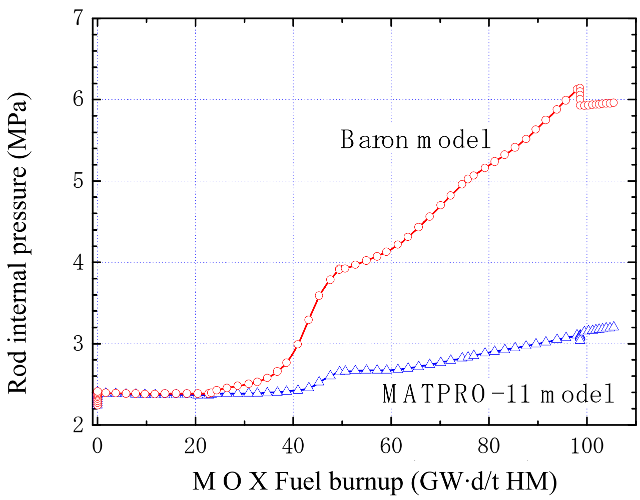


FIG. X-8. Rod internal pressure change with burn-up increase.

In the analysis, two different models for thermal conductivity are used to investigate their effects on the results. One is the Baron model [X-9], which is somewhat dependent on reduction of the thermal conductivity on progress of the burn-up. The other is the MATPRO-11 model [X-10], which is not dependent on the burn-up. The calculated internal pressures increase gradually, but the Baron model produces much higher values. This is because the model produces lower thermal conductivities of the fuel and hence, higher fuel temperatures, resulting in an increased FP gas release from the fuel pellets. Although the internal pressure at the end of the fuel life is about 6 MPa, it does not exceed the coolant pressure of 7.2 MPa. Being so, the cladding will never "lift-off" even at very high burn-ups.

These results suggest that the MOX fuel rod has no particular thermal behaviour modes that would raise concerns over safety and reliability. However, the behaviour of MOX fuel with such a high Pu content has been neither fully understood nor analytically predicted in the region of very high burn-ups. Therefore, a precise characterization of input data and models for material properties is necessary to perform more precise analytical evaluations of fuel safety and reliability. In addition, modelling of the thermal conductivity degradation on the burn-up extension of the pellets, swelling of the FP gas pores generated around Pu-rich spots

and fuel rod deformation behaviour are major issues to be hereafter considered in the code analysis. For these issues, irradiation experiments on the MOX fuel are of vital importance.

Primary coolant system

Figure X-9 shows a vertical cross section of the reactor pressure vessel (RPV) and the limiting factors considered in the design. It is very important to minimize the RPV size because it is a major factor in the size of the primary containment vessel (PCV) and reactor building (R/B), consequently with a great effect on the construction costs of the R/B. Therefore, the RPV size is minimized by the short fuel length, low pressure loss in the core, low core flow conditions characteristic of an RMWR and by optimizing the chimney height and inside diameter of the chimney. The chimney height of 5.5 m is decisive to maintain the necessary natural circulation head; the chimney diameter of 5.4 m is decisive to cause the steam velocity at the chimney exit to be less than 0.7 m/s to stabilize the water surface [X-11]. The height of 2.5 m from the water surface to a main steam line nozzle is adopted to enable a gravitational water/steam separation, based on experimental data [X-11] and a semi-empirical theory [X-12]. The annulus gap of 1.3 m from the shroud to the RPV is adopted to limit the neutron irradiation rate for the RPV wall and to reduce the carry-under also. As a result, the height of the RPV is about 17.4 m and the diameter about 6.3 m.

Adopting the large capacity main steam isolation valve (MSIV) reduces the number of main steam (MS) lines and the number of MSIVs. The number of residual heat removal (RHR) systems is reduced to one line by substituting a clean up water system (CUW) for an RHR because of the use of the large capacity CUW heat exchanger adopted in large capacity BWRs.

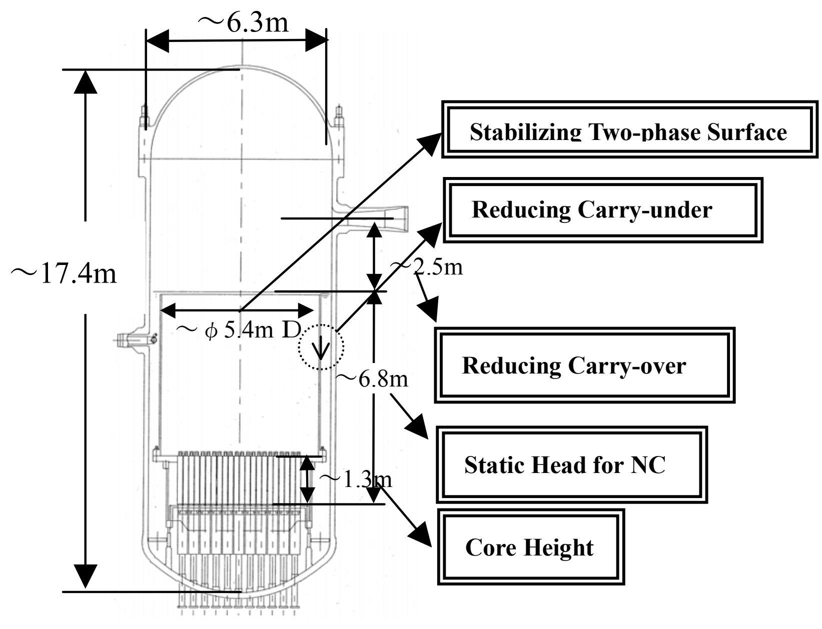


FIG. X-9. Cross section view of RPV.

Main heat transport system

The main heat transport system of a 300 MW(e) RMWR with specification of heat removal path in normal operation and in accidents is shown in Fig. X-10.

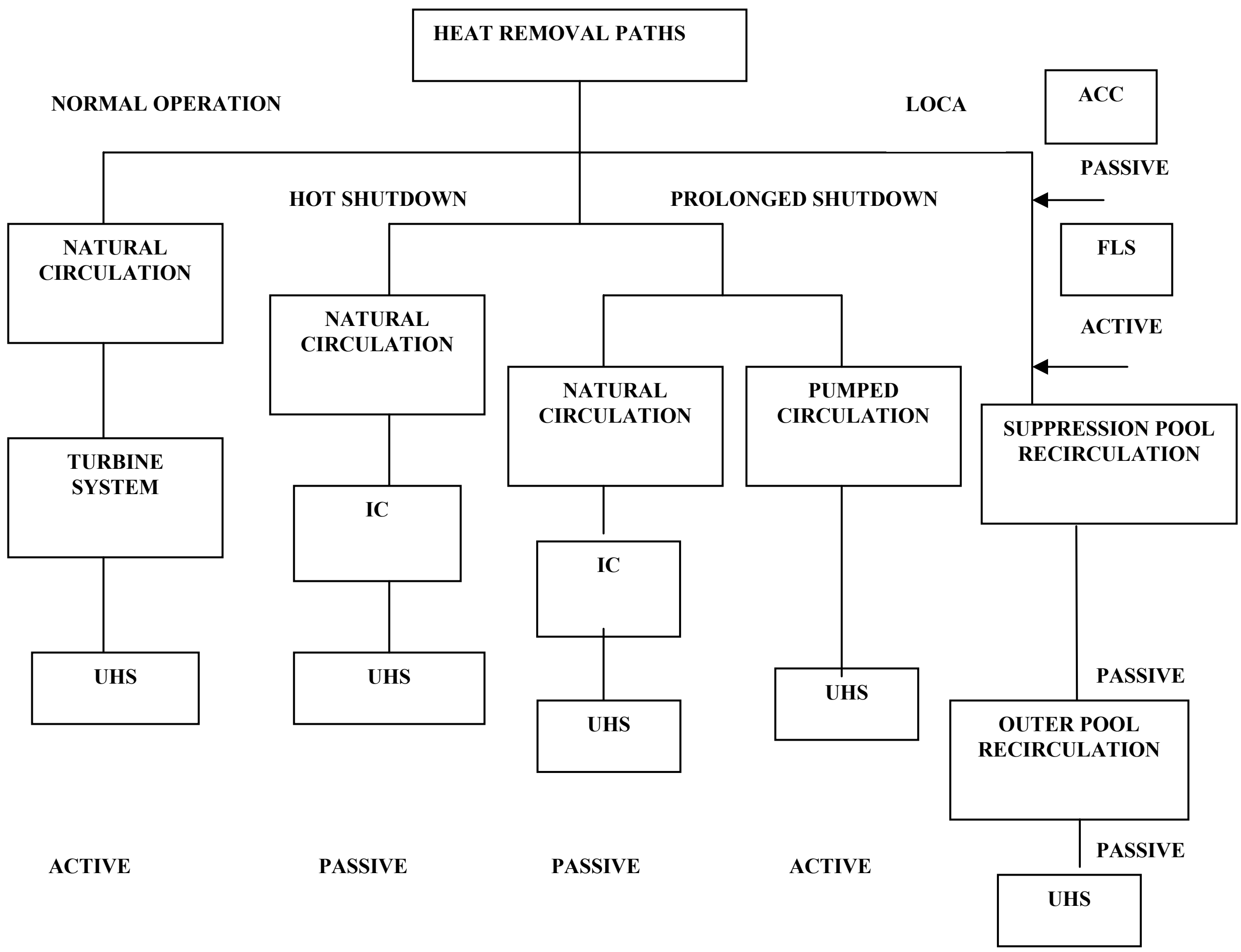


FIG. X-10. Main heat transport system of RMWR.

X-2.2. Description of the turbine generator plant and systems

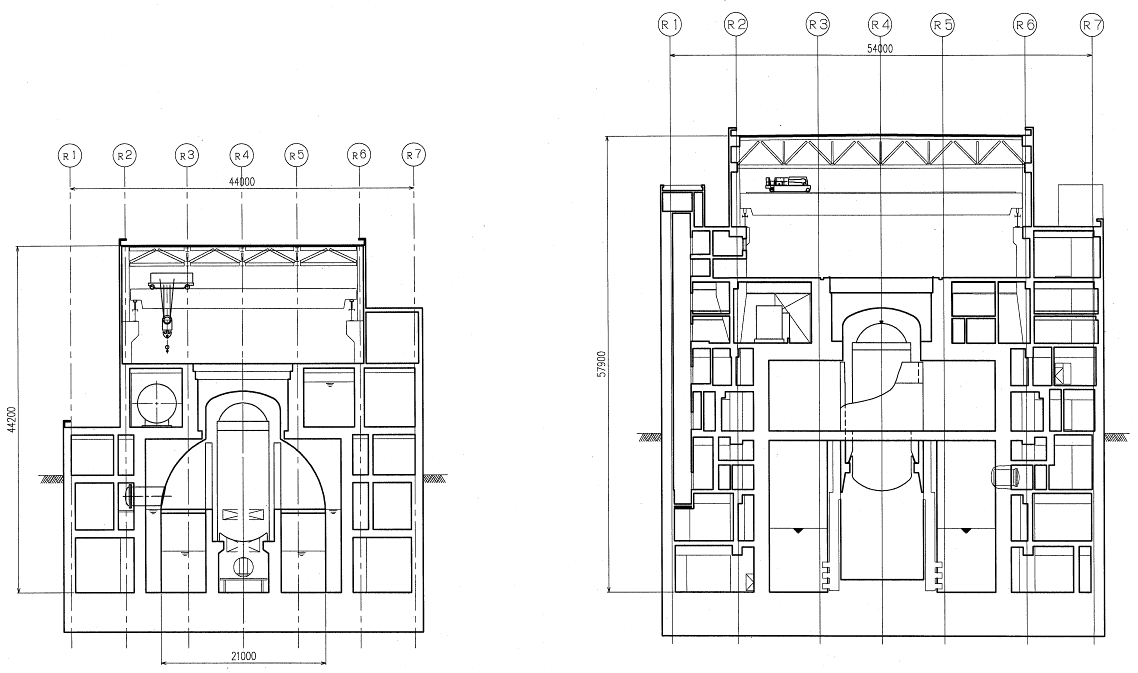
In the turbine system design, the TCDF–38 system composed of a single high-pressure turbine and single low-pressure turbine is adopted for the 300 MW(e) RMWR with the cascade heater drain system and without the moisture separator-reheater. This is because system simplification takes priority over improvements in the thermal efficiency. Furthermore, redundancy in the condensate and feedwater system (i.e., back-up pump and heater train) is eliminated because the risk of electric generation loss is lower in the small power plant.

X-2.3. Systems for non-electric applications

Non-electrical applications are not considered for a direct-cycle RMWR.

X-2.4. Plant layout

Figures X-11 and X-12 show the arrangements of the reactor building (R/B) and the turbine building (T/B) in the small RMWR as compared with the conventional ABWR of 1,356 MW(e), respectively. The reactor building design is improved by the simplified system composition based on the introduction of passive features. The reactor pressure vessel is located in the lower level, and hence, the design is also improved from the seismic point of view. The R/B volume and the T/B volume of the small RMWR are expected to achieve a capacity reduction of about 45% and about 37% compared with those of the ABWR.



Small RMWR ABWR

FIG. X-11. Comparison of reactor buildings between small RMWR and ABWR.

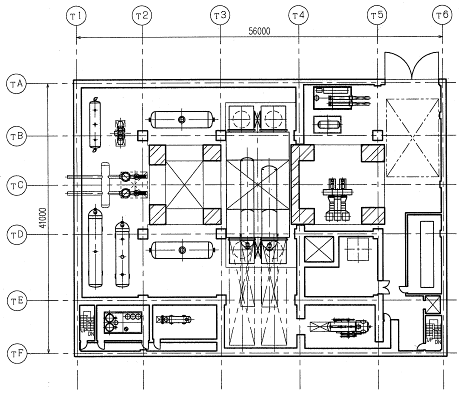
The reduction of the R/B volume is facilitated by the following design features of RMWR:

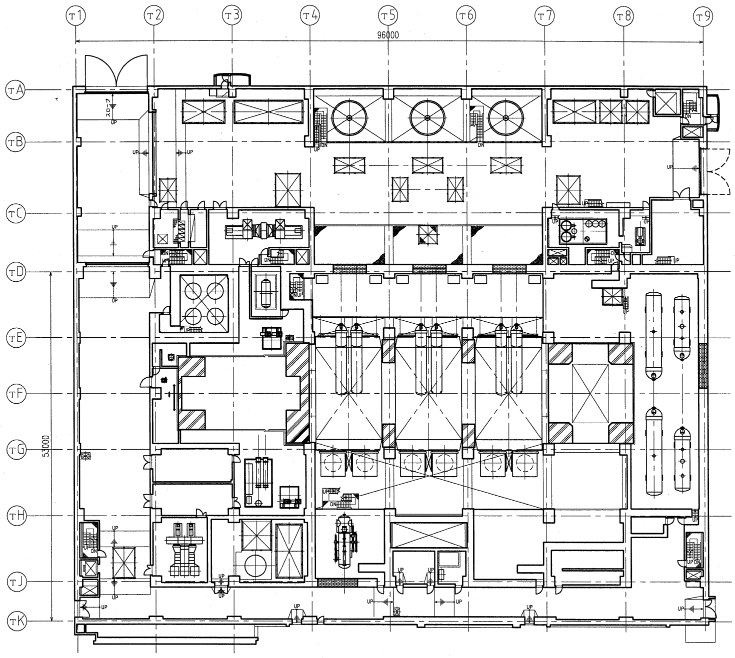
- Compactness of the RP.
- Reduction of the main steam lines from 4 to 2 lines.
- Elimination of active high pressure ECCS.
- Adoption of an air-cooled type emergency DG.
- Elimination of temporary space for the separator and dryer during maintenance.

The reduction of the T/B volume is facilitated by the following:

- Adoption of the TCDF-38 turbine.
- Elimination of the moisture separator-reheater.
- Elimination of redundancy in the condensate and feedwater system.
- Adoption of the cascade drain system.

The construction period will be shortened by the use of a steel containment vessel, a steel plate-reinforced concrete (SC) structure in building structures and by the implementation of a large block construction and modular construction approaches, in addition to the lower stories of the R/B.





Small RMWR   ABWR

FIG. X-12. Comparison of turbine buildings between small RMWR and ABWR.

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ANNEX XI

ADVANCED HEAVY WATER REACTOR (AHWR)

BARC, India

XI-1. General information, technical features and operating characteristics

XI-1.1. Introduction

The first phase of the Indian nuclear power programme [XI-1] is based on natural uranium fuelled, heavy water moderated pressure tube type reactors commonly designated as pressurized heavy water reactors (PHWRs), also known as CANDUs for such reactors of Canadian origin. Thirteen out of fifteen Indian nuclear power reactors under operation, and five out of eight Indian nuclear power reactors under construction, at the beginning of April 2005, are PHWRs. The first two of these reactors, Rajasthan units –1 and –2 are similar in design to the Canadian Douglas Point reactor. Rajasthan–1 was built at Rawatbhata in India with Canadian collaboration. This reactor started commercial operation in November 1972. Subsequently, the construction of Rajasthan–2 and design and construction of all future Indian PHWRs was done indigenously in India. The design of Indian PHWRs has progressively been improved and augmented to take into account the feedback from national as well as international experience with such reactors. A large infrastructure was set up at the Bhabha Atomic Research Centre, Mumbai to facilitate research, design, and development in several areas relevant to PHWRs. These areas include: materials technologies, critical components and new systems, reactor physics, thermal hydraulic and safety analysis codes, testing and qualification of reactor systems and equipment, and design and development of systems for in-service inspection and ageing management.

AHWR, being a pressure tube type heavy water moderated reactor, makes use of the PHWR specific technologies pertaining to pressure tube and low pressure moderator based design. These technologies are already developed and successfully demonstrated internationally. There are, however, several significant differences, between PHWR and AHWR. These differences are mainly related to the use of thorium based fuel [XI-2] with negative void coefficient of reactivity, the use of boiling light water in natural circulation mode as coolant, and to incorporation of several passive safety features aimed at achieving a grace period of three days and elimination of a need for emergency planning beyond the plant boundary. The concept of the reactor was developed in early nineties. Its basic design, and experimental development in areas required to establish feasibility of the basic design, have been completed at the Bhabha Atomic Research Centre (BARC). Several major experimental facilities have been set-up and used with this objective, and some others are under construction to produce additional data. The latter include a critical facility, with a capability to simulate the AHWR core lattice and fuel configurations, and a full height integral test loop to simulate the main heat transport (MHT) system of AHWR.

The design and development of this reactor has been fully funded by the government of India, Department of Atomic Energy (DAE). This work is mainly carried out at the BARC, a constituent unit of the DAE.

XI-1.2. Applications

The AHWR based nuclear power plant is designed to produce 300 MW(e) (gross) of electricity , along with 500 m^3/day of desalinated water. The plant can be configured to deliver higher desalination capacities with some reduction in electricity generation. More details are given in sections XI-1.4 and XI-2.

XI-1.3. Special features

AHWR is a land-based nuclear power station.

XI-1.4. Summary of major design and operating characteristics

The reactor is designed to produce 920 MW of thermal power, generating 300 MW(e) (gross), and 500 m^3/day of desalinated water. AHWR based plant can be operated in base load, as well as in load following mode. The target lifetime load factor and availability factors for AHWR are 80% and 90% respectively. Some major design characteristics of AHWR are given in Table XI-1.

TABLE XI-1 MAJOR DESIGN CHARACTERISTICS OF AHWR

ATTRIBUTES	DESIGN PARTICULARS
Core configuration	Vertical, pressure tube type
Fuel	Pu-ThO_2 MOX, and $^{233}UO_2$-ThO_2 MOX
Moderator	Heavy water
Coolant	Boiling light water
Number of coolant channels	452
Pressure tube inner diameter	120 mm
Pressure tube material	20% Cold worked Zr-2.5% Nb alloy
Lattice pitch	245 mm
Active fuel length	3.5 m
Calandria diameter	7.4 m
Calandria material	Stainless steel grade 304L
Steam pressure	7 MPa
Mode of core heat removal	Natural circulation
MHT loop height	39 m
Shut-down system-1 (SDS-1)	40 mechanical shut-off rods
Shut-down system-2 (SDS-2)	Liquid poison injection in moderator

A vertical cross-sectional view of the AHWR reactor block is shown in Fig. XI-1, and a simplified schematic diagram of the AHWR based nuclear power plant is given in Fig. XI- 2.

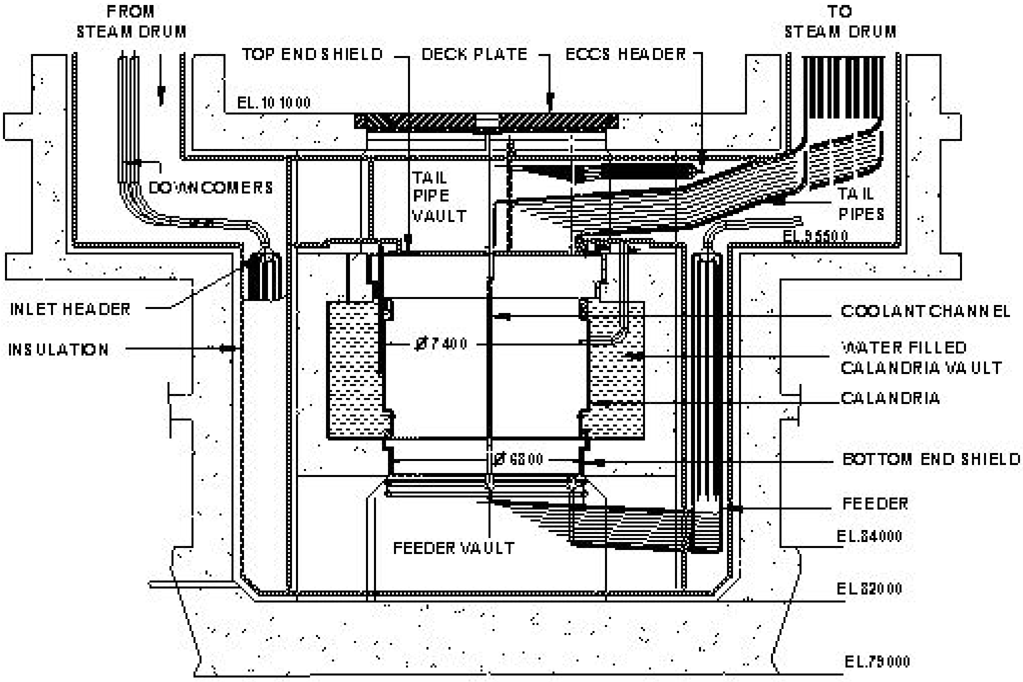

FIG. XI-1. AHWR reactor block.

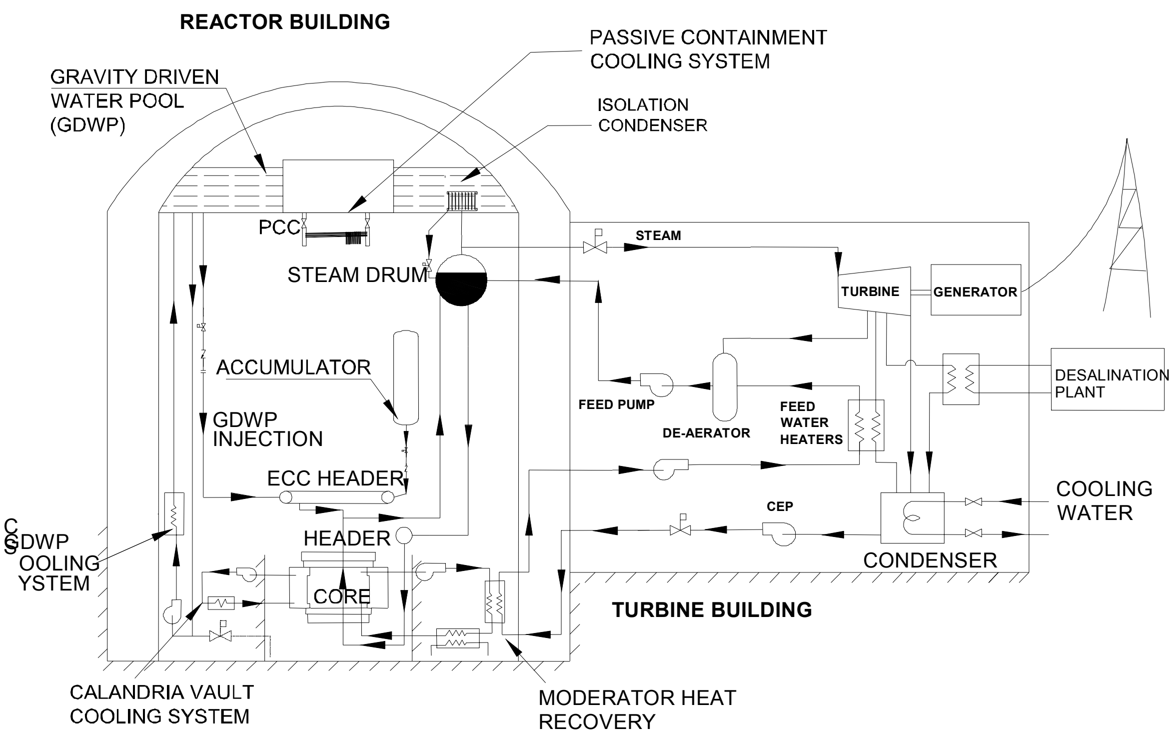

FIG. XI-2. Simplified schematic of AHWR.

The reactor core is housed in calandria, a cylindrical stainless steel vessel containing heavy water, which acts as moderator and reflector. The calandria, located below ground level, contains vertical coolant channels in which the boiling light water coolant picks up heat from fuel assemblies suspended inside the pressure tubes. The coolant circulation is driven by natural convection through tail pipes to steam drums, where steam is separated for running the turbine cycle. The four steam drums (only one shown for clarity), each catering to one-fourth of the core, receive feed water at stipulated temperature to provide optimum sub-cooling at reactor inlet.

Inside the calandria, a zircaloy–4 calandria tube surrounds each of the pressure tubes, to provide an annulus, open to air at the bottom, which separates the cold moderator from the hot pressure tube. An annulus gas monitoring system to detect any leakage from either pressure tube or calandria tube uses the gap between the pressure tube and the calandria tube

Down-comers, four from each steam drum, bring the flow to a circular inlet header, which distributes the flow to each of the 452 coolant channels through individual feeders. During shutdown, passive valves establish communication of steam drums with the isolation condensers submerged inside a 6000 m^3 capacity gravity driven water pool (GDWP) for decay heat removal, under hot shut-down condition. The pool acts as a heat sink for passive decay heat removal system. In the event of a loss of coolant accident (LOCA), four independent emergency core cooling system (ECCS) circuits (only one is shown for clarity) provide cooling of the core for at least 72 hours. A high pressure injection system using accumulators and a low pressure injection system using GDWP as source of water are passively brought into action, in a sequential manner, as the depressurisation of the MHT system progresses, following a LOCA.

A passive containment cooling system is incorporated to remove heat from the containment following LOCA. This system removes the energy released into the containment through cooling tubes, which are in hydraulic communication with the GDWP water. The containment steam condenses over the tubes and cooling water circulates through the tubes, aided by natural convection, to remove the heat of condensation to GDWP.

A fuelling machine located on top of the reactor block carries out on-line refuelling in the vertical channels. The reactor is confined within a double containment to minimize ground release of radioactivity. The inner primary containment is built of pre-stressed cement concrete whereas the outer secondary containment is built of reinforced cement concrete.

The reactor physics design of AHWR has been optimized to achieve the following objectives:

- Maximum (at least 60 percent) power production from Th/^{233}U.
- Negative coolant void reactivity coefficient.
- Minimization of initial plutonium inventory and its consumption.
- High fuel burn-up.

The axial flux peaking factor within the core is 1.4. Two different values of enrichment (typically, 2.5% in the top half and 4.0% in the lower half) in plutonium bearing fuel pins cause the axial flux peak to be located in a region below the high steam quality upper region in the coolant channel, thereby significantly improving the minimum critical heat flux ratio (MCHFR), as compared to what would be achievable with a uniform axial enrichment.

A summary of the reactivity effects in AHWR is provided in Table XI-2.

TABLE XI-2. REACTIVITY EFFECTS IN AHWR

CONDITION	REACTIVITY CHANGE (mk)
Temperature and void effects	
Channel temperature (300 K at cold critical to 558 K at hot standby)	+ 2.5
Moderator temperature (300 K to 353 K)	+ 3.0
Fuel temperature (558 K at hot standby to 898 K at full power)	- 6.5
Coolant void (density from 0.74 at hot standby to 0.55 g/cc at full power)	- 2.0
LOCA at full power (density change from 0.55 to 0.0 g/cc)	-4.0
Xenon load	
Equilibrium load	- 21.0
Transient load 30 min. after shutdown from full power	< - 1.0
Peak load 300 min. after shutdown from full power	- 7.0
Other neutron physical parameters	
Delayed neutron fraction, β (without photo neutrons)	0.003
Neutron life time, l, sec.	0.00022

As shown in the core map of the reactor (Fig. XI-4), the reactor regulation and protection system devices are housed in vertical channels at 53 lattice positions.

The reactor regulation system adjuster rods are classified in three groups on the basis of their function. The first group of four rods, called absorber rods (AR), is held normally in fully 'in' condition, and provides a 7 mk Xenon override capacity to facilitate reactor restart any time after shutdown. The second group of five shim rods (SRs) is normally kept in fully 'out' condition. These rods are required to provide up to 5 mk of negative reactivity for achieving power set back. The last group comprises four regulating rods (RRs). These rods are kept in partially 'in' condition to provide up to 8 mk of negative reactivity for fine control of reactivity.

In addition, for long-term sub-criticality control, there is a provision to add boron to the moderator.

On account of on-line refuelling, the excess reactivity in the core is always maintained at a low value. The excess reactivity in the core at any given time of operation under equilibrium conditions would range between 7 mk (ARs fully in and RRs fully out) and 15 mk (ARs and RRs fully in). After a long shutdown, with the Xenon load removed and ^{233}Pa converted to ^{233}U, the excess reactivity in the core will be approximately 35 mk**.**

The reactor protection system comprises two independent fast acting shutdown systems. Shutdown system–1 (SDS–1) is based on mechanical shut-off rods with boron carbide based absorbers in forty lattice positions, providing a total negative reactivity worth of 75 mk with all rods inserted, and a worth of 47 mk with two maximum worth rods not available. shutdown system–2 (SDS–2) is based on a liquid poison injection into the moderator. In addition, a pressurized addition of poison, passively driven by steam pressure, takes place in the event of over pressure in the MHT system.

AHWR is designed to operate in direct cycle mode, to supply nearly dry saturated steam at a nominal pressure of 7 MPa, yielding a thermodynamic efficiency of 33 percent with seawater temperature of 308 K.

Main thermal hydraulic characteristics of AHWR are given in Table XI-3.

TABLE XI-3: THERMAL HYDRAULIC CHARACTERISTICS OF AHWR

PARAMETER	CHARACTERISTICS
Circulation Type	Natural for normal operating as well as hot shut-down conditions
Coolant Conditions	Core inlet: 532 K, 2237 kg/s; Core outlet: 558 K, average exit quality 18.2%
Steam and feed water conditions	Steam at outlet from steam drum: 7 MPa, 558 K, 407.6 kg/s Feed water at inlet to steam drum: 403 K
Fuel temperatures during normal operation	For maximum rated channel: fuel centre line: 1213 K, Clad surface: 572 K The maximum permissible clad temperature is 673 K.

The minimum critical heat flux ratio (MCHFR) at 120 percent full power operation of the reactor is estimated as 1.68.

The reactor has a provision for on-line refuelling. On an average, 102 fuel assemblies will be replaced per year, assuming 300 effective full power days (EFPD) of operation per year, with an average residence time of a fuel assembly inside the core being approximately 4.5 years. The average discharge burn-up of fuel (without reconstitution) of AHWR is 24 000 MW·d/t. Assuming 300 effective full power days of operation per year, after attaining equilibrium core conditions, the annual material requirements for fuel have been worked out and are given in Table XI-4.

TABLE XI-4. ESTIMATED ANNUAL FUEL MATERIAL REQUIREMENTS FOR AHWR
(Assuming 300 EFPD of operation per year)

MATERIAL	REQUIREMENT/MW(e) INSTALLED	REMARKS
PuO_2	0.8 kg/yr	Discharged PuO_2 may be used in FBRs
$^{233}UO_2$	Self-sustaining	
Zircaloy-2	9.8 kg/yr	Possibility of recycling in the long term
ZrO_2	1.7 kg/yr	
Dy_2O_3	0.09 kg/yr	
ThO_2 (without recycling)	46 kg/yr	Only in the near term
ThO_2 (with recycling)	2.5 kg/yr	

The design life of all non-replaceable structures (including calandria vessel, end-shields, major piping and concrete structures) is 100 years. All components and equipment that will have lower design life will be easily replaceable during routine shutdowns. From this consideration, the zirconium alloy pressure tubes of AHWR have been made easily replaceable.

Seawater desalination is a feature incorporated in the design of AHWR. An integration of low temperature multi-effect distillation (LT–MED) desalination plant in the basic design is envisaged to produce the entire makeup demineralized. The capacity of desalination plant can be enhanced at the expense of some loss of electricity generation. It is estimated that each 1000 m^3/d of additional desalination capacity would require a reduction of 0.95 MW(e) in the gross electricity generated.

For the first AHWR preliminary estimates indicate that its overnight capital cost (base year 2001) will be Indian Rupees 53 000 (US $1170 at current rates) per kWe. The estimated construction period from first pour of concrete to criticality is six years. The unit energy cost for the electricity generated with this reactor is expected to be competitive with that for alternative energy options available in the time frames and at the geographical regions of deployment of this reactor.

XI-1.5. Outline of fuel cycle options

A closed nuclear fuel cycle is to be used for the AHWR reactor. Thorium, ^{233}U and plutonium will be recovered from the spent fuel. While thorium and ^{233}U will be used to manufacture fresh fuel for AHWR, reprocessed plutonium will be stored, and later used as fuel for a fast breeder reactor (FBR). The plutonium requirement for the reactor will be met by reprocessing of the spent fuel of PHWRs fuelled by natural uranium [XI-3]. A schematic of the near term fuel cycle for AHWR is given in Fig. XI-3.

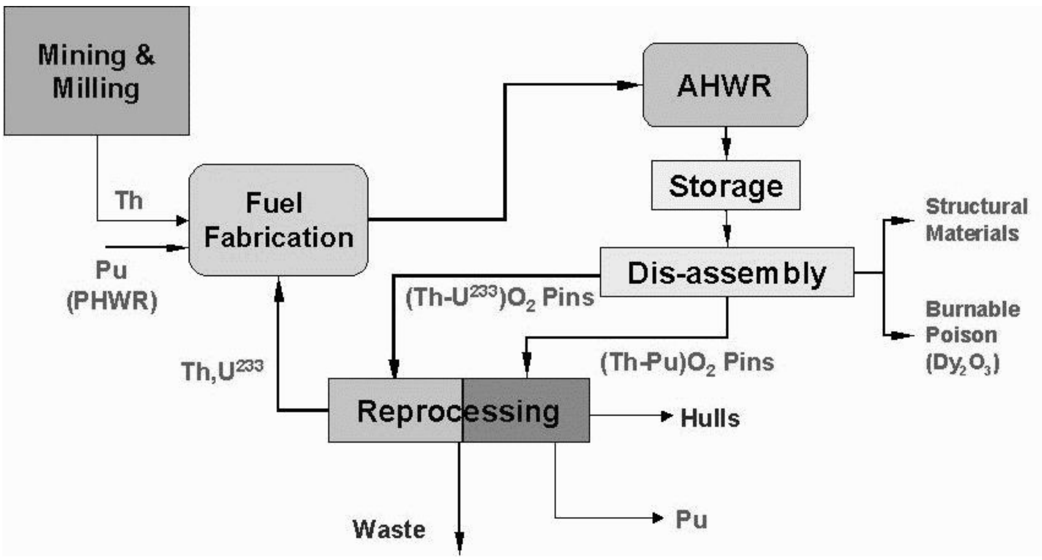

FIG. XI-3. Near-term fuel cycle for AHWR.

In the long-term, the fuel cycle will extend to include a synergy between PHWRs, FBRs and transmutation systems based on either accelerator driven systems or fast breeder reactors. A schematic of the long-term fuel cycle for AHWR is given in Fig. XI-4.

In principle, AHWR can be configured for a variety of alternative fuel cycle configurations. These include a plutonium burner mode using plutonium-thorium MOX and a fully thorium-^{233}U fuelled mode in a closed fuel cycle.

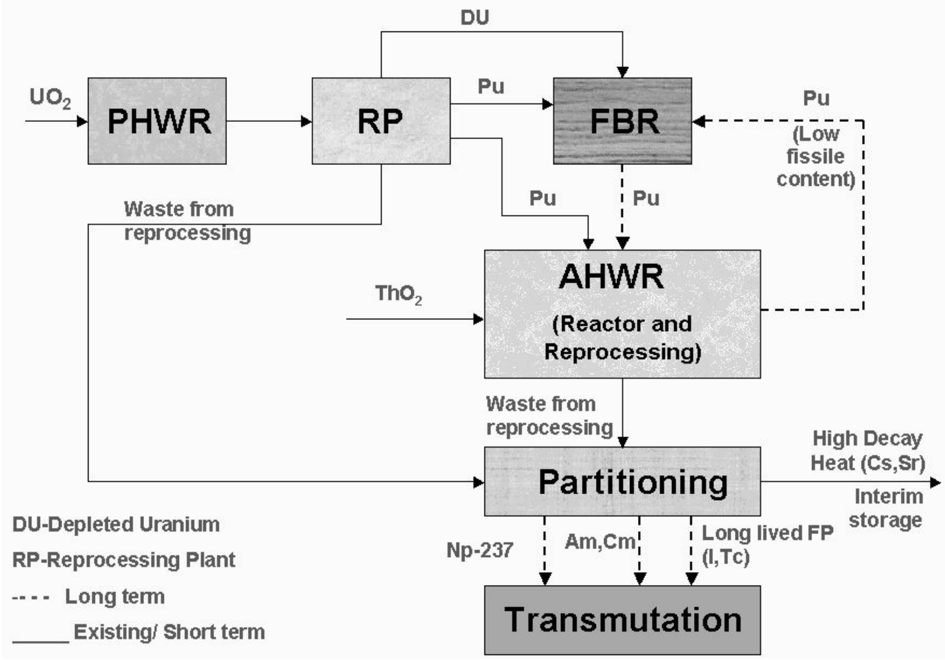

FIG. XI-4. Long-term fuel cycle for AHWR.

The back-end of the AHWR fuel cycle is being developed to work out technologies to cater to the special challenges associated with thorium based fuel. Thorium dioxide, or thoria, is a very inert material and one of the major challenges is to make it dissolve during the spent fuel reprocessing operations. Encouraging results have been obtained in laboratory experiments done at the BARC to find a solution to this problem. A process for three-stream reprocessing of thorium, uranium and plutonium has been developed and demonstrated on a pilot plant scale. This experience will be used to develop a scaled up version. Programmes have been identified to reach a desirable goal of using all the fissionable materials in the fuel cycle very efficiently while minimizing the radiological toxicity of the waste stream.

The fuel cycle facilities (fabrication and reprocessing) for AHWR will be co-located with the reactor at the same site.

XI-1.6. Technical features and technological approaches that are definitive for AHWR performance in particular areas

XI-1.6.1. Economics and maintainability

The advanced safety features and low capital cost of this reactor could facilitate its easy deployment in those developing countries that have limited financial and technological resources.
The main design features of this reactor leading to reduced capital cost per MW(e) are:

- Elimination of main coolant pump and associated other equipment.
- Substitution of steam generators with steam drums of simple design.
- Avoiding the use of costly heavy water in the main coolant system.
- Elimination of heavy water recovery and tritium management systems.
- 100 year design life of the reactor.
- Shop-assembled coolant channel assemblies.
- Direct cycle with moderator heat recovery to achieve better thermodynamic efficiency than in indirect cycle water-cooled systems within the same temperature limits.

The reactor has two important provisions for reducing O&M costs. The first one is the elimination of potential for heavy water leakages from main coolant system (such as in conventional PHWRs) as heavy water exists only in the low pressure moderator system. This saves the recurring cost for heavy water make up. The second important provision is the achievement of an easily replaceable coolant channel design. This not only eliminates the need for a long shutdown for replacement of the coolant channel as the design life of the pressure tube comes to an end, but also enables the re-use of end-fittings and other channel appendages saving considerable cost of their unnecessary replacement too.

The AHWR fuel has a higher core average discharge burn-up of 24 000 MW·d/t as compared to that possible in conventional PHWRs.

XI-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

On-line refuelling, and reprocessing and recycling of both fissile and fertile materials is foreseen to facilitate low consumption of fuel materials. R&D is in progress to examine a feasibility of the recycling of zirconium in clad material after isotopic denaturing.

The use of thorium as fuel practically leads to elimination of the generation of minor actinides from a non-plutonium bearing fuel of AHWR. The major design provisions of AHWR that have a potential to reduce dose levels in the reactor are: on-line refuelling, the use of light water instead of heavy water and hence the reduction in tritium activity, easily replaceable coolant channels, and the accessible design of layout and equipment to simplify in-service inspections.

AHWR is designed to operate in a closed nuclear fuel cycle, and the ^{232}U content in ^{233}U demands a remote and automated production of fuel. A closed fuel cycle involving the remote refabrication of fuel is considered an important attribute of several of the next generation nuclear fuel cycles now being internationally stipulated. With these stipulations, the incremental cost for a planned future incorporation of dry reprocessing in the AHWR fuel cycle is expected to be either low or nil.

XI-1.6.3. Safety and reliability

Safety concept and design philosophy

The emphasis in design has been to incorporate inherent and passive safety features to the maximum extent, as a part of the defence in depth strategy. The main objective has been to establish a case for elimination of a need for evacuation planning following any credible accident scenario in the plant. Another major objective has been to provide a grace period of at least 72 hours for absence of any operator or powered actions in the event of any credible accident scenario.

Provisions for simplicity and robustness of the design

Incorporation of several passive safety features, low power density in the core, good thermal characteristics of the thoria based fuel, and large coolant inventory are some of the important provisions for simplicity and robustness in design.

Active and passive systems and inherent safety features

The main features in these categories are listed below:

Inherent safety features

- Negative void coefficient of reactivity.
- Natural circulation driven heat removal during normal operation and hot shutdown condition.
- Double containment system.
- Four independent ECCS trains.
- Direct injection of ECCS water into the fuel cluster.

Passive systems

- Passive injection of high pressure and low pressure emergency core coolant through the use of one-way rupture disks and non-return valves.
- Passive containment isolation, following a large break LOCA, with a water seal.
- Passive shutdown by injection of poison in the moderator by use of system steam pressure in case of failure of wired systems of SDS-1 and SDS-2.

Structure of the defence-in-depth

Some major highlights of AHWR design, structured in accordance with the various levels of defence in depth are brought out below:

Level 1: Prevention of abnormal operation and failure

(A) Elimination of the hazard of loss of coolant flow:

Heat removal from the core under both normal full power operating condition as well as shutdown condition is by natural circulation of coolant. This eliminates the hazard of a loss of coolant flow.

(B) Reduction of the extent of overpower transient:

The characteristics of AHWR design, which help to achieve this target, are as follows:

- Slightly negative void co-efficient of reactivity.
- Low core power density.
- Negative fuel temperature coefficient of reactivity.
- Low excess reactivity.

(C) Continuous monitoring of plant state:

The condition of all important equipment items and components will be continuously monitored on line. For example, the annulus gas monitoring system is incorporated to monitor any postulated leakage from either the pressure tube or the calandria tube.

Level 2: Control of abnormal operation and detection of failure

The characteristics of AHWR design, which help achieve this objective, are as follows:

- An increased reliability of the control system achieved with the use of high reliability digital control using advanced information technology.

- Increased operator reliability achieved with the use of advanced displays and diagnostics using artificial intelligence and expert systems.
- Large coolant inventory in the main coolant system.

Level 3: Control of accidents within the design basis

The following features contribute to the achievement of this objective:

- Increased reliability of the ECC system, achieved through passive injection of cooling water (initially from an accumulator and later from the overhead GDWP) directly into a fuel cluster through four independent parallel trains.
- Increased reliability of a shutdown, achieved by providing two independent shutdown systems, one comprising the mechanical shut off rods and the other employing injection of a liquid poison into the low pressure moderator. Each of the systems is capable of shutting down the reactor independently.
- Further enhanced reliability of the shutdown, achieved by providing a passive shutdown device operated by steam pressure for injection of poison in case of extremely low probability case of failure of both mechanical shutoff rods and liquid poison shut down system,
- Increased reliability of decay heat removal, achieved through a passive decay heat removal system, which transfers the decay heat to GDWP by natural circulation.
- Large inventory of water inside the containment (about 6000 m^3 of water in the GDWP) provides a prolonged core cooling meeting the requirement of grace period.

Level 4: Control of severe plant conditions, including prevention of accident progression and mitigation of consequences of severe accidents

The following features contribute to the achievement of this objective:

- Use of moderator as heat sink.
- Presence of water in the calandria vault.
- Flooding of reactor cavity following a LOCA.

Level 5: Mitigation of radiological consequences of significant release of radioactive materials

The following features help in passively bringing down the containment pressure and in minimizing any releases from the containment following a large break LOCA:

- Double containment;
- Passive containment isolation;
- Vapour suppression in GDWP;
- Passive containment cooling.

Design basis accidents and beyond design basis accidents

A major objective of the design of AHWR has been to provide a capability to withstand a wide range of postulated events without exceeding the specified fuel temperature, thereby maintaining fuel integrity. The safety analysis of AHWR has identified an exhaustive list of 43 postulated initiating events. The events considered are categorized as follows:

- Decrease in coolant inventory (Loss of coolant accidents).
- Increase in coolant inventory.
- Increase in heat removal.
- Increase in system pressure / Decrease in heat removal.
- Decrease in coolant flow.
- Reactivity anomalies.
- Start-up shutdown transients.
- Multiple failure events.
- Failure of wired shutdown systems and other BDBAs.
- AHWR specific events (Defuelling, refuelling of AHWR channel).

Safety analysis included the analysis of 4 transients due to failure of wired systems of SDS-1 and SDS-2 and reactor shut down effected passively by injection of poison in the moderator by usage of system steam pressure.

The acceptance criteria for all design basis accidents are as follows:

Acceptance criteria

- Core coolability criteria for clad temperature to be less than 1473 K.
- Oxidation criteria of clad surface to be less than 17%.
- Maximum energy deposition in fuel for fuel shattering shall not exceed 200 Cal/gm.
- The maximum fuel temperature anywhere in the core shall not exceed UO_2 melting temperature through out the transient.

Fuel failure criteria

- Maximum energy deposition in fuel for fuel failure shall not exceed 140 cal/gm.
- Maximum clad surface temperature shall be 973 K.
- The radially averaged fuel enthalpy, anywhere in the core, shall not exceed 586 J/g.

Actual calculations indicate that in none of the design basis accident sequences mentioned above the fuel clad temperature exceeds 1073 K.

For the purpose of containment design, a double-ended guillotine rupture of the 600 mm diameter inlet header has been considered as the design basis accident. A large number of other accident scenarios would conventionally fall within the category of beyond design basis accidents (BDBA). However, even in these cases, including a case of a NPP blackout together with failures of both independent fast acting shut-down systems (SDS-1 and SDS-2), it has

been demonstrated that none of the acceptance criteria for design basis accidents as indicated above has been violated.

Provisions for safety under seismic conditions

The AHWR structures, systems and equipment are being designed for high level and low probability seismic events such as operating basis earthquake (OBE) and safe shutdown earthquake (SSE). These are also called S1 and S2 level earthquake respectively. Seismic instrumentation is also planned in accordance with the national and international standards.

Probability of unacceptable radioactivity release beyond the plant boundaries

It is expected that the probability of unacceptable radioactivity release beyond the plant boundaries will be less than 1×10^{-7}.

Measures planned in response to severe accidents

One of the important design objectives for AHWR is to eliminate the need for any intervention in public domain beyond the plant boundaries as a consequence of any postulated accident condition within the plant.

XI-1.6.4. Proliferation resistance

Some of the important technical features of AHWR, which reduce the attractiveness of its spent nuclear fuel material for use in any clandestine nuclear weapons programme, are as follows:

- The content of fissile plutonium in discharged fuel is very low (about 21% of the total Pu inventory).
- Radiation field from ^{233}U is very high due to the presence of ^{232}U.
- In the equilibrium condition, a high fraction of ^{234}U (up to about ten percent) will exist along with ^{233}U in the fuel.

The three factors indicated above also contribute to the prevention and discouragement of the diversion of AHWR based nuclear material for any clandestine nuclear weapons programme.

The technical features that prevent or discourage the production of weapon grade material in AHWR are:

- A self-sustaining design with respect to ^{233}U.
- Operation of reactor with low excess reactivity.

Provision for nuclear material accounting is an inherent part of the AHWR based nuclear fuel cycle, as has been the practice followed in the entire Indian nuclear programme. High gamma activity in the fresh as well as reprocessed AHWR fuel is expected to facilitate its verification with high efficiency and reliability.

XI-1.6.5. Technical features and technological approaches used to facilitate physical protection of AHWR

The double containment based reactor building of AHWR provides a significant barrier to the reactor damage arising out of external impacts.

XI-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of AHWR

AHWR is a concept developed in India with a developing country perspective. The design provides for a large component of local participation during construction. India has a large infrastructure for manpower training, which has been used and will continue to be available for providing specialized training in nuclear related areas to personnel from several IAEA member states, under IAEA programmes.

XI-1.8. Enabling technologies relevant to AHWR and their development status

A list of enabling technologies relevant to the AHWR based plant is given in Table XI-5 [XI-4].

TABLE XI-5. ENABLING TECHNOLOGIES RELEVANT TO AHWR BASED NUCLEAR POWER PLANT

MAIN OBJECTIVES	ENABLING TECHNOLOGIES	STATUS OF DEVELOPMENT
Use of thorium based fuel	Nuclear data for nuclides important for the thorium cycle	A critical facility is under construction
	Remote fuel fabrication technologies	Several options examined; technologies are developed on a demonstration scale
	Three stream reprocessing of fuel containing Pu, Th and U.	Laboratory level process development done
	Dry reprocessing of fuel	Early studies in progress
Negative void coefficient	Tight lattice pitch	Feasibility demonstrated
	Use of a scatterer cum absorber component within fuel cluster	Physical experiments to be done in critical facility
Optimum use of passive systems for core heat removal	Natural circulation driven main coolant system	Several ongoing and future experimental programmes. Large experimental facility (Integral Test Loop, ITL) under construction
	Isolation condensers	
	Large passive heat sink within containment	
	Passive valves	R&D in progress
Low peaking factor	Graded enrichment	Already a part of fuel manufacturing technology development
Increased burn-up of fuel	Provision for reconstitution of fuel	Though not a prerequisite for initial operation of AHWR, the technology is planned to be developed and demonstrated
	Laser isotopic denaturing of zirconium	Not a prerequisite for operation of AHWR. A future R&D programme has been planned
	Provision for on-line refuelling	Large experience with PHWRs exists

MAIN OBJECTIVES	ENABLING TECHNOLOGIES	STATUS OF DEVELOPMENT
Reduced O&M costs	Easily replaceable coolant channels	Rolled joint detachment technology developed
	100 year design life	Mainly achieved through material selection and selection of design approaches that facilitate easy management of ageing
	Nuclear desalination to provide demineralised water	Technology demonstrated
Reduced capital cost per MW installed	Moderator heat recovery	No R&D required
Enhanced safety following LOCA	Passive ECCS (Emergency core cooling system) of enhanced effectiveness	Large experimental facility (Integral test loop, ITL) under construction
	Fluidic device in ECCS	Ongoing experimental programme
	ECCS injection directly into fuel	Ongoing experimental programme
	Passive containment isolation	Large scale demonstration planned in a major facility under construction
	Core submergence	Passive feature, no R&D required
	One way rupture disk	R&D planned
	High reliability non-return valve	R&D in progress
Additional features to achieve low core damage frequency	Steam driven poison injection	Planned to be demonstrated in a large facility

XI-1.9. Status of R&D and planned schedule

The R&D for AHWR is fully supported by Government of India. The design and development of this reactor is mainly done by BARC. The basic design of the reactor and detailed design of its major nuclear systems have been completed [XI-5]. The research, design, and demonstration (RD&D) for AHWR has been and is being performed at the BARC with the financial support of the Government of India. The Nuclear Power Corporation of India Ltd. (NPCIL) has completed a peer review of the design in September 2003. The Indian Atomic Energy Regulatory Board (AERB) has been approached for initially carrying out a pre-licensing safety appraisal of AHWR. Subsequently, the regulatory clearances for different stages of construction, starting from plant siting and procurement of long delivery major equipment, will be progressively sought.

XI-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

First time use of technologies required for the front-end and back-end of the AHWR fuel cycle, and incorporation of several passive systems in the reactor design are the major

innovations in fuel cycle, reactor physics and engineering design of an AHWR based nuclear energy system. While the design is being qualified through extensive theoretical and experimental work, these technologies will be demonstrated for the first time in the first AHWR.

XI-1.11. List of other similar or relevant SMRs for which the design activities are o-going

No other similar SMRs are under design elsewhere.

XI-2. Design description and data for AHWR

XI-2.1. Description of the nuclear systems

Reactor core and fuel design

The circular fuel cluster of AHWR (Figure XI-5) contains thirty (Th-^{233}U) mixed oxide (MOX) pins and twenty-four (Th–Pu) MOX pins, along with a displacer rod at the centre containing dysprosia-zirconia pellets. A ferrule type spacer design, while giving minimum resistance to coolant flow, offers an option to reconstitute the plutonium pins, thus enhancing fuel burn-up and power from thorium. A central flow passage (indicated as water tube in the figure), axially located within the displacer rod, allows direct injection of ECCS water into the fuel cluster.

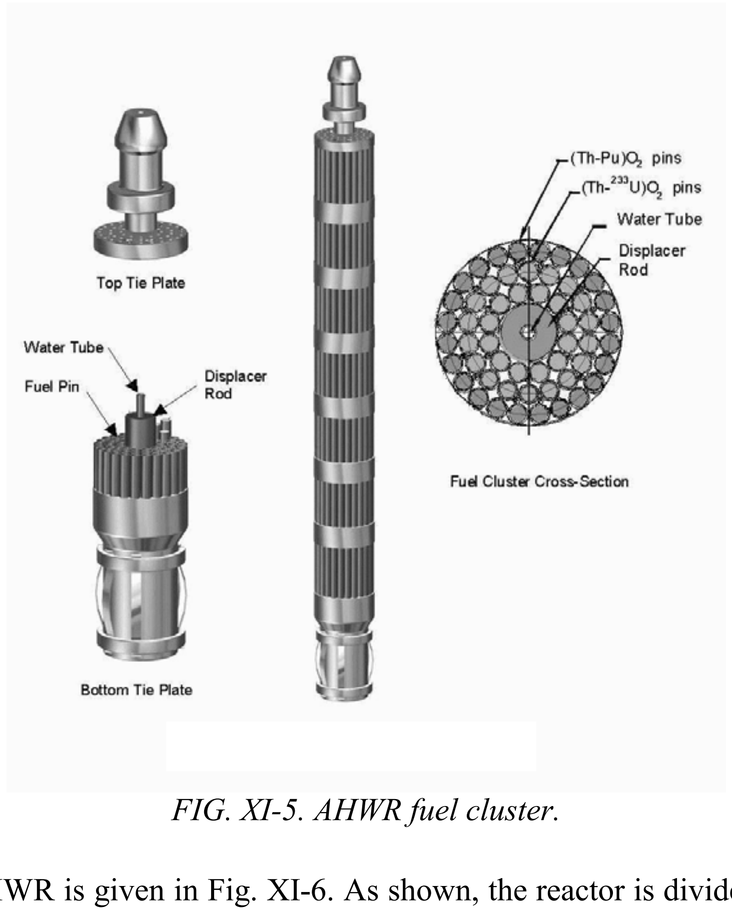

FIG. XI-5. AHWR fuel cluster.

A core map of AHWR is given in Fig. XI-6. As shown, the reactor is divided into three radial burn-up zones. The on-line refuelling scheme can be optimized to achieve a minimization of flux peaking by shuffling fuel across these burn-up zones. The radial flux peaking factor has a value of 1.3. The enrichment of ^{233}U bearing pins in the inner, middle, and outer burn-up regions is maintained, typically, as 3.0%, 3.75% and 3.25% respectively to achieve a reasonably flat radial flux distribution.

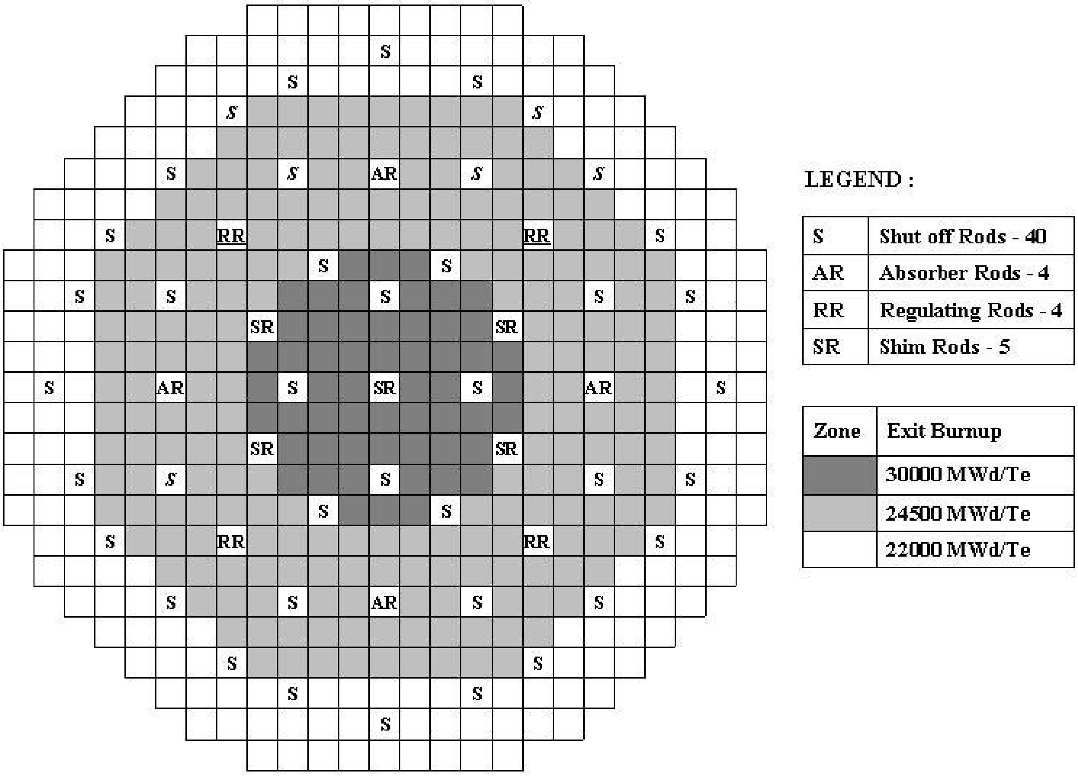

FIG. XI-6. AHWR core map.

Main heat transport system

The function of the MHT system is to remove nuclear heat from the reactor core in a natural circulation mode under reactor operating and shutdown conditions. The design objectives are to achieve a minimum critical heat flux ratio (MCHFR) of at least 1.5 at 120% full power, and to ensure thermal hydraulic stability under all operating conditions.

The MHT system transports heat from fuel rods to a steam drum using boiling light water as a coolant. The MHT system consists of a common circular reactor inlet header from which inlet feeders branch out to the fuel channels in the core. The outlets of the fuel channels are connected to tail pipes. The tail pipes carry steam-water mixture from the individual coolant channels to four steam drums. From each steam drum, four down-comers are connected to the inlet header.

The steam drum is a cylindrical vessel closed at both ends by the hemispherical heads. It incorporates longitudinal partition plates to prevent the mixing of the incoming steam with the sub-cooled feed water. The steam-water mixture (two phases) produced in the reactor core enters the steam drum through the tail pipes at its bottom. The steam separated from the steam-water mixture in four steam drums emerges out through two pipes, each pipe collecting steam from two steam drums. These pipes are connected to the steam chest of the turbine.

The steam turbine of AHWR is designed for a steam quality of 99.75%. During normal operation or in a bypass mode of operation the steam from the turbine exhaust is condensed in a condenser, which rejects the heat to seawater. The condensate is heated in heat exchangers by the moderator system. The feed water temperature is finally raised to 403 K through LP (low pressure) heaters and de-aerators using the steam bled from the turbine. The feed water pumps then pump the feed water into the steam drum where it mixes with the water separated from steam-water mixture.

The heat removal paths of AHWR under various operational states and in LOCA are shown in Fig. XI-7.

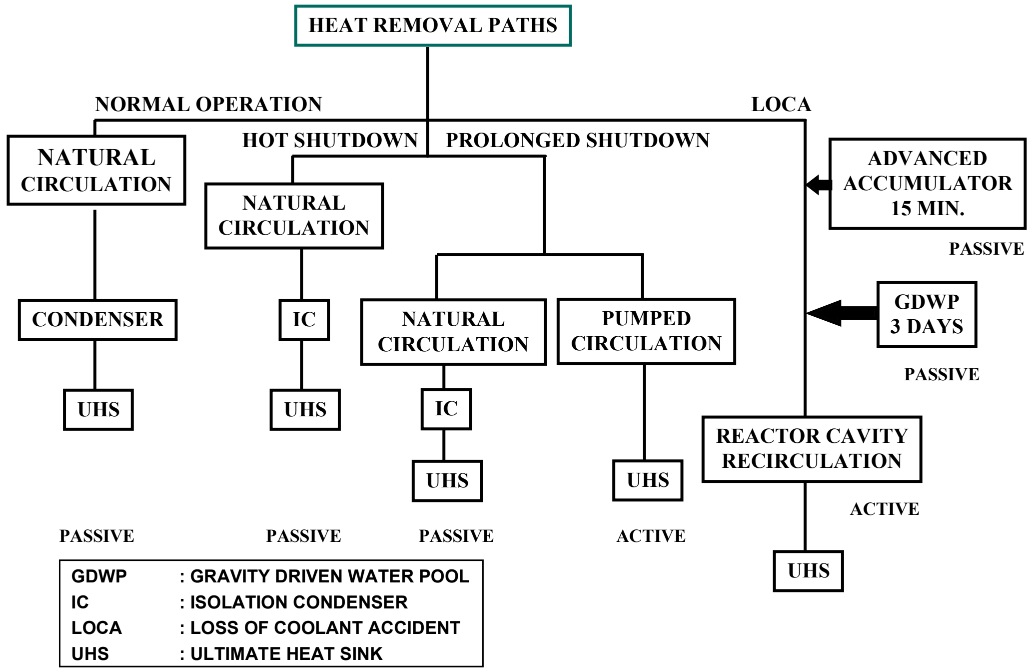

FIG. XI-7. Heat removal paths of AHWR.

XI-2.2. Description of the turbine generator plant and systems

Commercially available conventional equipment will be used in the turbine generator plant.

XI-2.3. Systems for non-electric applications

The demineralized (DM) water make up requirement of a 300 MW(e) AHWR is about 350 m^3/d. An additional requirement of about 150 m^3/d.of fresh water for drinking and other purposes is envisaged. It is therefore proposed to set up a 500 m^3/d low temperature multi effect distillation (LT–MED) seawater desalination plant utilizing low pressure steam from the turbine to meet the DM water requirements. Figure XI-8 provides a schematic flow sheet of the desalination plant of AHWR.

The nuclear desalination system consists of an isolation heat exchanger and a LT-MED desalination plant. The LT-MED desalination plant has four effects of a horizontal tube thin film (HTTF) type evaporator. Low pressure steam is used in the tube side as a heating medium. Feed seawater is sprayed on the outside of horizontal tubes by spray nozzles forming a thin film. Nucleate boiling takes place on the outside of the tubes. This type of boiling is more efficient than pool boiling due to a better heat transfer through thin film of seawater and the absence of any hydrostatic head over the boiling liquid.

This results in high heat transfer coefficients, and the heat transfer is possible with low temperature differences. An isolation heat exchanger is used for isolating a nuclear system from a desalination system. Low pressure steam from AHWR is used in isolation heat exchanger for generating steam for use in the desalination plant. The steam from the isolation heat exchanger is used in the first effect as a heating medium. Vapours generated in the effects are reused in the succeeding effects as heating media. The effects are maintained at a lower pressure than the preceding effect. The vapours generated in the last effect are condensed in

the final condenser. Condensate form the first effect is recycled back to the isolation heat exchanger. Condensate from all other effects and the final condenser is collected as product water.

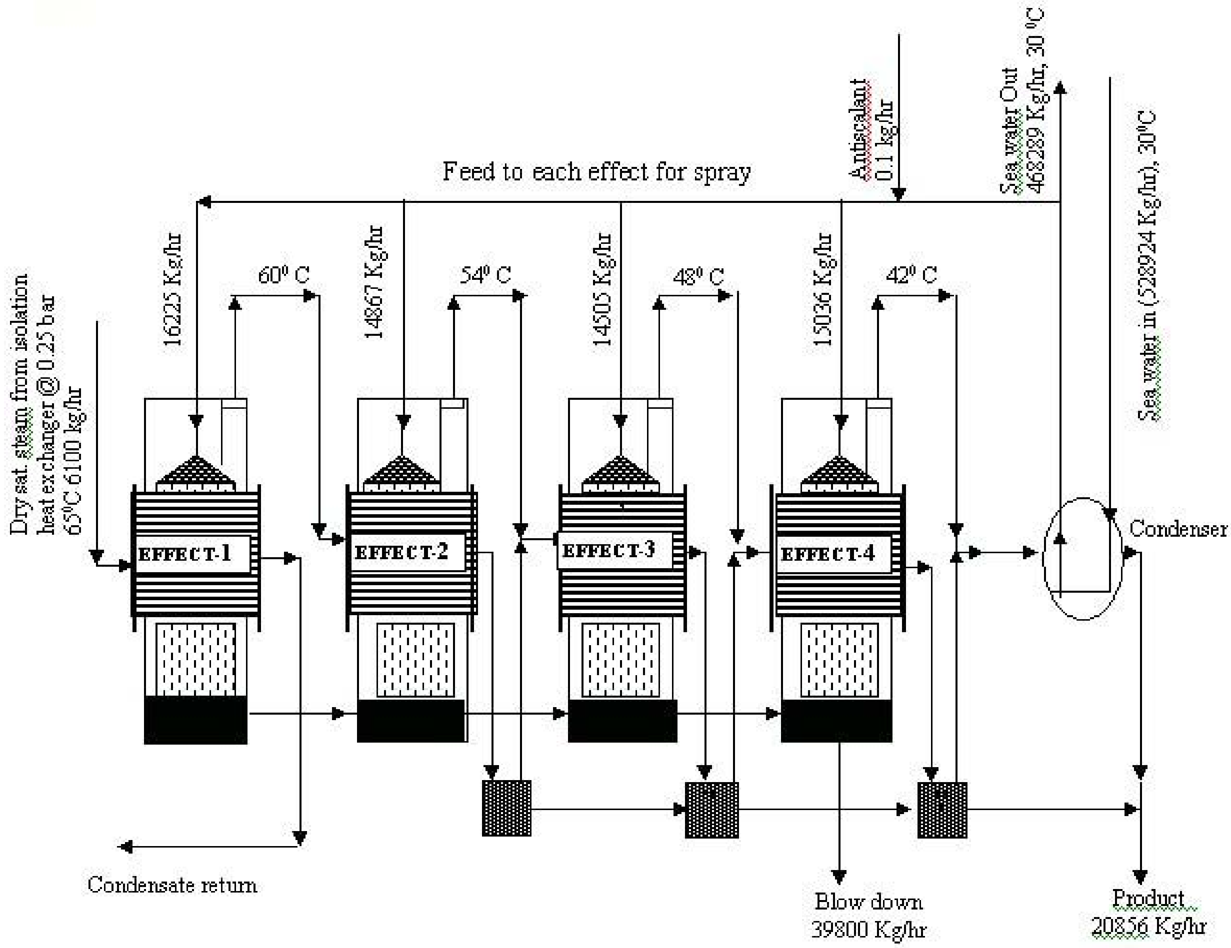

FIG. XI-8. Desalination plant of AHWR.

XI-2.4. Plant layout

The plant layout of Advanced Heavy Water Reactor (AHWR) is optimized for meeting various functional needs, as well as safety, radiation zoning, piping and cabling requirements, erection and construction requirements, and access and security considerations.

Floor plans of various buildings have been optimized based on the equipment and systems within, on functional needs, space utilization, radiation zoning, accessibility, serviceability, maintenance, transportation and ventilation aspects, and on access philosophy.

The considerations governing reactor building and containment layout are as follows:

- Minimization of primary containment volume.
- Effective utilization of space in the annulus between primary and secondary containments.
- Unrestricted entry to the reactor top for on-power fuel handling and transfer operations.

- Adequate shielding against radiation and preventing a spread of radioactive contamination during normal and accidental conditions.
- Provision of a large water inventory at a suitable height, capable of supporting a number of passive systems.
- Submergence of the reactor core under water before the exhaustion of the inventory of emergency core cooling system.
- Easy access to a maximum number of the equipment for operation and maintenance during normal and accidental situations.
- Fire prevention and control.

The general plant layout of AHWR, as proposed for a coastal site, is indicated in Fig. XI-9.

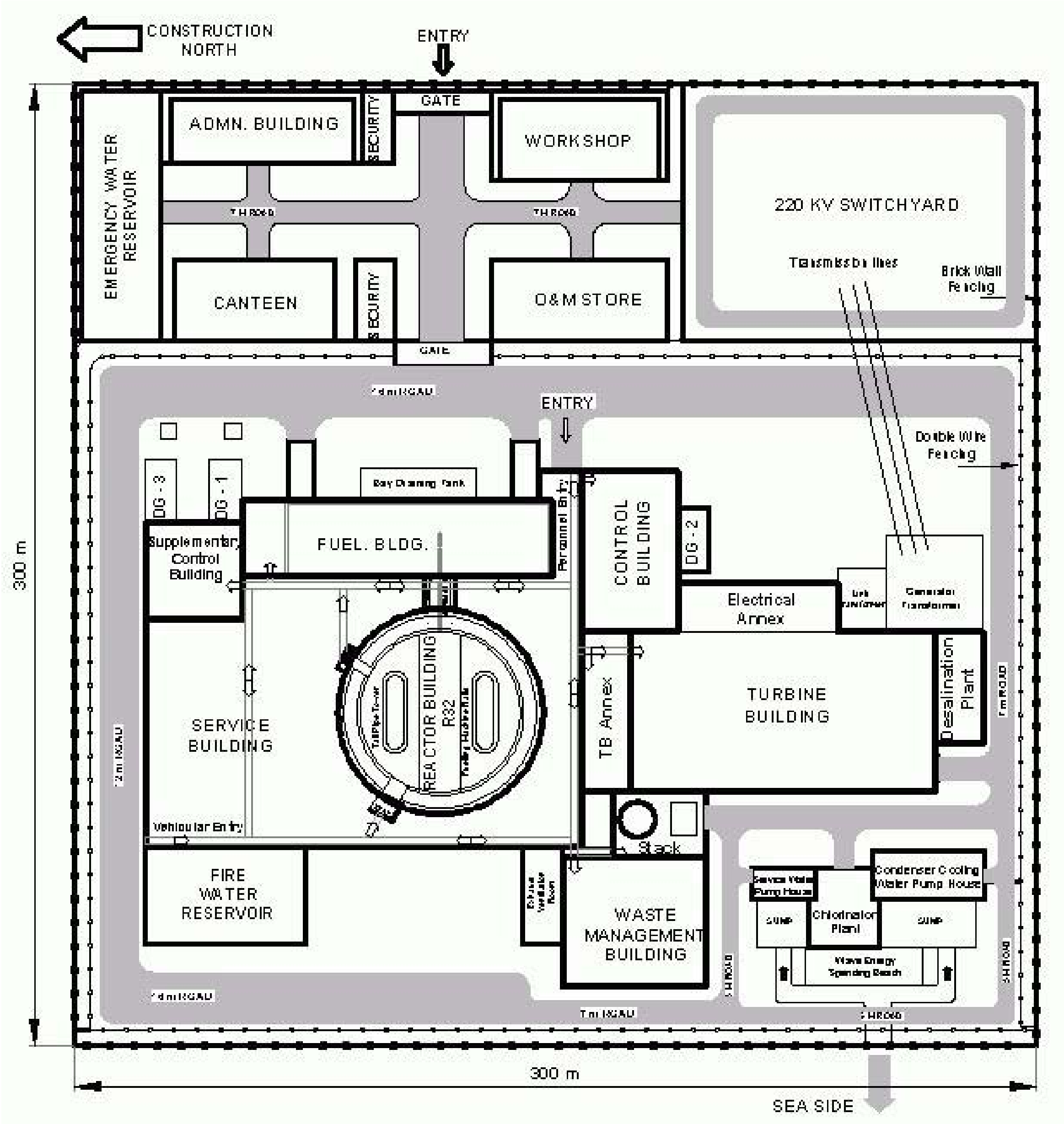

FIG. XI-9. Proposed layout of an AHWR based nuclear power plant for a coastal site.

The main plant buildings contained in an area of 300 m × 300 m are reactor building, control building, supplementary control building, service building, turbine building, turbine building annex, DG (diesel generator) building, fuel building, waste management building, fire water reservoir, emergency water reservoir, work-shop, desalination plant, condenser cooling water

& service water pump-houses, administrative building, stack & radiation monitoring rooms and switchyard.

The reactor building is located towards a seaside to facilitate the transportation of over dimensional consignments from the sea route, as well as land route. The fuel building, which accommodates fuel storage facilities, is adjacent to reactor building. The main control building housing Safety Group-1 C&I hardware is functionally and physically independent from the supplementary control building housing Safety Group-2 C&I hardware. The service building providing service facilities like access control, workshops, laboratories, plant services is encompassing the reactor building. The access to various buildings in nuclear island is provided from a single entry point in the service building as indicated in the sketch. The waste management building, the stack, the exhaust systems and seawater intake/discharge systems are located towards a seaside. The plant layout is designed with a dual-layered security arrangement to guard against an unauthorized entry. The nuclear island is isolated from administrative facilities by double-wire fencing. Provision is made for a space around buildings for the construction activities, including the space needed for movement and erection of tower cranes during construction. In order to restrict the contamination in various areas and to facilitate the control of the spread of radioactive contamination, the plant is divided into four zones of different contamination potential, and the zoning philosophy is maintained in individual floor plan designs as well.

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ANNEX XII

REACTOR FACILITY FOR DISTRICT HEATING WITH ATMOSPHERIC PRESSURE IN THE PRIMARY CIRCUIT (RUTA-70)

NIKIET, Russian Federation

XII-1. General information, technical features and operating characteristics

XII-1.1. Introduction

The RUTA-70 is a Russian abbreviation for the reactor facility of 70 MW(th) for district heating with atmospheric pressure in the primary circuit.

Single-purpose nuclear heating plants (NHP) are regarded as the possible sources of heat supply for district heating in Russia.

A challenging design in this area is the reactor facility RUTA designed specifically as a heat source for district heating systems.

A reliable and simple design of the RUTA is ensured, first of all, by the lack of excess pressure in the primary coolant (in the reactor pool). NHPs with such reactors are characterized by inherent safety features and could be located in the immediate vicinity of the heat users.

NHPs with the RUTA reactors are suitable for heat supply to urban areas with a population in the range of 10 to 100 thousand people.

In 1990, the Research and Development Institute of Power Engineering of the Rosatom of the Russian Federation (NIKIET) with the participation of IPPE (Obninsk, Russia) has developed the conceptual design of the RUTA plant with a 20 MW(th) output. In 1992, based on this design, the Russian organizations NIKIET, IPPE, VNIPIET and the Mining Institute of the Kola Research Centre of the Russian Academy of Sciences (MI KRC RAS) prepared a feasibility report called ‘Designing of the underground NHP with the RUTA reactor for district heating of Apatity, Murmansk region’, where it was suggested to use two reactors of 20 MW(th) each for the identified purpose.

When discussing these proposals with the town council of Apatity and with the authorities of the Murmansk region, it was considered feasible to develop the design of the RUTA NHP for Apatity and several smaller towns with the unit power output of the reactor increased to 50-60 MW(th).

The feasibility report, ‘Underground NHP with the RUTA reactors of 4 × 55 MW(th) output for district heating in Apatity, Murmansk region’, was prepared in 1994 by joint efforts of specialists from the NIKIET, IPPE, VNIPIET, MI KRC RAS and the Apatity heat and power plant (HPP).

The design of the NHP with the RUTA took the first prize in the competition of small sized NPP designs ‘SSNPP-91’ organized in 1991–1993 under the initiative of the joint stock company ‘Small power installations’ and under the auspices of the Nuclear Society of Russia.

The RUTA reactors were included in the list of facilities proposed by the Russian Federation as sources of thermal energy for NHPs and seawater desalination plants [XII-1].

In 2001, the State Research Centre of the Russian Federation IPPE proposed conducting a feasibility study on upgrading the district heating in Obninsk, Kaluga region, by constructing a NHP with the RUTA reactor. This proposal was approved by a decision of the Board for the programme of Obninsk development as a ‘science town.

The principal stakeholders of the RUTA are:

- The town council of Apatity, Murmansk region, Russian Federation.
- The town council of Obninsk, Kaluga region, Russian Federation.
- Rosatom - Nuclear Energy Agency of the Russian Federation.

XII-1.2. Applications

The reactor facility is an integral part of the single-purpose nuclear heating plant (NHP) intended for generation and transfer of thermal energy to a district heating system.

In addition, thermal energy generated by the reactor may be used for seawater and brackish water desalination provided that special-purpose equipment is added to the mix of NHP components.

XII-1.3. Special features

The reactor facility is a part of the ground based nuclear heating plant (NHP) designed similarly to research reactors of the pool type.

XII-1.4 Summary of major design and operating characteristics

Installed capacity

The NHP with the RUTA-70 is designed for heating water and supplying the users via an intermediate circuit with at least 70 MW(th) or 60 GCal/h of low-grade thermal energy as hot water.

Mode of operation

The NHP RUTA-70 may operate both in the base load and load follow modes.

Load factor/availability

Availability factor: ~ 95%.

Load factor:

- Base load - up to 95%[1]
- Load follow - 70%[2]

The major design characteristics of the reactor are presented in Table XII-1.

[1] Peak load is covered by another energy source.
[2] NHP covers peak load and the load factor depends on NHP location (for instance, in the Moscow region - 46%, in Polar regions - 70%).

TABLE XII-1. MAJOR DESIGN CHARACTERISTICS OF RUTA REACTOR

CHARACTERISTIC	VALUE
Type of fuel	UO_2
Fuel enrichment with uranium 235, %	3.0
Type of coolant (moderator)	Water (water)
Core height, cm	140
Equivalent diameter of the core, cm	147.2
Diameter and thickness of fuel cladding, mm	9.1 (0.69)
Fuel cladding material	Zr alloy
Number of loops in the intermediate (secondary) circuit	3

Simplified schematic diagram

RUTA is the name of the reactor facility of a pool type with a water-water reactor of 70 MW(th) and an integral layout of the primary circuit components.

The following approach has been adopted for designing the major reactor units and choosing the flow path geometry:

- In the power range from a minimum level and up to ≈50% $N_{Nom.}$, as well as in cooldown modes (both under normal and emergency conditions), the reactor core will be cooled and heat will be transferred from the primary heat exchangers to the secondary coolant by natural convection of the primary coolant.
- In the range of elevated heat loads (from ≈50% $N_{Nom.}$ to 100% $N_{Nom.}$), it is envisaged to start up pumps and uprate the reactor power at the expense of a higher flow rate in the internal circuit.

Forced convection is provided by axial pumps installed at the bypass of the natural convection circuit, upstream of the inlet to the downcomer.

Pumps in the reactor circuit and the use of forced coolant convection in the modes of operation at high power levels (50–100% $N_{Nom.}$) augment the primary coolant flow rate, increase temperatures in the downcomer and at the same time reduce heating in the core.

The reactor core is placed in the lower part of the reactor vessel, the vault, in the shell of the chimney section. Frontal and cross-sectional views of the RUTA-70 are given in Figs XII-1 and XII-2.

The distributing header is placed in the upper part of the shell of the chimney section. Pipelines of water supply to the primary heat exchangers are connected to the header from both sides. Heat exchangers are located in the upper part of the pool.

At the outlet of each heat exchanger, cooled coolant is supplied to the pipeline of an individual section connected to the intake header of the reactor downcomer, which at the same time is used as the suction header of the pump. Under the natural convection operating mode, the forced convection circuit is isolated by a check valve installed at the pipeline of the individual section. Using the same valve, the natural convection circuit can be isolated. The schematic diagram of this unit is given in Fig. XII-4. Downstream of the heat exchangers, coolant is directed via the suction header to the circulation pump that supplies water to a

group of heat exchangers located at one side of the pool. Water is returned from the pump head via the supply header. Pumps are connected to the bypass line of the natural convection circuit and are placed in a special compartment in close vicinity to the reactor pool (see Figs XII-2 and XII-3).

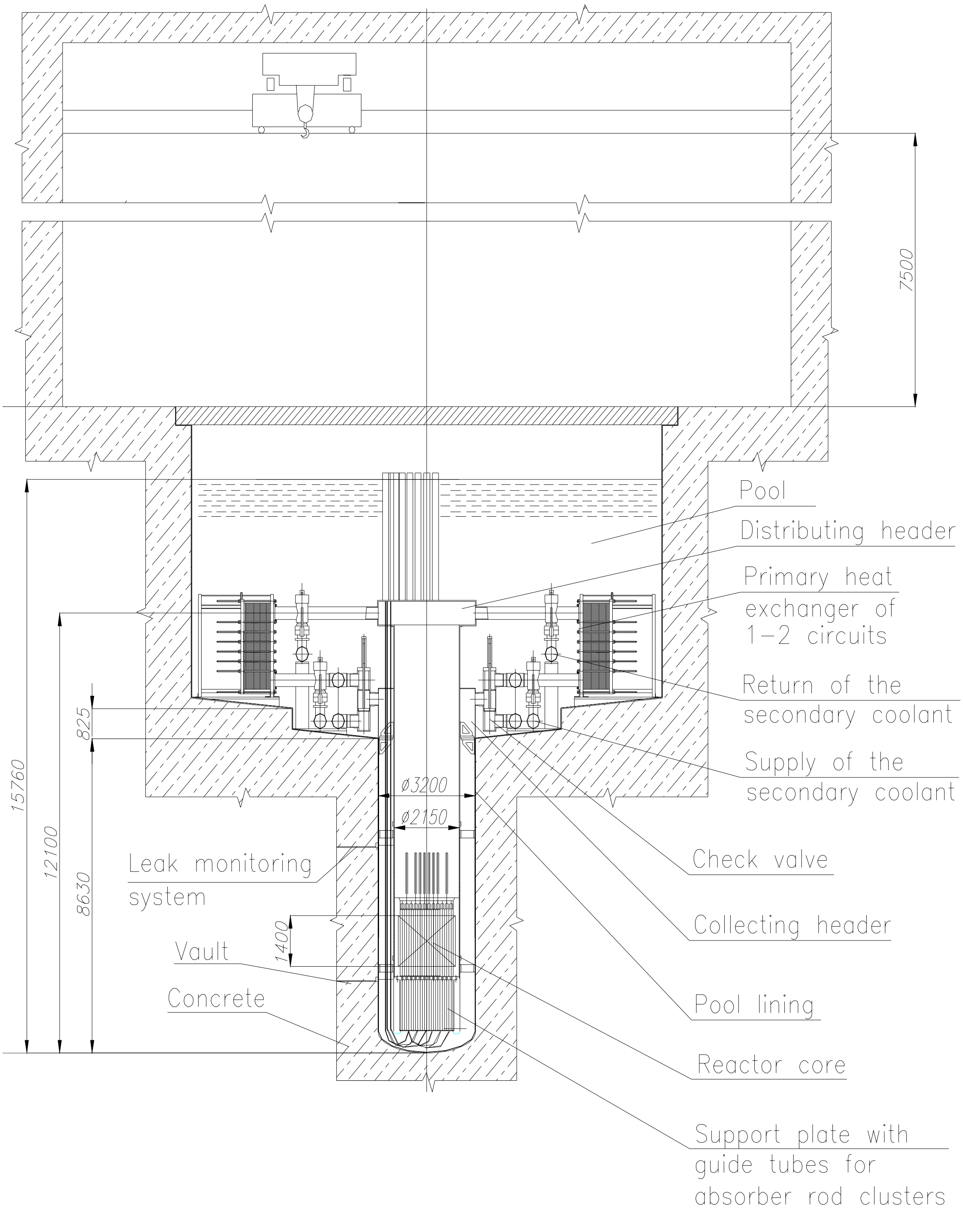


FIG. XII-1. RUTA-70 reactor (frontal view).

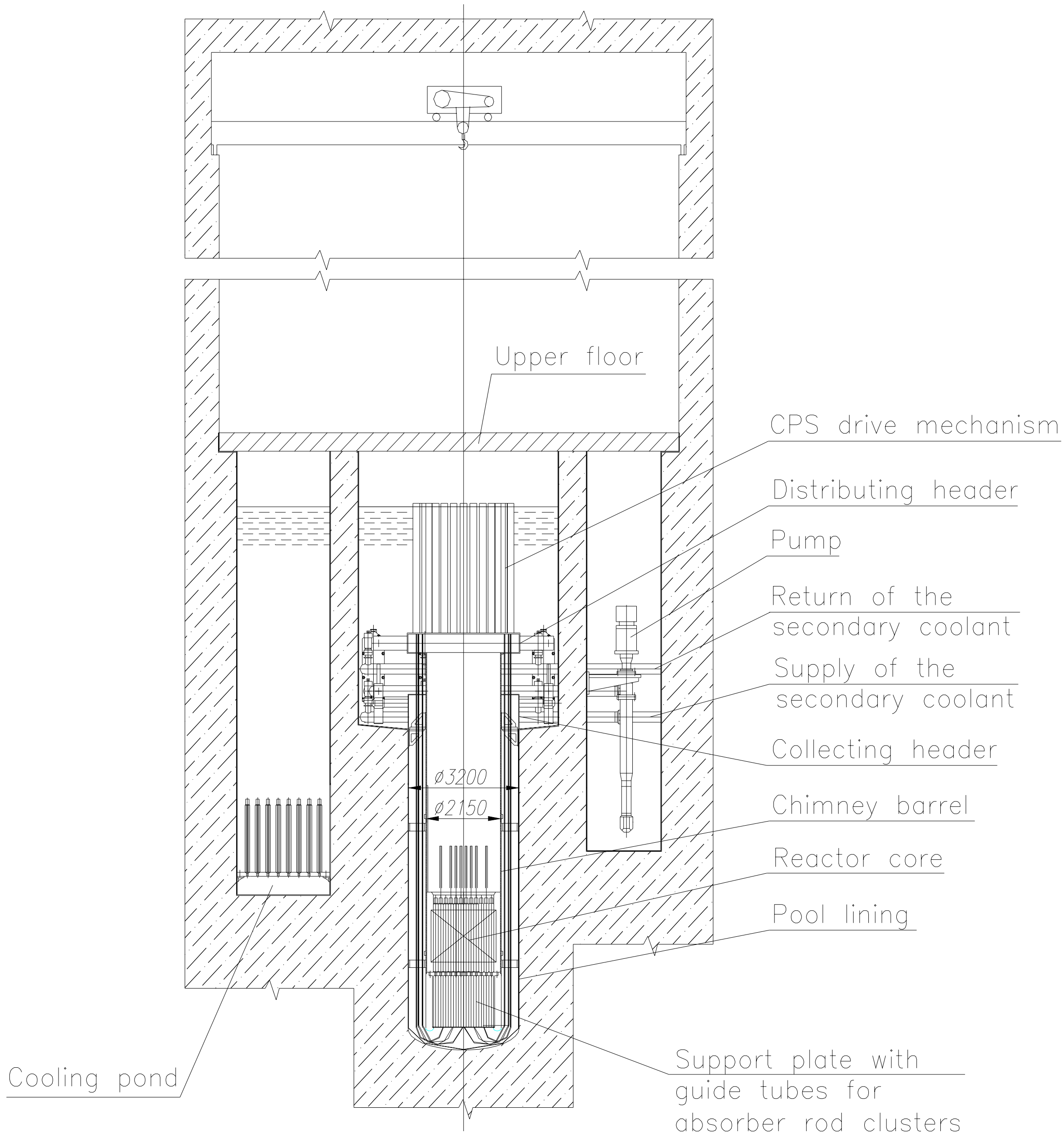


FIG. XII- 2. RUTA-70 reactor (cross-sectional view).

Pumps are located to allow easy access for inspection or replacement. It is envisaged that two axial pumps would be installed in the primary circuit. The pump characteristics are as follows: head 9.8 m, flow rate 1150 m^3/h, and rotational speed 1470 rpm.

A special compartment (see Figs XII-2 and XII-3) where the cooling pond is located is arranged near to the reactor pool. The cooling pond is designed to accommodate 126 fuel assemblies, the size of a full core plus its discharged part (one third of the full core) plus a 10% margin for damaged fuel assemblies.

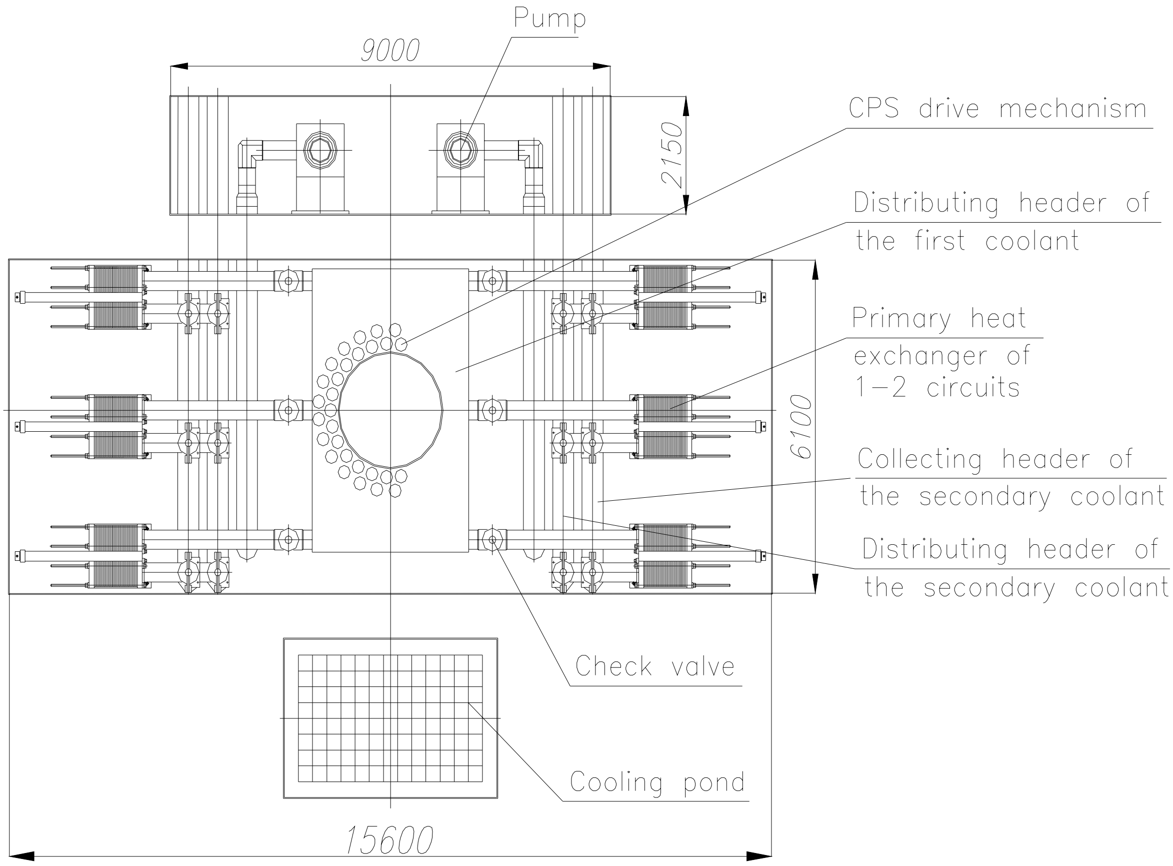


FIG. XII-3. RUTA-70 reactor (top view).

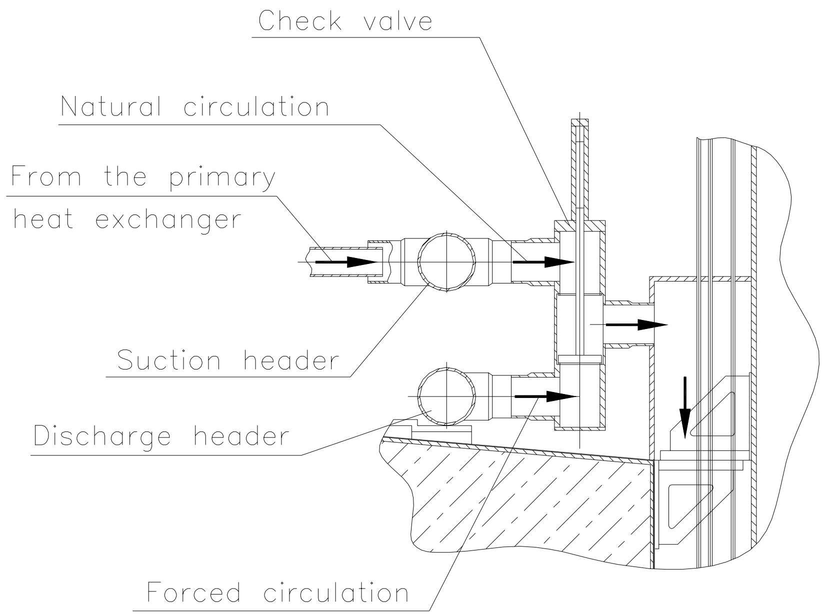


FIG. XII-4. The schematic diagram of circulation mode change-over.

The protective flooring composed of slabs is installed above the reactor pool to avoid possible damage to the primary components from external impacts. To prevent gas and vapour penetration to the reactor hall from the upper part of the reactor, joints of the protective slabs are gas-tight.

Neutron-physical characteristics

Neutron-physical characteristics of the reactor core of the RUTA-70 are shown in Table XII-2.

TABLE XII-2. NEUTRON-PHYSICAL CHARACTERISTICS

CHARACTERISTIC	VALUE
Type of fuel	UO_2
Reactivity margin in the cold state (20°C), %ΔK	7.6
Effect of heating from the cold state to the nominal parameters at average fuel temperature, % ΔKeff	(From 20°C to 90°C) T_U= 465°C -1.9
Effect of steady-state poisoning, % ΔK_{eff}	-1.9
Burn-up reactivity swing, % ΔK_{eff}	3.8
Void reactivity effect K_{Void}, ΔK/%void	$-1.2\ 10^{-3}$
Power reactivity coefficient K_{Power}, ΔK/ $N_{Nom.}$	$-2\ 10^{-4}$
Peaking factors: - In fuel assemblies - Spatially in the core	 1.5 2.3

Reactivity control mechanism

In the RUTA-70 design, the following mechanisms of reactivity control and power flattening are applied:

- Optimization of refueling.
- The use of burnable poison.
- Profiling of fuel loading.
- Movable control rods.

Cycle type and thermodynamic efficiency

The energy conversion cycle is not applied.

Thermal-hydraulic characteristics

The basic thermal-hydraulic characteristics of the RUTA-70 are provided in Table XII-3.

TABLE XII-3. BASIC THERMAL-HYDRAULIC CHARACTERISTICS

CHARACTERISTIC/ DESCRIPTION	VALUE
Circulation type: Up to 50% of nominal power 50÷100% of nominal power	Natural Forced
Temperature in the core (inlet/outlet), °C	75/101
Flow rate, kg/s (t/h) Under natural circulation (up to 50%$N_{Nom.}$) Under forced circulation (50÷100%$N_{Nom.}$)	309 (1114) 642 (2313)
Under nominal parameters: Average (maximum) coolant temperature in the core, °C Average (maximum) fuel cladding temperature, °C Average (maximum) fuel temperature, °C	87(109) 90(115) 470(785)
DNBR, °C	20

Maximum/average discharge fuel burn-up

Average burn-up of discharged fuel: 28.7 MW·day/kg U.
Maximum burn-up of discharged fuel: 37 MW·day/kg U.

Fuel lifetime/period between refuellings

Fuel lifetime: 2400 effective days.
Period between refuellings: 800 effective days.

Mass balances/ flows of fuel materials

Specific uranium consumption (3% enrichment): 1.45×10^{-3} g/kW h or 1.69 g/GCal.

Uranium consumption (3% enrichment) on an annual basis (at capacity factor 0.7): 623.2 kg/year

The specific consumption of natural uranium (at 0.25% of ^{235}U content in depleted uranium) is 8.66×10^{-3} g/kWh or 10.07 g/GCal.

The annual consumption of natural uranium (at capacity factor 0.7) is 3771 kg/year.

Design basis lifetime for reactor core, vessel and structures

The design lifetime of the NHP RUTA-70 and major equipment is 60 years.

The period of uninterruptible operation of the reactor equipment without attendance for maintenance and repair is no less than one year.

Design and operating characteristics of systems for non-electric applications

In the frame of the programme of the IAEA coordinated research on nuclear desalination in 1999-2000, NIKIET specialists have performed activities on the following subject, "Application of Russian small size reactors as the energy source for nuclear desalination facilities (NDF): Optimization of the reactor interfaces with desalination plants, the performance and economic indices of NDF". In particular, proposals have been made with regard to using the RUTA as a source of heat for evaporation based desalination plants; the schematic diagram of the reactor and desalination equipment interface has been developed and the economic evaluation of the NDF RUTA has been performed based on the "desalination economic evaluation program" (DEEP).

The NDF RUTA could be a reliable, safe and ecologically friendly plant due to technical features of the reactor. Only the standard domestic distillation desalination plants (DDP) are applied for desalination. These plants have been modified to operate in the range of heat parameters provided by the reactor. The output of a NDF equipped with a single RUTA-70 reactor is about 30 000 m^3/day of distilled water. The estimated price of one cubic meter of fresh water varies from US $0.9 to US $1.3 depending on local economic conditions. The schematic flow diagram of the desalination plant with the RUTA reactor is given in Fig. XII-10.

Economics

Tables XII-4 and XII-5 present the capital costs for construction of the NHP RUTA-70 and annual operating costs, respectively.

TABLE XII-4. CAPITAL COSTS FOR CONSTRUCTION OF THE NHP RUTA-70

CATEGORY OF COSTS	VALUE, US$ MILLION
Costs of NHP equipment	9.27
Cost of NHP equipment assembly	1.11
Construction costs	7.41
Other expenditures	1.78
TOTAL	19.57
Unit costs, US $/kW	279.60

TABLE XII-5. ANNUAL OPERATING COSTS FOR THE NHP RUTA-70

CATEGORY OF COSTS	VALUE, THOUSAND US$/YEAR
Wages including charges	166.8
Write-off for amortization for renovation and overhaul	587.1
Current repairs	489.2
Costs for general plant needs	299.1
Average annual fuel costs	230.3
TOTAL	1772.5

Duration of the NHP construction period is 3 years, including adaptation of the standard design to local site conditions.

XII-1.5. Outline of fuel cycle options

The standard fuel cycle option for the RUTA-70 NHP is a once-through fuel cycle with uranium dioxide fuel.

The alternative fuel cycle option is a once-through cycle with cermet fuel (micro-particles of fuel in a metallic matrix).

If decided, the standard fuel reprocessing method as used for VVER type reactors could be applied. In this, fuel reprocessing can be made centralized.

According to the design of the NHP RUTA-70, spent fuel assemblies should be stored in the cooling pond for 3 years after discharge from the reactor core and then transported to the fuel reprocessing plant without further long-term on-site storage.

XII-1.6. Technical features and technological approaches that are definitive for RUTA-70 performance in particular areas

XII-1.6.1. Economics and maintainability

Simplicity of design, low parameters, space-saving heat exchanging equipment and the use of passive systems contribute to the reduction of capital costs for construction of the NHP RUTA-70.

Due to low capital costs for construction and low operating costs, the NHP RUTA-70 could become an efficient option both as a nuclear heat plant and a source of thermal energy for thermal distillation. Regarding the Russian Federation, the marketing research has identified a number of regions where the district heating of the urban areas could be provided by NHPs with RUTA type reactors [XII-2], among them:

- The Arkhangelsk region.
- The Murmansk region.
- The Sverdlovsk region.
- The Kirovsk region.
- The Amursk region.
- The Primorski krai.

The O&M costs can be reduced due to the limited number of required operating personnel and low costs for the repair and consumables.

Fuel costs are reduced due to the low enrichment at a relatively high burn-up, due to the application of special structural materials characterized by low neutron absorption, as well as through the optimized pattern of partial refuellings.

XII-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

To improve fuel utilization, the reprocessing of spent nuclear fuel for re-use as MOX fuel in the RUTA-70 could be considered. This would require a transfer to the ¼ refuelling cycle. In addition to this, the RUTA-70 fuel has low specific power along with a substantial burn-up and a relatively long lifetime.

The reactor is designed to provide easy maintenance. There are no preconditions for equipment contamination and only partial refuelling once every three years is required.

The fuel reprocessing method similar to that applied to the VVER-440 reactors could be used.

XII-1.6.3. Safety and reliability

Safety concept and design philosophy

The safety concept of the RUTA-70 is as follows:

- Maximum use of inherent safety features.
- Consistent implementation of defence in depth strategy.
- The safety systems are designed to perform their functions based on such principles as:
  - Multichannelling.
  - Redundancy.
  - Spatial and functional independence.
  - Application of a single failure criterion.
  - Diversity.

The safety concept of the RUTA-70 is based on maximum possible use of inherent and passive safety features to secure a high level of the reactor safety. The acceptable safety level is understood as the level when the total effective annual internal and external radiation dose (caused by the effect of the nuclear energy source plus natural background levels) for the population under normal operating and emergency conditions does not exceed the natural background dose specified by sanitary regulations. In hypothetical situations postulated as low-probability beyond design basis accidents the additional radiation dose should not exceed the natural background dose.

In general, the advantages of tank or pool type reactors are as follows:

- No excess pressure in the reactor tank; this excludes an accident with an instantaneous rupture of the reactor tank and cessation of heat transfer from the core due to dryout.
- The high heat accumulating capability of water in the reactor pool ensures slow changing of coolant parameters during transient and emergency conditions and reliable heat transfer from the fuel elements, even if controlled heat transfer from the reactor is not available. Fuel temperatures are moderate.

In addition to the above properties that are inherent for reactors of a given type, technical solutions in the design of the RUTA-70 ensure the following important reactor safety features:

- The stabilizing reactivity feedback caused by negative reactivity temperature coefficients for the fuel and coolant as well as the void reactivity coefficient mean that heating up the core structural components, including fuel, or water boiling in the core would eventually result in a spontaneous reduction or self-limitation of the reactor power irrespective of the positions of control rods, including scram rods.
- In the range of operating modes at power levels from minimum controlled level to 50% $N_{Nom.}$, heat is transferred from the core by natural coolant circulation in the reactor tank to ensure reliable cooling of the nuclear fuel and passive cooldown of the reactor in shutdown.

- The possible natural circulation of coolant in the secondary circuit would provide residual heat transfer from the shutdown reactor and passive cooling of the facility in case of a plant blackout.

In the process of developing the pool type reactor for energy application, a focus has been made on consistent implementation of the defence-in-depth concept. This concept includes several levels of devices, systems and operations that provide execution of the basic safety functions:

- Reactor power control.
- Ensured heat transfer from the core.
- Confining radioactive products.

To ensure the function of reactor power control, two independent systems based on diverse drive mechanisms are provided for reactor shutdown. One system acts as an accident protection system, while the actuated second system is designed to provide guaranteed sub-criticality for an unlimited period of time and to be able to account for any reactivity effects including those in accidental states. Either system can operate under the failure of a minimum of one rod with maximum worth. In case of loss of power to the reactor control and protection system (RCP), all rods of this system can be inserted in the core under the effect of gravity.

Heat is transferred from the reactor core via two independent channels and each is designed by the three-circuit scheme, i.e. pressure gradually rises starting from the primary circuit via the secondary (intermediate) circuit and to the third (network) circuit, thus the possibility of fission product ingress to the coolant is excluded.

In emergency situations, residual heat is transferred by natural circulation of the coolant in the reactor tank and in the secondary circuit (in case of a blackout). Heat is removed from the secondary circuit convectors using the air system for emergency cooldown (ASEC) under forced or natural circulation of air in the convector compartments. Direct-acting devices open air louvers of the ASEC passively.

Provisions for simplicity and robustness of the design

Low operating parameters, low specific fuel power, a limited number of in-tank components, use of passive systems and inherent safety features contribute to simplicity and robustness of the RUTA-70 design.

Active and passive systems and inherent safety features

The RUTA-70 uses mostly passive systems to perform safety functions; they are listed below:

- An air system for emergency cooldown (ASEC).
- The reactor emergency protection system (insertion of the control rods in the core by the effect of gravity).
- A secondary circuit overpressure protection system.
- An overpressure protection system for air space in the reactor pool.
- A system for protection from external impacts.

The system for emergency makeup of the primary and secondary circuits is an active system.

Structure of the defence-in-depth

The defence in depth principle prescribes the following barriers to confine radioactive products within the specified limits:

- Fuel matrix.
- Fuel cladding.
- A reactor tank with a leak-tight cover and leak-tight heat exchange surfaces of the primary heat exchangers.
- Leak-tight boundaries of the reactor hall compartments and systems connected to the primary circuit.

Design basis accidents and beyond design basis accidents

The analysis of possible failures in the components of the RUTA-70 plant and human errors has been performed to define the list of the initiating events that may potentially result in occurrence of accidents. The initiating events were defines as follows:

- Inadvertent withdrawal of some control rods moved by the individual drives:
  - In a start-up from "cold" state.
  - On the power level.
- Stoppage of pump 1 during operation with forced circulation.
- Stoppage of pumps in the secondary circuit.
- Loss of integrity of the reactor pool.
- Failure to actuate the check valve during switching on/off the primary pump.
- Failures in the system of water purification in the reactor pool.
- Failures in the system of ventilation of the above-pool space of the reactor.
- A leak in the supply (return) pipelines of the primary pump located in a dry compartment.
- Loss of integrity of heat transfer surfaces of the primary heat exchangers.
- Loss of integrity of heat transfer surfaces of the network heat exchangers.
- Loss of integrity of pipelines in the system of water purification in the reactor pool.
- Loss of integrity of the system of ventilation of the above-pool space of the reactor.
- Loss of integrity of the secondary circuit inside (outside) the reactor.
- Loss of off-site power for a long period.
- Disconnection of network pumps or the loss of integrity of the circuit downstream of the isolation valves, resulting in pressure decreases in the heat network below the pressure of the secondary circuit.
- Loss of integrity of the network circuit in the line between the network heat exchanger and isolation valves.
- Accident in the process of fuel handling operations:
  - Falling of a fuel assembly into the reactor during refuelling.
  - Falling of the shielded cask with a fuel assembly onto the floor of the reactor hall.
  - Failures during release of the RCP rods.
  - Failures of handling and transportation equipment in the process of refuelling and fuel transportation.
  - Leak in the cooling pond.
- External initiating events:
  - Airplane crash.
  - Impact of a shock wave.
  - Seismic impact.

- Disturbances in the auxiliary systems resulting in violation of the safe operation limits.
- Disturbances of the reactor gas regime that may cause the hazard of an explosion.
- Fire in the compartments accompanied by failures of safety-related components and trains located therein.
- Flooding of the compartments and possible failures of electrical equipment of the safety systems in damaged compartments.

The initiating events presented above may be accepted for the design basis accident analysis however, it should be considered that not every situation related to these initiating events could progress into an accident with the release of radioactive substances over the operating limits specified in the design.

Below is the list of initiating events for the most severe beyond design basis accidents, covering the situations when the initiating event is aggravated by concurrent multiple failures of the safety systems or human errors that may result in accident progression beyond the design limits:

- A reactivity surge under complete failure of all reactivity control.
- A non-recoverable leak in the reactor tank.
- Loss of all controlled trains for heat removal from the reactor.
- An accident during handling and transport operations with discharged fuel resulting in a mechanical damage of the fuel elements.
- The accumulation and explosion of explosive mixtures in the air space of the reactor pool.

The beyond design basis accidents are characterized by a very low probability of the initiating event to occur however, they are considered to determine the ultimate dangerous states of the reactor and to identify the possible consequences.

Provisions for safety under seismic conditions

Seismic stability of the reactor and its systems is rated at no less than magnitude 8 per the MSK-64 scale.

The layout of the RUTA-70 and its equipment and structures meet as far as possible the following basic principles of the seismic resistant building code:

- Heavy equipment is mostly located at low elevations.
- The symmetry conditions are met and a more uniform distribution of mass and rigidity is achieved.
- Components and devices of category I of seismic resistance are placed in the compartments of category I of seismic resistance and are provided with aseismic fastenings.
- Major components of the RUTA-70 are fixed in place with high reliability and if required, provided with aseismic fastening to avoid displacement or impact to reactor components and building structures.

Probability of unacceptable radioactivity release beyond the plant boundaries

The probability of radioactivity release beyond the plant boundary is estimated to be less than 1×10^{-7} 1/year.

Measures planned in response to severe accidents

The computational analysis of beyond design basis accidents has revealed that:

- In case of multiple failures in the reactivity control systems and devices, safety can be ensured by self-control of reactor power (boiling - self-limitation of power), i.e. through the inherent safety features of the reactor.
- If all controlled trains of heat removal (i.e. all loops of the secondary circuit) are lost, heat losses via the external surface of the reactor pool to the surrounding environment (ground) are considered an additional train. Residual heat is accumulated in the pool water. The transient of pool water heatup in the aqueous mode before the onset of boiling takes several days. As soon as boiling starts, steam goes to the reactor hall where it is condensed (passive condensing facilities are provided). A reactor boil-off without makeup takes 18 to 20 days. Upon completion of this period residual heat is balanced by heat transfer to the ground. Core dryout is avoided. Moderate temperatures not exceeding the design limits characterize fuel elements. The maximum temperature of reinforced concrete during this accident will be 90°C; the adjacent layer of ground will be heated up to 50–70°C.
- An unrecoverable leak in the reactor pool that may cause dryout of the core can be excluded due to the concrete vessel which is designed to withstand external events, including maximum design basis earthquake and water filtration to the ground in a beyond design basis accident.
- Radiolysis hydrogen in the air space of the reactor may be accumulated in the event of failure of the gas system. The probability of an explosion in this situation is rather low because of the increased vapour content of the air however, the reactor cover has been provided with rupture devices to release the medium from beneath the reactor cover to the reactor hall.

XII-1.6.4. Proliferation resistance

The use of low-enriched fuel (3% of ^{235}U), a relatively long fuel lifetime and an insignificant concentration of fissile isotopes in the spent nuclear fuel make it an unattractive choice for use in nuclear weapon programmes.

Marking of the fuel assemblies, automatic verification and registering during fuel transfer facilitate nuclear material accounting and verification.

The core design and technical features of operation and refuelling contribute to the prevention of an undeclared production of direct-use material.

XII-1.6.5. Technical features and technological approaches used to facilitate physical protection of RUTA-70

Inherent safety features are incorporated in the RUTA-70 design to protect the reactor from internal impacts. The containment (protective enclosure) assures the protection from selected external impacts.

XII-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of RUTA-70

All services related to the fuel cycle are rendered by the organizations within the jurisdiction of the Russian Federation Agency for Atomic Energy (Rosatom). The Russian legislation at

the moment does not stipulate leasing of nuclear heat plants (NHPs) to private owners; the NPP operators in Russia can only be state-owned enterprises.

In case of selling of a NHP RUTA-70 abroad, its operation would be regulated by legislation of the buyer.

XII-1.8. List of enabling technologies relevant to RUTA-70 and status of their development

The list of enabling technologies and status of their development are summarized in Table XII-6.

TABLE XII-6. ENABLING TECHNOLOGIES OF RUTA-70 AND STATUS OF THEIR DEVELOPMENT.

DESIGN AREA	ENABLING TECHNOLOGY	STATUS OF DEVELOPMENT
Advanced fuel	Use of cermet fuel	Under investigation
Control rod drives	Use of U-type control rod drives	Under investigation
Reduction of capital costs	Use of more efficient (space-saving) heat exchangers	R&D efforts are not needed.
Reduction of operating costs	Changeover to operating life of 100 years.	R&D efforts are needed to select the appropriate materials.

XII-1.9. Status of R&D and planned schedule

The RUTA-70 nuclear heat plant does not require support by national R&D programmes because proven technical solutions and equipment are used to the maximum extent.

The RUTA-70 is supported by the programme of development of Obninsk, Kaluga region, as the “science town”.

The Russian institutes and organizations involved in the RUTA-70 design are NIKIET (leader), IPPE, VNIPIET, MI KRC RAS, and ‘Atomenergoproekt’ (AEP).

The time period required for deployment of the RUTA-70 is estimated as ~ 3 years in the Russian Federation and ~ 4÷5 years outside the Russian Federation.

At present, the design stage of the RUTA-70 is that of preliminary design.

Simplicity of the RUTA-70 design and the use of available equipment proven by series production and operation at existing nuclear installations reduce the costs and duration of R&D.

The R&D completed include:

- Optimization of core design (type of fuel, enrichment, fuel burn-up, fuel lifetime, thermal physics of the core, etc.).
- Optimization of the natural circulation circuit under normal operating conditions and nominal parameters.
- Primary heat exchanger design development and justification, by calculations.

The R&D still required and planned include:

- Investigation of the circuit for heat removal from the core (under natural and forced circulation) in normal and emergency conditions.

- The investigation and justification of the serviceability of U-type control rods and drive mechanisms.
- Justification and study of the possibility to use alternative fuel, e.g. cermet fuel.
- Verification and certification of the computer codes.

In the Russian Federation, licensing of a design is performed in five stages, with Gosatomnadzor of Russia (GAN RF) being involved in all of them. These stages are:

- A declaration of intent.
- The justification of investments.
- The design (feasibility study) of a NHP.
- Working drawings, construction.
- Start-up activities and commissioning.

The first two stages for the RUTA plant were completed in 1992.

The R&D costs needed to deploy a NHP with the RUTA-70 are estimated to be around US $4 million.

XII-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

The cogeneration plant RUTA is designed for district heating in small towns and communities.

The RUTA reactor can also be used as a heat source for desalination plants. These applications may require the plant to be located in the immediate vicinity of populated areas.

A demonstration prototype is needed to justify reliability of a NHP for its construction within the town limits, as well as to optimize the use of a new type of fuel, etc. The prototype (or a pilot unit) of the NHP RUTA-70 is envisaged for construction on the site of the Federal State Unitary Enterprise SSC IPPE in Obninsk.

XII-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

SLOWPOKE could be mentioned as the reactor similar to the RUTA-70.

XII-2. Design description and data for RUTA-70

XII-2.1. Description of the nuclear systems

Reactor core and fuel design

The reactor core is located in the lower part of the vessel-vault and is composed of 91 hexagonal fuel assemblies with fuel rods of the VVER-440 type containing uranium dioxide fuel in a zirconium cladding. The structural material of the fuel assemblies is zirconium alloy. Fuel assemblies are placed in a triangular lattice with the pitch of 147 mm and form a regular and symmetrical system. The reactor core height is 1400 mm; the equivalent diameter of the core is 1420 mm.

The fuel assembly is placed in a zirconium casing of 144 mm “turn-key” size and contains 127 fuel elements, including 12 fuel elements in which Gd_2O_3 is integrated with UO_2 as a burnable absorber. Each fuel assembly has six guide tubes for the absorber rods (AR) of the

RCP cluster system and a central instrumentation tube (for in-core sensors or for the mechanism moving the RCP cluster). The pitch of the fuel element lattice and the ARs in a fuel assembly is 12.2 mm.

In the core there are 42 reactor control and protection system (RCP) rods composing two shutdown systems with diverse actuators. One of these systems intended specifically for core emergency protection (EP) includes 12 rods. The second shutdown system performing the concurrent functions of shutdown and control includes a group of 6 automatic regulators (ACRs) and 4 groups of a total of 24 control rods, for remote manual reactivity control (manual rods - MCRs). In response to the scram signal, all control rods of the second shutdown system also perform the functions of emergency protection. Before bringing the reactor to power level, 12 EP control rods are withdrawn from the core and not involved in power control operations. MCRs are used to compensate for relatively fast reactivity changes such as heatup and xenon poisoning of the reactor therefore, most of MCRs will be withdrawn under nominal operating parameters. MCRs and scram rods may take the intermediate position in the core performing the functions of power control and forming the radial power profile. The slow transients of reactivity change (such as burn-up of fuel and burnable poison) are also controlled by the group of ACRs plus the required groups of MCRs. Due to a rather large total worth of RCP rods, they follow the special-purpose logic of movement with accounting for rods combining into the groups.

The location map of fuel assemblies and RCP rods is given in Fig. XII-5; a cross-section of the fuel assembly is shown in Fig. XII-6.

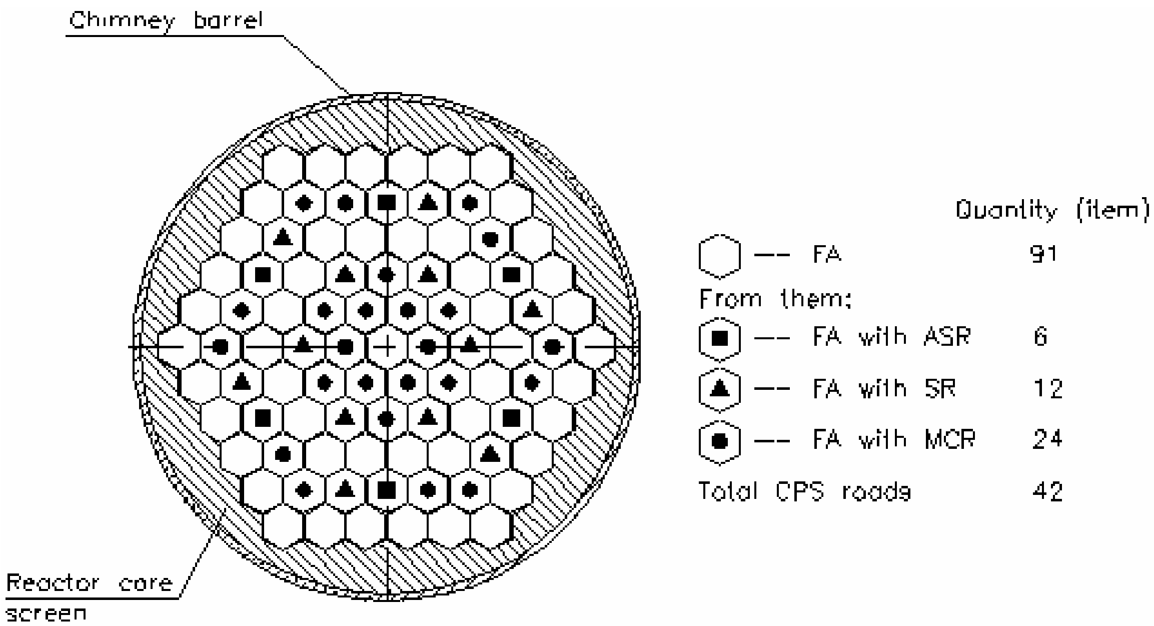


FIG. XII-5. The allocation map of core fuel assemblies showing the positions of control rods.

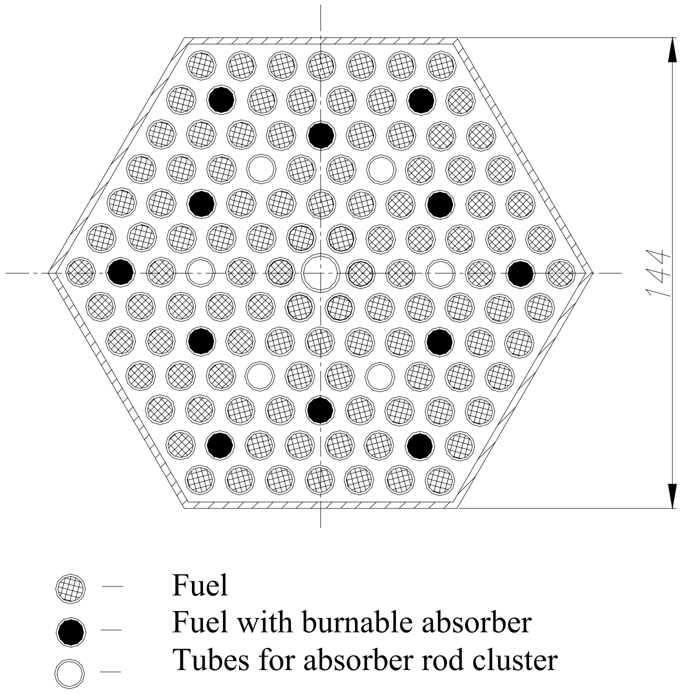


FIG. XII-6. Cross-section of fuel assembly.

Reactor control and protection system (RCP) actuator

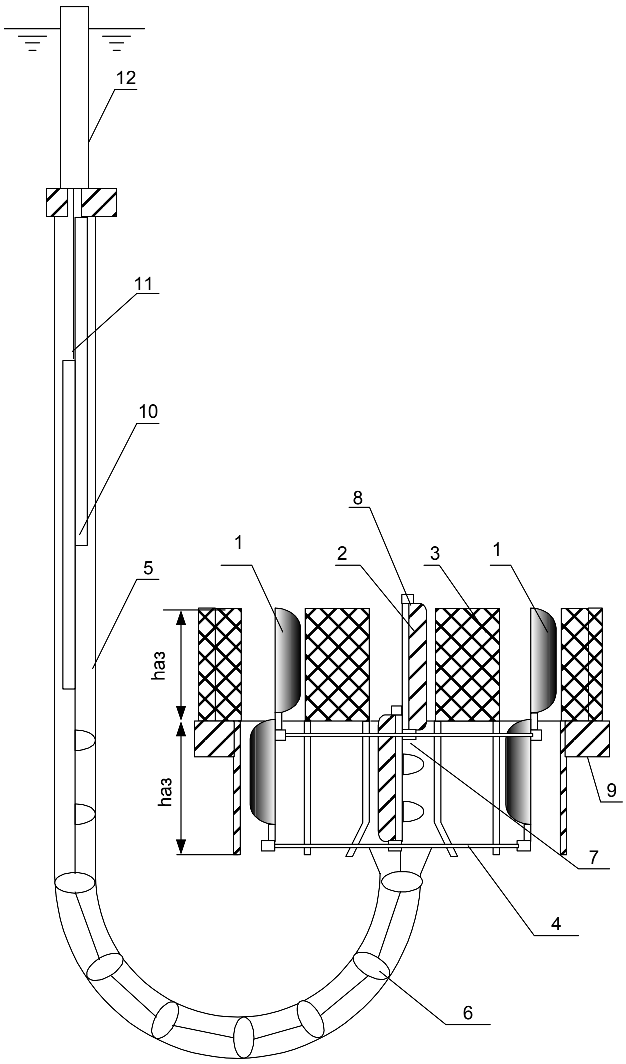
RCP actuators based on two diverse principles of action have been chosen for the RUTA-70:

- A multi-position mechanical RCP actuator for automatic (ACR) and manual control rods (MCR).
- A two-position hydrodynamic RCP actuator for scram rods (SR).

The multi-position mechanical RCP actuator is shown in Fig. XII-7.

The working parts are designed with similar absorber rods combined in a cluster assembly. The absorber rods of a cluster assembly are mounted and moved inside six guide tubes of the fuel assembly. The load-bearing components including the gripper of the tie-rod (catching tie-rod and the working part), the release of a gripper and the gripper of the working part are located in the central tube of the fuel assembly.

The guide tube is an embedded structure in the downcomer of the core cooling circuit with the lower part (below the support plate) bent at an 180° angle with the radius equal to half the distance from a vertical guide tube to the location of a fuel assembly with the control rod. A flexible tie-rod of the push-pull type in the guide tube is removable.



To the left of the guide tube axis - position "rods withdrawn"; to the right of the axis - position "rods inserted

1-absorber rods	2-load-bearing component
3-fuel assembly	4-traverse of the cluster
5-pipeline for supply of control action	6-flexible tie-rod
7-gripper of tie-rod	8-release of tie-rod gripper (gripper of the cluster)
9-support plate	10-load
11-drive tie-rod	12-drive

FIG. XII-7. Multi-position mechanical RCP actuator.

The drive is on the support metal structure (on top of a distributing header, see Figs. XII-1, XII-2 and XII-3).

A load intended to push the cluster is attached to the drive rod connected with the flexible tie-rod in case of a loss of power to the drive.

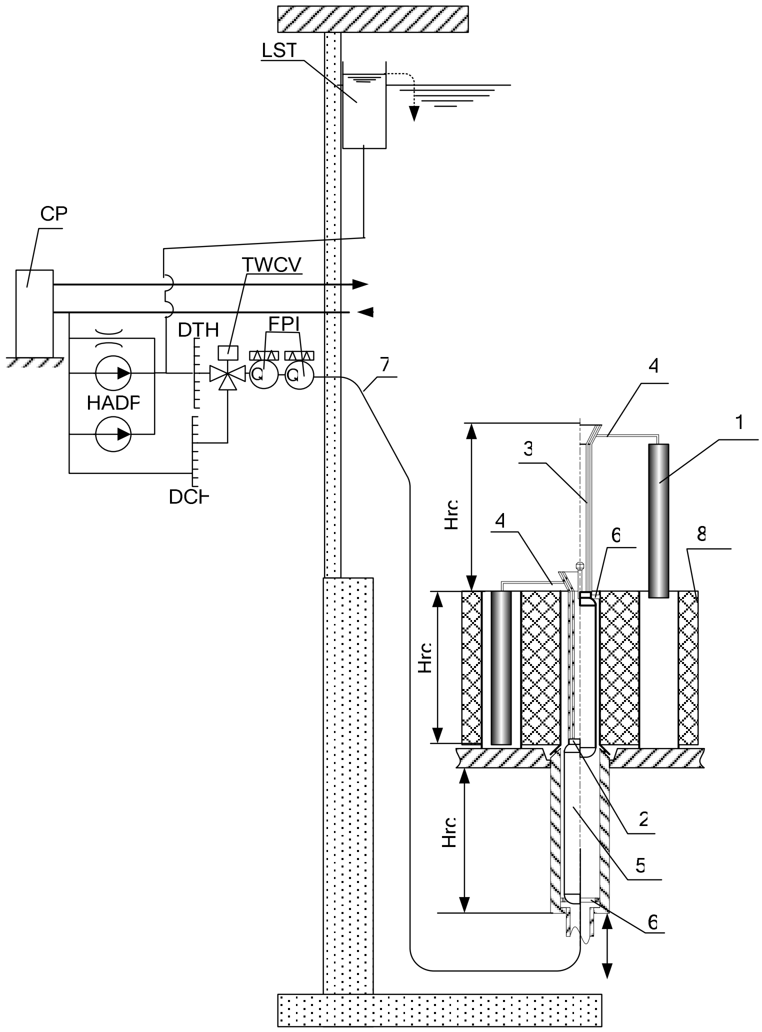
In case of upward motion of the MMA drive, the cluster is removed from the core. In case of downward motion of the drive (and in the event of loss of power), the gravity effect produced by the load will push the cluster inside the core.

Thus, the passive principle of absorber insertion into the core without any active force is achieved in case of loss of power.

The flexible tie-rod and the cluster assembly are engaged remotely. The gripper may be opened/closed by an axially moving release device driven by the refuelling machine.

The hydrodynamic RCP actuator (HAD) is shown in Fig. XII-8.

The working part is designed with absorber rods combined in a cluster. ARs are placed in six guide tubes of the fuel assembly. The central tube of the fuel assembly is used as the channel for a hoisting rod (HR) in the core. The lower part of the central tube is sealed to prevent leaks from the channel with the HR (the requirements for leak-tightness of this joint are not strict, however, leaks should not have impact on the parameters of cluster motion). A rest with an orifice is arranged in the upper part of the channel.



To the right of the axis - position "rods withdrawn"; to the left of the axis - position "rods inserted"

CP - circulation pump	HADP - HAD pump
DTH - distributing header	DCH - discharge header
TPFR - three-way crossover valve	FPI - flow position indicator of the cluster
LST - level stabilizing tank	1-absorber rod
2-gripper	3-release of gripper
4-traverse of the cluster	5-load-bearing rod

6-seat 7-U-type supply tube 8-Fuel assembly

FIG. XII-8. Two-position hydrodynamic CPS actuator.

The hoisting rod (HR) has a cylindrical shape. The end elements of the HR together with the channel rests form a primary transducer of the flow indicator of the cluster position in the core.

Above the HR is the load-bearing structure for engaging the cluster. The load-bearing structure includes a gripper (engaging the HR and cluster), a bar with a cluster clutch and a release for the gripper control.

Water to move the cluster is supplied to the HR channel via the supply pipeline at the pressure of the level stabilizing tank.

The level stabilizing tank is an overflow type installed beneath the water level in the pool. The tank is supplied by pumps of the HAD. Three-position flow regulators are installed in the supply pipelines to change the direction of water flow in the channel (for the cluster insertion or withdrawal). To monitor the position of a cluster, flow position indicators (flow meters of forward and backward fluid directions) are installed in the core.

Main heat transport system

Heat is transferred from the secondary circuit to:

- The third (network) circuit - under normal power operation of the reactor.
- The network circuit or service water supply system with cooling towers - under scheduled cooldown modes.

Under emergency cooldown of the shutdown reactor heat is transferred from the secondary circuit to:

- The third (network) circuit, if it remains serviceable.
- Air in converters of the air system of emergency cooldown, if the third circuit is isolated or in the event of NHP blackout.

The passively actuated air system for emergency cooldown (ASEC) provides residual heat removal to the ultimate heat sink (atmospheric air). ASEC is envisaged for reactor cooldown in case of loss of auxiliary power. Each loop of the secondary circuit has an ASEC subsystem (train); the ASEC is connected at the bypass line of the network heat exchangers.

Cooldown is provided by the combined exhaust and intake ventilation of the compartments, as well as passively, under natural coolant circulation in the secondary circuit and natural circulation of atmospheric air. The ASEC consists of water-air heat exchangers designed as a system of parallel ribbed pipelines (air convectors) located in separate compartments of the secondary circuit, outside the reactor leak-tight area, at elevations that ensure head for natural circulation sufficient for residual heat removal and reactor cooldown in long-term loss of power.

The scheme of heat removal from the core is given in Fig. XII-9.

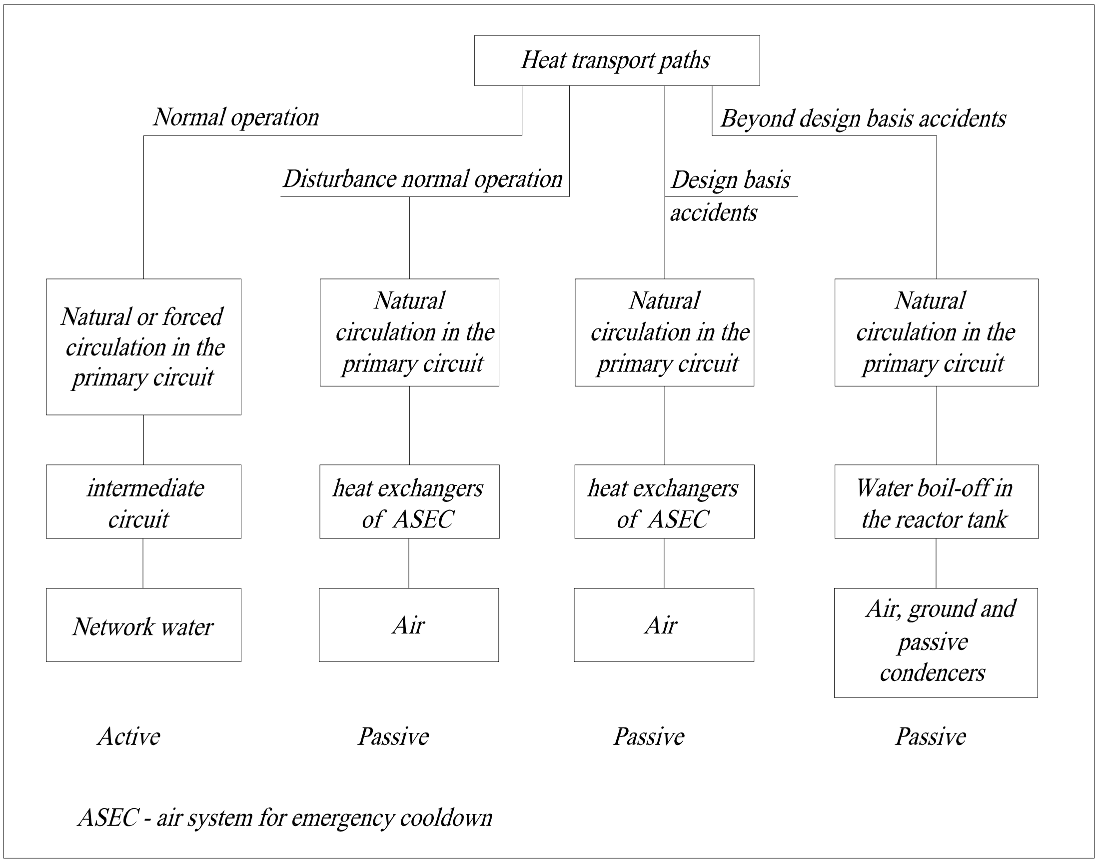


FIG. XII-9. Heat removal paths from RUTA-70 core in normal operation and in accidents.

Intermediate circuit

The secondary (intermediate) circuit removes heat from the reactor and transfers it to the third circuit of users, i.e. the heat network.

The secondary circuit consists of two autonomous loops each of which includes three primary heat exchangers, the secondary (network) heat exchangers, three circulation pumps and an air pressurizer.

The secondary circuit pipelines within the boundaries of the reactor pool are made of corrosion resistant steel. To reduce costs, pipelines outside the reactor pool may be manufactured of bi-metallic tubes (i.e., tubes of carbon steel clad from the inside with corrosion-resistant steel).

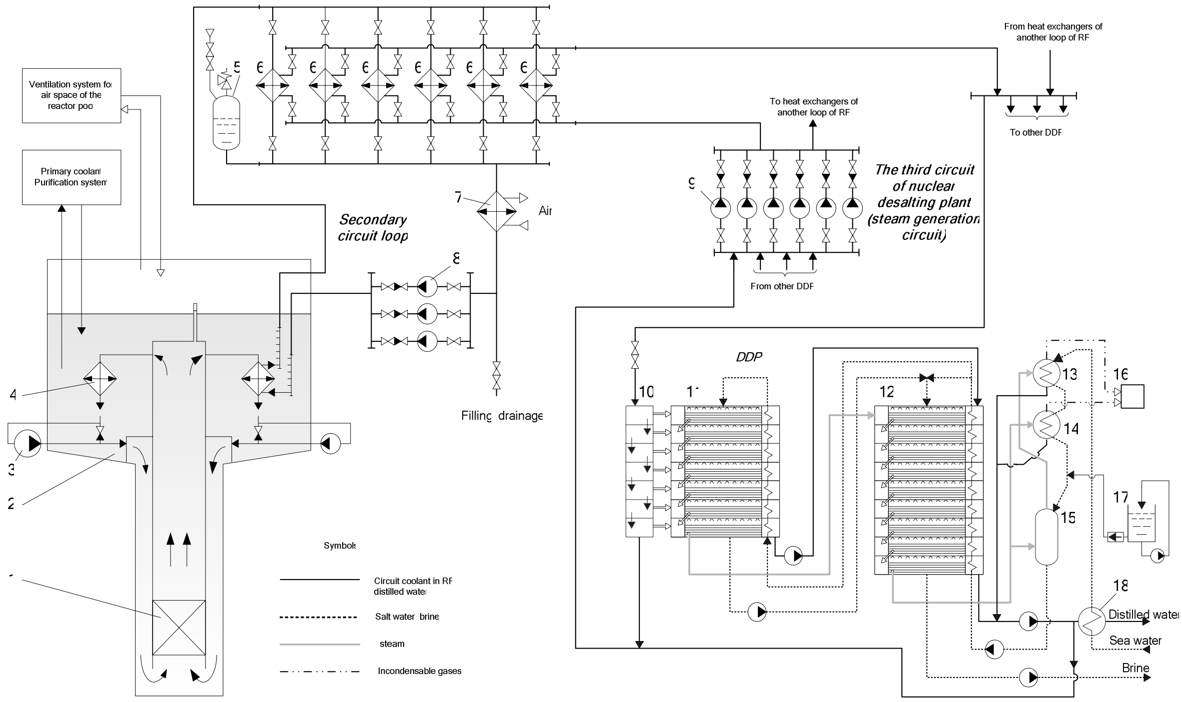
Hot secondary coolant from the receive header of the primary heat exchangers is fed via the rising connecting pipeline from the reactor to the distributing header of the intermediate heat exchangers. Being distributed among the heat exchangers, coolant is cooled, and the heat removed from the reactor is transferred to the network coolant; it is then collected in headers and supplied to suction of the secondary circulation pumps via the downcomer pipeline.

XII-2.2. Description of the turbine generator plant and systems

The turbine and associated systems are not used in the NHP RUTA-70.

XII-2.3. Systems for non-electric applications

A schematic diagram of the desalination plant with the RUTA reactor is shown in Fig. XII-10.



1-Reactor core	2-Reactor pool
3-Primary circulation pump	4-Primary heat exchanger
5-Pressurizer of the secondary circuit	6-Heat exchanger of 2/3 circuits
7-Heat exchanger of air system for emergency cooldown	8-Secondary circulation pump
9-Circulation pump of the third circuit	10-Self-evaporator
11,12-Units of evaporating stages of DOU (EU1 and EU2);	
13-Condenser of deaerator vapour	14-Condenser of last stage vapour
15-Deaerator	16-Water ejector unit
17-System for disincrustant dosing	18-Cooler of distilled water

FIG. XII-10. Desalination plant coupled with RUTA reactor.

XII-2.4. Plant layout

The layout of a nuclear heat plant (NHP) with the RUTA-70 reactor is shown in Fig. XII-11. The depicted components are as follows:

- Administration building.
  - Unit control room.
  - Group of premises for amenities and sanitary purposes.
- Auxiliary services.
  - Restricted access workshops.
  - Transportation and handling platform.
  - Fresh fuel storage facility and preparation.
  - Storage facility for receivers of incombustible gases.
  - Pump station (for fire fighting).

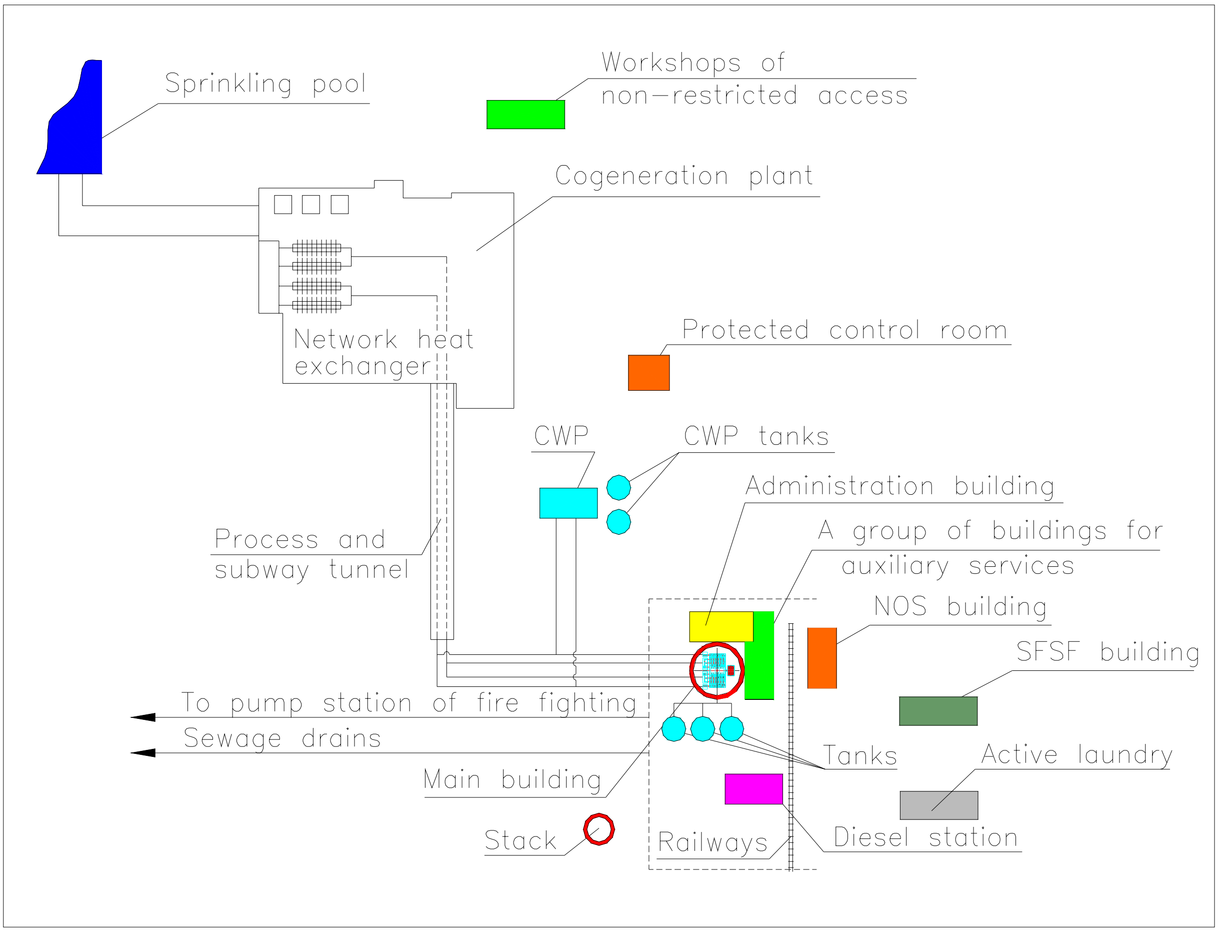


FIG. XII-11. Layout of NHP RUTA-70.

- Main building.
  - Reactor compartment.
  - Cooling pond.
  - Pump stations for the primary and secondary circuits.
  - Process systems and active water treatment.
  - Stand-by diesel power station and storage batteries.
  - Interim storage of solid and liquid radioactive waste (radwaste).
  - Transportation and handling hall.
  - Primary circuit purification system.
  - Air space ventilation system for the reactor pool, etc.
- Chemical water purification (CWP) building.
- Chemical water purification tanks.
- Spent nuclear fuel storage facility (SFSF).
- Building for the protected control room.
- Building for the nitrogen-oxygen station (NOS).
- Diesel station.
- Non-restricted access workshops, etc.

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ANNEX XIII

CONCEPT OF A PASSIVE-SAFETY REACTOR (KAMADO)

CRIEPI, Japan

XIII-1. General information, technical features and operating characteristics

XIII-1.1.Introduction

The concept of a passive-safety reactor “KAMADO” (in Japanese: a Japanese traditional kitchen range for cooking with firewood) was proposed in 2001 by the Central Research Institute of Electric Power Industry (CRIEPI), Japan. The KAMADO concept is based on a synthesis of the design approaches used in light water reactors, the heavy water reactor FUGEN [XIII-1] and pool type research reactors.

The design objective of the KAMADO is to develop a nuclear reactor with negligible possibility of a core meltdown accident. The KAMADO concept provides for a simple plant system design without a reactor pressure vessel, emergency core cooling system (ECCS), re-circulation systems (as in BWRs), etc. Therefore, construction cost per electric power generated is expected to be sufficiently low, comparable with conventional large scale LWRs [XIII-2, XIII-3].

The R&D for this reactor concept has been fully performed and funded by the CRIEPI.

XIII-1.2. Applications

The KAMADO is designed to produce 300 MW(e), assuming a generating efficiency of 33%. Modular composition is easy for this reactor concept, with two or more reactor cores being installed in a single reactor water pool with a single steam turbine system. Therefore the total output is flexible (MW to GW).

Since the KAMADO has a reactor water pool at atmospheric pressure and low temperature similar to pool type research reactors, the irradiation by neutrons and γ-rays around the reactor core is available for purposeful use. With the γ-ray heating around the reactor core, very high temperature (>800ºC) steam can be produced; this very high temperature steam is then directed to the outside of the reactor water pool, Fig. XIII-1, and could be transported well away from the reactor to be used for hydrogen production based on a thermochemical process. It is expected that several thousands of m^3/ hour of hydrogen could be produced using highly efficient hydrogen production technology and the equipment installed outside of the nuclear reactor with a 1000 MW(th) core.

XIII-1.3. Special features

The KAMADO is designed as a land based nuclear power station. However, its use within a floating power plant is not excluded.

The KAMADO has a simple plant system design without a reactor pressure vessel (RPV), ECCS, etc., which makes the construction and transportation of the components to a site essentially more simple.

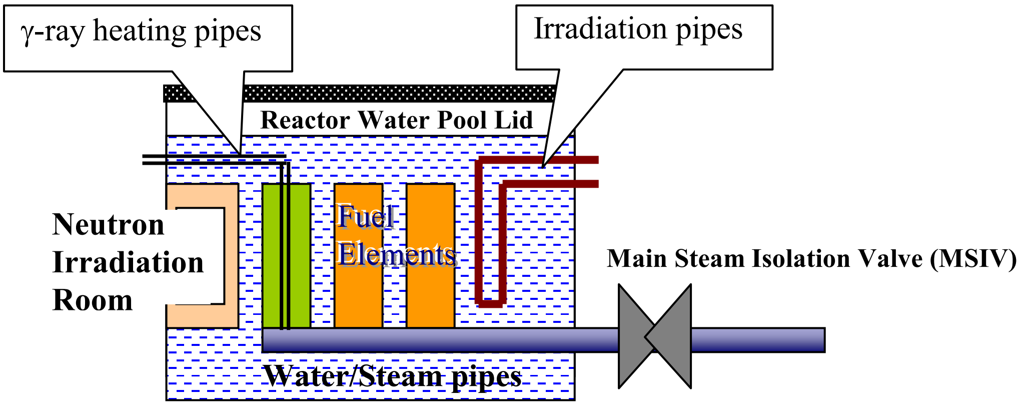


FIG. XIII-1. Concept of a purposeful use of neutron and γ-ray irradiation in KAMADO.

TABLE XIII-1. MAJOR DESIGN CHARACTERISTICS OF KAMADO

ATTRIBUTES	DESIGN PARTICULARS
Core configuration	Vertical, pressure tube type
Fuel	Less than 5% enriched UO_2 (MOX option is available too)
Cladding	Zircaloy
Moderator	Graphite and light water
Coolant	Boiling light water (Inlet/outlet temperatures are about 200ºC / 300ºC)
Number of coolant channels	229 cooling water/steam pressure tubes per fuel element
Pressure tube inner diameter	About 10 mm
Pressure tube material	Zircaloy-2
Number of fuel rods per fuel element	60
Fuel rod lattice pitch	13.1 mm
Active fuel length	3 m
Fuel element lattice pitch	243 mm
Number of fuel elements	278
Core size	4.5 m × 4.5 m
Steam pressure	About 7 MPa
Mode of core heat removal	Forced circulation
Primary shutdown system	Mechanical shut-off rods
Secondary shutdown system	Liquid poison injection in the reactor pool

XIII-1.4. Summary of major design and operating characteristics

Installed capacity

The KAMADO is designed to produce 1000 MW(th), generating 300 MW(e).

Mode of operation

The KAMADO can be operated in base load, as well as load-follow modes.

Load factor / Availability

The KAMADO targets 90% load factor and availability, like conventional LWRs.

Some tentative design characteristics of the KAMADO are given in Table XIII-1.

Simplified schematic diagram

A vertical cross-section of the KAMADO fuel element is shown in Fig. XIII-2. Cooling water is fed from the bottom of the fuel element; heated steam is also discharged from bottom of the fuel element. In a cooling water tube, water is heated by fuel rods through graphite and changed to steam. Within a steam tube, steam flows downward and is superheated. In the present design, standard LWR type fuel rods are used to save costs associated with the development of new fuel rods.

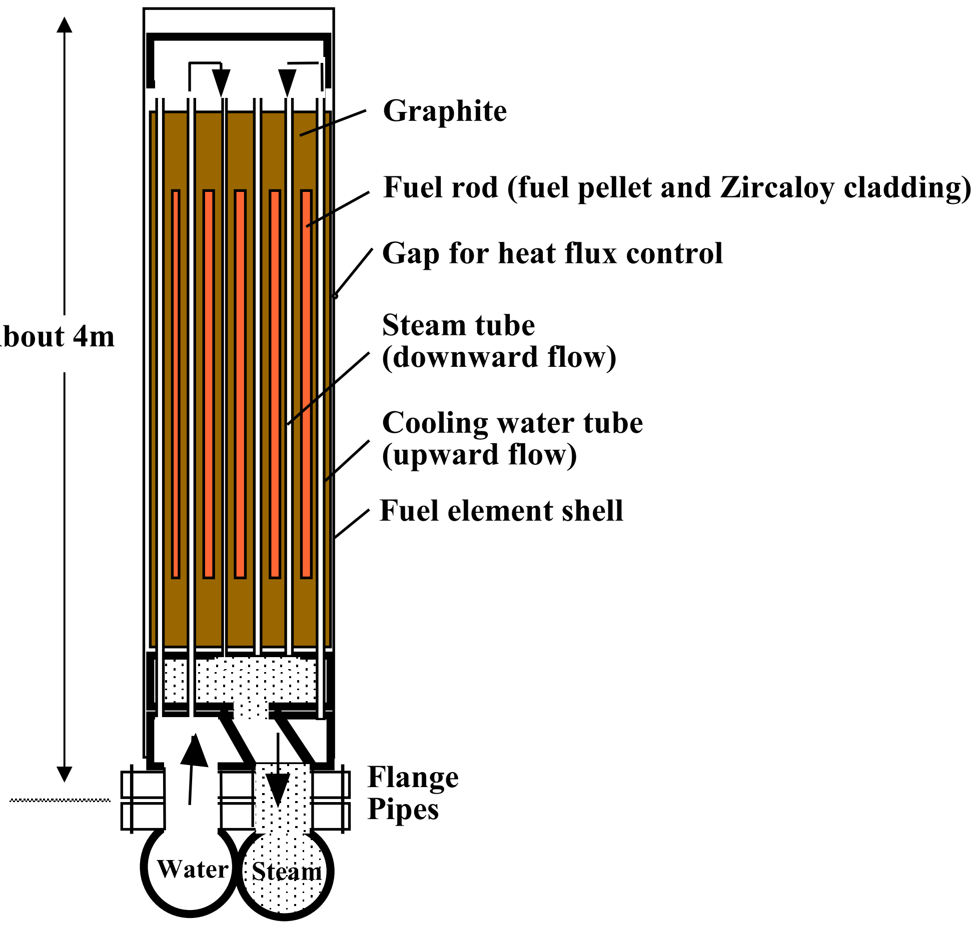


FIG.XIII-2. Conceptual design of a fuel element.

Fuel elements are arranged in a reactor water pool. A reactor core consists of two or more fuel elements, control rods, etc. within the pool. Steam pipes are bundled and led to a steam turbine, Fig.XIII-3. Cross-type control rods are inserted between the fuel elements from the top.

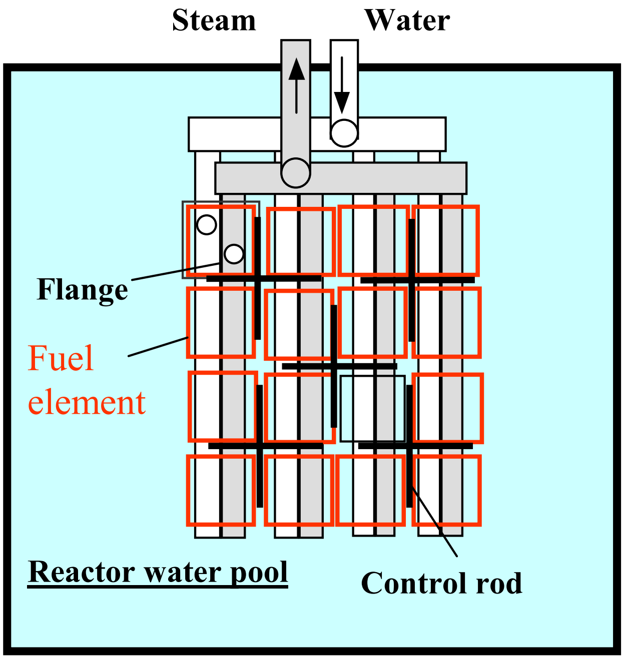


FIG.XIII-3. Arrangement of fuel elements in a reactor water pool.

Figure XIII-4 illustrates the basic concept of the present KAMADO design. The heat generated by a fuel rod is not removed directly with cooling water but via graphite blocks of a fuel element, i.e., cooling water is heated by high temperature graphite of the fuel element. In case of a loss of coolant or flow, such as a pipe break, turbine trip, etc., the decay heat is removed by passive heat transfer from surfaces of the fuel elements to the reactor water pool operating at atmospheric pressure and low temperature.

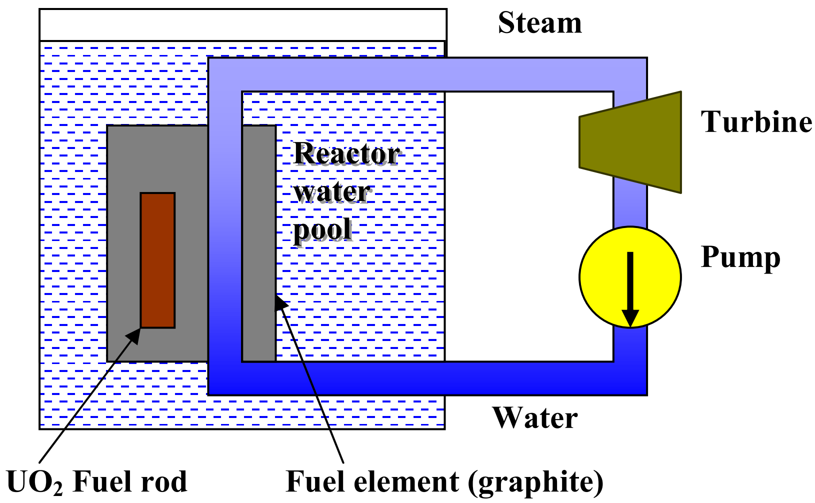


FIG. XIII-4. Basic concept of KAMADO.

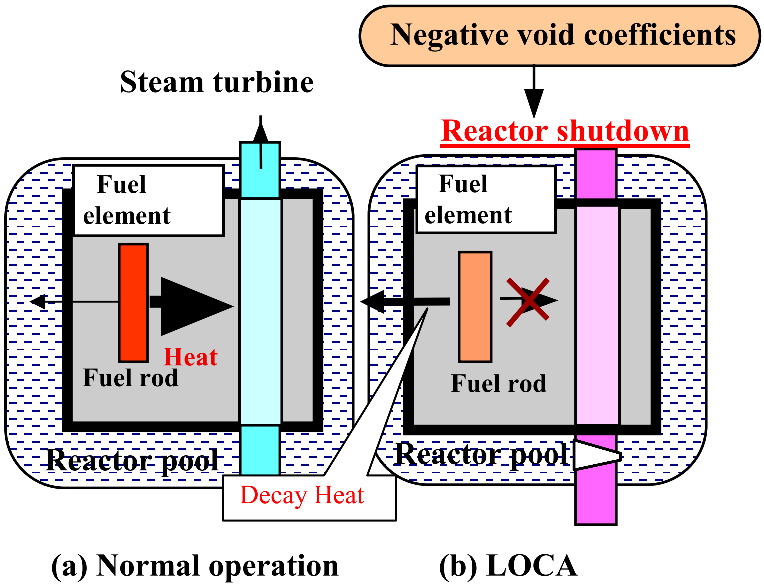


FIG. XIII-5. Heat removal paths in normal operation and in LOCA.

Neutron-physical characteristics

The neutron energy spectrum of the KAMADO fuel element is a little softer than that of a BWR with 8×8 type of fuel, because of the neutron moderation on graphite. Since burn-up reactivity swing is similar to that of a BWR with 8×8 type fuel, the KAMADO fuel element could, probably, reach high burn-ups (more than 55 MW·d/kg U) with a 5% enriched UO_2 fuel.

The void coefficient of reactivity is an important inherent safety characteristic of reactor core. The calculations performed with continuous energy Monte Carlo code MVP show negative void coefficient of –15% Δk/k (at 40% void, BOL) and temperature coefficient of -2.3E-4% Δk/k/ºC for graphite. These coefficients are rated sufficient to secure passive shutdown of the reactor core in accidents.

Reactivity control mechanism

In the KAMADO there are two independent and diverse systems of reactivity control. The mechanical control rods are used to compensate reactivity changes due to fuel burn-up and operational reactivity changes; the primary reactor shutdown is assumed to be passive. A liquid poison injection system in the reactor pool is available as a secondary shutdown system.

Cycle type and thermodynamic efficiency

The KAMADO is a direct flow reactor, i.e., uses a direct cycle with superheated steam at core outlet. Different from BWRs, there are no steam recirculation system and separators. The thermodynamic efficiency (target) is 33%.

Thermal-hydraulic characteristics

Basic thermal hydraulic characteristics of the KAMADO are assumed to be similar to those of a BWR, excluding minimum critical power ratio (MCPR). Since a fuel rod is not directly cooled with water, the limitation of MCPR is less strict in the KAMADO concept.

Maximum/average discharge burn-up

The KAMADO fuel elements have a possibility to reach high burn-ups (more than 55 MW·d/kg U, or about 6% FIMA) with 5% enriched UO_2 fuel similar to that of other LWRs.

Fuel lifetime/period between refuellings

The KAMADO refuelling concept is similar to that of other LWRs. Since the KAMADO concept has a simple plant system design, plant maintenance becomes easy too. Therefore shorter refuelling/outage time is expected. Assuming 360 effective full power days (EFPD) of operation and 40 days of refuelling, the load factor of 90% could be achieved.

Mass balances/flows of fuel materials

The inventory of heavy metals is about 40 t in a 300 MW(e) KAMADO plant. In case of a 4-batch refuelling, about 10 t of heavy metals (enriched uranium) is loaded / unloaded annually. The annual consumption of natural uranium depends on enrichment of the fuel. In case of a high fuel burn-up (more than 55 MW·d/kg U), about 100 t of natural uranium is necessary for a 300 MW(e) reactor annually (Fig. XIII-6). That is equal to 330 000 kg/GW(e) year.

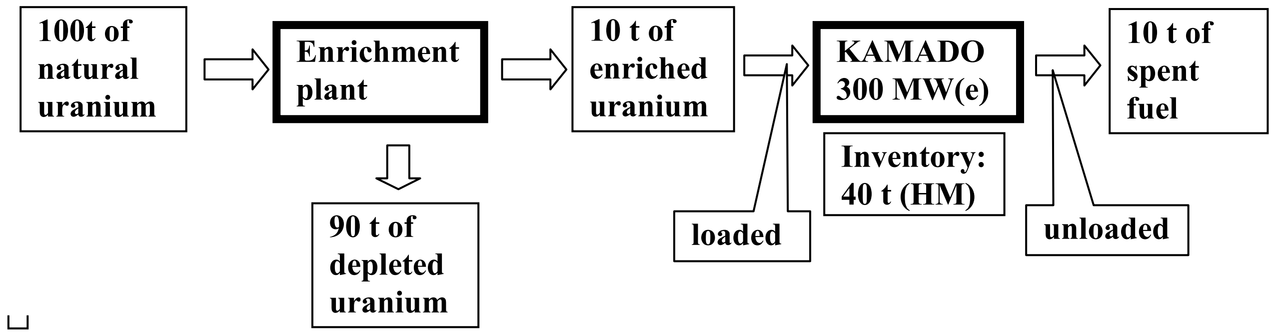


FIG. XIII-6. Annual mass balance of uranium (fuel burn-up ~ 55 MW·d/kg U).

Design basis lifetime for reactor core, vessel and structures

The KAMADO design has no reactor pressure vessel, ECCS, steam re-circulation systems and separators, concrete radiation shielding, etc. Heavily irradiated equipment components and structures are limited to fuel elements and control rods, which are replaced periodically.

Economics

A small nuclear power plant has the disadvantages of economy of scale. For a 300 MW(e) reactor, the scale factor is around 1.4–1.7 when a 1000 MW(e) unit is used as reference. The cost reduction target for the KAMADO is around 60%, based on radical plant design simplification. Therefore, the KAMADO plant of 300 MW(e) could have a generation cost per kW-h almost compatible to conventional large scale LWRs.

XIII-1.5. Outline of fuel cycle options

The KAMADO fuel cycle concept could be similar to that of other LWRs. Spent fuel from the KAMADO is suitable for storage at the reactor site (AR) or away from the site (AFR) since fuel elements have the function of spent fuel canisters. In addition, since a fuel element has cooling water tubes, spent fuel storage is possible in dry areas as well as in a pool.

Spent fuel also could be transported to reprocessing plants and reprocessed. However, different from spent fuel of conventional LWRs, the KAMADO fuel elements include graphite blocks. Therefore, a newly designed transport cask is necessary for the KAMADO spent fuel. In reprocessing plants, combustion or a mechanical destruction of the graphite blocks is necessary before starting the dissolution process.

XIII-1.6. Technical features and technological approaches that are definitive for KAMADO performance in particular areas

XIII-1.6.1. Economics and maintainability

The KAMADO has simple plant system design eliminating many components present in conventional LWRs. The targets for cost reduction in certain components of the KAMADO are given in Table XIII-2.

Since the reactor basic shutdown and decay heat removal are passive, there is no need in dedicated engineered safety systems.

An incremental capacity increase is possible with this reactor concept in which two or more reactor cores are installed in a single reactor water pool. For this reason, even if the output of one reactor core is 300 MW(e), a large total output is possible.

On the total, construction cost per electric power generation is expected to be sufficiently low compared with conventional large scale LWRs.

TABLE XIII-2. TARGETS FOR COST REDUCTION

SYSTEMS	COST REDUCTION TARGET AND APPROACHES
Primary cooling system	25% by eliminating reactor pressure vessel, steam recirculation systems, separators, etc.
Engineered (active) safety systems	100% by eliminating ECCS and all other engineered systems
Other reactor systems	60% by reducing the radioactive waste treatment systems, etc.
Turbine system	Not changed
I&C system	Not changed
Buildings	50% by unifying reactor building and turbine building
Others	Not changed
Total	60% of a conventional LWR of the same unit power

Since the control rod drive (CRD) mechanisms can be located above the reactor pool (no radiation, room temperature area), easy maintenance of the CRDs is achievable.

XIII-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

In the KAMADO only fuel elements and control rods are strongly irradiated, which contributes to minimizing the volume of wastes.

Since the KAMADO is designed to have a negligible possibility of core meltdown, a potential radiation exposure in accidents could be essentially reduced or eliminated.

XIII-1.6.3. Safety and reliability

Safety concept and design philosophy

The design objective of the KAMADO is to develop a reactor with negligible probability of a core meltdown accident. To achieve this goal the KAMADO design strongly relies on the inherent safety features related to optimal core neutronics and the confinement of radioactive materials, and also makes use of passive systems for decay heat removal. An important feature of this design is passive decay heat removal achieved for the fuel elements installed in the reactor water pool operating at atmospheric pressure (1 atmosphere) and low temperature (< 60ºC).

Provisions for simplicity and robustness of the design

The KAMADO has a simple plant system design without engineered safety systems. The KAMADO design ensures high margin to fuel failure in accidents.

Active and passive systems and inherent safety features

The inherent safety features are:

- Negative void and temperature reactivity coefficients (providing a passive shutdown capability);
- High heat capacity of the fuel elements and the reactor water pool;
- Low (atmospheric) pressure and temperature of the water pool.

The design features of the KAMADO provide a passive decay heat removal capability with all components of the reactor core and water pool acting as a passive decay heat removal system. In this, the residual heat removal (RHR) system is reduced to a water pool cooling system, which could be made non-safety-grade and passive.

The KAMADO has a sufficient negative void reactivity coefficient and negative temperature reactivity coefficients of fuel and graphite. In case of a loss of coolant or flow, the reactor will be shut down passively. Increases in the fuel element temperatures will be suppressed during several seconds after LOCA initiation by heat transfer to the graphite of fuel elements (Fig. XIII-5), which has high enough heat capacity. Decay heat will be transferred passively to the reactor pool and, therefore, high temperature of the fuel elements can be avoided in LOCA.

The reactor water pool has enough heat capacity to absorb decay heat for 3 days without relying on operators. In case of a malfunction of the cooling system of the reactor pool, its temperature would increase slowly and operators will have enough time to shut down the reactor manually.

The active safety systems are:

- A liquid boron injection system of the reactor pool, which is a reserve shutdown system.
- Main Steam Isolation Valve (MSIV).
- Hydrogen combustion system.

The reactor pool lid and the MSIV, additionally enhance the reactor capability to confine radioactive materials, Fig. XIII-1.

Hydrogen generated in the reactor pool (mainly due to the radiolysis caused by gamma-rays) could be treated by a combustion system similar to that used in conventional LWRs.

Since water of the reactor pool is important for the KAMADO safety concept, loss of water from the reactor pool should be prevented through appropriate design measures, such as double walls of the reactor pool, a monitoring system of water leakage, etc. Location of the reactor pool below ground level could inherently prevent accidents with the loss of pool water.

Structure of the defence-in-depth

The KAMADO has adopted the defence-in-depth concept with multiple barriers, such as fuel pellets, cladding, fuel elements, a shielded reactor pool and a shielded reactor building.

Design basis accidents and beyond design basis accidents

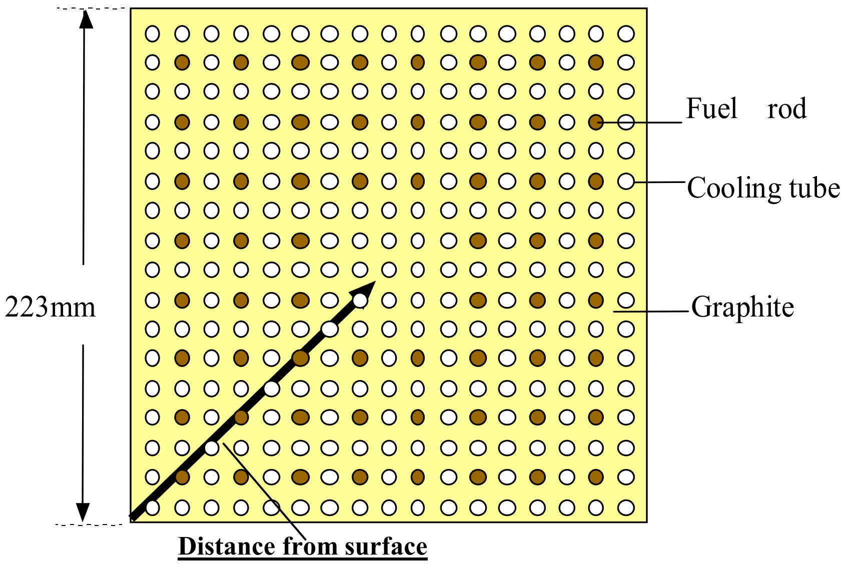
A tentative list of the design basis accidents includes:

- Loss of coolant accident (LOCA).
- Loss of flow accident.
- Malfunction of Main Steam Isolation Valve (MSIV).
- Blockage of pipes or tubes.
- Reactivity induced accidents.
- Malfunction of the cooling system of the reactor pool.

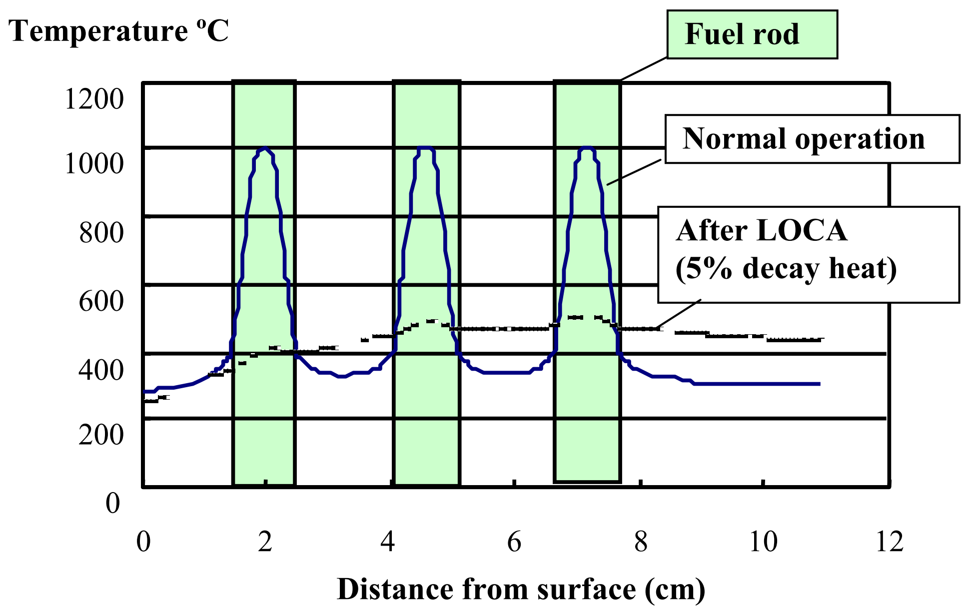
In case of a loss of coolant/flow, the reactor will be shut down passively by the negative reactivity coefficients. Additionally, reactivity induced accidents can be controlled by such design features as gravity driven safety rods. In case of a malfunction of MSIV or a blockage of pipes or tubes, the reactivity is increased through the collapse of voids, since the void coefficient is negative. On the other hand, this results in the increase of the fuel element temperature, leading to a negative reactivity insertion. The subsequent transient progressions are expected to be similar to the loss of flow accident, providing that detailed examinations should be performed.

The temperature distribution in the KAMADO fuel element in normal operation and in LOCA was calculated with a 2D thermal diffusion equation code for the design parameters given in Table XIII-1. The thermal conductivity values of 35.0, 0.673 and 2.53 W/m-K were used for the graphite, fuel and water respectively. The linear heat rate of a fuel rod is 20 kW/m; the heat transfer coefficients of 100.0 and 3000.0 W/m^2-K were assumed for fuel element and cooling water tube surfaces, respectively. The heat transfer coefficient of a fuel element surface to the reactor pool is reduced remarkably considering the gap near the fuel element surface. This gap restricts the heat leakage to the water pool to a few percent during normal operation. In normal operation, the temperature of the graphite is estimated to be 350–400ºC (Fig. XIII-7); after LOCA (the decay heat is 5% of normal power in 13 seconds after the

reactor is stopped), the maximum temperature of the graphite is estimated to be less than 500ºC without heat removal through the cooling water tubes. These calculations have not taken into account the conductivity of a thermal gap neither between fuel pellet and cladding nor between fuel cladding and graphite. Therefore, the fuel temperature might be understated by about 100ºC.



(a) Arrangement of fuel rods within a fuel element



(b) Temperature distributions

FIG.XIII-7. Arrangement of fuel rods (a) and temperature distribution in fuel elements (b).

The beyond design basis accidents may include:

- Anticipated transient without scram (ATWS);
- Total NPP blackout.

In ATWS, e.g. even if the reactor does not stop its operation after LOCA, the temperature of the fuel elements will be passively suppressed and kept under the melting points of the fuel and graphite. Since the KAMADO has passive shutdown and decay heat removal capabilities, core meltdown is not expected in case of a total NPP blackout (loss of internal and external power supply).

Though the detailed safety and accident analyses have not been performed yet, the preliminary evaluations indicate that the KAMADO might be designed to essentially exclude a core meltdown accident.

Measures planned in response to severe accidents

The reactor water pool has enough heat capacity to absorb decay heat for more than 3 days without operator intervention. If necessary, external water can be injected into the reactor water pool from outside of the reactor building.

XIII-1.6.4. Proliferation resistance

No information provided.

XIII-1.6.5. Technical features and technological approaches used to facilitate physical protection of KAMADO

No information provided.

XIII-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of KAMADO

No information was provided.

XIII-1.8. List of enabling technologies relevant to KAMADO and status of their development

The KAMADO concept is similar to conventional LWRs in fuel rods, control rods, main steam lines, and radioactive treatment systems. However, a fuel element with graphite block and a primary cooling system are the main innovative components to be developed and demonstrated. The fuel elements (with graphite blocks) and the reactor pool technology could, perhaps, be developed and demonstrated using an experimental base provided by pool type research reactors and high flux reactors. Primary cooling system could be validated using thermal-hydraulic facilities with mock-up heating.

The list of enabling technologies for the KAMADO is given in Table XIII-3.

XIII-1.9. Status of R&D and planned schedule

The preliminary conceptual design of the KAMADO is in progress. All activities on design and technology development for the KAMADO are performed and funded by the CRIEPI of Japan.

It is foreseen that the KAMADO concept could be developed and demonstrated using the pool type research reactors, high flux reactors and cold (non-radioactive) experimental facilities.

XIII-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

As listed in Table XIII-3, the innovative fuel elements and the primary cooling system with a new technology for the direct flow production of superheated steam would require a substantial amount of RD&D.

Table XIII-3. List of enabling technologies for KAMADO

TECHNOLOGY	STATUS OF DEVELOPMENT
Fuel rods	Available, similar to LWR fuel rods
Fuel elements	Research design and demonstration (RD&D) necessary
Control Rods / CRD	Available, similar to BWR control rods An option to locate CRD mechanisms above the reactor water pool should be examined.
Primary cooling system	RD&D necessary to prove reliability of a direct flow reactor with fuel elements including fuel rods, graphite blocks, and water and steam pipes.
Radioactive waste treatment system	Available, similar to that of a BWR
Reactor water pool	RD&D necessary. Some experience of pool type research reactors and high flux reactors could be of relevance.
Residual heat removal (RHR) system	Available, RHR is reduced to water pool cooling system, which is not a safety grade system and could be designed using available technologies
Hydrogen production system using process steam*	RD&D necessary
Turbine system	Available, standard equipment can be used
Electrical system	Available, standard equipment can be used
Building	Available, building could be similar to that of a BWR.

* Hydrogen production system is optional. The equipment using a highly efficient hydrogen production technology is to be installed away from the nuclear reactor.

XIII-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

No other similar SMRs are under design elsewhere.

REFERENCES

[XIII-1] MITA, S., et al., Development of the Advanced Thermal Reactor in Japan, Nuclear Engineering Design 144, No.2 (1993), 283–292.

[XIII-2] MATSUMURA, T., et al., New concept of a small passive-safety reactor with UO2-Graphite-Water Core, Advanced Nuclear Power Plants (Proc. Int. Congress, Hollywood, FL. USA, June 9–13, 2002) ICAPP'02 (1038).

[XIII-3] MATSUMURA, T., et al., Ability of new concept passive-safety reactor KAMADO - Safety, economy and hydrogen production, GENES4/ANP2003 (1092), Kyoto, Japan (2003).

DESIGN DESCRIPTIONS OF GAS COOLED SMRs

PEBBLE BED MODULAR REACTOR (PBMR)

PBMR (Pty) Ltd., South Africa

XIV-1. General information, technical features and operating characteristics

XIV-1.1. Introduction

In 1993, the pebble bed modular reactor (PBMR) was identified by ESKOM, the electric utility of South Africa, as a leading option for the installation of new generating capacity to their electric grid. This innovative nuclear power plant incorporates a closed cycle primary coolant system utilizing helium to transport heat energy directly from the modular pebble bed reactor to a recuperative power conversion unit with a single-shaft turbine/compressor/generator. This replacement of the steam cycle that is common in present nuclear power plants (NPP) with a direct gas cycle provides the benefits of simplification and a substantial increase in overall system efficiency with the attendant lowering of capital and operational costs.

Although the historical development of this plant is interrelated to other types of high temperature gas cooled reactors (HTGRs), the principle focus herein is on the pebble bed (spherical) fuel element type reactor. The long-term development of this reactor type began in Germany by the KFA Nuclear Research Center (now FZJ). Two pebble bed plants were constructed in Germany, the 46 MW(th)/15 MW(e) Arbeitsgemeinshaft Versuchsreaktor (AVR) and the 750 MW(th)/296 MW(e) thorium high temperature reactor (THTR-300). Basically, these steam/electric plants validated the temperature and fission product retention capabilities of the ceramic (TRISO) coated fuel particle and the safety characteristics of the HTGR. Most notable of the operational achievements was with the AVR in sustaining long-term operation at an average core outlet temperature of 950ºC, and in demonstration of safety such as extended loss of forced cooling on the core. Reference [XIV-1] provides more details on the AVR and THTR-300 plants.

The next evolution of the pebble bed plant began in the early 1980s with development of the modular reactor. This small reactor added the unique characteristic of being able to cool the core entirely by passive heat transfer mechanisms following postulated accidents without exceeding the failure temperature of the coated particles, which is key to the normal safety characteristics of all HTGRs. Originally, the focus of the modular HTGR was on the steam cycle and included designs by Germany, the Russian Federation and the USA. These designs all incorporated the TRISO ceramic coated fuel particles and utilized steel vessels to house the primary system.

Design of the present direct cycle gas turbine modular plant such as the PBMR began in the early 1990s. This plant incorporates the basic safety attributes of the modular pebble bed reactor with the direct improvement of not being tied to the complexities and low efficiencies associated with the steam cycle. Also attendant with the modular direct cycle PBMR is a high degree of standardization for this relatively small, simplified design. This approach allows the benefits of plant modularization and shop fabrication with corresponding improvements in quality control, reduction of construction schedules, and optimization of manufacturing procedures and processes; all resulting in improvements in schedule and capital costs.

A listing of the more prominent test facilities and R&D performed to support the PBMR includes:

- The ASTRA critical facility at the Russian Research Centre “Kurchatov Institute” (RRC KI), Moscow, the Russian Federation was utilized to perform benchmark analysis of the pebble bed core physics and neutronic codes [XIV-2].
- The NACOK (Naturzug im Core mit Korrosion) facility at the Research Centre, Juelich, Germany is used to investigate the oxidation of graphite for postulated air ingress accidents.
- The micro-model facility at the university of the Northwest, Potchefstroom, South Africa has demonstrated the operation of the closed cycle, three shaft recuperative Brayton cycle power conversion system [XIV-2].
- The IVV-2M at NIKIET, the Russian Federation and the SAFARI-I at NECSA, South Africa reactors are being utilized to confirm fuel performance through the full burn-up range.
- The heat transfer test facility [under design at the university of Northwest, Potchefstroom, South Africa is to be utilized for performance of separate effect tests of heat transfer mechanisms in an annular pebble bed. Tests will provide heat transfer correlations for analytical codes and more in-depth knowledge of heat transfer phenomena in pebble beds.
- A prototype of the burn-up measuring system is being used to validate the capability to measure the fuel burn-up value of each sphere.
- Participation in the IAEA’s coordinated research programme on “Evaluation of high temperature gas cooled reactor performance” is providing independent validation of the codes and models being used in the design of the PBMR.
- The control rod drive (CRD) test mechanism verifies CRD behaviour including torque and friction of the chain drive mechanism.

The PBMR project is unique in that a principle partner, ESKOM, is both a key participant in the plant development as well as a committed purchaser of the initial unit(s).

The principal shareholders in the PBMR are incorporated within the South African company, PBMR (Pty) Ltd. These shareholders include:

- ESKOM, National South Africa Electric Utility.
- Industrial Development Corporation of South Africa, a national development financial institution.
- British Nuclear Fuel plc, a global nuclear fuel cycle company, (parent of Westinghouse nuclear) solely owned by the government of the United Kingdom.

XIV-1.2. Applications

Initial development of the PBMR is to utilize a helium cooled pebble bed HTGR of 400 MW(th) to generate approximately 165 MW(e) of electricity with a conservative net plant efficiency of ≥41%. Construction of a PBMR demonstration unit is projected to start in early 2007 at the present Koeberg NPP site near Cape Town, South Africa.

Utilization of the PBMR for co-generation applications including electricity generation, process heat production, desalination and hydrogen production are being considered.

XIV-1.3. Special features

The PBMR is a land based nuclear power plant. All major components including the reactor and power conversion system vessels and associated internal components are being sized and designed for off-site prefabrication capability. The transportability design intent is for the PBMR to be commercially available to a broad latitude of customers worldwide.

The intent of the PBMR developers is to offer power plants that consist of multiples of individual units.

XIV-1.4. Summary of major design and operating characteristics

Initial development of the PBMR was to include a power conversion unit (PCU) consisting of three separate rotating shafts in a vertical configuration and equipped with magnetic bearings. These machines included a high pressure turbo-compressor unit, a low pressure turbo-compressor unit and a turbine-generator unit. The decision was recently made to replace the three shafts PCU with a single rotating machine. This major design change was initiated due to recent developments in the application of dry gas seals that allowed use of conventional oil bearings, and improvements in gear reduction capability for high capacity applications.

The magnitude of this design change to a single shaft PCU will necessitate a thorough review of all plant parameters within the primary coolant system and associated sub-systems. However, it is anticipated that the design parameters for the reactor plant will remain consistent to those previously determined for the 400 MW(th) reactor. Table XIV-1 provides major design and operating characteristics of the 400 MW(th)/165 MW(e) plant.

TABLE XIV-1. MAJOR DESIGN AND OPERATING CHARACTERISTICS OF THE PBMR

Installed thermal capacity	400 MW(th)
Installed electric capacity	165 MW(e)
Mode of operation	Load following between 100-20-100%
Load factor	100% base load
Availability	≥95%
Core and fuel characteristics	
Fuel	TRISO ceramic coated U-235 in graphite spheres
Fuel enrichment	9.6% UO_2 for equilibrium core
Primary coolant	Helium
Moderator	Graphite
Core type	Pebble bed in an annular core right circular cylinder arrangement with a fixed central reflector
Core outer diameter	3.7 m
Outer diameter of central reflector	2.0 m

TABLE XIV-1. (continued 1)

Fuel effective cylindrical height (flattened top to the top / start of cone)	11.0 m
Total fuel volume	83.7 m^3
Primary system characteristics	
Vessel material	Carbon steel
Reactor pressure vessel wall thickness	0.18 m
Cycle type	Direct
Number of circuits	1
Neutron & physical characteristics	
Temperature reactivity effect	Variable, but always negative (about -3.0×10^{-5}/ ºC during equilibrium operation)
Void reactivity effect	Negligible
Burn-up reactivity swing	0
Peaking factor	3.10 (P_{max}/P_{avg} = 2.746 kW/0.885 kW [permissible is 4.5 kW/Pebble])
Average core power density	4.78 MW/m^3
Reactivity control mechanisms	
Number of independent active control and protection systems	2
Number of control rods	24 Positioned symmetrically and equally spaced in outside core reflector
Number of reserve shutdown system	8 Positioned symmetrically and equally in fixed central reflector
Reactivity control system material	B_4C and steel alloy
Burnable poison	None
Cycle type and efficiency	
Cycle type	Direct
Net cycle efficiency	>41%
Thermal-hydraulic characteristics	
Circulation type	Forced
Core outlet temperature	900ºC
Core inlet temperature	488ºC
Primary coolant mass flow rate	184.8 kg/s
Primary coolant pressure	9 MPa

TABLE XIV-1. (continued 2)

Maximum operating fuel temperature	1130ºC
Maximum fuel temperature limit	~1600ºC
Maximum component temperatures on a DLOFC accident	See Fig. XIV-5
Average fuel temperature at full power	1035ºC
Average central reflector temperature at full power	~650ºC
Maximum RPV temperature at full power	280ºC
Fuel burn-up and material balance characteristics	
Target burn-up	>90 000 MW·d/t U
Average fuel discharge burn-up	~95 000
Refuelling Type	On-line
Average daily fuel sphere circulation	2833
U_3O_8 requirement	259.4 kg/GW(t) day
Separative work	187.3 kg SWU/GW(t) day
Projected graphite needs	234 kg C/GW(t) day
Projected helium needs	14.6 kg/GW(t) day
Number of passes a sphere makes through core	6
Demonstration plant lifetime	35 Full power years
Fixed central reflector lifetime	22 Full power years
Economics	
Overnight construction cost (for nth Plant)	<1500 US$/kW installed (2004)
Combined fuel and O&M costs (for nth plant)	9 mills/kW h

The PBMR functions as a direct Brayton cycle, with primary coolant helium flowing downward through the core and exiting at 900ºC. The helium then enters the turbine relinquishing energy to drive the electric generator and compressors.

After leaving the turbine, the helium then passes consecutively through the hot side of the recuperator, then the pre-cooler, the low pressure compressor, intercooler, high pressure compressor and on to the low temperature side of the recuperator before re-entering the reactor vessel at 488ºC. Figures XIV-1 and XIV-2 provide a schematic representation of the PBMR flow path and conceptual primary system, respectively.

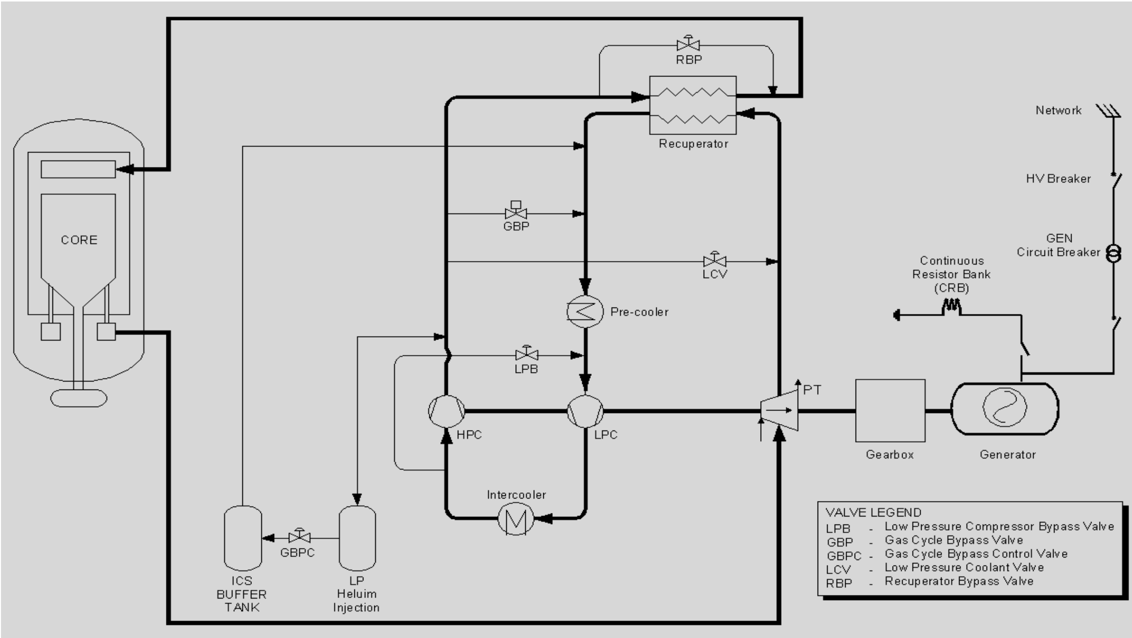


FIG. XIV-1. PBMR direct Brayton cycle flow path [XIV-3].

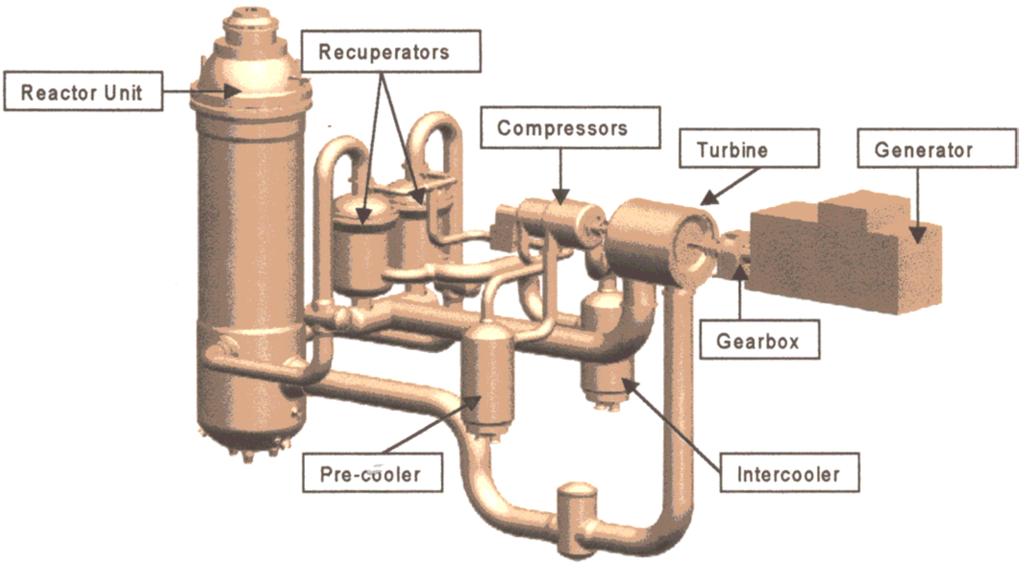


FIG. XIV-2. Conceptual layout of the PBMR primary system [XIV-3].

Power is adjusted by regulating the mass flow rate of gas inside the primary circuit. This is achieved by a combination of compressor bypass and system pressure changes. Increasing the pressure results in an increase in mass flow rate, which results in an increase in the power removed from the core. Power reduction is achieved by removing gas from the circuit. A helium inventory control system is used to provide an increase or decrease in system pressure [XIV-4].

The PBMR reactor core is basically a long right circular cylinder with a fuel effective height of 11.0 m and a diameter of 3.7 m. Twenty-four reactivity control system rod holes are equally spaced outside the core. For shutdown purposes and for minor reactivity adjustments, two diverse reactivity control systems are used.

The one system, the reactivity control system (RCS), consists of 12 control rods and 12 shutdown rods. The design of both systems is identical and the positions during operation will be, within a small band, the same. The driving systems are also identical – each system has a stepper motor with a gearbox that finally drives a chain wheel that positively locates the chain on which the control rod is supported. When inserted into the reflector, the control rods move to a depth of 6.5 m below the bottom of the top reflector and the shutdown rods move to a depth of 10 m. The RCS rods move in borings in the side reflector.

The other system, the reserve shutdown system (RSS), consists of 10 mm diameter B_4C absorber spheres, which during a shutdown operation, are dropped into the eight borings in the central reflector. The spheres are normally stored in containers at the top of the core structures, and are released by opening a valve system [XIV-4]. The RCS and RSS drive/actuating mechanisms are located on top of the pressure boundary. These need not be accessible for on-line maintenance [XIV-5].

XIV-1.5. Outline of fuel cycle options

On-line refuelling is another key feature of the PBMR. Fresh fuel elements are added to the top of the reactor while used fuel pebbles are removed at the bottom to keep the reactor at full power. On average, each fuel pebble makes six passes, back through the reactor before being finally discharged to the spent fuel storage tanks. The aim is to operate uninterrupted for six years before the reactor is shut down for scheduled maintenance [XIV-6].

Figure XIV-3 is a representation of the fuel handling and storage system. The fuelling system has 3 feeding and 3 defuelling points. The operating pressure is up to 9 MPa with a temperature of 20 to 260ºC. The fuel storage capacity includes 8 spent fuel tanks with a total spent fuel storage capacity of 6 000 000 spheres. The spent fuel storage period is up to 80 years.

The fuelling system design also makes provision for unloading of the entire core into a used fuel tank should it become necessary for special maintenance. Provision is also made for initially filling the core cavity with graphite spheres and for doing the approach to critical loading on top of the bed of graphite spheres. The fuel handling system can also separate fuel from graphite spheres by a gross gamma activity measuring system. The enrichment of the fuel used for the initial core will be lower than the enrichment of the fuel used during equilibrium core conditions. During operation with the initial core, the lower enrichment fuel (start-up fuel) will be intermixed with graphite spheres in a specific ratio. The fuel handling system can also continuously change this ratio during the time when the initial core transitions into the equilibrium core. A separate tank is provided in which all of the graphite spheres can be stored [XIV-4].

External fuel cycle

Regarding the external fuel cycle, all spent fuel will be stored on-site below the reactor in spent fuel tanks. Upon final discharge of the last spent fuel load, i.e. after 40 years of operation the fuel can be placed in safe store for a further 40 years. After this period has lapsed the spent fuel could be suitably classified as medium level waste, which can be packaged for final disposal in a suitable repository. It is foreseen that this waste will be treated

in accordance with the classification order discussed in the IAEA guidelines of improvement of safety assessment methodology, features, events, and processes listing [XIV-5].

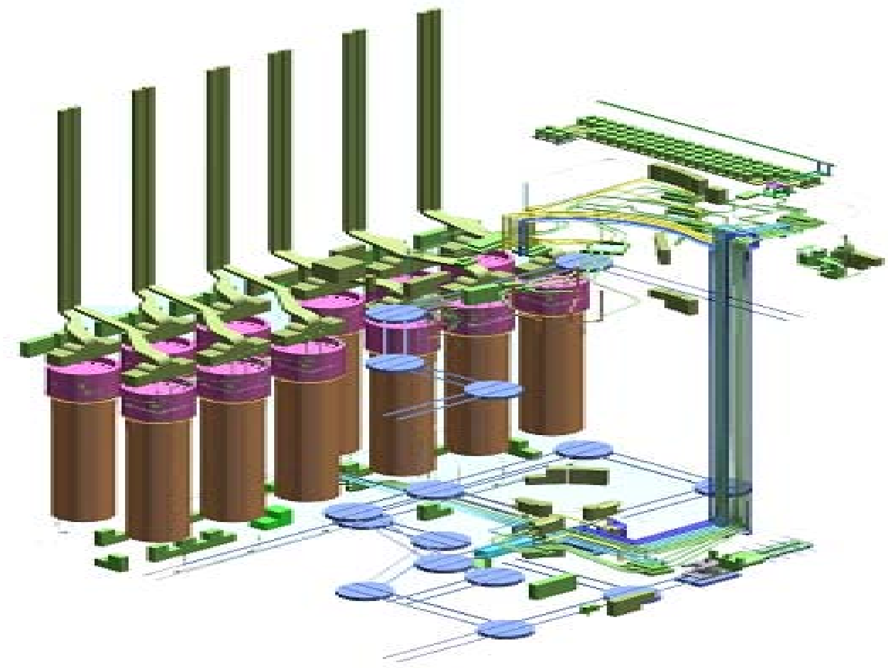


FIG. XIV- 3. Fuel handling & storage system [XIV-7].

XIV-1.6. Technical features and technological approaches that are definitive for PBMR performance in particular areas

XIV-1.6.1. Economics and maintainability

In 2000, the PBMR (Pty) Ltd. company was formed with international investment partners to build and market PBMR-based power plants. Since the technology had not previously been commercialized, the intention to build and operate a single module to serve as a demonstration plant and as a launch platform for local and international sales, and an associated fuel plant, was publicized. Successful completion of the demonstration phase will be followed by commercialization, with ESKOM likely to be the first customer [XIV-6].

The long-term marketing approach taken by the partners of PBMR (Pty) Ltd is to sell plants comprized of multiple of modules. This has led the PBMR partners to develop early relationships with strategic suppliers for key equipment. A partial list of strategic suppliers is shown in Table XIV-2. PBMR (Pty) Ltd works closely with these suppliers to develop equipment and subsystem designs that are both easier to manufacture and are more cost effective. PBMR (Pty) Ltd ensures that there is a collaborative working arrangement where tradeoffs between major equipment, plant structures and/or support systems are considered in the optimization of the plant.

This approach also results in sourcing of the key equipment with the same strategic suppliers for a series of plants. By doing this, the suppliers can justify making the first-of-a-kind investments in design engineering, fabrication, test and start-up equipment (e.g. fabrication

fixtures, test stands, etc.) that are necessary to achieve the repetitive quality and cost targets which will make the PBMR design commercially successful [XIV-6].

TABLE XIV-2. PBMR STRATEGIC SUPPLIERS

SUPPLIER	EQUIPMENT/SERVICE/SYSTEM
Mitsubishi Heavy Industries (Japan)	Turbo machinery / Core barrel assembly
Nukem (Germany)	Fuel technology
SGL (Germany)	Graphite
GEA/Heatric (Germany/UK)	Recuperator
IST Nuclear (South Africa)	Nuclear auxiliary systems
Westinghouse (USA)	Instrumentation
ENSA (Spain)	Pressure boundary
Sargent & Lundy (USA)	Architect/Engineering services

Numerous reviews by independent organizations have been performed to establish the economic goals for the PBMR. A partial listing of reviewers includes AEA Technology, Mitsubishi Heavy Industries, Fuji Electric, Bechtel, McKinsey and Sumitomo Corporation. Economic and plant development targets for the PBMR include:

- The overnight construction cost is expected to be less than <1500 US$/installed kW·h (in 2004$).
- Construction of the first demonstration module is projected at 30 to 34 months, with a 24-month construction period for commercial modules.
- Combined fuel and O&M costs are projected at 9 mills/kW·h.

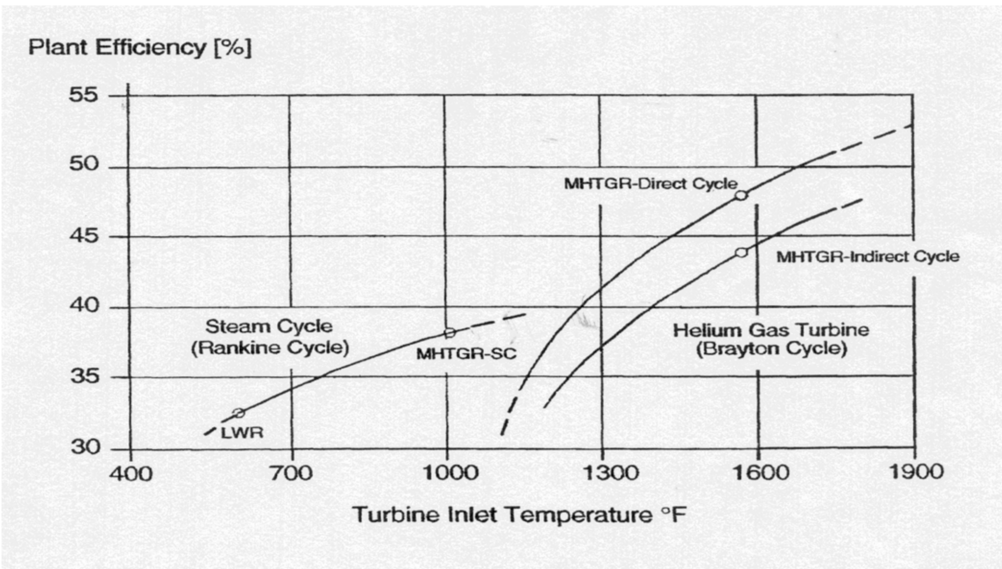


FIG. XIV-4. Plant efficiency comparison [XIV-8].

The single most prominent factor in projecting the low capital and O&M costs for the PBMR is the plant simplification and high efficiency brought about by incorporation of the Brayton cycle, Figure XIV-4.

The direct conversion of the heat energy from reactor to the power conversion unit without an intermediate exchange of coolant medium provides for this substantial reduction in costs. Also, incorporation of the modular reactor has added the unique characteristic of being able to cool the reactor entirely by passive heat transfer mechanisms following postulated accidents without exceeding the failure temperature of the coated particles. This allows for simplification of safety systems that are passive in nature rather then the need for complex safety systems commonly present in LWRs.

Although the initial demonstration unit will be a single PBMR module, the commercialization strategy is to provide complete plants that incorporate eight, four or two modules. An eight module plant would provide an electricity power block of 1320 MW(e). This multi-module business strategy relies heavily on standardization. Plant modularization and system/component standardization are key requirements of the PBMR (Pty) Ltd developers for the commercialization of the PBMR.

The benefits of plant modularization and shop fabrication are many-fold. They include establishing multiple parallel construction paths, utilizing a stable and well-trained work force, use of a controlled environment for quality control and work support functions, and the ability to learn from repetitive work while incorporating the lessons learned. In addition it becomes easier to localize the manufacturing for global sourcing because the volume from the series of plants can support the required investments to set up the shops, fabricate the “jigs”, and develop the optimized procedures and manufacturing drawings.

While most of the LWR building programmes worldwide are based on substantial “in hole” labour, building the plant in “stick” form, i.e., pipe by pipe and rebar by rebar, the PBMR design will rely much more heavily on bringing large manufactured components and composite subparts of systems in pre-assembled units to the site and installing them there. This is a natural consequence of the compact design of the PBMR with its direct helium power conversion cycle, pneumatic fuel handling and storage system, helium inventory system, etc. These systems are primarily composed of manufactured components with connecting pipes and valves, which will also be shop-fabricated and installed in-plant as large equipment/mechanical modules [XIV-6].

The maintainability of the PBMR is enhanced by its simplicity of design, reliance on passive systems rather then complex active systems for reactor safety and the use of the TRISO coated fuel particle. Fewer systems/components and use of passive systems equate to a reduction in plant staffing and overall maintenance procedures and requirements. The radiological cleanliness of the plant brought about by use of the TRISO coated fuel particle in a helium environment has been excellent, thereby lowering contamination and reducing the costs and manpower needs in maintaining and operating the plant.

XIV-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

From the standpoint of sustainability, the next phase of projected PBMR sustainability is two-fold, construction of a demonstration unit at the Koeberg NPP site and construction of a pilot fuel manufacturing facility at the nuclear laboratory facility, Pelindaba, near Pretoria, South Africa. These facilities are intended to assure the capability of the PBMR to operate as the

designers are projecting and warranting, and also to provide an established supply of fresh fuel elements for initial commercial plants of the future. Larger scale, commercial fuel plants will be built to support sales of commercial PBMRs.

A high degree of fuel utilization is met with the on-line refuelling system. The burn-up on each sphere is measured after passing through the core. Each element is then re-circulated until the burn-up is optimized. At this time, the fuel is passed to the spent fuel storage tanks and new fuel inserted in the reactor.

The PBMR utilizes the TRISO coated fuel particles. The high temperature and long-term performance of this coated fuel has been demonstrated primarily by the AVR and the Ft. St. Vrain plants, respectively, to allow plant radiological cleanliness that is considerably lower than previously demonstrated by NPPs throughout the past three decades. This, coupled by the high system efficiency of the PBMR and its corresponding need for less fuel to generate an equivalent amount of electricity, are distinct attributes for minimizing adverse plant environmental impact and optimising waste management.

XIV-1.6.3. Safety and reliability

Safety concept and design philosophy

The safety design philosophy statement for the PBMR [XIV-9] is built on the premise that the PBMR will produce nuclear power in a manner so that:

- No credit is given to active mechanical components within 24 hours following an accident.
- There is a strong negative temperature coefficient.
- The heat transfer is through a static structure.
- Fuel damage does not initiate a challenge to the primary pressure boundary (PPB).
- There is no fuel damage induced by uncooled heatup.
- There is no credible reactivity addition that would lead to fuel damage.
- The static structure survives the worst possible PPB failure (1.95 m).
- Building strength is designed to stop external events.
- Defence in depth is designed into all primary safety functions (example, heat transfer through the core structure is backed up by active cooling).
- A PPB leak below 10 mm is fully filtered [XIV-10].

In the design of any nuclear installation the following three conditions need to be addressed adequately to guarantee the protection of the population at large in the event of an accident or upset condition:

(1) It must be possible to shut down the nuclear installation at any given time. In addition to the reactivity shut down system a strong negative temperature coefficient of reactivity should be able to shut the reactor down from any operational condition, i.e. by physical means alone.

(2) The decay heat must be transported from the fuel at any given time to ensure that the fuel will never overheat or be damaged. Usually this is done by specially devised decay heat removal systems. It is advantageous, however, if the possible level of decay heat is limited by the design. This implies that the decay heat removal does not dependent on the availability of a decay heat removal system but on the laws of nature alone.

(3) The integrity of any barriers preventing the leakage of fission products into the environment must be guaranteed by design. The number of barriers should be diverse and redundant to provide for safety also during hypothetical considerations.

In longer-term innovative designs, such as Generation IV reactors an inherent safety feature is defined as the intrinsic physical capability of the installation, which should encompass the abovementioned protection systems.

The safety design philosophy statement for the PBMR includes:

(1) The fundamental safety philosophy is based on the premise that the TRISO coated fuel will adequately retain its integrity to contain radioactive fission products under normal and accident conditions and thereby allow radiological safety to be assured. This is achieved by relying on fuel, whose performance has been demonstrated under simulated normal and accident conditions, and whose integrity will therefore not be challenged even under accident conditions.

    To ensure this fuel integrity will be maintained, the plant design for normal and accident conditions:

    - Includes sufficient heat removal capability such that fuel temperatures will remain in the proven safe region.
    - Limits chemical and other physical attack on the fuel.
    - Provides adequate measures to ensure the shut down of the reactor and to control reactivity.

    The plant is designed to avoid the need for early operator intervention or the early functioning of any systems with moving mechanical parts in order to maintain nuclear safety.

(2) Appropriate analysis demonstrates that this safety philosophy has been meet with adequate margins. The design has been systematically analyzed to ensure that all potential accident and operating conditions have been identified and considered. This analysis will be updated with any changes to the design during its life and reviewed periodically.

(3) The design is such that any single failure of an element of the safety case will not invalidate the above fundamental safety design philosophy. This is achieved by applying the defence in depth principle.

(4) The design will ensure for all pathways that any dose received by the operators and public and releases to the environment in normal operations, as well as risks from accident conditions, will not only meet all regulatory limits and constraints but will also be as low as reasonably achievable (ALARA).

(5) An extensive test and commissioning programme will demonstrate the performance of all systems, structures, components and materials important to safety. This programme will ensure that any physical phenomena that have a unique application to the safety of the PBMR design are adequately demonstrated on the first module.

(6) To support the safety of the plant the PBMR will operate inside a series of defined programmes throughout its operating life. These will include:

    - Operations.
    - Radiation protection.

- Maintenance.
- Inspection and testing.

The plant design facilitates and makes provision for these programmes.

(1) The PBMR design minimizes the generation of radioactive waste throughout its lifecycle (including decommissioning) and includes appropriate processing, conditioning, handling and storage systems.

(2) Over its entire lifecycle the PBMR is supported by a quality management system [XIV-11].

The plant's fundamental safety characteristics include:

- The utilization of a small normal operational excess reactivity, made possible by continuous fuelling and defuelling.
- The radionuclide retention capability of the fuel elements containing coated fuel particles, even at high temperatures.
- The large negative temperature coefficient of reactivity of the fuel.
- The neutron transparency of helium, used as the reactor coolant and working fluid in the gas turbine.
- The large passive heat removal capability of the reactor design, due to the slender core and uninsulated reactor vessel [XIV-11, XIV-12].

Active and passive systems and inherent safety features

Active systems

There are no active safety systems in the PBMR, which would be required to operate in the event of a major accident [XIV-9].

Passive systems

The long and narrow design of the reactor allows for optimal passive heat removal from the core even under conditions with no coolant flow and the reactor depressurized. Heat flow through conduction and radiation to the RPV, and subsequent removal through the passive heat removal system in the reactor cavity, will limit the maximum fuel temperature and the vessel temperature so that both remain in the safe region.

The following passive safety systems are part of the PBMR design:

- The reactor control system and reserve shutdown system are fail-safe and introduce a cessation of power to reduce reliance on the negative temperature coefficient. The latter is not a system but an attribute of the core and core structures.
- Passive heat removal is ensured from the fuel to the RPV via the core structures and on to the reactor cavity cooling system (RCCS).
- The RCCS is a system of water pipes fed from qualified water tanks to provide 72 hours of cooling in case the active circulation fails which add investment protection for the RPV and the citadel wall. The RCCS is designed to passively remove the heat produced from the reactor during normal operation and any upset condition.
- Analysis shows that the reactor cavity walls can remove enough decay heat in the first 200 hours to ensure both the fuel and RPV do not exceed design limits regardless of the RCCS performance.

- The pressure relief system will ensure that the confinement pressure does not exceed design limits by venting the pressure through a series of rupture panels connecting to a relief shaft. A passive closing mechanism is provided to restore building leak tightness criteria and enable filtered releases.

Inherent safety features

There is no batch refuelling in the PBMR. The small excess reactivity at normal operation is a result of a core that is always in the equilibrium state due to continuous fuelling and defuelling. This means that no excess reactivity is needed to allow operations over a prolonged operating cycle (6 years to allow maintenance on the turbines). Excess reactivity is therefore solely designed to allow for Xenon fluctuations and load following conditions [XIV-9]. As such the design is aimed at maintaining the resultant fuel temperature below the fuel design limits in a hypothetical event of total reactivity insertion available in the reactor.

The fuel kernel coatings consisting of multiple layers of PyC and SiC provide the high temperature radionuclide retention capability. These coated fuel particles have demonstrated excellent capability in containing radiologically significant gaseous and solid fission products under elevated temperature conditions.

The large negative temperature coefficient of the fuel throughout all operating conditions is a result of the low enriched uranium fuel in the graphite matrix. This is caused by the temperature dependence of the resonance absorption in the fertile material ^{238}U. This, together with the negative moderator temperature coefficient, adds up to a strong total negative reactivity coefficient for temperature, which means that the reactor will quickly and inherently counteract a rise in temperature with a reduction in power [XIV-11, XIV-12].

The large RPV surface area designed to passively remove the decay heat resulting from long-term power operation followed by a depressurized loss of forced cooling (DLOFC) event.

Structure of the defence-in-depth

The neutron transparency of helium means that the void coefficient for reactivity of the helium coolant is zero and that the loss of coolant cannot cause a reactivity accident [XIV-9]. The chemical inertness of helium, which holds true for even very high temperatures, dictates that it will not aggravate an accident by chemically reacting with the graphite or fuel. The use of a single phase cooling medium has additional advantages; flashing and boiling of the coolant are impossible, no coolant level measurements are required, no cavitation of pumps can occur and pressure measurements are more certain.

The primary gas envelope can also be considered a barrier against radionuclide release. However, for the short-lived fission gases, the dominant removal mechanism is radioactive decay. For the condensable fission products, the dominant removal mechanism is deposition or plate-out on the various helium wetted surfaces in the primary circuit. The primary pressure boundary, consisting of conventional steel pressure vessels, is designed to ASME Section III. Through-wall cracks are considered unlikely. The chemically inert helium coolant minimizes corrosion and eliminates the complications associated with internal cladding, and only materials for which extensive data exist are to be used in the construction of the vessels.

The reactor building is a reinforced concrete, vented confinement building. No leak-tight requirement is placed on this building. In the event of a break in the primary boundary, it is only the slight gas-borne activity in the primary coolant and a portion of the activity deposited on the surfaces of the primary system that may be released into the reactor building. If the vent opens, natural removal mechanisms (including radioactive decay, condensation, fallout, and

plate-out) reduce the concentration of the radionuclides in the building atmosphere, reducing off site releases [XIV-11, XIV-12].

The protective barriers to confine and retain fission products include:

- The TRISO coated fuel particles are the primary barrier to retention of all fission products.
- The graphite encasing the fuel particles is a secondary barrier and will retain most fission products, although graphite is transparent to noble gasses.
- The helium circuit pressure boundary serves as a physical barrier for retaining short-lived fission gases until the gases decay.
- The reactor building serves as another physical barrier [XIV-13].

Probability of unacceptable radioactivity release beyond plant boundaries

The basic philosophy is that there shall be no identifiable accident that would result in the need to evacuate or shelter people living near the emergency planning zone of 400 meters. The design shall be that an ALARA target of 10% of the regulatory limit for all AOO and DBA shall be attainable for both the public and the personnel. The NNR limits for the public are:

Normal operations and AOO - <250 μSv/annum per site (frequency $1–10^{-2}$).

DBA (10^{-2} to 10^{-6} per event) - 50 mSv;

Beyond 10^{-6} per event a mortality risk is applied.

PBMR expects (based on preliminary results) that there is no identifiable accident, including DBA, for which the DBA target cannot be achieved [XIV-9].

Design basis accidents and beyond design basis accidents

Figure XIV-5 provides PBMR system representative temperatures following a LOFC accident.

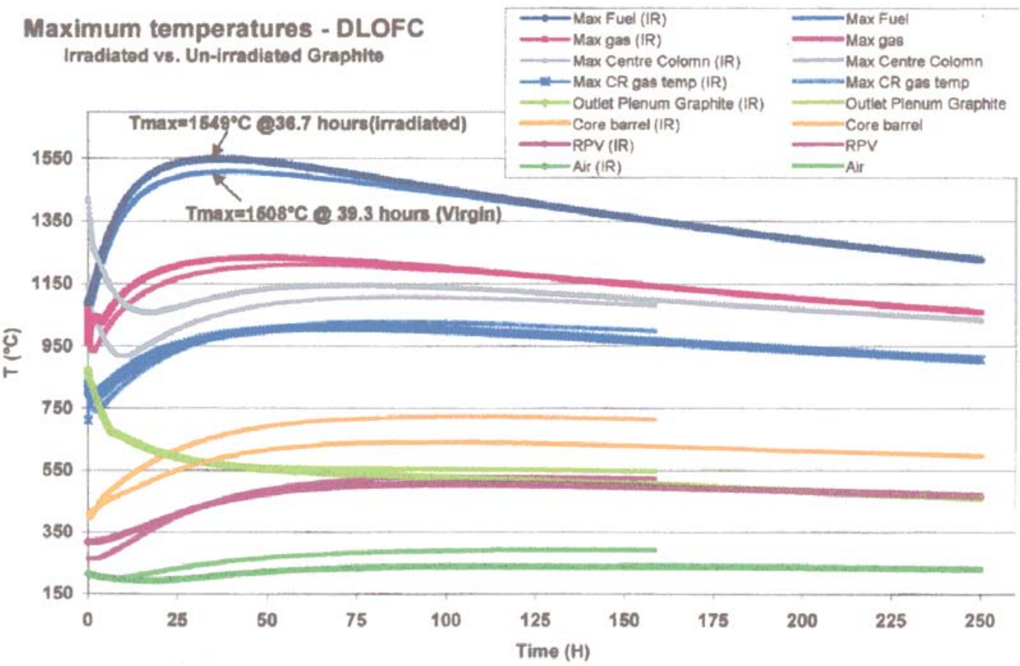


FIG. XIV-5. Temperatures following a simulated DLOFC accident [XIV-10].

An extensive code/modelling programme has been in progress throughout the past decade to evaluate and assure the accuracy of the safety premise upon which the PBMR design is based.

The PBMR design basis accident (DBA) categories that are investigated include: plant trips (PLOFC), primary boundary leaks and ruptures (DLOFC), reactivity, air ingress, seismic, internal missile, aircraft crash, loss of external heat sink, loss of all electrical power supplies, fire, flood and water ingress. Table XIV-3 provides a listing of event types and associated categories for accidents under investigation.

TABLE XIV-3. PBMR FREQUENCY CATEGORIES AND EVENT TYPES [XIV-9]

EVENT TYPE	POSTULATED INITIATING EVENT	FREQUENCY CATEGORY	NOTES
Leaks	Leaks up to 10 mm in diameter	AOO	
Breaks	Breaks between 10 mm and 230 mm	DBA	Including SSE (DBA) and control rod ejection (BDBA)
	Breaks between 230 mm and 1950 mm	BDBA	
Reactivity transients	Group control rod withdrawal	DBA	During these events the HPB remains intact – no release of activity
	RSS removal	DBA	
	Overcooling	DBA	Due to the failure of any valves or any internal pipe work
	Drop of the top reflector on the core	BDBA	
	Inadvertent scram	DBA	
Other reactivity events	Xe oscillations	AOO	
	Water/steam moderation	DBA	
	Inadvertent insertion of one RSS SAS channel	DBA	
	Inadvertent insertion of one control rod	DBA	
Transients	CCS cooling	AOO	
	PLOFC	DBA	

Measures planned in response to severe accidents

No off-site emergency planning is required for the PBMR.

XIV-1.6.4. Proliferation resistance

A primary deterrent to the diversion of nuclear material lies with the basic design of the spherical fuel pebble. The structure of each pebble would require the disassembling of first the outer graphite coating and then the very strong individual particle coatings. Only then would the UO_2 be accessible, but only in minute quantities. The 9.6% new fuel enrichment is a further deterrent and is well within the guidelines for non-proliferation.

Diversion of nuclear material that has been irradiated is further deterred by high depletion in the spent fuel (Figure XIV-6) and the need for remote handling and processing. Access to this material would require entering the PBMR structure. This represents the additional obstacles of penetration of the outside building and spent fuel cavity and, finally, the spent fuel canisters, with the attendant radiological and security considerations. Also, the very low metal loading of each sphere (9 g) makes it necessary to obtain very large numbers of spheres ($>10^5$) to create a significant quantity of nuclear materials. Finally, the design of the fuel handling and storage system creates a completely enclosed system for fuel accountability assurance. Numerous monitoring points and mechanisms have been included in the design to provide remote and independent material accountability at all times.

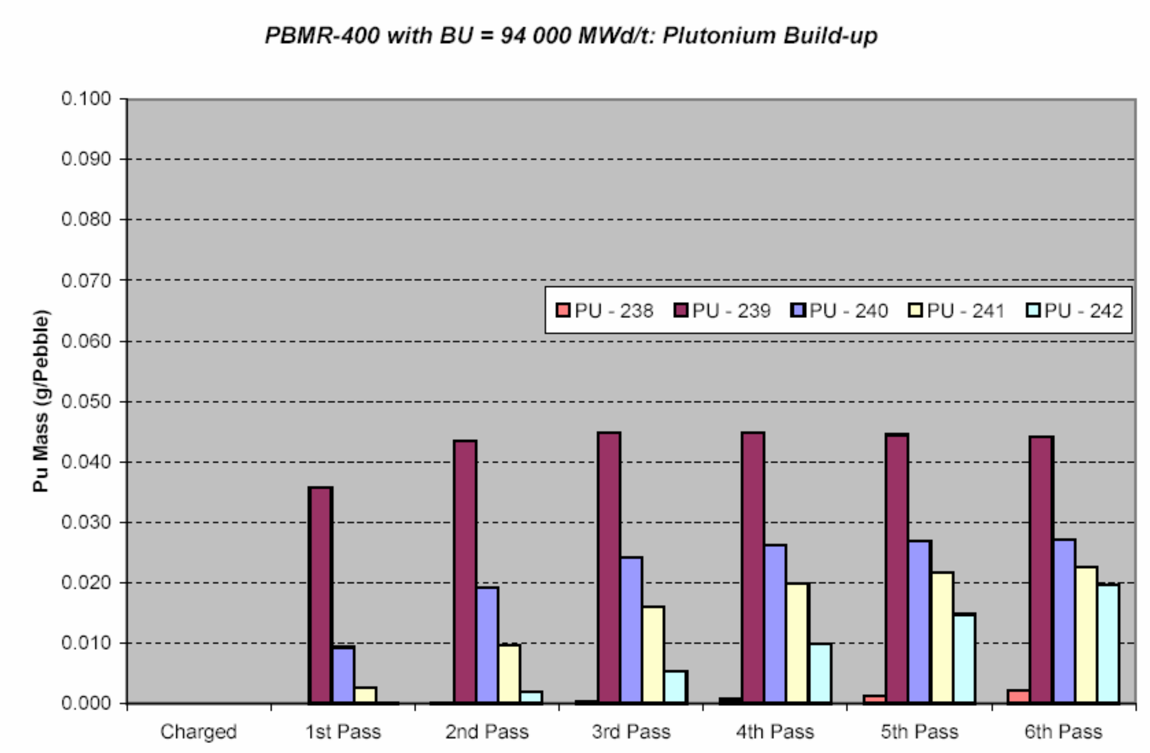


FIG. XIV-6. Plutonium build-up vs. fuel pass [XIV-5].

XIV-1.6.5. Technical features and technological approaches used to facilitate physical protection of PBMR

The reactor building is a single building constructed from concrete. Approximately half of the building will be constructed below ground level. The building is also designed to protect the reactor and equipment from external accidents such as external natural or human-induced events, as well as internal process caused events [XIV-4].

Physical protection of each PBMR module is enhanced by the added strength of the physical building. This building is designed for a seismic acceleration of 0.4 g horizontal and would withstand an aircraft crash of <2.7 ton without penetration on the outside building. Crash of a Boeing 777 aircraft has been analysed to penetrate the outside building barrier, but nuclear safety would not be compromised. Figure XIV-7 provides an overview of the building surrounding a single PBMR module.

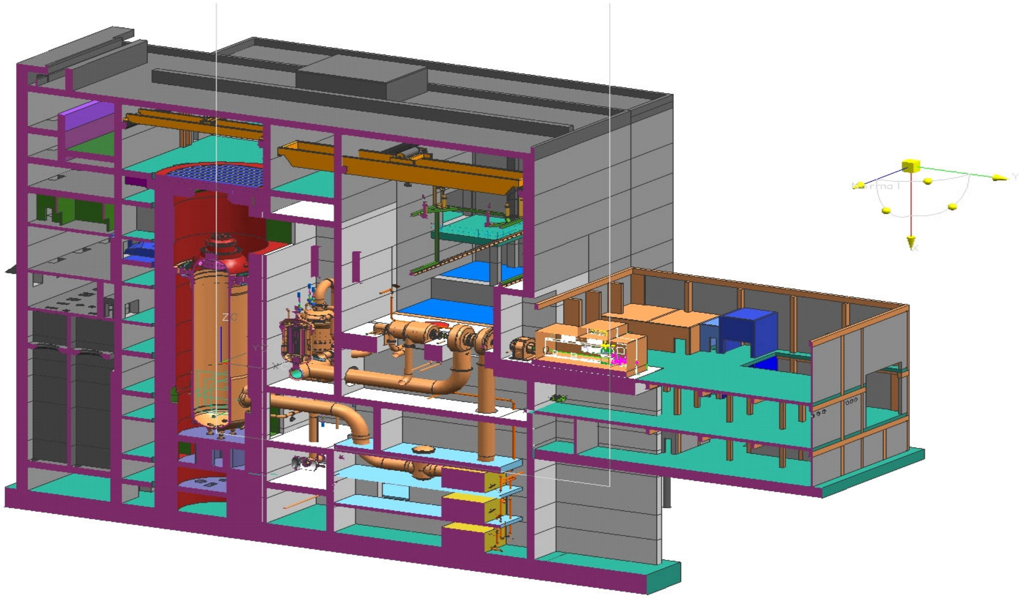


FIG. XIV-7. PBMR single module building [XIV-14].

XIV-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of PBMR

ESKOM is pursuing the PBMR both as capacity additions for their electrical grid and as a commercial offering in the international market place. In this regard, the project has been analyzed from the perspective of its value to the nation of South Africa, ESKOM and the investors. In studying the PBMR, a base case was established to allow analysis with a model, which assumed 10 PBMR units/yr. for local construction and 20 units/yr. for export. The 10 units/yr. for the local market was based on the long-term growth trend at ESKOM of 3.53% (1980–1993), which equates to 1500 MW/yr. on a base of 41 000 MW. This equates to the long-term medium to high growth assumptions of ESKOM [XIV-11].

The PBMR studies were originally initiated in order to meet a future need for distributed electrical generation at a cost competitive to South Africa's current coal generation. The combination of advantages of the PBMR over any other identified options were determined by ESKOM to be the capability for distributed generation, short construction period, small unit size, excellent load following, competitive economics and low impact to the environment [XIV-15]. The PBMR project was initiated solely on the basis of the utility requirements. It was then recognized that the economic advantages of the PBMR would not be limited to the South African grid. Unlike ESKOM's other low cost options (coal and hydro), the PBMR costs are virtually independent of location. The base load cost is very low compared to overseas costs. Therefore, it was determined by ESKOM that this represents an excellent export possibility [XIV-11].

XIV-1.8. List of enabling technologies relevant to PBMR and status of their development

Table XIV-4 provides the status of a sample listing of PBMR related research and development activities with associated facilities.

TABLE XIV-4. ENABLING TECHNOLOGIES FOR PBMR DEVELOPMENT

OBJECTIVE	FACILITY/TEST	STATUS
Investigate neutron physics of the PBMR pebble bed reactor	ASTRA critical facility	Initial testing completed
R&D areas include: Reactor physics & fuel cycles; HTR fuel technology development; materials development; gas turbine power conversion system feasibility issues and HTR plant safety	HTR Technology Network. Partly funded by European Commission	European multi-phase programme for HTR technology base. All phases are currently in work
Investigate magnetic and catcher bearing reliability in high pressure, high temperature helium environment. Maintainability of primary system components to assure ALARA.	Electric Power Research Institute (EPRI)	Initial evaluation on catcher bearing completed. EPRI-Charlotte evaluating ALARA/Ag issues.
Investigation of oxidation of hot graphite cores by oxygen with natural circulation following air ingress events	Natural convection in core with corrosion (NACOK)	Initial tests completed. Additional testing in late 2004-5
Testing of reactor system components in high temperature, high pressure helium environment	Helium test facility (HTF)	Design completed. Procurement/ construction start late 2004
Separate effects tests of heat transfer mechanisms in annular pebble bed.	Heat transfer test facility (HTTF)	Conceptual design complete, pending contract placement
Demonstrate sphere transport system	Air test loop	Initial tests completed
Test for valve operation in high temperature helium environment	Test valves	Initial tests completed
Demonstrated effectiveness of sphere handling for the reactor unit	Core unloading device	Initial tests completed
Validate PBMR codes and models to specific benchmark problems and test facilities such as HTTR and HTR-10.	IAEA CRP on Evaluation of HTGR performance	Validation analyses to be Completed by late 2004
To evaluate behaviour including torque and friction in CRDM	Control Rod Drive test facility	Initial testing completed
Validate capability to measure fuel burn-up in each sphere	Burn-up measurement system prototype	Initial tests completed
Develop pebble bed powered hydrogen production NGNP	Idaho National Engineering & Environmental Laboratory	Point design completed
Validate power conversion unit operational performance with physical model to validate primary thermal-hydraulic code.	Potchefstroom University, PCU Micro-Model	Primary testing completed
Support gas reactor fuel qualification, specifically the TRISO coated fuel	The U.S. Department of Energy	Test programme in planning phase

XIV-1.9. Status of R&D and planned schedule

Initial PBMR development is focused on completion of the detailed design and engineering for a demonstration unit to be located at the Koeberg NPP site north of Cape Town, South Africa. In support of this phase, the South African government has (as of June, 2004) approved a contract of ~260 million rands to the South African technology group, IST, for the design of three key systems on the demonstration plant at Koeberg.

Overall, the PBMR (Pty) Ltd. is formed to oversee commercialization of the PBMR and is comprised of ESKOM (30%), the Industrial Development Corporation of South Africa (25%) and British Nuclear Fuel Limited (22.5%). A stake of 10% is earmarked for black empowerment and the final 12.5% for a foreign partner. The full cost of the demonstration plant is estimated at ~US$ 1 billion. Full funding of the demonstration plant is yet to be completely realized.

The South African government has designated the PBMR a national strategic project with a cabinet level committee appointed in February 2004.

The environmental impact assessments were completed in 2002 with positive record of decisions issued in 2003 for the demonstration plant and pilot fuel plant. In January 2005, the EIA was remanded back to the DEAT by the courts to address procedural flaws. The additional reviews should be completed in mid-2005. The demonstration plant site preparation is scheduled to begin at the Koeberg NPP site in the first quarter of 2007 with fuel loading anticipated for mid-2010. The commercial acceptance by ESKOM is scheduled for early 2011 [XIV-16].

Development goals related to PBMR application for Generation IV

PBMR development goals related to Generation IV include the following [from XIV-17]:

- To improve commercial competitiveness.
- To minimize waste production.
- To diversify into other fields of application.
- Enhance fuel performance.
- Increase reactor outlet temperature to enable the PBMR to operate as a very high temperature reactor (VHTR).

To achieve PBMR-VHTR development will require R & D in the following areas:

(1) The enhancement of fuel performance that includes a.) maximum accident fuel temperature of ~2000ºC, b.) fuel burn-up of >200 GW d/t U, c.) same or better fission product retention capability, and d.) improved resistance to oxidation.

(2) An increase in reactor outlet temperature to 1,200ºC with corresponding high temperature qualification in primary system materials and components.

This development will allow:

- An increase in reactor power to >600 MW(th) for the same core geometry and a coolant mass flow rate which will increase the commercial plant competitiveness.
- A reduction of spent fuel produced through higher fuel burn-up.

- A reduction of contamination through better fission product retention even at high temperatures.
- Higher reactor outlet temperature to enhance application to other missions.

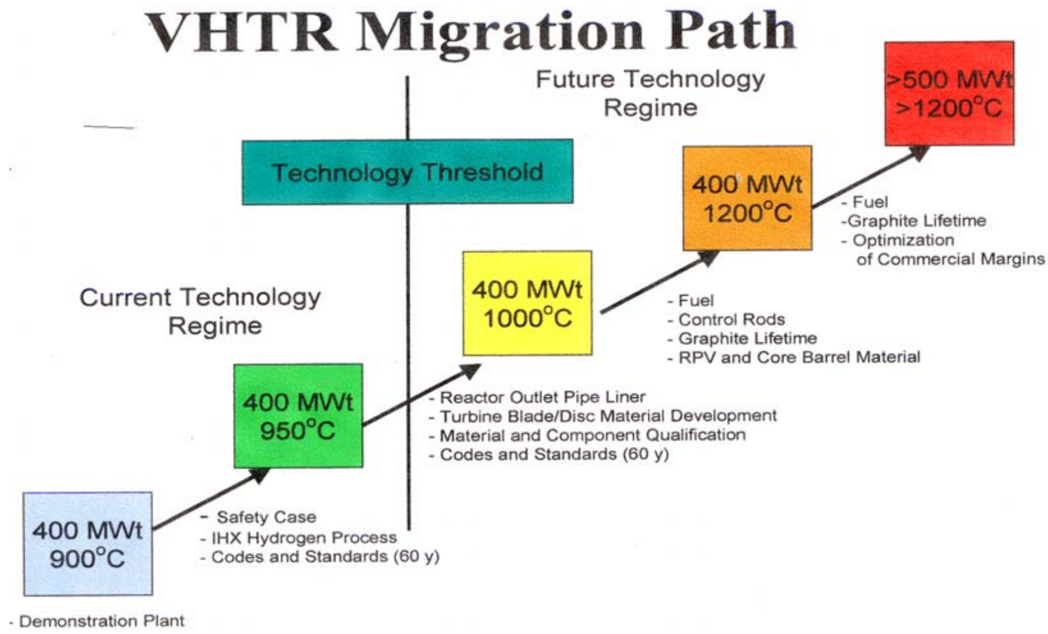


FIG. XIV-8. Conceptual development path from demonstration PBMR to VHTR [XIV-10].

Figure XIV-8 depicts a possible long range development path from the 400 MW(th) PBMR demonstration plant with a 900°C average core outlet temperature into the future technology regime of a 600 MW(th) PBMR of 1200°C outlet temperature.

XIV-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

The innovative nature of the PBMR is represented throughout the entire primary coolant system as follows:

- The primary system will facilitate the direct movement of heat energy from the pebble bed reactor to the power conversion system by the use of high pressure, high temperature helium. This will be a first-of-a-kind NPP application. There will not be an intervening coolant such as steam.
- The application of a relatively small reactor with its very tall annular pebble bed core has never been demonstrated.
- Components such as the compact plate-fin recuperator and the dry helium seals on the turbomachine have never been used under the helium conditions of the PBMR.
- The passive nature of this 400 MW(th) modular core and its anticipated capability to recover from all normal operating and reasonable transient conditions and accidents without the intervention of active safety systems represents a new innovative NPP feature.
- A commercialization strategy that is unique for large nuclear power plants. One that relies on standardization and modularization to the extent that shop fabrication of major systems can be performed remote from the actual plant site.

XIV-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

Considerable international attention has been focused on the development of the gas turbine modular HTGR plant. Included are the GT-MHR, GTHTR300 and ACACIA plants and a brief overview of their design intent. Details associated with the design of these plants can be found in this TECDOC and also in reference [XIV-11].

The Gas Turbine-Modular Helium Reactor (GT-MHR)
The primary designer of this plant is OKBM, the Russian Federation, with support from General Atomics and the DOE of the U.S., and Framatome of France. This closed cycle gas turbine HTGR utilizes TRISO coated fuel particles in prismatic fuel elements. The conceptual design is now completed on this 600 MW(th)/293 MW(e) plant which is currently under development for the destruction of weapons plutonium, but with the longer-term goal of commercial development utilizing a uranium fuelled core. An aggressive development programme is underway on this plant, primarily in the Russian Federation. This HTGR as well as others such as the PBMR are sharing in the research and development activities currently being performed throughout the world such as with Europe’s HTR Technology Network and the IAEA’s HTGR related coordinated research programmes.

GTHTR-300 plant
The Japan Atomic Energy Research Institute is developing this modular HTGR closed cycle gas turbine plant for electric generation. Conceptual design of this 600 MWt is under development. The power conversion system includes a vertical heat exchanger vessel and a horizontal turbo-machine vessel to allow for bearing support and stable rotor operation. The cycle configuration has been simplified in comparison to the GT-MHR by elimination of a compressor unit and the corresponding intercooler. The overall net plant efficiency for this simplified unit is 45.4%.

ACACIA plant
ECN Nuclear Research in the Netherlands is developing a conceptual design of an HTGR for the combined generation of heat and power for industry within the Netherlands as well as for possible export. The ACACIA plant utilizes a 40 MWth pebble bed HTGR to produce 14 MW of electricity and 17 tonnes of 10 bar, 220°C. steam per hour. The electric generation system utilizes a basic closed cycle gas turbine, which receives helium from the HTGR at 800°C and 2.3MPa. After the recuperator, a secondary helium loop removes heat from the primary system via an intermediate heat exchanger (precooler), which then transfers energy to the steam / feed water system for industrial use [XIV-18].

XIV-2. Design description and data for PBMR

XIV-2.1. Description of the nuclear systems

Reactor core and fuel design

The PBMR is an annular core that contains a 2.0 m diameter fixed graphite column with 8 equally spaced holes to accept the boron carbide reserve shutdown pellets. The fixed centre column was chosen in order to reduce the amount of core bypass flow to lower the fuel temperature. The graphite column includes an interlocking design and, if required, can be replaced with remote handling equipment. Figure XIV-9 provides a horizontal cross section diagram depicting the core and centre column.

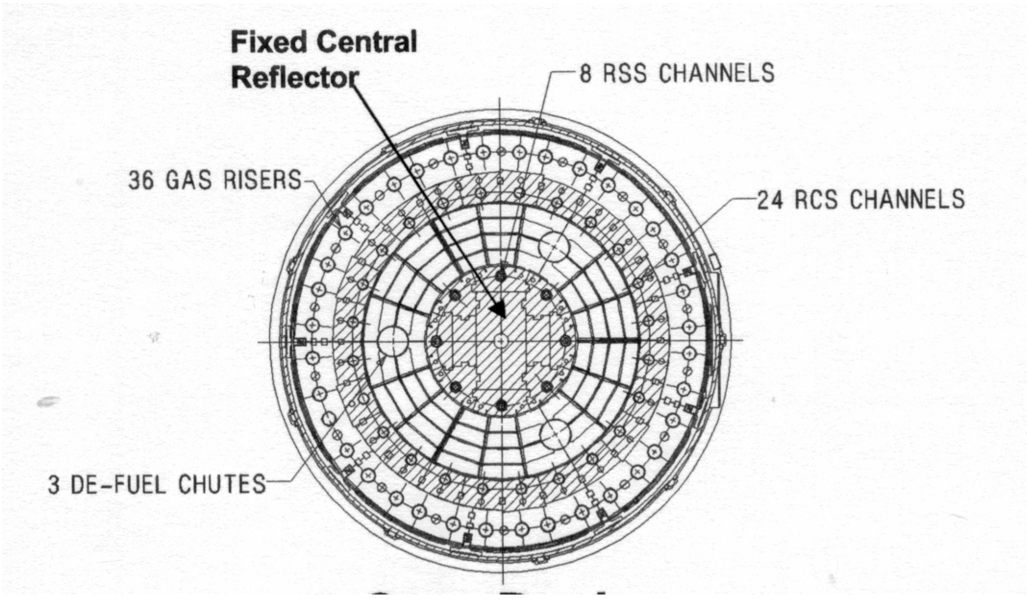


FIG. XIV-9. Horizontal cross-section of PBMR core [XIV-14].

The core is comprised of ~452 000 fuel spheres or "pebbles" (Fig. XIV-10). The fuel spheres have a diameter of 60 mm, and each sphere contains nominally 15 000 UO_2 TRISO coated micro spheres imbedded in a graphite matrix. The fuelling scheme employed is the continuous on-line multi-pass method similar to the designs used in Germany. The spent fuel system consists of a core unloading device in each of the three defuelling chutes from where the fuel is moved pneumatically to the burn-up assaying equipment located at a level above the reactor unit. After the burn-up has been determined, the fuel is routed either to the spent fuel tanks or back to the core, depending on its burn-up. The fuel spheres are reloaded into the core through three fuelling lines. The spent fuel tanks have sufficient capacity to hold all the spent fuel generated during the entire operating life of the facility [XIV-4].

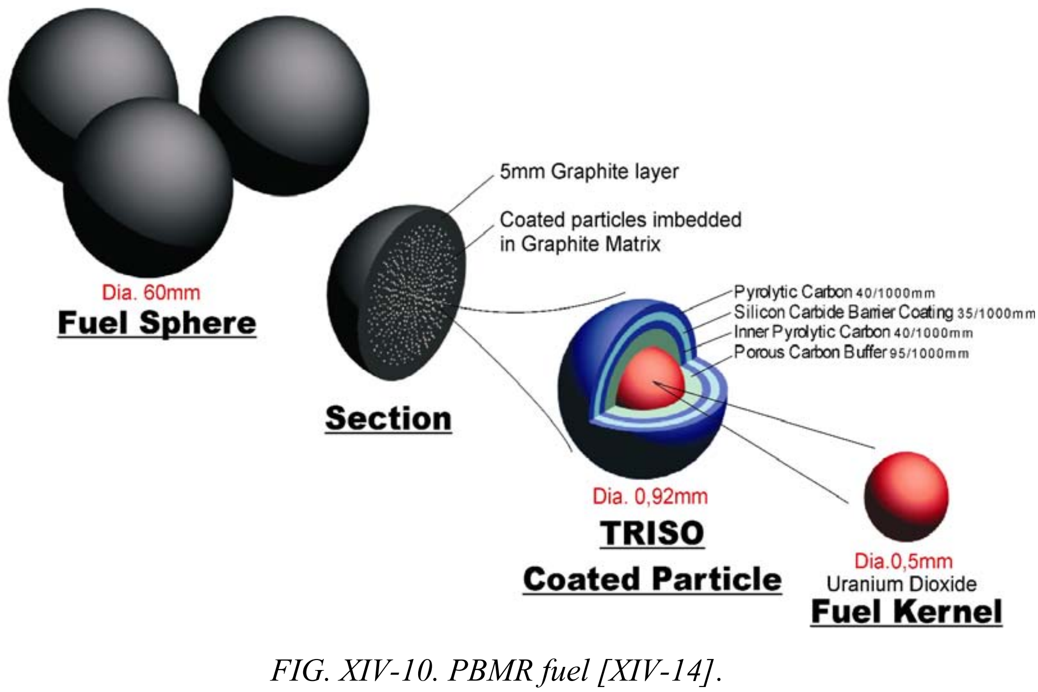


FIG. XIV-10. PBMR fuel [XIV-14].

The reactor fuel is continually replenished with fresh and reusable fuel pebbles added at the top of the core while used fuel is removed from the bottom. Each used pebble exiting the core is measured to determine the remaining amount of fissionable material and returned to the reactor if it contains usable fissionable material. Each fuel pebble is cycled through the core ~6 times [XIV-13].

TABLE XIV-5. FUEL ELEMENT CHARACTERISTICS

DESCRIPTION	VALUE
Fuel pebble	
Fuel pebble outer radius	3.0 cm
Thickness of fuel free zone	0.5 cm
Total heavy metal loading per fuel pebble (equilibrium fuel)	9 g
Carbon content	189 g/fuel sphere
Average pebble bed packing fraction	0.61
Number of spheres in core	~455 000
Coated Particle	
Fuel kernel diameter	500 micron
Kernel material type	UO_2
UO_2 density	10.4
Kernel Coating Material	C / C / SiC / C

Main heat transport system

Figures XIV-1 and XIV-2 provide a schematic diagram and a side view rendition of the primary system, respectively. Figures XIV-11 and XIV-12 are renditions of the PBMR primary system as seen from above and the reactor pressure vessel, respectively.

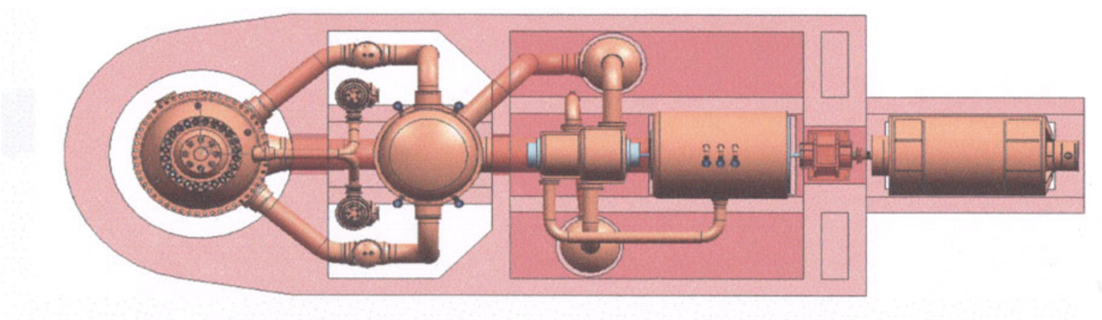


FIG. XIV-11. PBMR primary system as seen from above [XIV-19].

The heat removal path under normal operation is the PCU with the primary energy user being the turbine and compressors. Also within this path are the precooler and intercooler heat exchangers. These coolers utilize water as the cooling medium on the secondary side.

Two other cooling systems are available for utilization or removal of primary system energy: the core conditioning system (CCS) and the reactor cavity cooling system (RCCS). The CCS serves the functions of removal of core decay heat when the Brayton cycle is not operating and the provision of helium flow through the core for reactor heat-up purposes during start-up operations [XIV-11].

The RCCS removes heat transferred from the reactor vessel to the cavity around the vessel. The basic functions and requirements of the RCCS are:

- To provide investment protection by preventing thermal radiation from impinging directly onto the concrete walls of the reactor cavity.
- To remove all heat from the reactor cavity during normal operation, thereby maintaining the concrete surfaces of the cavity below their design temperature limits being nominally 65°C under normal operating conditions.
- To remove all decay and residual heat generated in the reactor cavity during a pressurized and depressurized loss of forced core cooling event.
- In the event of the loss of active pumping capacity of the secondary cooling system, to remove heat from the reactor cavity passively, and to release this heat to the atmosphere in the form of steam. This passive operation continues for a minimum period of 96 hours.
- To switch from active to passive operation without any mechanical, electrical or human intervention.

It is not a primary function of the RCCS to ensure that the fuel does not exceed its maximum allowable temperature, but together with the design of the heat transfer path from the fuel to the outer surface of the reactor pressure vessel (RPV), the RCCS is providing a heat sink for continuous removal of heat transferred from the RPV during normal operation, and in a postulated loss of forced cooling event [XIV-4].

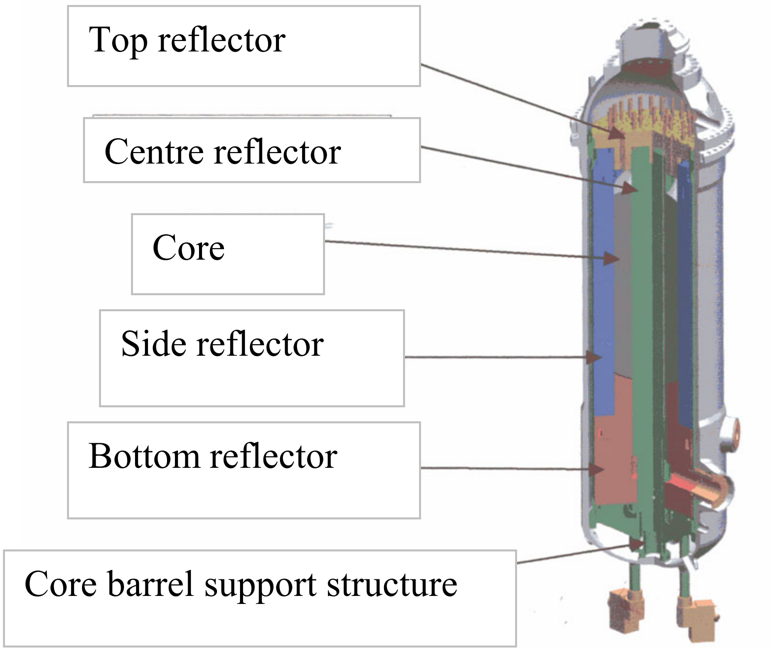


FIG. XIV-12. PBMR core [XIV-3].

Figures XIV-13 and XIV-14 provide simplified schemes for emergency and passive heat removal paths in normal operation and in accidents.

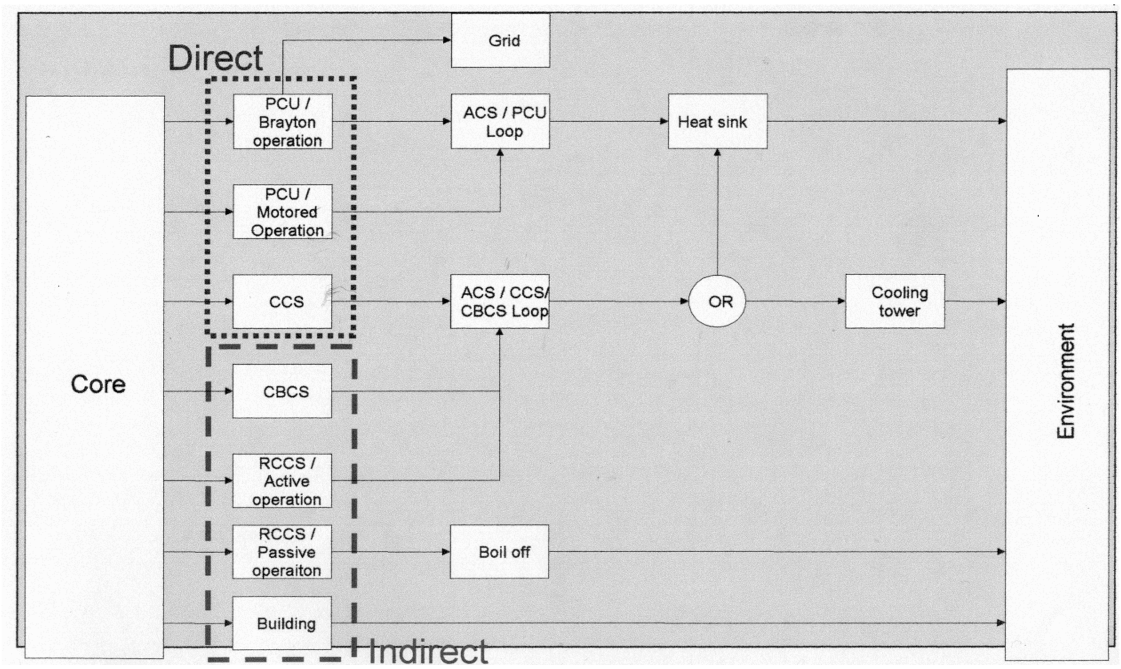


FIG. XIV-13. Emergency heat removal paths [XIV-9].

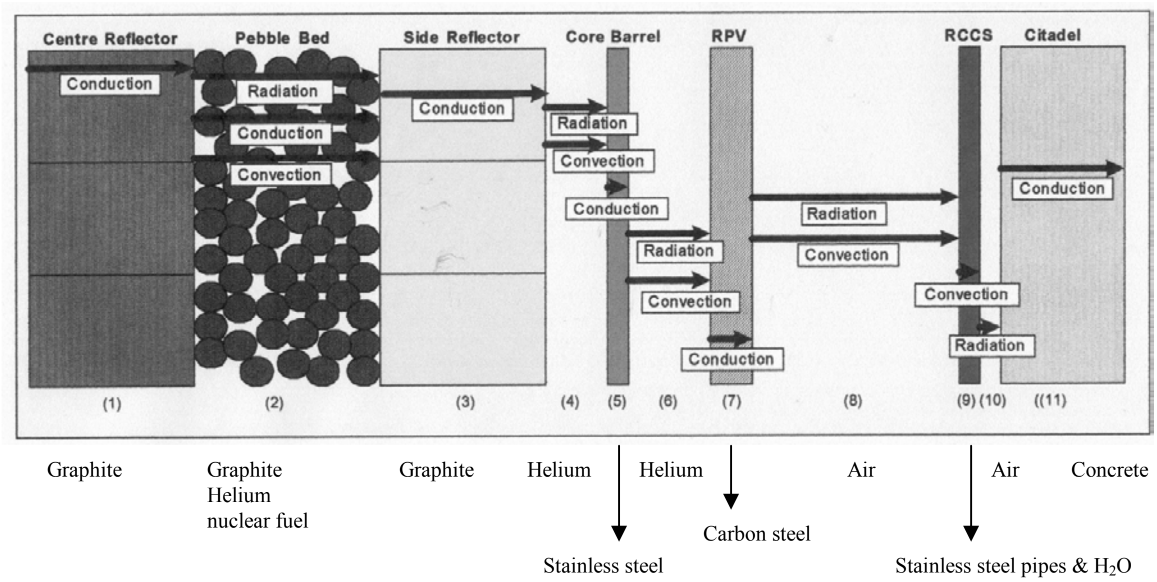


FIG. XIV-14. Passive heat removal paths [XIV-9].

XIV-2.2. Description of the turbine generator plant and systems

The original design of the PBMR primary system included three separate, vertically orientated rotating machines. These included the low pressure turbo-compressor, the high pressure turbo-compressor and the turbine generator. A recent major decision by the developers, PBMR (Pty) Ltd., resulted in usurping the 3-shaft PCU for a single high speed rotating machine comprised of two compressors, turbine, reduction gear and generator in a horizontal configuration with oil bearings and dry gas seals. This major design change was initiated due to recent developments in the application of dry gas seals that allowed use of conventional oil bearings, and improvements in gear reduction capability for high capacity applications. A representation of this machine is provided in Fig. XIV-15.

The advantages of the single machine over the three shaft PCU include:

- Approximately 1,000 EMB penetrations of pressure boundary eliminated.
- Elimination of potential power turbine generator unstable operations during trip and subsequent restart.
- Less complex control system.
- Easier to balance shaft thrust forces.
- No large resistor bank required to maintain load on trip.
- Elimination of start-up blower system
- Conversion from 50 to 60 Hz simplified.
- Improved maintenance very similar to combustion gas turbine systems.
- No special rotor balancing facilities required; conventional commissioning.
- Reduced cost of turbo machinery equipment.
- Significantly lower R&D required, i.e. lower development risk [XIV-14].

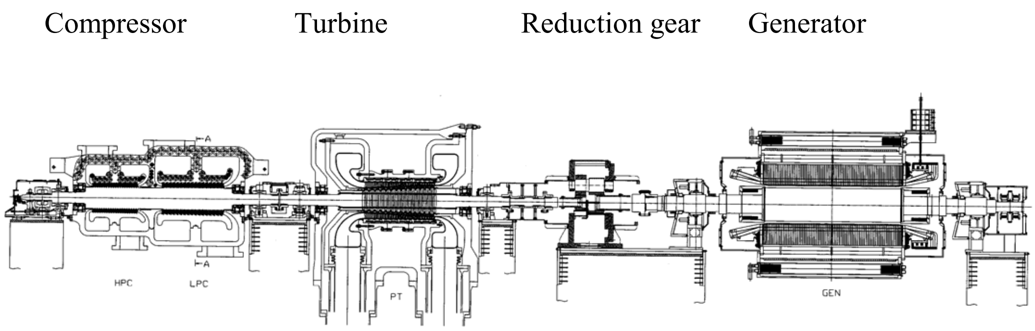


FIG. XIV-15. PBMR power conversion machine (by Mitsubishi Heavy Industries) [XIV-3].

XIV-2.3. Systems for non-electric applications

Utilization of the PBMR for the production of hydrogen is being considered by a number of entities including Westinghouse and the Idaho National Laboratory (INL).

Figure XIV-16 depicts the coupling of the PBMR with the sulphuric acid decomposition reactor and vaporizer and the HI decomposition reactor. A 0.25 mile separation would exist between the PBMR and the hydrogen plant. The PBMR provides heated helium to the hydrogen plant reactors. The hydrogen plant circulates H_2SO_4 and productions from the HI reactor at low temperatures and includes three major chemical reactions.

Figure XIV-17 depicts the PBMR/Westinghouse process interface for the production of hydrogen. This is a co-generation application for the PBMR, supplying electricity and heated helium. A 0.25 mile interface exists between the PBMR and the hydrogen production process. However, this process utilizes an electrolizer rather then the HI reactor, and circulates H_2SO_4 and other products from the decomposition reactor and vaporizer at low temperatures and requires only a single chemical reaction and single heat transmission.

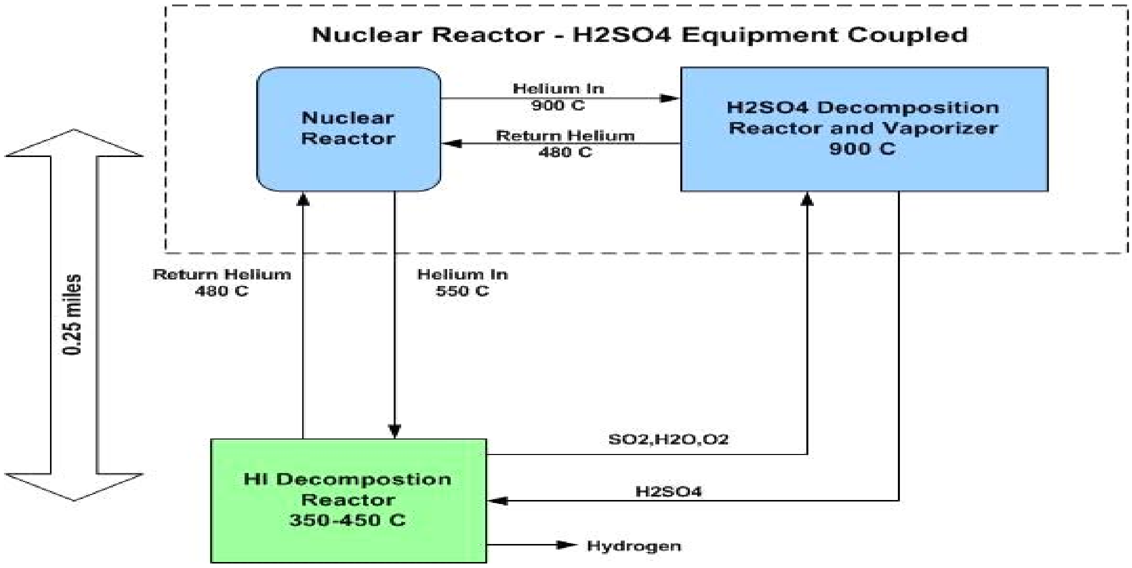


FIG. XIV-16. I/S Process interface with the PBMR [XIV-3].

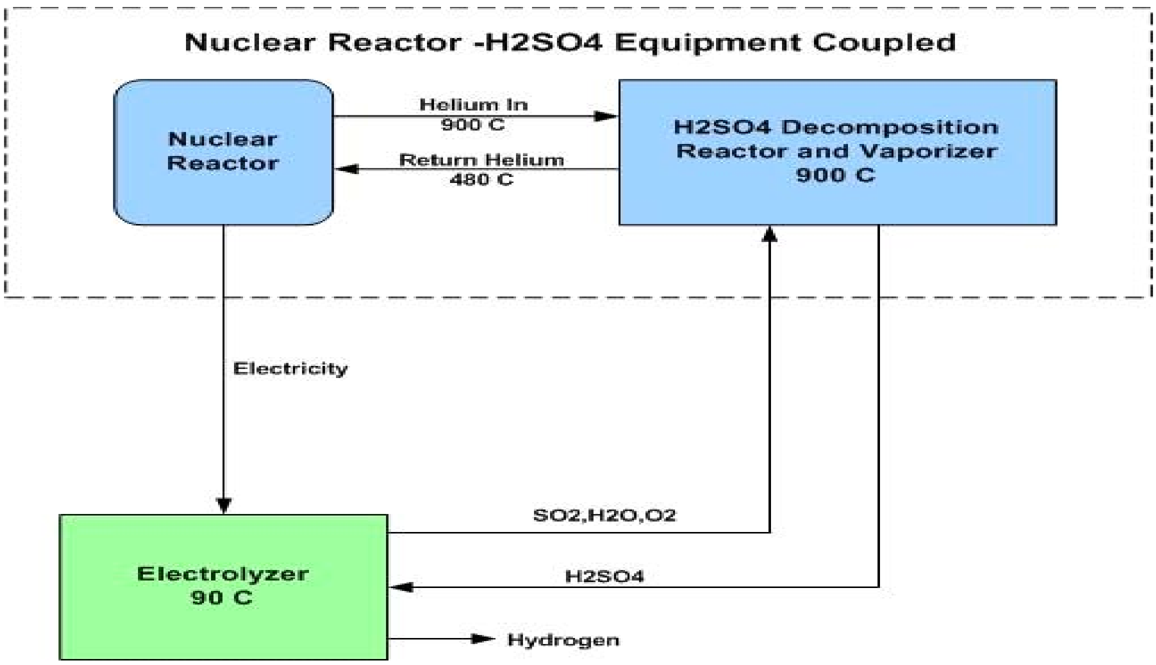


FIG. XIV-17. PBMR/Westinghouse hydrogen/electricity co-generation plant [XIV-3].

The INL has performed an assessment of both the pebble bed and prismatic fuel type helium gas cooled reactor. This is for the next generation nuclear plant (NGNP) to demonstrate emissions-free nuclear assisted electricity and hydrogen production. These designs will meet the three basic requirements that have been set for the NGNP: a coolant outlet temperature of 1,000°C, passive safety, and a total power output consistent with that expected for commercial HTGRs.

Two point design studies of a passively safe pebble bed NGNP have been developed at the INL, a 300 and a 600 MW(th) module. A modified pebble design that improves both the fuel utilization and safety was identified (the fuel zone radius in the pebble was adjusted to optimize the fuel-to-moderator ratio). Design optimization calculations were performed with a generic algorithm that automatically selects a sequence of design parameter sets to meet specified fitness criteria increasingly well. And, finally, cross-sections were calculated more accurately for pebble bed reactors, and research needs were identified for the further refinement of the cross section calculations [XIV-20]. Table XIV-6 provides a summary of the characteristics for the 268 MW(th) PBMR and the NGNP-300 and NGNP-600 pebble bed cores.

TABLE XIV-6. FEATURES OF 268 MW(th) PBMR*, NGNP-300 AND 600 MW(th) CORES [XIV-20]

DESCRIPTION	PBMR*	NGNP-300	NGNP-600
Power (MW)	268	300	600
Inlet temperature (°C)	503	600	600
Outlet temperature (°C)	908	1000	1000
Coolant flow rate (kg/s)	126	144	288
Active core volume (m^3)	81.8	79.8	119.4
Inner reflector radius (cm)	~ 87	40	150
Core radius (cm)	175	175	250
Outer reflector thickness (cm)	75	76	76
Active core height (m)	8.4	8.75	9.75
Mean pebble temperature (°C)	806	862	863
Peak pebble temperature (°C)	1041	1027	1028
Peak pebble power (W)	1397	1301	1791
Mean core power density (W/cm^3)	3.3	3.8	5.03
Peak core power density (W/cm^3)	6.8	7.9	9.3
FUEL UTILIZATION			
Description	*PBMR**	*NGNP-300*	*NGNP-600*
K_{eff}	1073	1073	1073
Discharge burn-up (MW·d/kg U)	80.1	94.3	82.6
Enrichment	8%	8%	8%
Number of particles per pebble	15 000	13 271	13 106
Heavy metal mass daily throughput (g/day)	3334	3183	7260
Heavy metal mass throughput/MW·d	12.5	10.6	12.1

* The values indicated in Table XIV-6 are for the smaller 268 MW(th) unit rather then the commercial 400 MW(th) plant.

XIV-2.4. Plant layout

The commercial intent of the PBMR developers is to offer plants with a multiple of individual units. These multiples are tentatively given as 2, 4 or 8 modules per plant. The PBMR multi-module design is very compact from the perspective of the main building footprint. Figure XIV-18 illustrates the differences in size between a large modern LWR power block (to provide 1350 MW(e)) and an "8-pack" PBMR plant of comparable output (1320 MW(e)) [XIV-6].

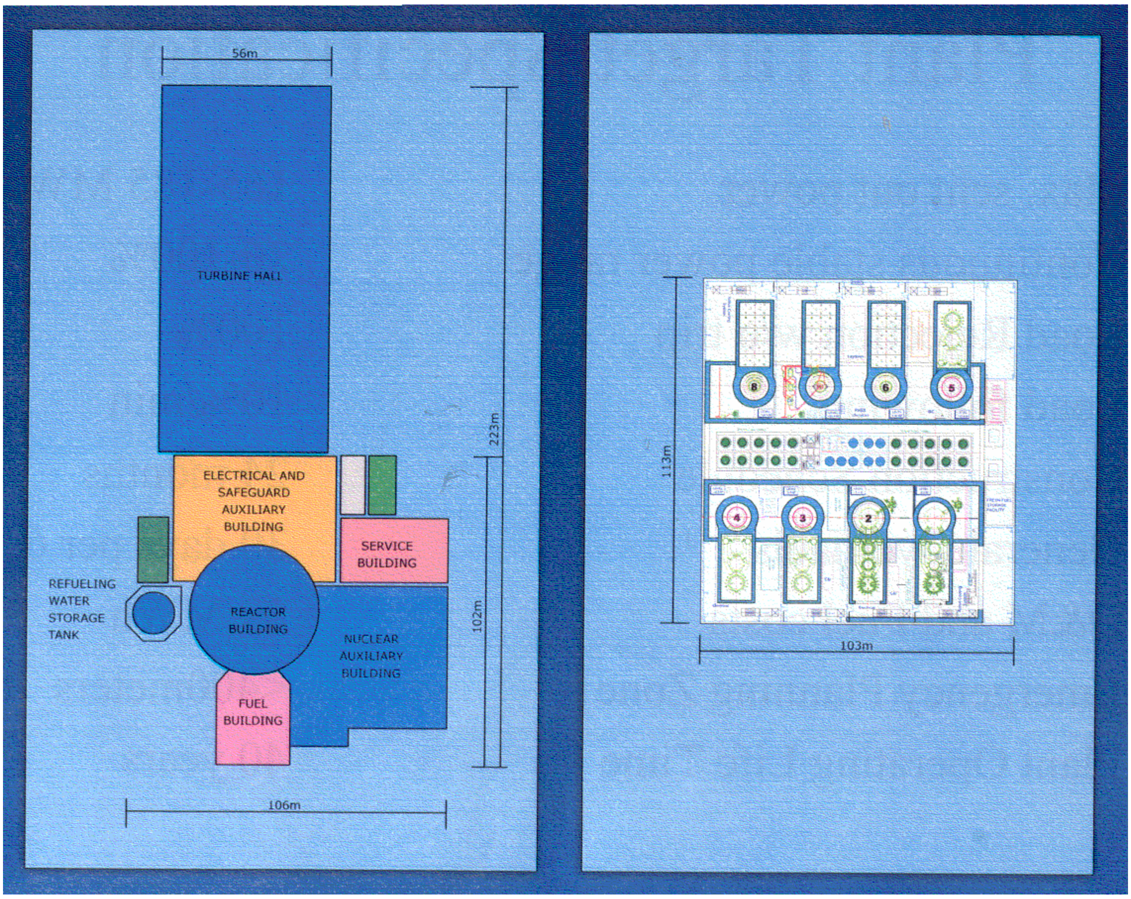


FIG. XIV-18. Comparison of large LWR footprint with a PBMR "8-Pack" [XIV-10].

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ANNEX XV

GAS TURBINE MODULAR HELIUM REACTOR (GT–MHR)

General Atomics, United States of America

XV-1. General information, technical features and operating characteristics

XV-1.1. Introduction

The Gas Turbine — Modular Helium Reactor (GT–MHR) couples a high temperature gas cooled reactor (HTGR) with a Brayton power conversion cycle to produce electricity at high efficiency. Because of its capability to produce high coolant outlet temperatures (at least 850°C with potential for still higher temperature), the modular helium reactor system can also efficiently produce hydrogen by high temperature electrolysis or thermochemical water splitting.

The HTGR concept evolved from early air-cooled and CO_2-cooled reactors. The use of helium in lieu of air or CO_2 as the coolant in combination with a graphite moderator offered enhanced neutronic and thermal efficiencies. The combination of helium cooling and graphite moderator makes possible production of high temperature nuclear heat, and hence the name, high temperature gas cooled reactor (HTGR).

To-date, seven HTGR plants have been built and operated as shown in Table XV-1. The research facilities and reactors are summarized in more detail in [XV-1]. The first HTGR plant was the 20 MW(th) Dragon test reactor in the UK. The Dragon was followed by construction of two relatively low power plants, the 115 MW(th) Peach Bottom I (PB–1) in the US and the 49 MW(th) AVR in Germany. PB–1 and AVR demonstrated electricity generation from HTGR nuclear heat using the Rankine (steam) cycle. These two plants were followed by the construction of two medium sized steam cycle plants, the 842 MW(th) Fort St. Vrain (FSV) plant in the US and the 750 MW(th) Thorium High Temperature Reactor (THTR) plant in Germany. In addition to demonstrating the use of helium coolant (with outlet temperatures as high as 950°C) and graphite moderator, these early plants also demonstrated coated particle fuel, a fuel form that employs ceramic coatings for containment of fission products at high temperature which is a key feature of HTGRs.

In the US, General Atomics used the HTGR technology from these early plants to design several large, 2000-3000 MW(th), HTGR plants and orders were received for 10 of these large HTGR plants. The large HTGR plant orders were cancelled, along with the cancellation of orders for a large number of other nuclear power plants, following the oil embargo in the early 1970s and the ensuing energy conservative measures that dramatically reduced energy demand and the need for new electricity generation capacity. Most recently, two additional HTGR test reactors have been constructed and are successfully operating, the 30 MW(th) high temperature test reactor (HTTR) in Japan and the 10 MW(th) high temperature reactor (HTR 10) in China (Table XV-1), with design outlet temperatures of 950ºC and 900ºC respectively.

The US modular HTGR concept began in 1984 when the US Congress challenged the HTGR industry to investigate the potential for using HTGR technology to develop a “simpler, safer” nuclear power plant (NPP) design. The goal was to develop a passively safe HTGR plant that was also economically competitive.

TABLE XV-1. HTGR PLANTS CONSTRUCTED AND OPERATED

FEATURE	DRAGON	PEACH BOTTOM	AVR	FORT ST. VRAIN	THTR	HTTR	HTR–10
Location	UK	USA	Germany	USA	Germany	Japan	China
Power, MW(th)/MW(e)	20/ -	115/40	46/15	842/330	750/300	30/ -	10/ -
Fuel elements	Cylindrical	Cylindrical	Spherical	Hexagonal	Spherical	Hexagonal	Spherical
He temperature (inlet / outlet), °C	350/750	377/750	270/950	400/775	270/750	395/950	300/900
He pressure, Bar	20	22.5	11	48	40	40	20
Power density, MW/m^3	14	8.3	2.3	6.3	6	2.5	2
Fuel coating	TRISO(a)	BISO(b)	BISO(b)	TRISO(a)	BISO(b)	TRISO(a)	TRISO(a)
Fuel kernel	Carbide	Carbide	Oxide	Carbide	Oxide	Oxide	Oxide
Fuel enrichment	LEU(c) / HEU(d)	HEU(d)	HEU(d)	HEU(d)	HEU(d)	LEU(c)	LEU(c)
Reactor vessel	Steel	Steel	Steel	PCRV(e)	PCRV(e)	Steel	Steel
Operation years	1965–1975	1967–1974	1968–1988	1979–1989	1985–1989	1998–	1998–

(a) TRISO refers to a fuel coating system that uses three types of coatings, low density pyrolithic carbon, high density pyrolithic carbon and silicon carbide
(b) BISO refers to a fuel coating system that uses two types of coatings, low density pyrolithic carbon and high density pyrolithic carbon
(c) LEU means low enriched uranium (<20% U^{235})
(d) HEU means high enriched uranium (>20% U^{235})
(e) PCRV means pre-stressed concrete reactor vessel

Like most nuclear power plants up to that time, HTGR plants had been designed with reactor core length-to-diameter (L/D) ratios of about 1 for neutron economy. Detailed evaluations showed that low power density HTGR cores with L/Ds of 2 or 3, or more, were effective for rejecting decay heat passively. In the long slender, low power density HTGR cores, it was found that decay heat could be transferred passively by natural means (conduction, convection and thermal radiation) to a steel reactor vessel wall and then thermally radiated (passively) from the vessel wall to surrounding reactor cavity walls for conduction to a naturally circulating cooling system or to ground itself.

To maintain the coated particle fuel temperatures below damage limits during passive decay heat removal, the core physical size had to be limited, and the maximum reactor power capacity was found to be about 200 MW(th) for a solid cylindrical core geometry. However, a 200 MW(th) power plant was not projected to be economically competitive. This led to the development of an annular core concept to enable larger cores and therefore, higher reactor powers. The first modular high temperature gas cooled reactor (MHTGR) designed with an annular core had a power of 350 MW(th). When coupled with a steam cycle power conversion system, the plant had a net thermal efficiency of 38% and was economically competitive (marginally) at that time (late 1980s). To improve economics while maintaining passive safety, the core power was subsequently raised to 450 MW(th) and then to the current reference core power of 600 MW(th). The resultant modular HTGR design, now known as the modular helium reactor (MHR), represents a fundamental change in reactor design and safety philosophy.

The latest evolution made for the purpose of economics has been replacement of the Rankine steam cycle power conversion system with a high efficiency Brayton (gas turbine) cycle power conversion system to boost the thermal conversion efficiency to ~48%. The coupling of the MHR with the gas turbine cycle forms the GT–MHR. The GT–MHR retains all of the MHR passive safety characteristics but is projected to have more attractive economics than any other generation alternative.

In the mid 1990s, the GT–MHR concept became the focus of a joint effort between the United States and the Russian Federation as an option for the destruction of weapons grade plutonium in conjunction with electricity production. This mission has advanced the development of the GT–MHR concept, with the recognition that the resulting design could be deployed for commercial electricity production using low enriched uranium in a modified core with minimal overall plant design modifications.

Primary stakeholders in the plutonium consumption GT–MHR project include the National Nuclear Security Administration (USA) and Rosatom and Experimental Design Bureau of Machine Building (OKBM, the Russian Federation). Additional supporting stakeholders, with interest in technology development and commercial applications, include multiple institutions contracted by OKBM in the Russian Federation; the Electric Power Research Institute, General Atomics, and Oak Ridge National Laboratory in the United States; and participants from Japan and the European Union through the International Science and Technology Centre.

In the USA, the technology embodied in the GT–MHR concept has been recognized to have high potential, with modest further development work, to meet the requirements for the next generation nuclear plant (NGNP) demonstration project planned to be built at the Idaho National Laboratory (INL).

XV-1.2. Applications

The GT–MHR concept is designed for high efficiency electric power production. Because of its capability to produce high coolant outlet temperatures (at least 850°C with potential for still higher temperature), the modular helium reactor system can also efficiently produce hydrogen by high temperature electrolysis or thermochemical water splitting.

Variations of this concept have been investigated and shown to be attractive for cogeneration applications including seawater desalination [XV-2] and low temperature process heat applications.

The thermodynamic characteristics of the heat rejection from the GT–MHR power cycle are such that waste heat can be used for seawater desalination without reducing the efficiency of electric power generation. Also, under these conditions the relative amount of fresh water and electricity production are well balanced to the needs of an urban population.

XV-1.3. Special features

Special features of the GT–MHR concept include the following

- Incremental capacity addition: The GT–MHR can be deployed as incremental 268 MW(e) modules in a multi-module site application, with common auxiliaries and central control of up to 4 modules for a total capacity of 1145 MW(e). This allows sequencing of new capacity additions to more closely match demand while obtaining the economic benefits of large-scale generation as the capacity expands.
- Flexible siting: The waste heat per unit electrical generation is about half that of a conventional water reactor due to the high thermal efficiency of the GT–MHR power conversion system. This allows greater siting flexibility due to reduced cooling water demands. Electricity generation using air blast heat exchangers cooling is also possible.
- Simplified operational licensing requirements: The passive safety characteristics allow the GT–MHR to safely respond to all incidents within the design basis without need for short-term operator actions or AC powered systems. This simplifies and reduces the scope of operator training, and equipment qualification and surveillance required to meet safety requirements.

XV-1.4. Summary of major design and operating characteristics

The GT–MHR (Fig. XV-1) couples a gas-cooled modular helium reactor (MHR), contained in one pressure vessel, with a high efficiency Brayton cycle gas turbine (GT) power conversion system (PCS) contained in an adjacent pressure vessel.

As a direct cycle gas turbine power plant, the GT–MHR has no intermediate or secondary circuits as would be associated with steam cycle power plants. To illustrate the primary components and their function in the power cycle, a simplified process flow diagram is provided in Fig. XV-2.

The GT–MHR direct Brayton cycle (gas turbine) power conversion system contains a gas turbine, an electric generator, and gas compressors located on a common, ~29 m long vertically orientated shaft supported by magnetic bearings. The power conversion system also includes recuperator, pre-cooler and intercooler heat exchangers. Heated helium flows directly from the MHR into a gas turbine to drive the generator and gas compressors, Fig. XV-2. From

the turbine exhaust, the helium flows through the hot side of the recuperator, through the pre-cooler and then passes through low and high pressure compressors with inter-cooling. From the high pressure compressor outlet, the helium flows through the cold, high pressure side of the recuperator where it is heated for return to the reactor.

The use of the direct Brayton cycle to produce electricity results in a net plant efficiency of approximately 48% (Fig. XV-3). This efficiency is ~50% higher than that in current LWR nuclear power plants.

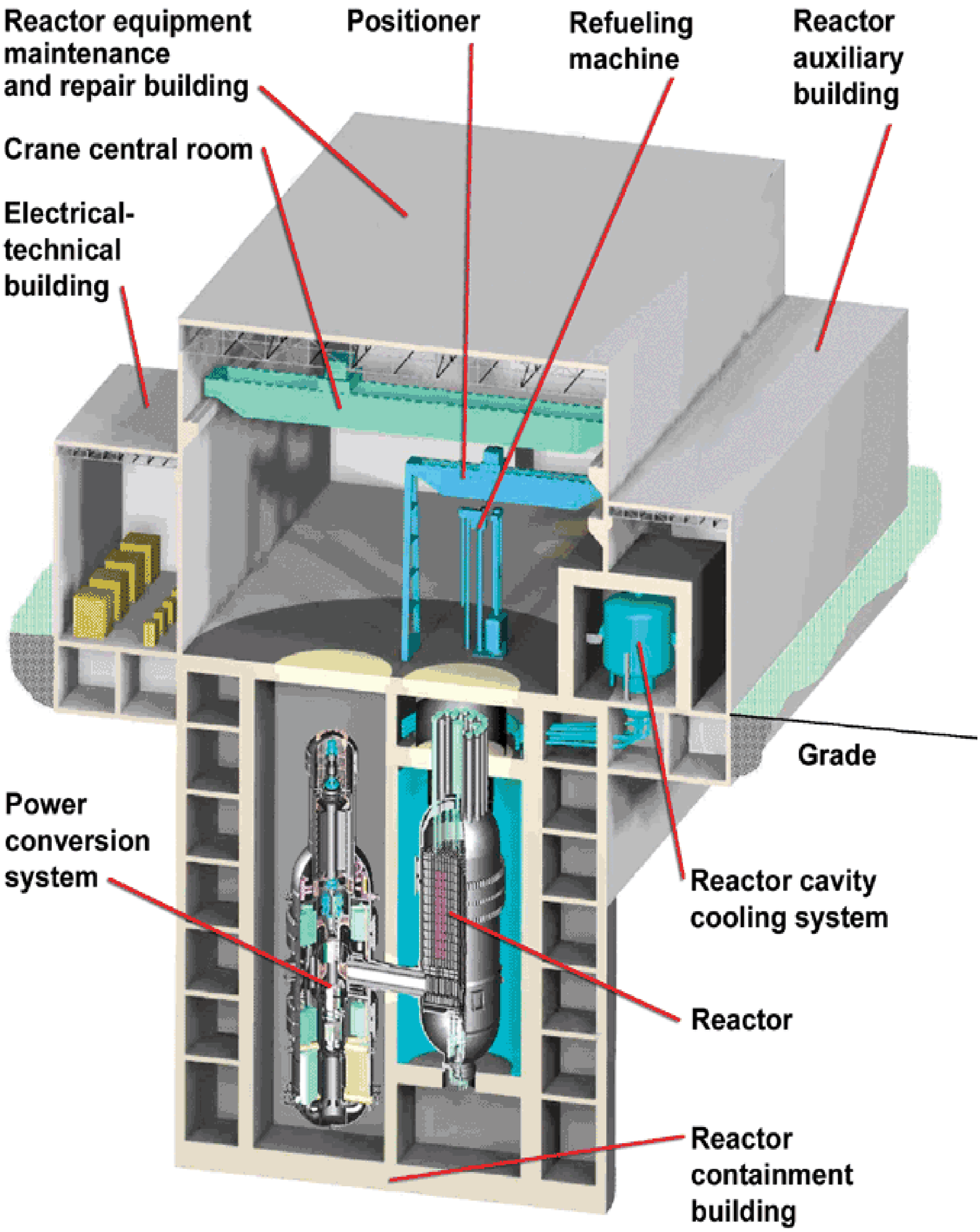


FIG. XV-1. GT–MHR reactor module.

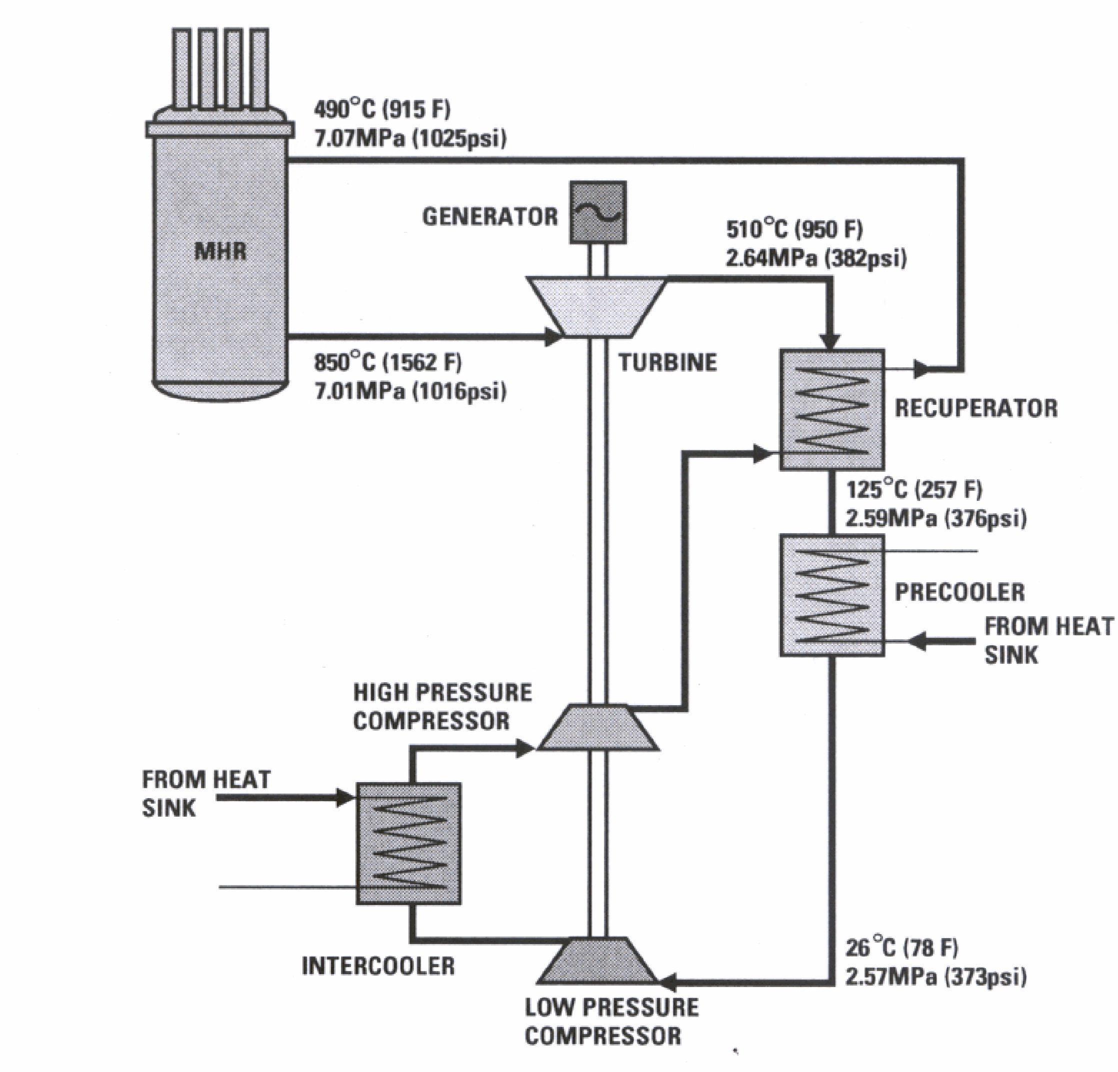


FIG. XV-2. GT–MHR power cycle diagram [XV-1].

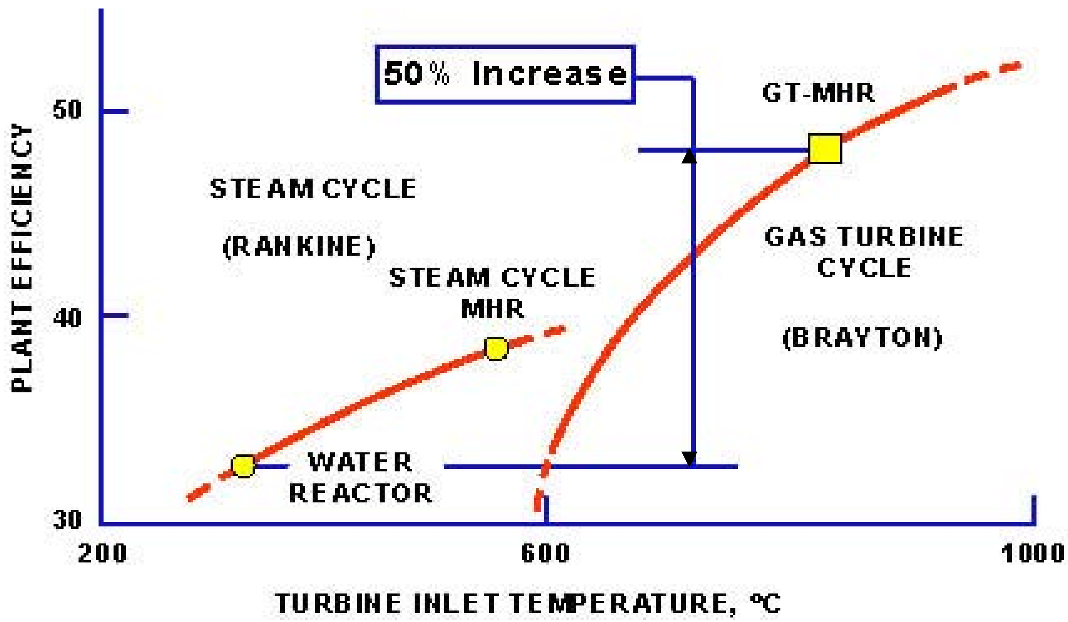


FIG. XV-3. Comparison of thermal efficiencies.

The selection of (1) the direct cycle PCS and (2) integrated vertical shaft PCS arrangement was made on the basis of achieving optimum economics from consideration of several alternatives. There are several alternative high efficiency Brayton cycle PCSs and arrangements that could be used. Some of these would require less development effort but would have higher capital cost and electricity generation cost. One alternative would be to substitute two half-sized, direct-coupled turbines in the first-generation plant, with the intent to advance to the single turbine having less capital cost when the technology is considered sufficiently mature. Another alternative is to use an indirect Brayton cycle PCS. In this cycle, the PCS would be located in a secondary loop, nominally outside of the safety envelope, coupled with the primary loop through an intermediate heat exchanger (IHX). Locating the PCS outside the safety envelope enables use of more conventional (more proven) technology such as horizontally oriented turbines and compressors, oil-lubricated bearings, and multiple shaft machines. The IHX would, however, require development and the cycle thermal efficiency would be lower than for the direct cycle. Nominally, the plant capital cost would be higher (due to more equipment required, the IHX plus a primary system circulator) but there are opportunities for minimizing the PCS capital costs because of the greater potential for more complete modularization and factory assembly of this equipment.

The GT–MHR gas turbine power conversion system (PCS) has been made possible by key technology developments during the last several years in large aircraft and industrial gas turbines; large active magnetic bearings; compact, highly effective gas-to-gas heat exchangers; and high strength, high temperature steel alloy vessels.

The major design and operating characteristics of the GT–MHR module are presented in Table XV-2 and Fig. XV-4.

TABLE XV-2. SUMMARY OF MAJOR DESIGN AND OPERATING CHARACTERISTICS

CHARACTERISTIC	VALUE
Rated thermal power	600 MW(th)
Plant efficiency	48%
Rated electric power	287 MW(e)
Type of fuel	Ceramic coated particles in prismatic blocks, UCO kernels
Fuel enrichment	Fissile particles –19.8% Fertile particles –0.7%
Coolant	Helium
Moderator	Graphite
Reactor internals structural materials	Fuel blocks, replaceable reflectors, core support - H–451 graphite Permanent reflector, core lateral restraint – HLM graphite Metallic internals – alloy 800H Design life, permanent internals – 60 years
Core	Annular ceramic (hexagonal blocks) Height – 7.9 m Fuelled annulus inner diameter – 3 m
	Fuelled annulus outer diameter – 4.8 m
Reactor vessel	Material – steel (9 Cr or 2¼Cr, 1 Mo) Height – 31.2 m Inner diameter – 7.3 m Wall thickness – 0.27 m Outer diameter (including flanges) – 8.6 m Design lifetime – 60 years
Cycle type	Direct closed cycle gas turbine (Brayton cycle)

CHARACTERISTIC	VALUE
Number of coolant circuits	1
NPP style	Modular, loop
Neutronics characteristics	Temperature reactivity coefficient – < -3×10^{-5} δk/k/ºC (Fig. XV-4) Coolant void & density coefficients – negligible Burn-up reactivity swing (with lumped burnable poison) – 3.5% Maximum axial (F_z) peaking factor - ~1.3 Maximum horizontal (F_{xy}) peaking factor - ~1.35 Power flattening by varying fuel loadings (fissile and fertile particles), fixed burnable poison, reload fuel shuffling, reflector control rods The LEU GT–MHR burnable poison consists of pyrolithic carbon (pyrocarbon) coated B_4C granules in graphite rods loaded in place of fuel in the assemblies in the corners of fuel assemblies[1]
Types of reactivity control, reactor protection[1]	Control rods (clad boron carbide compacts) in reflector for normal operational control and hot shutdown. Reflector rod worth (%δk/k BOC/EOC) – 12.0/13.6[2] Control rods (clad boron carbide compacts) in active core in addition to reflector rods to achieve cold shutdown All rods inserted (%δk/k BOC/EOC) – 27.3/32.4[2] Reserve shutdown using pyrocarbon coated boron carbide pellets Reserve shutdown worth (%δk/k BOC/EOC) – 12.9/15.7[2]
Thermal-hydraulic characteristics	Forced circulation – driven by power turbine Core inlet / outlet temperatures, °C 491 / 850 Core inlet / outlet pressure, MPa 7.07 / 7.02 Coolant flow rate (kg/s) 318 Turbine inlet / outlet temperatures, °C 848 / 511 Turbine inlet / outlet pressures, MPa 7.01 / 2.64 Recuperator hot side inlet / outlet temperature, °C 511 / 125 Recuperator cold side inlet / outlet temperature, °C 105 / 491 Temperatures (°C): Fuel (normal operation): Average 850; Maximum 1250[2]; Limit 1250[3] Fuel (accident conditions): Average 1250[3]; Maximum 1550; Limit 1600 Fuel element graphite: Average 830; Maximum 1200[4]; Limit variable[5]
Burn-up cycle (equilibrium)	Heavy metal loading, t / GW(th) 7.5 Discharge burn-up Segment average, MW·d/kg 117 Fissile particle maximum, % FIMA 26 Fertile particle maximum, % FIMA 7 Refuelling interval, effective full power days (EFPD) 460 Fraction of core refuelled: 1/2

CHARACTERISTIC	VALUE
	Segment U loading (tonnes): 2.26
U_3O_8 consumption in equilibrium fuel cycle, t/ GW(e) year	246
Hydrogen production	200 tons H_2/day at 600 MW(th)
Desalination	42,000 m^3/day from multi-effect distillation (MED) at 600 MW(th) with electricity cogeneration
Economics (estimated by designer, including data from [XV-3])	Cost estimates (2002 US$, unless otherwise indicated) Lead plant capital cost: 1 460 US$/kW(e) N[th] plant costs Capital: ~1000 US$/kW(e) (2003 US$) O&M: 3 US$/MW/h Fuel: 7.4 US$/MW/h 20-year levelized busbar generation cost: 0.031 US$/kW/h (2003 US$) Construction period, yr.: ~3 for 1[st] module of modular plant

1 Use of B_4C is based on early conceptual design of LEU core. Erbium and gadolinium may be considered in future design development

2 Control rod and reserve shutdown system reactivity for a plutonium core; they will be similar for LEU core.

3 Fuel maximum operational temperature and temperature limit are time-averaged temperatures specified to assure fuel integrity during accidents.

4 Graphite maximum operational temperature is time-averaged value.

5 Operational limit is a function of local temperature history, temperature gradient and fluence to assure dimensional changes are within assumed design values.

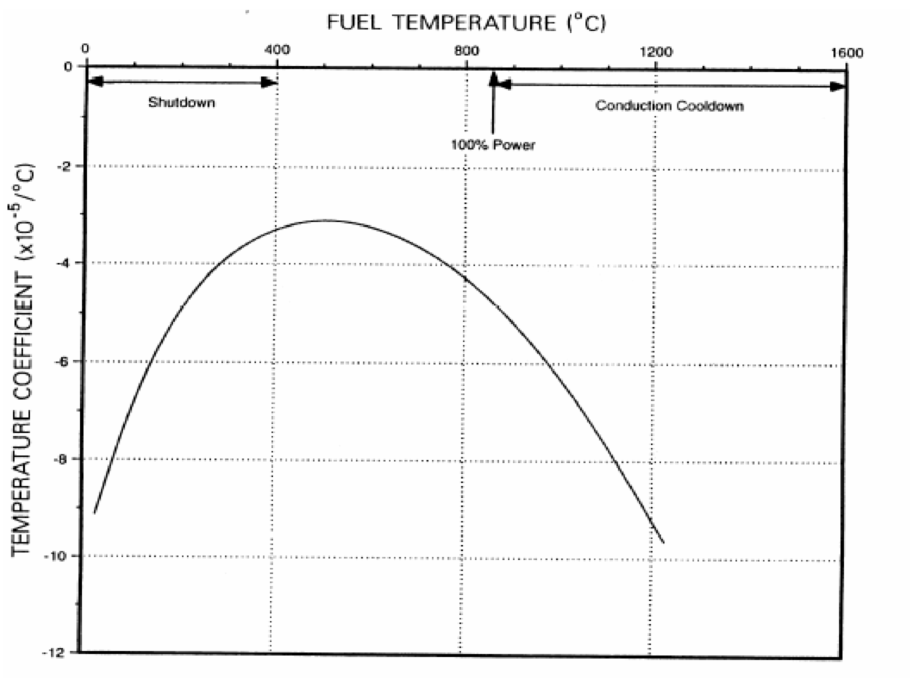


FIG. XV-4. Typical equilibrium cycle temperature coefficient.

XV-1.5. Outline of fuel cycle options

The standard fuel cycle for the commercial GT–MHR utilizes low enriched uranium (LEU) in a once-through mode without reprocessing. Two types of fuel particles are combined in varying ratios according to the fuel zoning scheme: a fissile particle with enrichment slightly below the defined maximum LEU limit of 20%, and a fertile particle with natural uranium. By varying the ratio of fissile and fertile particles, the effective enrichment of fuel elements can be varied, and the fuel element-loading configuration can be better optimized to flatten core power distribution during a fuel cycle.

The GT–MHR reactor design can accommodate alternative fuel cycles if supported by external infrastructure. A fuel cycle utilizing recycled water reactor plutonium can be accommodated, and will be effectively demonstrated by the GT–MHR plutonium consumption project in the Russian Federation. Thorium can be used as an alternative to natural uranium in the fertile particles. If reprocessing is supported in the future, fissile particles can incorporate recycled ^{233}U from thorium fertile particles to reduce the separative work required to produce fissile particles.

If reprocessing of coated particle fuel is desired in the future, reprocessing methods have been identified and studied [XV-4]. Typically, reprocessing would involve removal of the fuel compacts from the graphite block, burning of the graphite/pyrocarbon materials to expose the silicon carbon coatings, crushing of the silicon carbide coating in a mechanical process, removal of remaining graphite/pyrocarbon materials by heating in air, and subsequent processing steps equivalent to water reactor fuel reprocessing.

In the reference once-through cycle, the fuel would be supplied by fuel manufacturing organizations, likely to be organizations currently supplying water reactor fuel. Each fuel supplier would be required to qualify its product by irradiation and post irradiation testing of sufficient quantities of fuel representative of the production fuel, and by certification of compliance with fuel fabrication process and product specifications.

The reference GT–MHR spent fuel characteristics support disposal by direct burial of the fuel element after a period of above ground dry storage [XV-5]. Research and analysis has shown the graphite fuel elements and ceramic particle coatings to be sufficiently stable under repository conditions to retain radionuclides over geologic time scales.

XV-1.6. Technical features and technological approaches that are definitive for GT–MHR performance in particular areas

XV-1.6.1. Economics and maintainability

The GT–MHR economics design objective is a busbar generation cost (20 year levelized) less than the least cost generation alternative. Subcomponents of this objective include:

– An overnight capital installation cost of less than US$1000/kW(e).
– A construction period of ≤3 years for the first module of a reference 4-module GT-MHR plant with successively shorter periods for sequentially constructed follow-on modules.

The GT–MHR is projected to have economic advantages over other plants for the addition of new base load generation capacity. The economic competitiveness of the GT–MHR is a consequence of the use of the direct Brayton cycle power conversion system and the broad implementation of inherent safety features and passive safety systems. The direct Brayton

cycle provides high thermal conversion efficiency and eliminates extensive power conversion equipment required by the Rankine (steam) power conversion cycle. Reduction in the complexity of the power conversion equipment reduces both capital and operation and maintenance (O&M) costs. Strong reliance on the inherent and passive safety design features eliminates the need for extensive safety related equipment that also reduces both capital and O&M costs.

The overnight capital cost for the N^{th}-of-a-kind reference GT–MHR plant containing four standardized reactor modules is projected to be ~1000 $/kW(e) (2003 US$). The N^{th}-of-a-kind plant costs are the costs estimated for the 8^{th} plant built assuming the eight plants are built one after another resulting in the cost economies from bulk material orders for multiple plants and construction cost efficiencies resulting from the sequential deployment of plant construction resources (manpower and equipment). The full capital cost of a standard four-module GT-MHR is not all at risk prior to the generation of revenue from the sale of electricity because the four modules are designed to be deployed sequentially. The highest value of investment-at-risk prior to generation of revenue is the cost of the first module plus the required balance of plant infrastructure, approximately half the value of the full plant.

The construction period required for the first module of the N^{th}-of-a-kind standardized GT-MHR plant is estimated to be ~3 years based on experience information from serial construction of identical design nuclear plants (in France as well as in the USA), and assuming the use of the 10CFR52 one-step licensing process as well as modern (computerized) plant construction practices. The GT–MHR N^{th}-of-a-kind plant 20 year levelized busbar generation cost is projected to be 3.1 cents/kWh (2003 US$) including capital, O&M, fuel, waste disposition and decommissioning.

XV-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

The GT–MHR environmental impact design objectives, relative to the impacts of LWRs, are:

– Reduced thermal discharge.
– Reduced heavy metal wastes.
– Reduced risk of repository spent fuel radionuclide migration to the biosphere.

A comparison of resource requirements and environmental impacts between a 4-module GT-MHR plant and a large pressurized water reactor (PWR) is provided in Table XV-3.

TABLE XV-3. RESOURCE CONSUMPTION AND ENVIRONMENTAL IMPACT COMPARISON

PLANT PARAMETERS	LARGE PWR	GT–MHR
• Thermal power, MW(th)	3931	4×600
• Electric power, MW(e)	1384	1145
• 60 year power generation, GW(e)–yr.	72.2	59.8
THERMAL DISCHARGE		
• Heat rejection, GW(th)/GW(e)	1.8	1.1
• Cooling water required, 10^4 Acre-Ft/GW(e)–yr.	2.4	1.4

PLANT PARAMETERS	LARGE PWR	GT–MHR
EQUILIBRIUM FUEL CYCLE		
• Heavy metal loading, MT/GW(th)	29.9	7.5
• Uranium enrichment, %	4.2	15.5 (average)
• SWU demand, 10^3 kg-SWU/GW(e) y	135	208
• U_3O_8 consumption, t/GW(e) yr	170	182
• Full power days per cycle	477	474
SPENT FUEL DISCHARGE		
• Discharged heavy metal, t/GW(e) y	20.6	5.3
• Discharged Pu, kg/GW(e) yr	268	96
• Discharged ^{239}Pu, kg/GW(e) yr	149	36

The thermal discharge (waste heat) from the GT–MHR is significantly less than in a PWR plant because of its greater thermal efficiency. If this waste heat is discharged using conventional power plant water heat rejection systems, the GT–MHR requires <40% of the water coolant per unit of electricity produced. Alternatively, because of its significantly lesser waste heat, the GT–MHR waste heat can be rejected directly to the atmosphere using air cooled heat rejection systems such that no water coolant resources are needed. Because of this capability, the use of the GT–MHR in arid regions is possible.

The GT–MHR produces less heavy metal radioactive waste per unit energy produced because of the plant's high thermal efficiency, high fuel burn-up and lower fertile fuel inventory. Similarly, The GT–MHR produces less total plutonium and ^{239}Pu (materials of proliferation concern) per unit of energy produced.

The TRISO fuel particle coating system, which provides containment of fission products under reactor operating conditions, also provides an excellent barrier for containment of the radionuclides for storage and geologic disposal of spent fuel. Experimental studies have shown the corrosion rates of the TRISO coatings are very low under both dry and wet conditions. The coatings are ideal for a multiple-barrier, waste management system. The measured corrosion rates indicate the TRISO coating system could maintain its integrity for a million years or more in a geologic repository environment.

The capability to achieve high burn-up (known as deep-burn capability) and high radionuclide containment integrity of TRISO particles offer potential for improvements in nuclear spent fuel management. A high degree of degradation of plutonium and other long-life fissile actinides can be achieved by the deep-burn capability. Nuclear design analyses of the GT-MHR deep-burn concept indicate that, in one pass through the reactor, virtually complete destruction can be accomplished of weapons-usable materials (Plutonium-239), and up to 60% of all transuranic waste, including near total destruction of Neptunium-237 (the most mobile actinide in a repository environment) and its precursor, Americium-241. The resultant particles contain significantly reduced quantities of long-life radionuclides and very degraded fissile materials that can then be placed in a geologic repository with high assurance the residual products have insufficient interest for intentional retrieval and will not migrate into the biosphere by natural processes before decay renders them benign.

The flexibility of possible fuel cycles for the GT–MHR supports future options with increased fuel utilization, including efficient use of thorium, if enabled by an infrastructure supporting fuel reprocessing [XV-5].

XV-1.6.3. Safety and reliability

Safety concept and design philosophy

The GT–MHR safety design objective is to provide the capability to reject core decay heat relying only on passive (natural) means of heat transfer (conduction, convection, and radiation) without the use of any active safety systems.

The GT–MHR safety concept is centred on retention of the radionuclides in the fuel under all normal and postulated accident conditions to a sufficient degree that doses at the site boundary will be within the US Environmental Protection Agency's radionuclide Protective Action Guidelines, without reliance on AC powered systems or operator action. This objective is achieved through incorporation of the following:

- Multiple ceramic coatings on uranium oxycarbide micro spheres, capable of retaining safety significant radionuclides within the envelope of service conditions experienced by the fuel under normal operation and accident conditions for all particles whose properties are within the range allowed by the fuel specification.
- A fuel specification quality requirement for the maximum fraction of particles exceeding specification limits, as well as heavy metal contamination or failed as-manufactured particles, in order to meet dose limits.
- Fuel service conditions during postulated accidents, including simultaneous cooling system depressurization and loss of forced circulation, limited by passive heat loss mechanisms and negative reactivity feedback to within specified allowable conditions.

Provisions for simplicity and robustness of the design

Not requiring AC powered safety systems eliminates the need for complex active systems with sensors, controls, actuators, backup power, etc., that must be qualified to start up and operate reliably over the full range of conditions (e.g. fire, seismic events), which might be encountered. The passive conduction, convection and radiation heat transfer mechanisms used to limit fuel temperatures under accident conditions are simple and robust.

Active and passive systems and inherent safety features

The GT-MHR inherent safety features are as follows:

- Helium coolant, which is single phase, inert, has only minute reactivity effects and does not become radioactive;
- Graphite core, which provides high heat capacity, slow thermal response, and structural stability to very high temperatures;
- Refractory coated particle fuel, which retains fission products at temperatures much higher than normal operation and postulated accident conditions;
- Negative temperature coefficient of reactivity, which inherently shuts down the core above normal operating temperatures; and
- An annular, low power density core (6.5 watts/cc) in an uninsulated steel reactor vessel surrounded by a natural circulation reactor cavity cooling system (RCCS).

The GT–MHR has two non-safety grade, diverse active heat removal systems, the power conversion system and the shutdown cooling system that can be used for the removal of decay heat.

In the event that neither of these active systems is available, an independent passive means is provided for the removal of core decay heat. This is the reactor cavity cooling system (RCCS) that surrounds the reactor vessel (Fig. XV-5).

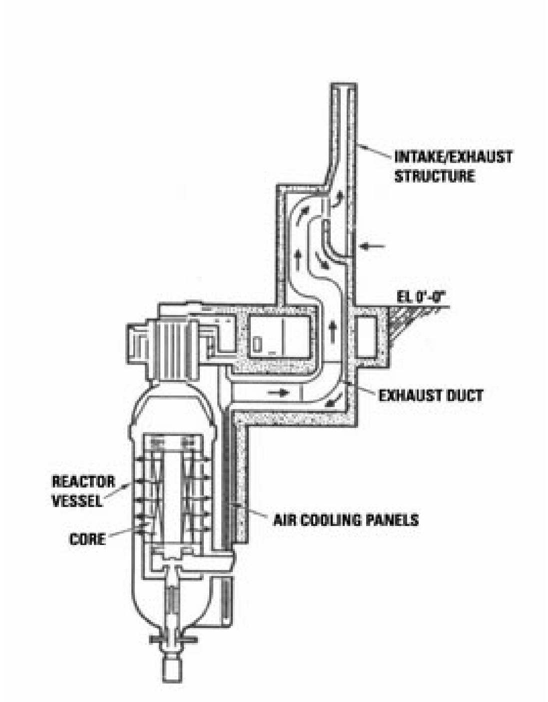


FIG. XV-5. Passive reactor cavity cooling system.

For passive removal of decay heat, the core power density and the annular core configuration have been designed such that the decay heat can be removed by conduction to the pressure vessel (Fig. XV-6) and transferred by radiation from the vessel to the natural circulation RCCS without exceeding the fuel particle temperature limit (Fig. XV-7).

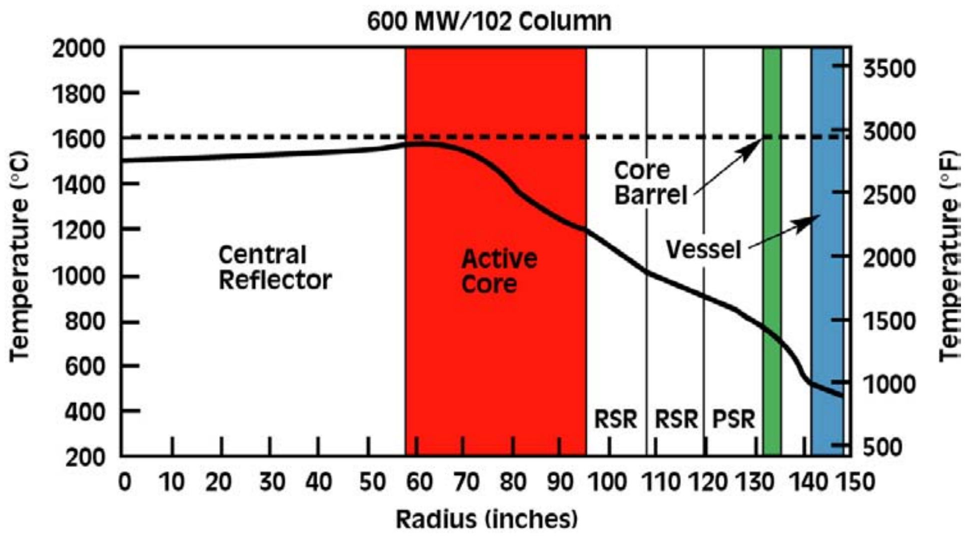


FIG. XV-6. Radial temperature gradient during after-heat rejection to RCCS.

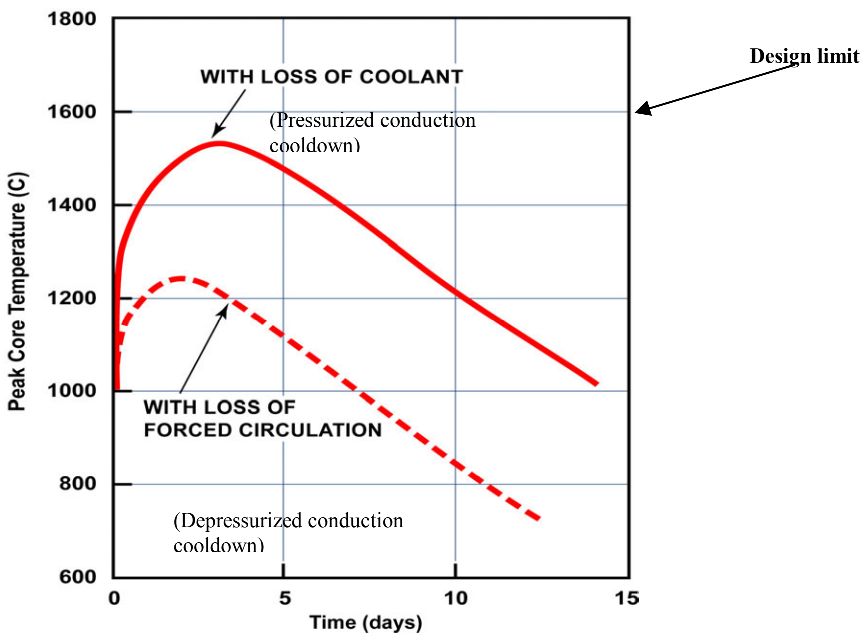


FIG. XV-7. Core heat-up temperatures with passive after heat rejection.

Even if the RCCS is assumed to fail, passive heat conduction from the core, thermal radiation from the vessel, and conduction into the silo walls and surrounding earth (Fig. XV-8) is sufficient to maintain peak core temperatures to below the design limit. As a result, radionuclides are retained within the refractory coated fuel particles without the need for active systems or operator action.

Structure of the defence-in-depth

An important aspect of defence in depth is the construction and operation of the plant in a manner such that challenges to plant safety are minimized and the capability to effectively respond to challenges is assured. The emphasis on inherent features and passive systems discussed above simplify the training, operation, maintenance and surveillance activities necessary to provide assurance of safety by reducing the scope and complexity of safety systems.

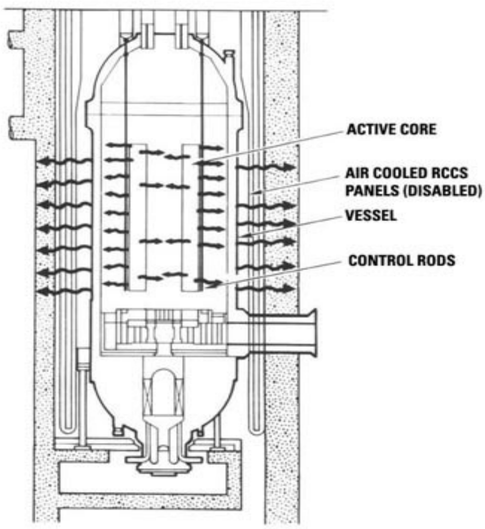


FIG. XV-8. Passive radiation and conduction of after heat to silo containment.

Defence in depth is also assured by redundant and diverse barriers to the release of radionuclides in combination with provisions to assure the integrity of the barriers. Redundant barriers to control release of radionuclides include the following (relative effectiveness of individual barriers varies depending on the chemical and physical properties of radionuclides):

- Fuel kernel – diffusion barrier retaining all or a major fraction of each of the radionuclides deposited in the kernel.
- Inner and outer pyrocarbon coatings – effective diffusion barrier for gaseous fission products and short half-life metallic fission products.
- Silicon carbide coating – effective diffusion barrier for all radionuclides except silver, which is important for personnel exposure but not significant with regard to offsite dose.
- Compact matrix – effective sink for some radionuclides due to sorption.
- Graphite block – partial diffusion barrier with effectiveness varying by radionuclide.
- Reactor coolant pressure boundary – effective sink due to plateout of some radionuclides, effective diffusion barrier for all radionuclides.
- Reactor building – effective barrier to long term releases due to heatup following depressurization accident.

Provisions to assure barrier integrity include:

- Quality control and quality assurance of the as manufactured coated particle by compliance with detailed process and product specifications.
- Qualification of the coated particle fuel for specified service conditions by irradiation and accident testing of representative samples.
- Passive safety design to assure that fuel damage limits are not exceeded for all accident conditions within the design basis.
- Coolant activity monitoring for compliance with plant technical specifications.
- Monitoring of primary coolant system for leakage.

Design basis accidents and beyond design basis accidents

Design basis accidents have been identified based on probabilistic analysis of a full spectrum of events to identify limiting events within specified probability regions, covering the normally identified categories of events – loss of cooling, loss of coolant, reactivity events, and external events. The most challenging event within the design basis is a loss of forced circulation in conjunction with a depressurization of the primary system. In this case, reactor power is terminated due to insertion of control rods or negative reactivity feedback from increasing core temperatures. A typical transient response for cases with loss of helium forced cooling (pressurized), loss of helium pressure and forced cooling (depressurized) and loss of helium pressure, forced cooling and heat removal by the reactor cavity cooling system (conduction to ground) is shown in Fig. XV-9.

In all three cases in Fig. XV-9, long term core heatup occurs following loss of the normal heat sink. The decay heat slowly decreases with time after the event initiation, while heat loss from the reactor vessel increases with increasing temperature. The heatup is reversed after the heat loss from the uninsulated reactor vessel exceeds the decay heat, with fuel temperature reaching a maximum more than a day after initiation of the event. The pressurized case, indicating response to a loss of heat sink or station blackout, results in a maximum fuel temperature in the upper portion of the core that is approximately the same as the maximum

fuel temperature for normal operation, which occurs near the bottom of the core. The depressurized case results in a higher maximum fuel temperature due to reduced convection heat transfer in the reactor vessel at the much lower helium pressure.

In the absence of positive reactivity insertions associated with the chemically and neutronically inert helium coolant, and limits on possible cool down events due to the large heat capacity of the reactor core and internals, control rod withdrawal events result in the maximum reactivity insertion. The design of the control rod drive mechanisms, and their location in the reactor cavity, preclude a control rod ejection event, thus the limiting overpower event is that associated with an inadvertent control rod withdrawal. Power and fuel temperature response to a control rod withdrawal event, with shutdown by the safety rods and with shutdown by the reserve shutdown system assuming failure of the safety rods, are shown in Fig. XV-10.

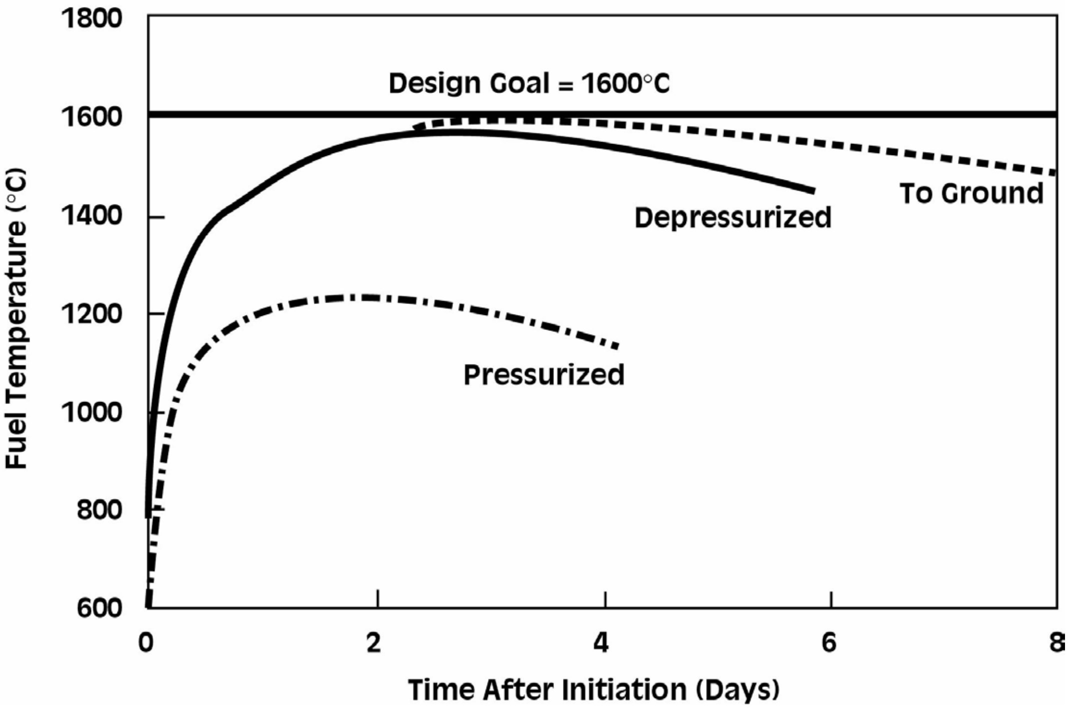


FIG. XV-9. GT–MHR pressurized and depressurized loss of cooling [XV-6].

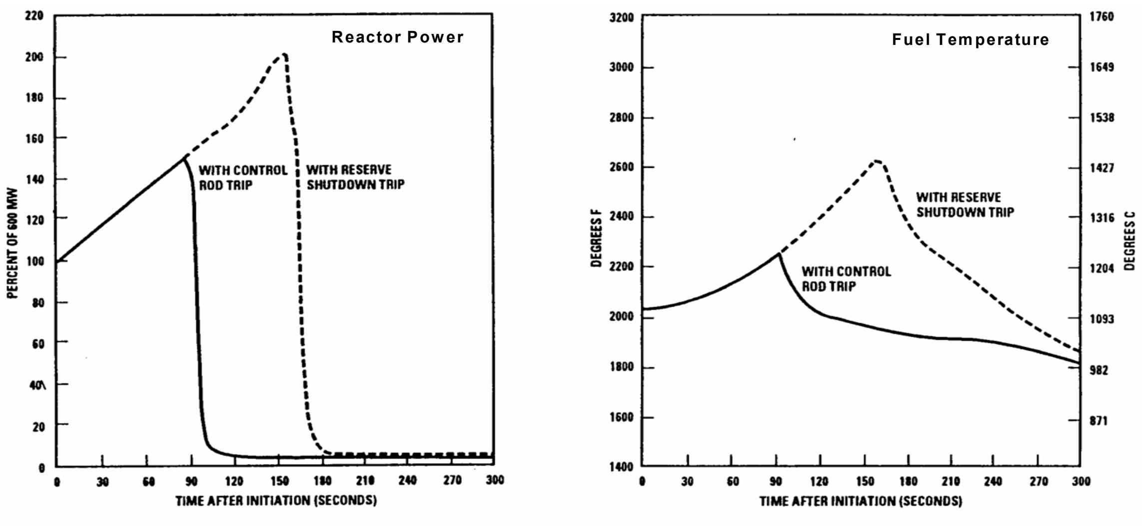


FIG. XV-10. Response to transient overpower event.

The reactor power increases in a slow and smooth fashion mediated in part by the negative temperature coefficient, with the increase terminated by negative reactivity insertion from the safety rods or reserve shutdown system. In both cases, the increases in fuel temperatures are well below fuel damage limits.

Identification of beyond design basis events to be considered in the GT–MHR licensing process is an objective of future licensing interactions. Much of the discussion to date has been directed toward extended air ingress events associated with large primary system failures. Tests conducted in Germany, where irradiated fuel spheres and particles were heated in air to temperatures ranging from 1300 to 1620ºC [XV-7], resulted in complete removal of the outer pyrocarbon coating by oxidation. The results showed effective retention of fission products for several days at the low end of the temperature range, with increasing release at higher temperatures. Analysis of air ingress events have shown the progression to be limited to oxidation of graphite in the regions near the air entrance for several days by air flow resistance and limited oxygen availability.

Seismic events are addressed in accordance with standard practices for nuclear licensing, with seismic accelerations and spectra dependent on location of the site. In the certification of a standard design, and in the design of standard components, seismic conditions enveloping a wide range of sites are used (in the development of standard seismic requirements for the GT-MHR, an objective of encompassing 85% of US sites was used, resulting in safe shutdown earthquake horizontal and vertical accelerations of 0.3 g). The structural elements associated with maintaining a safe geometry and preserving the path for passive heat removal (e.g. reactor internals and core, reactor vessel and supports, reactor cavity cooling system) will be subjected to nuclear safety grade analysis, procurement, installation and surveillance requirements to assure their ability to function in accordance with the assumptions of the safety analysis during and after a seismic event.

The use of high quality coated particle fuel and graphite structural materials with high temperature capability, low core power density, high reactor heat capacity, inert coolant and passive safety design effectively eliminates the possibility of loss of core structural integrity or a major release of radioactivity from the fuel. In the event of a substantial failure of the primary coolant pressure boundary, the reactor building is designed to release the low activity helium coolant and serve as a low pressure filtered confinement to retain the longer term (days) limited radionuclide releases calculated for the core under the heatup conditions resulting from the depressurization. Thus the potential for failure of a containment penetration or other leak pathway due to stresses associated with pressurization, resulting in a rapid release of high levels of radiation, is eliminated.

Because of the accident at Chernobyl in 1986, the role of graphite in reactor safety has received increased attention. However, the consequences of the Chernobyl accident were caused by massive fuel failure and not by graphite oxidation that occurred during the accident. Decay heat from the nuclear fuel was sufficient to maintain relatively high graphite temperatures for an extended period of time, causing the graphite to radiate the "red glow" that was observed during the accident. High-purity, nuclear-grade graphite reacts very slowly with oxygen and would be classified as non-combustible by conventional standards. In fact, graphite powder is a class D fire extinguishing material for combustible metals, including zirconium. For the GT–MHR, the oxidation resistance and heat capacity of graphite serves to mitigate, not exacerbate the radiological consequences of a hypothetical severe accident that allows air into the reactor vessel.

XV-1.6.4. Proliferation resistance

The GT–MHR proliferation resistance design objective is a plant and fuel system that has high resistance to sabotage and to diversion of either weapons usable special nuclear materials or radioactive materials.

The GT–MHR fuel form presents formidable challenges to diversion of materials for weapons production, either as fresh or spent fuel.

In the case of fresh fuel, access to the uranium in the fuel would require handling and disassembly of the large, heavy graphite fuel assemblies to reach the fuel compacts, and processing of the compacts and fuel particles to reach the uranium.

The particles in the compact would have to be deconsolidated, then sorted into fissile and fertile particles (to prevent dilution of the higher enriched fissile particle uranium by the natural uranium fertile particles), burn off of the outer pyrocarbon layer, crushing of the extremely hard, strong silicon carbide coating, and further processing to remove the inner pyrocarbon and buffer layers. After obtaining the uranium oxycarbide, the maximum fissile particle enrichment of 19.8% falls within non-proliferation guidelines and thus would require further chemical processing and enrichment. The technology and process development, facility construction and operation required to conduct the above activities would be time consuming, costly, and readily detectible due to the scale required by the low fuel volume fraction in the hexagonal graphite fuel elements.

Attempts at materials diversion from spent fuel would encounter all the challenges discussed above, plus performing the required activities in a high radiation field. The enrichment step could be eliminated by chemically separating out plutonium. However, the isotopic content of the plutonium in the spent fuel is not attractive for weapons use due to the neutronic characteristics of the GT–MHR LEU cycle. The quantity of fissile material (plutonium and uranium) per GT–MHR spent fuel element is low (50 times more volume of spent GT–MHR fuel elements would have to be diverted than spent light water reactor fuel elements to obtain the same quantity of plutonium-239).

No process has yet been developed to separate the residual fissionable material from GT-MHR spent fuel. While development of such a process is entirely feasible (and potentially desirable sometime in the future) there is no existing, readily available process technology such as for spent light water reactor fuel. Until such time as when the technology becomes readily available, the lack of the technology provides an enhanced proliferation resistance.

Attempts to alter the GT–MHR fuel cycle to produce plutonium with an isotopic content suitable for weapons material would be difficult and readily detectible. This would require frequent refuelling and use of fuel loadings that differ substantially from normal fuel loadings, with corresponding differences in neutronic properties. The safety and operational requirements of the GT–MHR will require strict compliance with requirements for fuel handling and placement in the core to insure core power and temperature distributions are within the limiting conditions assumed in the safety analysis. The fuel accountability and inspection requirements would preclude the major alterations in the fuel cycle required to produce the desired plutonium isotopic content, and would preclude diversion of spent fuel for clandestine reprocessing.

XV-1.6.5. Technical features and technological approaches used to facilitate physical protection of GT–MHR

The slow, stable response of the GT–MHR to internally or externally initiated transients, in combination with passive safety features, provides a strong defence against internal or external threats to the plant. These characteristics arise from the inert coolant, large reactor heat capacity, low power density, strong negative reactivity feedback coefficients and passive decay heat removal by conduction, radiation and convection.

The absence of a dependence on AC powered active safety systems, and the inherent characteristics of the GT–MHR greatly reduce the potential for challenges to the radionuclide retention function from internal sabotage or external attacks, including aircraft crashes. There are no active systems external to the reactor cavity that can be destroyed or deactivated to compromise plant safety. Even events which could challenge the capability of the passive reactor cavity cooling system to function would not result in large radionuclide releases because heat conduction to the ground around the reactor cavity and continuing falloff in decay heat power would limit fuel temperatures to acceptable levels.

The reactor cavity, which is below grade level and constructed of thick reinforced concrete walls, can withstand aircraft impact while maintaining the integrity of components within the cavity. Short-term (hours) disruptions in cooling, which could result from combustion of fuel following aircraft impact, would not substantially increase the maximum fuel temperature due to the high heat capacity of the reactor system.

XV-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of GT–MHR

Development of the GT–MHR has already benefited substantially from international cooperation on research and development. The following coordinated research projects (CRPs) conducted by the IAEA in the 1990s have resulted in exchanges of technical data and analyses that have advanced the GT–MHR design and reduced development risks:

- Validation of safety related physics calculations for low-enriched high temperature gas cooled reactors.
- Validation of predictive methods for fuel and fission product behaviour in gas cooled reactors.
- Heat transport and afterheat removal for gas cooled reactors under accident conditions.

The following ongoing CRPs are continuing to advance the state of the technology supporting the GT–MHR:

- Evaluation of high temperature gas cooled reactor performance.
- Advances in HTGR fuel technology.

The ongoing joint United States/Russian Federation project to develop and construct a version of the GT–MHR to consume surplus weapons plutonium is an important element of commercial GT–MHR development. The major systems, structures and components of the GT–MHR, including the power conversion system, reactor vessel and internals, and reactor building, can be developed and demonstrated through this project. The primary alterations to the plutonium consumption design are expected to be in the reactor core and possibly the reactor cavity cooling system, with the remainder of the commercial GT–MHR drawing directly from the plutonium consumption version.

The Generation IV International Forum (GIF), initiated by the United States Department of Energy, is another vehicle for fostering multinational collaboration on research and development in support of advanced nuclear power concepts, including the GT–MHR [XV-8]. The GT–MHR was recognized to satisfy the Generation IV goals of passive safety, competitive economics, enhanced proliferation resistance, and improved environmental characteristics including reduced waste and better fuel utilization than the current generation of nuclear power plants. The next generation nuclear plant (NGNP) project, under development in the US with input from GIF, is directed toward construction of a demonstration nuclear plant for electricity generation and hydrogen production. A variant of the commercial GT–MHR is a leading NGNP candidate. In combination with the activities discussed above, the NGNP project could address development needs of the GT–MHR.

The modular design of the GT–MHR, enabling deployment in incremental capacity additions of 287 MW(e), reduces the capital required and improves matching of generation and load through phased deployment of multi-module plants. This is particularly valuable for developing countries, where electricity grid capacities are smaller and less able to absorb larger incremental additions. Standardization of the GT–MHR plant, along with simplification of operational safety and licensing requirements, will facilitate development of regulatory organizations and training and qualification of operating staffs through standard certification programmes. These simplifications, resulting from the inherent and passive characteristics of the GT–MHR, along with the high degree of proliferation resistance, will facilitate development of infrastructure and establishment of plant construction and operation organizations in developing countries.

Options for GT–MHR deployment could include purchase of a turnkey plant with standardized acceptance criteria, and maintenance and operational requirements; leasing of the plant with active ongoing operation and maintenance support from the supplier; or a power purchase contract with plant operation by a multinational generating company. Fuel supply and spent fuel disposition arrangements with a fuel manufacturer can be established in accordance with the needs of individual plants.

XV-1.8. List of enabling technologies relevant to GT–MHR and status of their development

The technology development needs of the GT–MHR were addressed in the US DOE Generation IV roadmap development activity [XV-8]. The major elements are summarized below:

Fuel development and qualification

The ability of the TRISO coated particle fuel to retain radionuclides to a very high degree and high level of confidence under normal operating conditions and postulated accidents is an essential element of the GT–MHR concept. This ability has been effectively demonstrated in a general sense for service conditions (burn-up, fast fluence, operating temperatures and temperature gradients, accident condition temperatures) representative of a pebble bed modular HTGR design (the HTR-MODUL) by an extensive fuel fabrication and testing programme conducted in Germany. However, the service condition envelope for the GT-MHR substantially exceeds the HTR-MODUL envelope for burnup (26% FIMA vs. 11% FIMA), and also has a higher operating temperature and temperature gradient. As a result, the GT-MHR designers have selected a particle design with a smaller kernel diameter (350 vs. 500 μm) and different kernel composition (UCO vs. UO_2). Thus a fuel development and demonstration programme is required to establish a fuel fabrication process and a qualified fuel product for GT–MHR service conditions. Because of the extended time

requirements associated with test fuel fabrication, irradiation, and post irradiation testing and examination, this is a critical path element of the GT–MHR schedule.

Fuel performance and fission product transport analysis methods and codes

The calculation of radionuclide release from the core and retention in the transport pathways to the site boundary is necessary to demonstrate compliance with offsite dose limits. The objective of meeting the low dose limits associated with protective action guidelines (relative to the traditional limits for accidents, e.g., 10CFR100) requires verified and validated methods for calculating release and transport of all radionuclides of potential safety significance. Considerable effort has been applied to this need in conjunction with past HTGR technology development, but development and qualification of codes and methods specific to the needs of the GT–MHR is required.

Reactor physics and thermal hydraulics analysis methods and codes

Design and safety analysis of the GT–MHR requires verified and validated methods for calculating the coolant flow, power and material temperature distributions of the core, reactor internals and reactor vessel during both normal operation and accident conditions. The methods must address phenomena that affect these distributions, including bypass flows through control rod channels, gaps between fuel and reflector blocks, variations of coolant density with temperature, cross flow at interfaces between fuel and reflector blocks, and power distributions including the effects of fuel zoning, shuffling and depletion during a cycle. Methods that have been developed for earlier HTGR designs will need to be updated and qualified for application to the GT–MHR.

Metallic materials

Materials must be developed and/or qualified for the GT–MHR service conditions for use in the reactor vessel, reactor internals, hot duct, turbine and recuperator. For the reactor vessel, internals and hot duct, qualification will require data on performance of the materials under irradiation conditions representative of GT–MHR service over the component design life. Data needed include effects of irradiation on tensile strength, low-cycle fatigue, fracture toughness, creep and relaxation, creep-fatigue strength and high-cycle fatigue strength.

Graphite materials and ceramics

The majority of the reactor core and internals structural components are made from graphite. A large body of data and analysis is available regarding the general behaviour of graphite under irradiation and temperature conditions similar to those of the GT–MHR [XV-9]. However, data are required for performance of the graphites specified for the GT–MHR to address multi-axial strength, fatigue strength, mechanical properties, irradiation-induced dimensional change, irradiation induced creep, thermal properties, fracture mechanics, corrosion and oxidation, and coke source qualification. Data are also needed for other ceramic components, including carbon/carbon composite materials for control rods (if used) and hard ceramic insulation used under the graphite core support structure.

Reactor system component design and validation

Major reactor system components require detailed design and validation through testing of scale models and assemblies, and in some cases demonstration testing of prototypical components. These components include the reactor internals and hot duct, neutron control components, safety instrumentation, the shutdown cooling system circulator and heat exchanger, the reactor cavity cooling system, fuel handling equipment and reactor service equipment.

Power conversion system (PCS) component design and validation

Major power conversion system components also require detailed design and validation. These components include the turbomachine (helium turbocompressor and generator), recuperator, and precooler/intercooler. In addition, an integrated test of the PCS is needed to confirm the performance of prototype components under normal operation and plant transient conditions.

A goal for the NGNP is a core outlet temperature of 1000°C to provide higher efficiencies for either electricity generation or hydrogen production. The primary challenges the 1000°C NGNP outlet temperature poses are related to the reactor fuel and metallic materials. Core designs having lower peaking factors may be required so peak fuel temperature limits are not exceeded, or a revised fuel particle coating system (e.g. ZrC instead of SiC coating) may be required having higher temperature limits.

Advanced materials for NGNP project

The key NGNP metallic material challenges are the reactor vessel and the thermal barrier structural materials. An alternate, higher temperature reactor vessel material may be required because the core inlet temperature, that governs the reactor vessel temperature, will most likely be higher for the higher core outlet temperature. Higher temperature vessel materials are available but have not been fabricated in the vessel sizes required by the NGNP. The largest comparable vessel for which there is significant fabrication experience is the ABWR vessel fabricated from a lower service temperature material. The indicated size requirement for the NGNP is a reactor vessel ~1.5 times the weight of the ABWR vessel and fabricated from a material with a higher service temperature. In the GT–MHR, thermal barriers containing insulation are used to protect metallic structural components from the hot outlet temperature coolant gas but the thermal barriers make use of metallic materials for holding the thermal insulation. For the NGNP 1000°C outlet temperature, alternative materials (e.g., carbon-carbon composites) may be required in place of these thermal barrier metallic materials.

XV-1.9. Status of R&D and planned schedule

The development and deployment of commercial GT–MHRs is based upon leveraging an ongoing international project to develop and deploy a multi-module GT–MHR designed to consume excess weapons grade plutonium in the Russian Federation. The commercial GT-MHR design would utilize the technology development conducted in support of the plutonium consumption version, with the majority of additional development focused on fabrication and qualification of LEU fuel. The institutions involved in the R&D, design and deployment of the plutonium consumption GT–MHR include the following:

- The Russian Federation – Rosatom, OKBM, RRC KI, VNIINM Bochvar, SPA "Lutch", SCC, VNIPIET, NIIAR, SNTC, ISTC.
- The United States – DOE/NNSA, EPRI, General Atomics, ORNL.
- The European Union and Japanese participation via ISTC.

Many of the above institutions would also be involved in transfer of technology from the plutonium consumption version to the commercial GT–MHR. In addition, the GT–MHR Utility Advisory Board (UAB) has been actively supporting the commercialization of the GT-MHR. Membership of the UAB has included:

- The Russian Federation – Rosenergoatom.
- The United States – Entergy, NMC, Dominion, OPPD, PSEG, Southern California Edison, and Constellation Energy.

Efforts in support of the commercial version of the GT–MHR by the above institutions have resulted in the production of licensing and deployment plans that are the basis for the schedule discussions below. Planning for a possible demonstration plant in the US (the NGNP project), begun in 2003, may result in a restructuring of the GT–MHR commercialization strategy.

A schedule produced for the plutonium consumption GT–MHR indicates that the prototype could begin full power operation nine years after completion of the preliminary design. This would include the following elements:

- Complete design and development in the Russian Federation – 3 years.
- Russian regulatory review (in parallel with above) – 4 years.
- Prototype construction in the Russian Federation – 4 years.
- Fuel load, ascent to power and demonstration testing – 1 year.

The commercial GT–MHR schedule for a US deployment would parallel the plutonium consumption version schedule summarized above, lagging the deployment in the Russian Federation by about one year with the following elements:

- Convert Russian design to US standards (in parallel with Russian design) – 4 years.
- Site plan and plant order – 1 year.
- Site preparation and long lead material orders – 1 year.
- Module 1 construction – 3.5 years.
- Fuel load, ascent to power and demonstration testing – 1 year.

With the exception of fabrication and qualification of low enriched fuel for the commercial GT–MHR, the technology development and demonstration would be conducted in conjunction with the development and deployment of the plutonium consumption version. Construction of a fuel fabrication pilot plant, and fabrication and irradiation of proof test fuel is estimated to take eight years and could be conducted in parallel with the plant schedule.

Pre-application licensing interactions with the US Nuclear Regulatory Commission began in 2001, including submittal of a Licensing Plan [XV-10]. The plan identified the following licensing stages and durations:

- Pre-application phase, resulting in NRC licensability statement – 24 months.
- Early site permit – NRC review, issue site permit, lagging but in parallel with pre-application phase – 24 months.
- First module application, resulting in combined license for 1st module – 36 months.
- Design certification application beginning in parallel with 1st module application, resulting in certification of standard design – 72 months.

An extended pre-application licensing review of the MHTGR was conducted from the mid 1980s through early 1990s, culminating in a pre-application safety evaluation report issued by the US NRC [XV-11]. Many of the licensing issues addressed in the MHTGR review are applicable to the GT–MHR. The GT–MHR pre-application licensing interactions with the U.S. Nuclear Regulatory Commission resulted in NRC requests for additional information to support their technical review. Further interactions on the GT–MHR to address the NRC requests for additional information have been deferred, pending the initiation of financing of activities on the NGNP. The GT–MHR is one of several plant options included in early site permit applications by several US generating companies, with the support of the US DOE.

Funding support for the development of the plutonium consumption version of the GT–MHR is continuing through the DOE NNSA in the United States and Rosatom in the Russian Federation, with additional technology development support from the EU and Japan through ISTC. Funding support for pursuing early site permits that include the GT–MHR as an option is provided by the DOE under the NP 2010 initiative and by participating generating companies. As noted earlier, variation of the GT–MHR is expected to be a candidate option for the next generation nuclear plant project under development for design and construction of a high temperature gas cooled reactor in Idaho.

From a technology development standpoint, the path forward for deployment of the GT–MHR technology is necessarily a demonstration project, such as the NGNP project because of a number of heretofore-unproven characteristics embodied in the design. The most prominent of these include items such as the safety design approach, fuel operating conditions (burn-up, fluence, temperature), power conversion system design (vertical shaft, magnetic bearing suspension), and pressure vessel design (size, operating temperatures). The combination of these unproven characteristics result in there being relatively large uncertainties in the risks during all stages of first deployment (licensing, construction cost and schedule, start-up and operation). The risk uncertainty makes attempts to obtain project financing by private industry extremely difficult.

The potential benefits of the GT–MHR for the generation of electricity coupled with the potential for efficient production of hydrogen provide significant incentives for a government-sponsored demonstration programme such as the proposed INL NGNP demonstration project. The INL NGNP demonstration project is a first step toward the development of the next generation of nuclear power resulting from the extensive, multi-national evaluation of advanced nuclear power generation options conducted in the Generation IV programme. The demonstration project objectives are currently being planned to:

- Demonstrate a full-scale prototype gas cooled reactor to produce very high temperature process heat for efficient electricity generation and hydrogen production;
- Demonstrate high temperature, high efficiency Brayton cycle power production at, or near, full scale;
- Demonstrate high efficiency hydrogen production using high temperature process heat produced by nuclear energy;
- Demonstrate by test the safety capabilities of advanced gas cooled reactors;
- Obtain an NRC License, under 10CFR Part 50, for construction and operation to provide a basis for future deployment and licensing of commercial plants.

Demonstrating that the fuel satisfies performance requirements will necessarily have to be an integral part of the demonstration project. Test fuel will have to be fabricated, irradiated and accident tested to provide the performance data needed for licensing the plant by the NRC.

XV-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

A prismatic block core, with fuel and reflector blocks essentially identical to the GT–MHR design, was built and operated to power the Fort St. Vrain generating station. Open cycle gas turbines are operating at temperatures enveloping the GT–MHR conditions. Component testing will provide additional experience and understanding in areas not covered by Fort St. Vrain and open cycle gas turbines. However, Fort St. Vrain differed from the GT–MHR in important aspects (e.g. use of a pre-stressed concrete reactor vessel vs. a steel reactor vessel,

active vs. passive safety systems, steam cycle power conversion system vs. a gas turbine). Likewise, existing large gas turbines for power generation differ from the GT–MHR PCS in important aspects (e.g. atmospheric air vs. high pressure helium, open cycle vs. recuperated closed cycle, magnetic bearings on vertical shaft vs. gas bearings on a horizontal shaft). Achieving competitive economics for the GT–MHR is based on replication of a certified standard design. Arriving at a standard design with sufficient confidence to commit to widespread deployment requires the construction and operation of a demonstration unit.

XV-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

The GT–MHR shares certain technologies and design approaches with other prismatic block or pebble bed fuel high temperature gas cooled reactor designs described in this report, e.g. GTHTR300 (Japan), PBMR–400 (South Africa), HTR–PM (China), etc.

XV-2. Design description and data for GT–MHR

XV-2.1. Description of the nuclear systems

Reactor core and fuel design

The nuclear systems of the GT–MHR are described in Section 4.3.3 of an IAEA-TECDOC on gas cooled reactor technology [XV-1], and briefly summarized below. While the reactor core information provided in the TECDOC includes both the plutonium and low enriched uranium (LEU) designs, this discussion will be directed to the LEU design, as it is the basis for GT-MHR commercialisation efforts.

The reactor core is comprised of hexagonal graphite fuel elements containing coated fuel particles in compacts loaded into holes drilled in the graphite blocks, with additional holes in the blocks for coolant flow. The fuel elements and coated particle fuel are illustrated in Fig. XV-11.

The GT–MHR refractory coated particle fuel (Fig. XV-11), identified as TRISO coated particle fuel, consists of a spherical kernel of either fissile (LEU) fuel or fertile material (natural U), as appropriate for the application, encapsulated in multiple layers of refractory coatings. The multiple coating layers form a miniature, highly corrosion resistant pressure vessel and an essentially impermeable barrier to the release of gaseous and metallic fission products. For the GT–MHR design, the TRISO coated fuel particles are mixed with a carbonaceous matrix and bonded into cylindrical fuel compacts nominally of 12.5 mm outer diameter, 50 mm long and loaded into hexagonal fuel blocks 360 mm across flats × 800 mm long (Fig. XV-11).

The TRISO coatings provide a high temperature, high integrity structure for retention of fission products to very high burn-ups. The coatings do not start to thermally degrade until temperatures approaching 2000°C are reached (Fig. XV-12). Normal operating temperatures do not exceed about 1250°C and worst-case accident temperatures are maintained below 1600°C for a core with an outlet coolant design temperature of 850ºC. Extensive tests in the United States, Europe, and Japan have demonstrated the performance potential of this fuel, but tests still need to be done to demonstrate it satisfies GT–MHR performance requirements for normal operating and accident conditions.

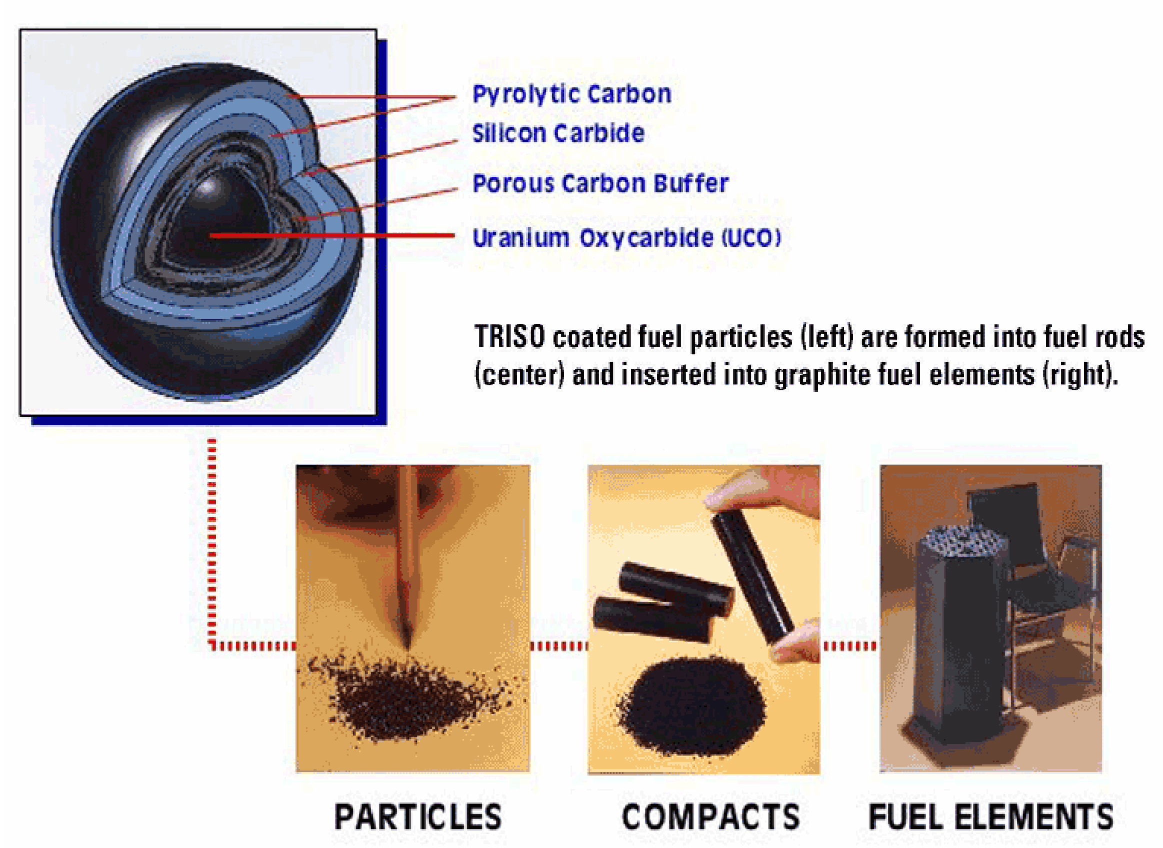


FIG. XV-11. GT–MHR fuel element.

The GT–MHR core design consists of an array of 1020 hexagonal fuel elements surrounded by identically sized solid graphite reflector elements vertically supported at the bottom by a core support grid plate structure and laterally supported by a core barrel. The fuel elements are stacked 10 high in an annular arrangement of 102 columns to form the active core (Fig. XV-13). The core is enclosed in a steel reactor pressure vessel. Control rod mechanisms are located in the reactor vessel top head, and a shutdown cooling system provided for maintenance purposes only is contained in the bottom head. Figure XV-14 provides a plan view of the fuel and reflector elements within the reactor internals.

In the reference GT–MHR design, mixed mean helium outlet temperature is 850°C. The hot outlet helium flows from the reactor core to the power conversion system (PCS) through a hot duct located in the center of the cross-vessel; helium is cooled to 490°C in the PCS and returns to the reactor through the annulus formed between the cross-vessel outer shell and the central hot duct. The cooled helium flows up to an inlet plenum at the top of the core through the annulus between the reactor vessel and the core barrel. From the top inlet plenum, the helium is heated by flowing downward through coolant channels in the fuel elements, collected in a bottom outlet plenum and guided into the cross-vessel hot duct. All the core components exposed to the heated helium are either graphite or thermally insulated from exposure to the high temperature helium. Graphite has high strength, does not readily combust and has dimensional stability to very high temperatures (~2300°C).

Summary level nuclear system design data are provided in Table XV-4. Figure XV-15 shows a cutaway of the reactor vessel and internals. Detailed design data and discussion of the reactor system and fuel is provided in section 4.3.3.2 of [XV-1].

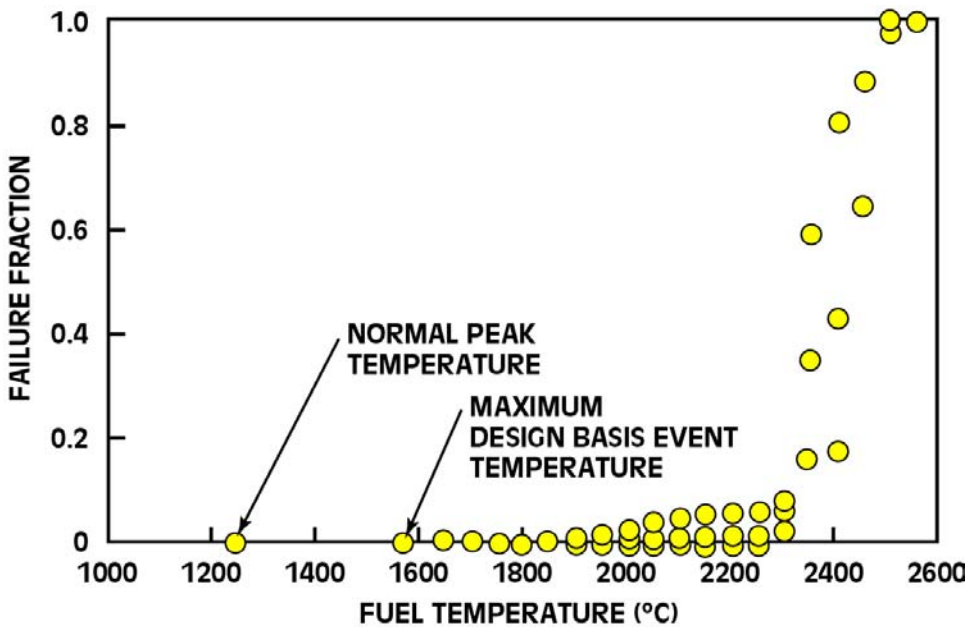


FIG. XV-12. TRISO coated particle fuel temperature capability.

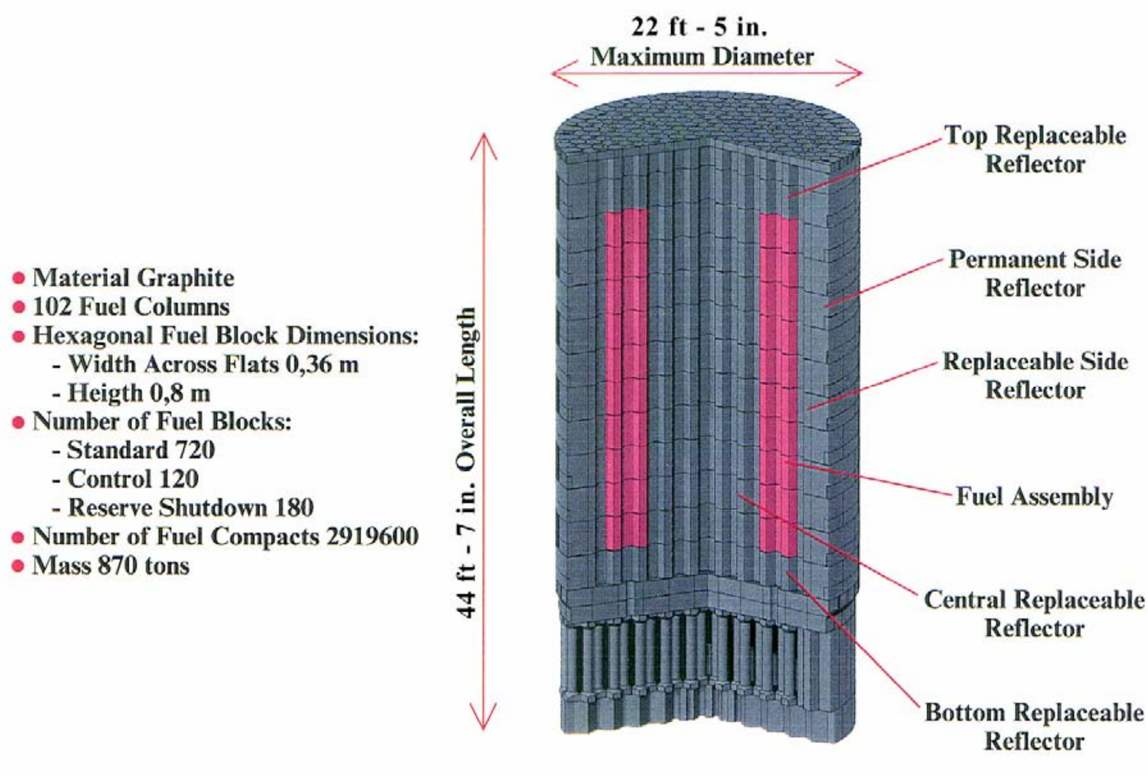


FIG. XV-13. GT–MHR graphite reactor internals.

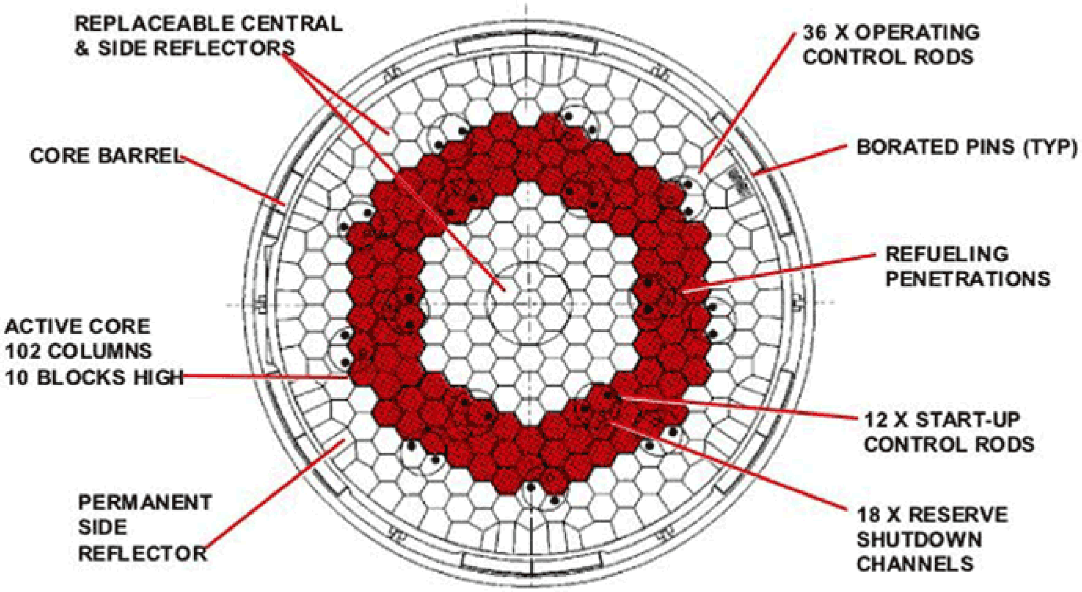


FIG. XV-14. Core cross section.

TABLE XV-4. SUMMARY OF NUCLEAR SYSTEM DESIGN DATA

DESCRIPTION	VALUE	
Fuel particle dimensions, μm	Fissile	Fertile
Kernel diameter	350	500
Buffer thickness	100	65
Inner pyrocarbon thickness	35	35
Silicon carbide thickness	35	35
Outer pyrocarbon thickness	40	40
Fuel assembly dimensions, mm		
Height	800	
Width across flats	360	
Number of fuel assemblies	1020	
Number of control rods	48	
Number of reserve shutdown channels	18	
Core average power density, w/cc	6.6	

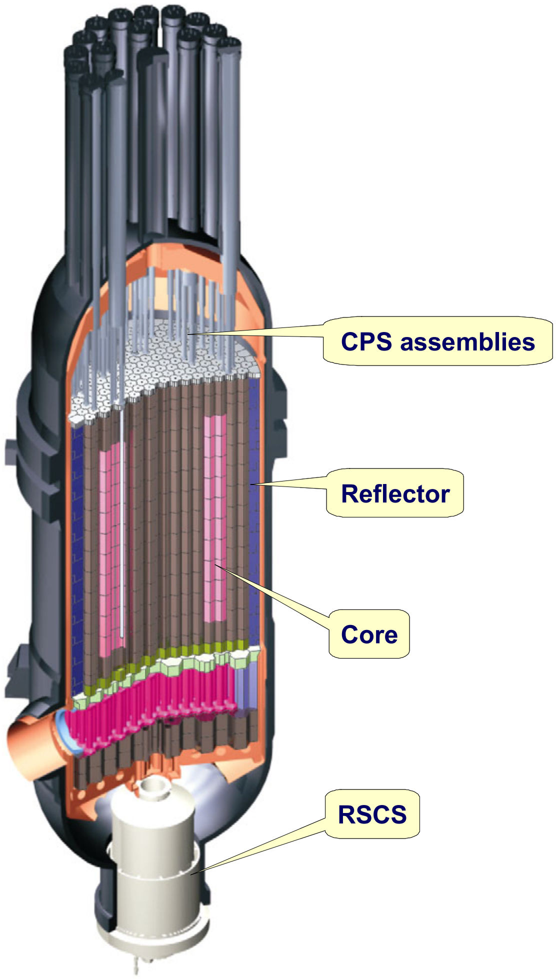


FIG. XV-15. Reactor system & vessel.

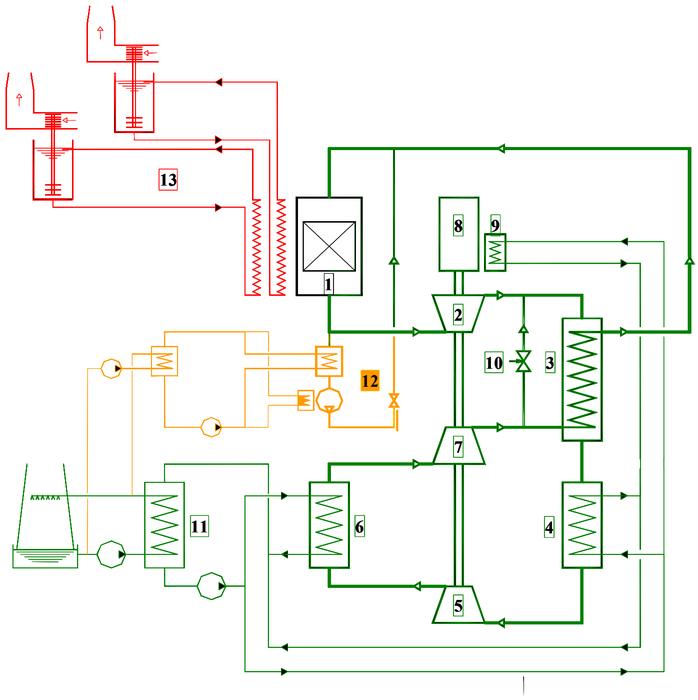
Main heat transport system

The systems for normal heat removal at power, normal shutdown heat removal, and passive decay heat removal are shown in Fig. XV-16. The reactor cavity cooling system is a passive natural circulation system that is normally operating and can be monitored to assure continuous operability.

The electricity generation concept [XV-12] is illustrated in Figures XV-2 and XV-16. Coolant flows into the reactor vessel through the annulus of the cross duct, up along the inside of the vessel, down through the core and out the centre of the cross duct. Figure XV-17 shows a cutaway of the power conversion vessel and internal components. More design data is provided in the section XV-2.2.

XV-2.2. Description of the turbine generator plant and systems

Updated values of the primary parameters of the power conversion system (PCS) that have been developed in the Russian Federation are given in Table XV-5; the general arrangement is shown in Fig. XV-17.



COMPONENTS AND HEAT REMOVAL PATH

1 - reactor

Normal heat removal

2 - turbine

3 - recuperator

4,6 - Precooler and intercooler

5,7 - High and low pressure compressors

8 - generator

9 - Gas cooler

10 - Bypass valves

11 - Normal heat removal system

Normal shutdown heat removal

12 - Shutdown cooling system

Passive decay heat removal

13 - Reactor cavity cooling system

FIG. XV-16. Schematic diagram of main heat removal paths.

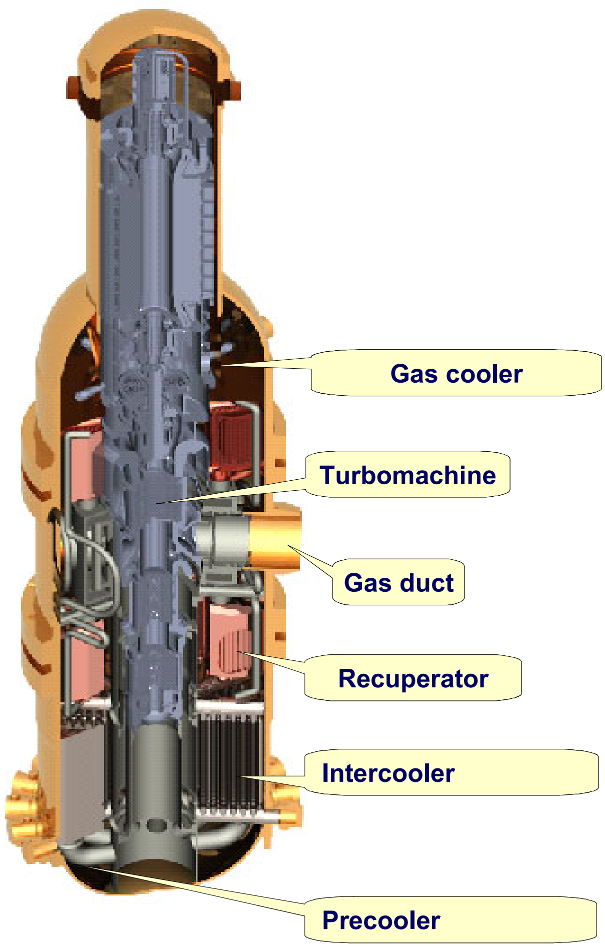


FIG. XV-17. Power conversion system & vessel.

TABLE XV-5. SUMMARY TURBOMACHINE DATA

DESCRIPTION	VALUE
Rotational speed, rpm	4400
Number of stages	
Power turbine	9
Low pressure compressor	13
High pressure compressor	10
Rotor length, m	13.5
Rotor mass, tonnes	30

XV-2.3. Systems for non-electric applications

The GT–MHR concept can be applied to both high and low temperature process heat applications, greatly extending its potential. Low temperature applications such as district heating and desalination are addressed using waste heat in a cogeneration mode.

The GT–MHR is well suited for the desalination of seawater in a cogeneration mode. By transferring the higher temperature energy from the precooler and intercooler (Fig. XV-2) to the desalination system (e.g. multi-effect distillation), and the lower temperature energy to the site heat sink (e.g. cooling tower), the desalination system will have no impact on plant thermal efficiency.

High temperature applications include coal gasification, hydrogen production from methane or water, and high temperature process steam. High temperature applications of gas cooled reactors have been addressed in two IAEA-TECDOCs [XV-13, XV-14], summarized below.

Hydrogen can be produced from nuclear energy by several means. Electricity from nuclear power can separate water into hydrogen and oxygen by electrolysis. The net efficiency is the product of the efficiency of the reactor in producing electricity, times the efficiency of the electrolysis cell, which, at the high pressure needed for distribution and utilization, is about 75–80%. If a GT–MHR with 48% electrical efficiency is used to produce the electricity, the net efficiency of hydrogen production could be about 36–38%. Electrolysis at high temperature, providing some of the energy directly as heat, promises efficiencies of about 50% at 900°C. Thermochemical water-splitting processes similarly offer the promise of heat-to-hydrogen efficiencies of ~50% at high temperatures. Thermochemical water splitting is the conversion of water into hydrogen and oxygen by a series of thermally driven chemical reactions that could use nuclear energy as the heat source.

The sulphur-iodine (S-I) thermo chemical water-splitting cycle has been determined to be one of the most promising methods for coupling to a nuclear reactor [XV-2]. The S-I cycle (Fig. XV-18) consists of three chemical reactions, which sum to the dissociation of water. Only water and high temperature process heat are input to the cycle and only hydrogen, oxygen and low temperature heat are output. All the chemical reagents are regenerated and recycled. There are no effluents. An intermediate helium heat transfer loop would be used between the GT–MHR coolant loop and the hydrogen production system. At the standard GT–MHR outlet temperature of 850°C, a maximum temperature of 825°C is estimated for the process heat to the process, which yields ~43% efficiency. At a reactor outlet temperature of 950°C and a 50ºC temperature drop across an intermediate heat exchanger, an efficiency of ~52% is estimated.

The economics of an N[th]-of-a-kind hydrogen production plant using the S-I thermochemical cycle coupled to the GT–MHR have been estimated. Cost data for the S-I process were used along with the same GT–MHR cost data as used in evaluation of the electric generation economics. The resultant estimated hydrogen production cost is about \$1.60/kg. The cost of producing hydrogen from natural gas by steam reformation of methane depends strongly on the cost of the natural gas, which is used for both the feedstock and the energy source. At the current natural gas cost of about \$6/MBtu, steam reformation can produce hydrogen for about \$1.40/kg, but produces CO_2 emissions. If carbon capture and sequestration is required, an estimated cost of up to \$100/ton of CO_2 could add as much as 20 cents/kg of H_2 to the cost of hydrogen from methane. Nuclear production of hydrogen using the GT–MHR could thus be competitive at today's prices for natural gas. As the price of natural gas rises with increasing demand and decreasing reserves, nuclear production of hydrogen would become more and more cost effective while producing no greenhouse gas emissions.

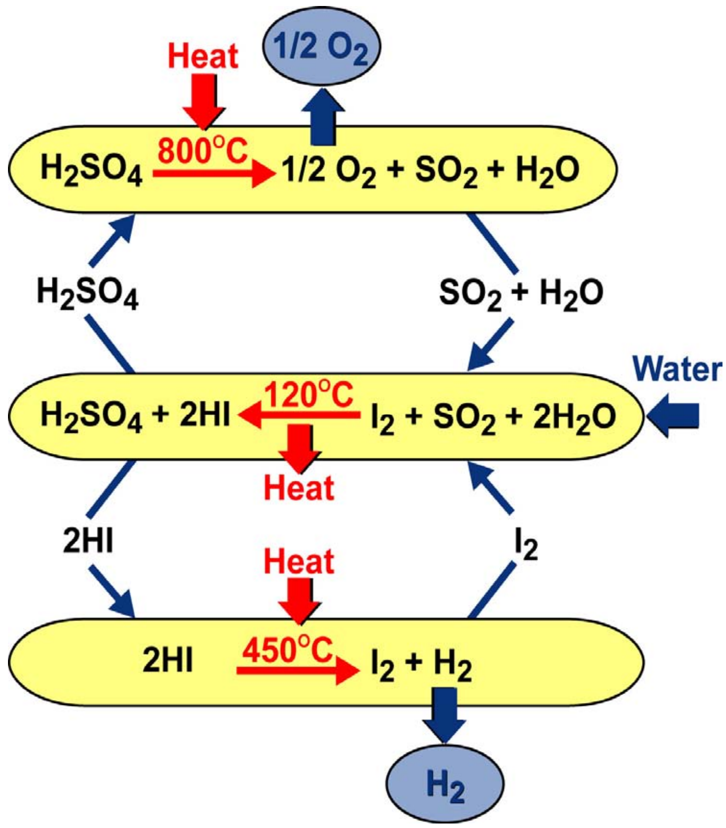


FIG. XV-18. Sulphur-iodide hydrogen production process.

XV-2.4. Plant layout

The GT–MHR is designed to be deployed in four module plants, with all four modules controlled from a central control room. A perspective view of a four module GT–MHR, having a total generation capacity of 1148 MW(e), is shown in Fig. XV-19.

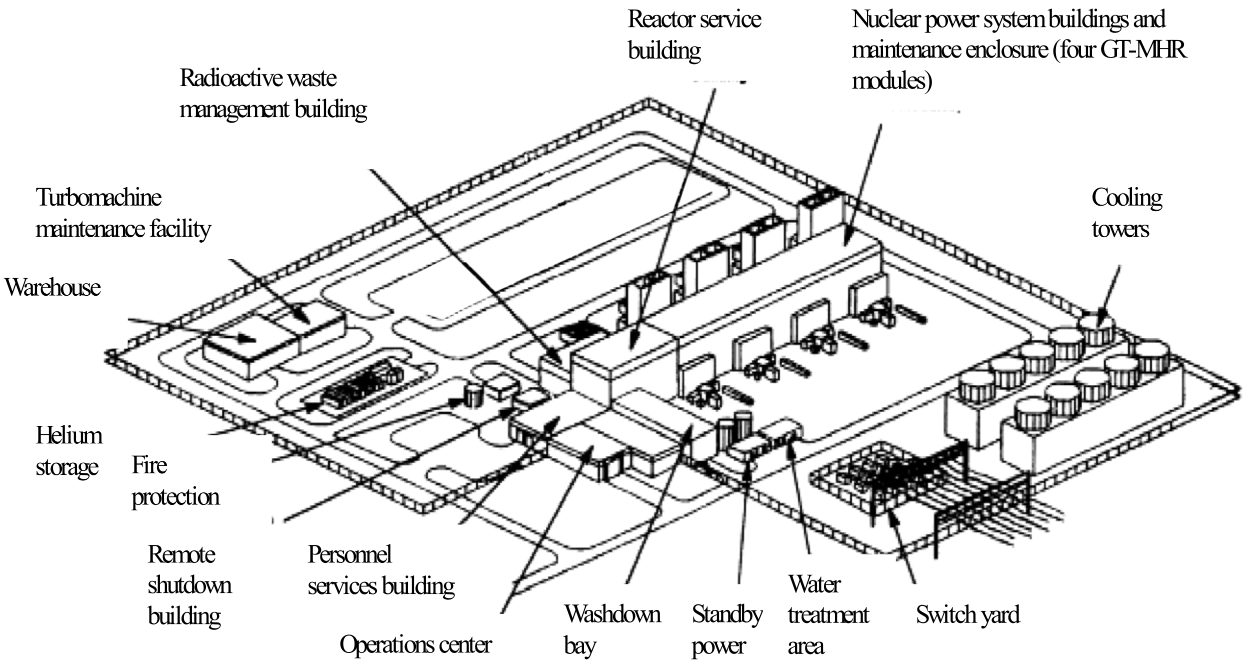


FIG XV-19. GT–MHR plant layout.

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GAS TURBINE HIGH TEMPERATURE REACTOR 300 (GTHTR300)
JAERI, Japan

XVI-1. General information, technical features and operating characteristics

XVI-1.1. Introduction

In Japan, the development of High Temperature Gas Cooled Reactor (HTGR) technology has been conducted for over 20 years. The High Temperature Engineering Test Reactor (HTTR) [XVII-1] with outlet gas temperatures of 950°C and a thermal power of 30 MW was constructed at Oarai Research Centre in the Japan Atomic Energy Research Institute (JAERI). First criticality was attained in 1998 [XVII-2] and full power operation with outlet gas temperatures of 850°C was completed on December 7, 2001. Since then, a safety demonstration test has been conducted and operational data to establish and upgrade the HTGR technology base will be accumulated over the next several years.

In parallel to the development and successful operation of the HTTR, since 1997 JAERI had undertaken a feasibility study on various types of HTGRs with Gas Turbines (HTGR-GT). As a result of this study, JAERI selected a block-fuel-type HTGR with a direct cycle gas turbine system as the best candidate for a future commercial reactor purely from an economic and technological point of view [XVI-3]. Since 2001, JAERI has been designing an original Japan gas turbine high temperature reactor, Gas Turbine High Temperature Reactor 300 (GTHTR300). The greatly simplified design is based on salient features of the HTGR with a closed helium gas turbine and enables the GTHTR300, a highly efficient and economically competitive reactor, to be deployed in the early 2010s. Also, the GTHTR300 takes full advantage of experience accumulated in the design, construction and operation of the HTTR and fossil gas turbine systems to reduce technological development necessary to complete a reactor and electric generation system. The original features of this system are: a reactor core design based on a newly proposed refuelling scheme named sandwich shuffling; use of conventional steel material for a reactor pressure vessel; an innovative plant flow scheme and a horizontally-installed gas turbine unit. The GTHTR300 can be continuously operated without the refuelling for two years.

The principal stakeholder in the GTHTR300 is Japan Atomic Energy Research Institute (JAERI).

XVI-1.2. Applications

The GTHTR300 is being developed to provide passively safe and economic electric power generation and to enable other attractive cogeneration applications including high temperature process heat for hydrogen production and process steam and low temperature heat for seawater desalination and district heating. The GTHTR300 design is described more fully for the electricity generation system in reference [XVI-4] and for the electricity and hydrogen cogeneration system in reference [XVI-5].

XVI-1.3. Special features

A GTHTR300 power station may contain one or more modular reactor systems with nominal generating capacity of 300MW(e) per modular unit.

XVI-1.4. Summary of major design and operating characteristics

The GTHTR300 design for the electricity generation system shown in Fig. XVI-1 consists of three subsystem modules including a 600MW(th) prismatic fuel reactor module, a horizontal gas turbine generator (GTG) module and a vertical heat exchanger (HTX) module. The cycle process flow path and main parameters are depicted in Fig. XVI-2. The prismatic reactor has an annular core in which the thermal power is maximized within passive reactor safety requirements and within the bulk limit for reactor pressure vessel construction. The fuel design is improved from the pin-in-block fuel element of the HTTR test reactor that JAERI developed and is now operating successfully. The reactor thermal power is maximized to 600 MW(th) commensurate with the passive reactor safety requirements and reactor pressure vessel construction bulk limit.

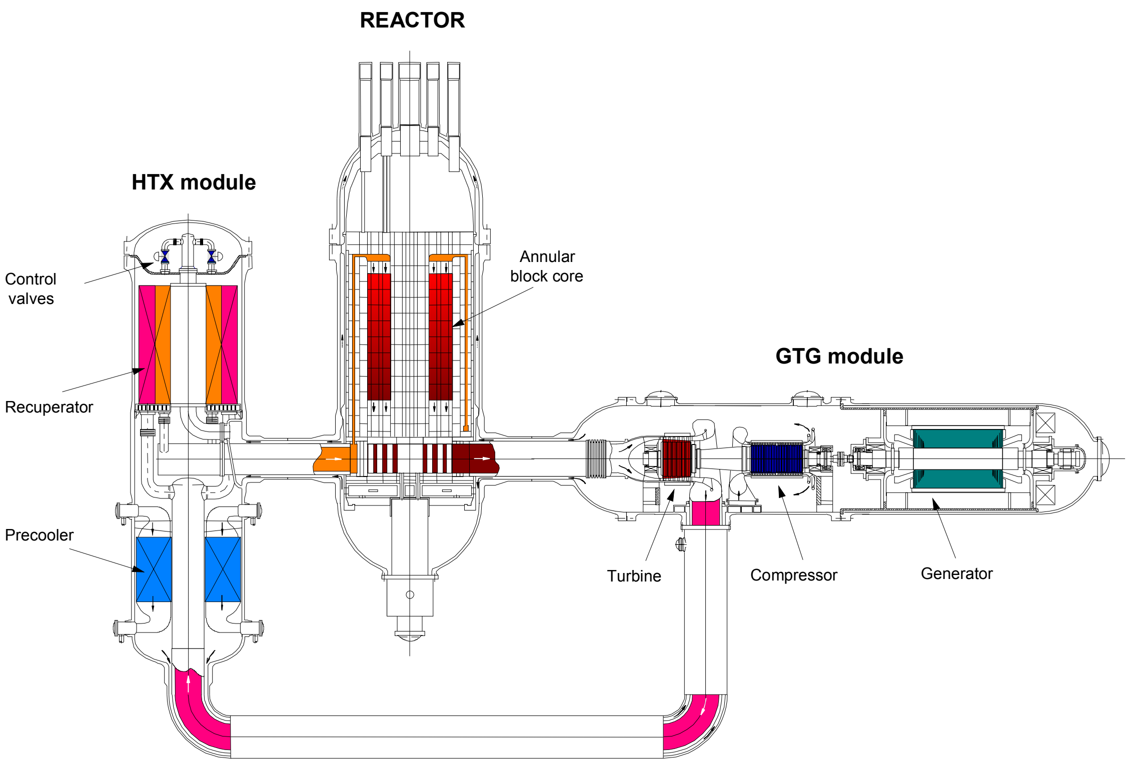


FIG. XVI-1. GTHTR300 power plant primary system arrangement.

For timely development of a successful system, the GTHTR300 assumes a system design approach of simplicity, economic competitiveness and originality, namely *SECO,* as follows:

- *Simplicity* — Greatly simplified design solutions are incorporated to minimize significant technical development requirements or risk and to yield the kind of system simplicity characterized by the least number of components and a modular system layout facilitating low-cost construction and improved maintainability;
- *Economic competitiveness* — The key to economic performance is a combination of system simplicity with maximized scale (reactor thermal power) and high generating efficiency. The targeted cost of power generation is ¥4/kW·h (~3.5¢/kWh), which is competitive with other nuclear system options in Japan;
- *Originality* — The design and economic goals identified above represent a unique set of system requirements, making the development of an original plant system imperative.

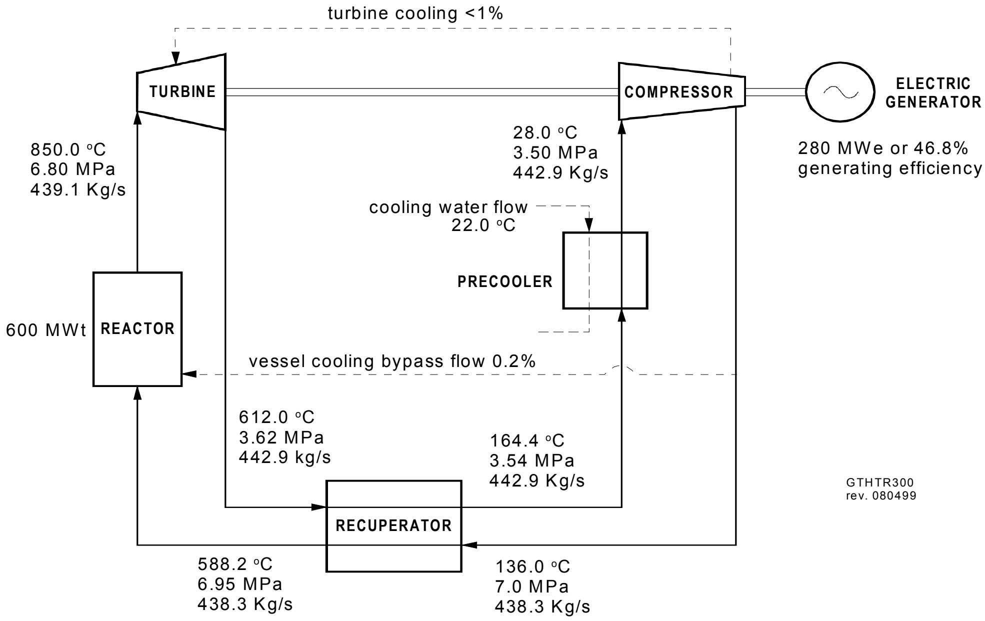


FIG. XVI-2. Cycle process flow path and major operating parameters.

A summary of the GTHTR300 design and operating characteristics is presented in Table XVI-1.

TABLE XVI-1. SUMMARY TABLE OF DESIGN AND OPERATING CHARACTERISTICS [XVI-4, XVI-6]

CHARACTERISTIC	VALUE
Rated thermal power	600MW(th)
Rated electric power	274MW(e)
Load factor	93%
Targeted availability	90%
Type of fuel	Pin-in-block type fuel based on TRISO coated particles with UO_2 kernels
Fuel enrichment	14 weight %
Coolant	Helium
Moderator	Graphite
Structural materials of reactor internals	Fuel blocks, replaceable reflectors, core support, permanent reflector, core lateral restraint - Graphite corresponding to IG11 Metallic internals - NCF800H
Core	Annular prismatic (Hexagonal blocks) Height - 8.0 m Fuelled annulus inner diameter - 3.7 m Fuelled annulus outer diameter - 5.5 m
Cycle type	Direct closed cycle gas turbine
Number of coolant circuits	1

CHARACTERISTIC	VALUE
Reactor vessel	Material - Steel (SA533) Height - 24.3 m Inner diameter - 7.6 m Wall thickness - 0.17 m Design lifetime - 60 years
Neutron-physical characteristics	Temperature reactivity coefficient - < -6 $10^{-5}\Delta k/k/°C$ Coolant void & density coefficients - Negligible Excess reactivity - 17.7 %Δk/k Maximum axial peaking factor - 1.7 Maximum horizontal peaking factor - 1.4 Power flattening by control rods and adequately arranging burnable poisons. The burnable poison consists of B_4C/C compacts, stored in a hole under the dowel pin arranged at the top of three corners in a block.
Types of reactivity control and reactor protection	Control rods (absorber: B_4C/C compacts) for normal operational control and shutdown Shutdown margin >1.4 %Δk/k Reserve shutdown system using B_4C/C pellets Shutdown margin >2.3 %Δk/k
Thermal-hydraulic characteristics	Forced circulation driven by power turbine Fuel temperature (normal operation) — Maximum 1398°C (with uncertainties); Limit 1400°C Fuel temperature (accident conditions) — Maximum 1595°C (with uncertainties); Limit 1600°C Coolant temperature — Inlet 587°C; Outlet 850°C Coolant flow rate - 438 kg/s Coolant Pressure - 7.0 MPa
Fuel Cycle	Discharge burn-up Segment average - 120 (MW·d/t) Segment maximum - 155 (MW·d/t) Refuelling pattern Sandwich shuffling Refuelling interval - 730/24 (EFPD/months Fraction of core refuelled - 1/2 Full fresh U inventory - 7.09 (tons)
Design lifetime for permanent internals	60 years
Economics	Capital cost (target) Yen/kW(e) - less than 200 thousand Electric cost (target) - 4 Yen/kW·h

XVI-1.5. Outline of fuel cycle options

HTGR fuels are expected to be suitable for a once through fuel cycle with final disposal after interim storage because of high burn-up and low quantity of residual fissile materials. However, since the national policy for nuclear energy in Japan mandates fuel cycles with reprocessing of spent nuclear fuels, the standard fuel cycle for GTHTR300 uses low-enriched uranium in a reprocessing mode. The GTHTR300 is shown to maintain a minimal-peaked power distribution at an enrichment of only 14%.

The feasibility of GTHTR300 spent fuel reprocessing was investigated [XVI-7]. In HTGR fuel elements, fuel particles consisting of UO_2 fuel kernels with TRISO coatings are dispersed in a graphite matrix. Since reprocessing of UO_2 fuel kernels is basically same as that of light water reactor (LWR) fuels, that is, using the PUREX process, an additional process to supply HTGR fuel to reprocessing plants for LWR fuels was devised as a head-end process. The 1st step of the head-end process is to remove the matrix graphite and carbon outer coating layer of the fuel particles. The 2nd step is to remove the SiC layer of coated fuel particles. In the 3rd step, two inner graphite layers are roasted and removed from UO_2 fuel kernels; UO_2 is oxidized and U_3O_8 is obtained by this step. In the conditioning process after the head-end process, U_3O_8 is dissolved with nitric acid and diluted with depleted uranium solution to meet the conditions of the PUREX process for reprocessing for LWR fuels in Japan.

XVI-1.6. Technical features and technological approaches that are definitive for GTHTR300 performance in particular areas

XVI-1.6.1. Economics and maintainability

The features of GTHTR300 that secure excellent economics and maintainability are:

- A GTHTR300 power station may contain one or more modular reactor systems with a nominal generating capacity of 300 MW(e) per unit.
- The fuel element is the same type of pin-in-block design demonstrated in the HTTR and improved with a sleeveless fuel rod design. The low-enriched uranium fuel cycle has design characteristics of high (120 GW·d/t) burn-up and low power peaking factor for an extended (2-year) refuelling interval.
- The reactor pressure vessel, which is cooled in an intrinsic flow scheme, makes use of the conventional steel SA533/SA508.
- The system design utilizes horizontally installed turbines with a non-intercooled Brayton cycle.
- All turbomachine and heat exchanging components are appropriately sized and arranged in systems to facilitate modular construction and maintenance.

Due to these salient features, the capital cost of the GTHTR300 is less than a target cost of 200 thousands Yen/kW(e) and the electric generation costs are close to a target cost of 4 Yen/kW·h.

XVI-1.6.2. Provisions for sustainability, waste management and minimum adverse environmental impacts

The quantity of heavy metals in the spent fuel of the GTHTR300 is less than that in LWRs. In the fuel reprocessing, many graphite blocks separated from fuel blocks become the waste and waste fixing CO_2 gas is generated in large quantities after their burning; carbon-14 in the

waste should be dealt with carefully. However, the waste of the GTHTR300 is almost low-level waste; the amount of high-level waste is far less than in LWRs and since the GTHTR300 has high heat efficiency and a long residence time of fuel in the core plus a lower quantity of loaded fuel. There is no generation of high-level metal waste such as claddings and elements of fuel assembly in LWRs. For long-term storage of spent fuel, it is necessary to manage the atmosphere in storage vessels to prevent oxidation. However, since decay heat is low enough and the temperature of stored spent fuel is far less than the fuel temperature during reactor operation, stable long-term storage of spent fuel is possible.

XVI-1.6.3. Safety and reliability

Safety concept and design philosophy

Since HTGRs use ceramic-coated fuel particles and graphite core structures instead of metal cladded fuels or metallic reactor components, the core of HTGRs can endure high temperature conditions during abnormal events including accidents. The transient temperature behaviour during abnormal events is slow due to the large heat capacity and small power density of a graphite moderated reactor core. Safety concepts and the design philosophy of the GTHTR300 are based on these inherent safety characteristics of HTGRs.

A unique safety concept is proposed for the GTHTR300. Severe accidents are defined as any conditions beyond design basis accidents that cause core damage with fission product releases to the environment, although all severe accident sequences are very low in probability. The new safety concept in the GTHTR300 is to avoid most accidents and to achieve a probability of severe accidents of less than 10^{-8}/reactor-year, that is, at least two orders lower than for current reactors. Even in the worst event such as a double guillotine break of a primary concentric duct, fuel temperature exceeding its failure limit and excessive fuel oxidation by air ingress can be avoided because of inherent safety features and the passive removal system for decay heat. Furthermore, the double confinement building adds another design measure to keep the reactor safe in such accidents.

Nearly full-scale worst accident simulation tests can be carried out to obtain licensing before commercial operation (demonstrable safety) to convince the public and the regulators of this safety concept. Safety demonstration by accident simulation tests can be, and shall be, requisite in licensing of the GTHTR300.

Structure of the defence-in-depth and active and passive systems and inherent safety features

Defence-in-depth is a basic philosophy for the GTHTR300 as well as for the Light Water Reactors (LWRs). Various layers of defence are used to maintain safety. However, major differences of the safety philosophy between the GTHTR300 and the current LWRs are the following: the LWR uses highly reliable, redundant and diverse passive or active safety features. On the other hand, GTHTR300 safety would be maintained mostly due to inherent safety characteristics and potentially safe components.

Design basis and beyond design basis accidents

The safety evaluation of the GTHTR300 has been performed considering its inherent safety characteristics and salient engineering features. The classification of the events to be evaluated is based on the HTTR and LWR safety evaluation guidelines. In LWR and HTTR safety evaluations, the Anticipated Operation Occurrence (AOO) is defined as the off-normal event with its frequency of roughly lower than approximately 10^{-2}/reactor-year and the accident is defined as one with a frequency lower than approximately 10^{-4}/reactor-year. A deterministic evaluation will be carried out for these AOOs and accidents. On the other hand,

in the GTHTR300 safety evaluation, an accident is defined as an event with a frequency in the range from 10^{-2}/reactor-year to 10^{-8}/reactor-year. Examples of classification of abnormal events are listed in Table XVI-2; deterministic evaluations would be performed for these accidents. A full-scale probabilistic safety assessment (PSA) will not be forced for designing the GTHTR300; such evaluation is being performed for reference.

A large amount of fission products (FPs) released from a primary coolant pressure boundary would directly cause a large FP release into environment because there is no containment vessel (CV) in the GTHTR300. In the LWR, the probability of a large FP release into environment is decreased to less than approximately 10^{-8}/reactor-eary by the adoption of a CV and accident management. In this way, risk for the general public can be considered negligible. From the same point of view, the threshold value of the probability of a severe accident to be evaluated is fixed as 10^{-8}/reactor-year in the GTHTR300. Since unacceptable radioactivity release beyond the plant boundary can be prevented by this philosophy, a concept of "severe-accident-free" design could be applied to the GTHTR300.

TABLE XVI-2. EXAMPLES OF CLASSIFICATION OF ABNORMAL EVENTS

CATEGORY	EVENTS OR COMBINATIONS OF EVENTS INCLUDING FAILURE OF SAFETY FUNCTIONS	FREQUENCY (1/REACTOR-YEAR)
AOO	- Loss of generator load - Decrease of primary coolant pressure - Decrease of primary coolant flow rate - Control rod withdrawal at rated power operation	1.0 2×10^{-2} 1×10^{-2} 1×10^{-2}
Accident	- Turbine deblade - Small leakage of primary coolant - Failure of gaseous radioactive waste treatment facility - Large leakage of primary coolant - Turbine deblade + failure of protection bypass valve - Control rod withdrawal + failure of scram (control rods) - Small leakage of primary coolant + failure of scram (control rod)	1×10^{-3} 5×10^{-4} 1×10^{-4} 1×10^{-5} 1×10^{-5} 1×10^{-6} 5×10^{-8}
Beyond design basis event	- Large leakage of primary coolant + failure of scram (control rods) - Large leakage of primary coolant + failure of vessel cooling system	1×10^{-9} 1×10^{-12}

A containment vessel is not necessary in the GTHTR300 because it is designed as severe-accident-free reactor, that is, no significant fuel failure is assumed to occur even in accidents. However, the concrete reactor building is designed to provide a confinement structure with a low leak rate, to mitigate air ingress into the reactor core. This is to prevent the oxidation of fuel during a depressurization accident, induced by large break in the primary coolant boundary. The reactor building should be intact against any pressure attack in a depressurizing accident, that is, a kinetic blowout pressure and static pressure increase inside the building by helium gas release from the helium coolant boundary. No additional fuel particle failure occurs during an accident and the failure fraction of fuel particles during normal operation is very small. Therefore, the release of helium gas into environment during the initial phase of a depressurizing accident can be allowed and the confinement function of the reactor building remains intact during depressurization.

A vessel cooling system (VCS) is a residual heat removal system for the complete loss of forced cooling in a depressurizing accident. The VCS is designed as a system for passive heat removal by the natural circulation of air in the cooling panels installed at the outside of the reactor pressure vessel (RPV). Residual heat in the reactor core is transferred to the cooling panel by radiation from the outer surfaces of the RPV and by the natural convection of air in the cavity between the RPV and the cooling panels. The VCS is designed to keep the temperature of the fuel and the RPV lower than each temperature limitation during normal operation or even during an accident such as depressurization.

A result of the evaluation of the temperatures in the RPV and fuel during a depressurizing accident is shown in Fig. XVI-3. The fuel temperature does not exceed 1600°C, the criterion for safety evaluation. The VCS has a fundamental safety function; the characteristic of heat removal by the VCS will be demonstrated by safety tests of the HTTR under conditions simulating the loss of forced flow and reactivity insertion.

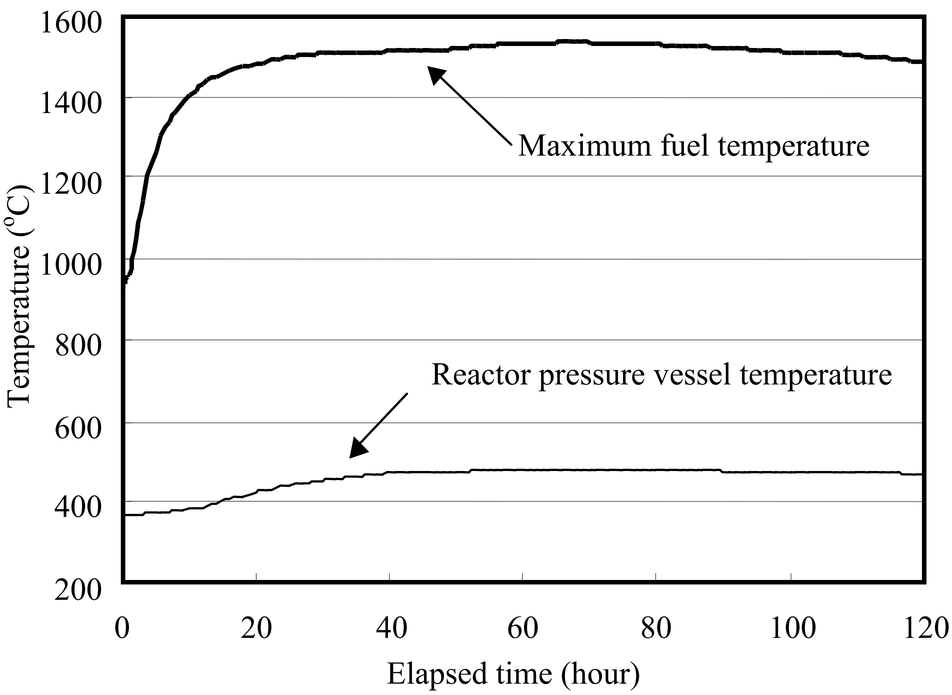


FIG. XVI-3. Reactor transient during depressurization accident.

A reactor shutdown system consists of two diverse and independent systems: a control rod (CRS) and a reserve shutdown system (RSS). The CRS has sufficient reactivity worth to shutdown the reactor from operating to a cold subcritical condition with a "one rod stuck margin". The RSS has the reactivity worth to reach the subcritical condition at high temperatures typical of operating conditions. Control rods can be dropped into channels in the reactor core by gravity, in case of emergency shutdown. In the RSS, B_4C/C pellets are dropped into a hole at the same reflector column as the control rod channel. From the point of view of reactivity initiated events, the sum of the reactivity worth of the control rods which are ejected at the same time by a one stand-pipe rupture is designed to be lower than 0.5%Δk/k, to prevent fuel failure by temperature increases during a stand- pipe rupture.

XVI-1.6.4. Proliferation resistance

Fresh fuel of the GTHTR300 consists of coated fuel particles with low enriched uranium that is on itself unattractive for weapon programmes. The radiation dose of spent fuel is too high

to be easily approached because of the high content of fission products due to high burn-up. Residual plutonium is degraded in several years of storage since plutonium-241, which has a short half-life, exits at a high ratio from residual fissile plutonium. Residual fuel including plutonium is dispersed in many fuel particles and is impossible to extract without destroying the hard coating layers of the fuel particles. For these reasons, weapons-usable materials are very difficult to extract. The GTHTR300 has the flexibility to select various fuels and burn-up cycles and it is possible to utilize the transmutation of plutonium to reduce the amount of weapons-usable materials in spent fuel.

XVI-1.6.5. Technical features and technological approaches used to facilitate physical protection of GTHTR300

Since HTGRs use ceramic coated fuel particles and graphite core structures, the core of GTHTR300 can endure high temperatures during abnormal events including accidents. Transient temperature behaviour during abnormal events is slow due to the large heat capacity and the low power density of a graphite moderated reactor core. Based on these inherent safety characteristics of HTGRs, the safety concept and design philosophy of the GTHTR300 provide a strong defence against external impacts and sabotage caused by the internal actions of the personnel.

XVI-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of GTHTR300

To ensure that the GTHTR300 meets the market demands in Japan, utility and user requirements were established in consultation with major domestic utilities and industries. In addition, the committees representing academics, utilities and industries have reviewed the design and development programme ongoing in JAERI on a regular basis. A deployment roadmap was prepared for the GTHTR300.

XVI-1.8. List of enabling technologies relevant to GTHTR300 and status of their development

Since the objective is to deploy the GTHTR300 in the near term, the design approach was to minimize development requirements by simplifying the system, using existing materials and design codes, and by following proven component practices. This design approach is noted in the plant systems that incorporate reactor technologies developed in the HTTR, Table XVI-3.

In addition to reducing new development, this approach substantially reduces cost and time in fulfilling system validations. As a result, commercial prototype demonstrations can be pursued at technical and financial risks acceptable to the vendors and utilities willing to share public interest in the GTHTR300 deployment.

XVI-1.9. Status of R&D and planned schedule [XVI-8, XVI-9, XVI-10]

The GTHTR300 project has completed the basic design phase consisting of system and component design, safety evaluation and economic assessment. After a design review by utilities, modification of the basic design will be conducted until 2007. The basic design and development of the GTHTR300 including R&D for the gas turbine system will be finished by the end of March 2008.

A deployment roadmap was prepared for the GTHTR300, moving from the ongoing plant design and development through prototype demonstration towards full deployment (Fig. XVI-4).

TABLE XVI-3. LIST OF MAJOR ENABLING TECHNOLOGIES FOR GTHTR300

Enabling technology	Status
Non-intercooled conversion cycle	The system configuration is simpler than the intercooled cycle. The demonstration of control and operation for this system will be necessary
Conventional carbon steel reactor pressure vessel	This material is used in RPV for the LWRs. No future R&D is necessary.
Horizontally oriented turbo-machinery	Almost all of the GT systems in Japan were mounted horizontally. No technical problems are foreseen.
Magnetic bearing for turbo-machinery	No operational and manufacturing experiences for this size of magnetic bearing are available. The R&D to confirm basic performance of the magnetic bearing is necessary.
Aerodynamic design for turbo-machinery	No helium based closed cycle system was ever constructed and operated except the German HHT project. R&D to develop the turbo-machinery design is necessary.
High burn-up fuel	Technologies used for the HTTR fuel will be directly applicable. However, the irradiation performance data should be accumulated.
Fuel recycling technology	The basic technology for the fuel recycling has already been developed. However, it should be demonstrated in a small-scale test facility.
Waste disposal	Graphite blocks discharged from the core every two years should be properly stored. An economical storage system will be developed.

The roadmap comprises three successive phases, each of which includes further carefully structured steps, to allow full system development in stages while minimizing technical risks and development costs.

The first phase ongoing in JAERI lasting through 2007 and receiving exclusive funding from the Ministry of Education, Culture, Sports, Science and Technology (MEXT) of Japan, covers the prototype plant design and associated basic R&D technology development. For each major component and subsystem employed in the design, proven technologies from the best industrial experience bases were assessed and adopted whenever appropriate and additional technology needs have been identified. Plans for necessary technology development are executed by thorough subcomponent and subsystem development and testing.

The second phase, the utility prototype plant demonstration, will be carried out over a period of ten years (2008-2018), mainly in the private sector, with public funds added to cover only one-time costs.

Success of the prototype plant development and demonstration will mark an important milestone towards full deployment of the GTHTR300. Full deployment during Phase III will exploit significant system advancement options. Through development and introduction of a few performance-enhancing technologies in fuel, material and power equipment, the GTHTR300 would be commercially deployed, achieving more than 50% net plant efficiency and a further 10% reduction in the cost of electricity. The system would be upgraded to deliver 950°C helium, which will contribute to the deployment of other attractive systems such as the GTHTR300C cogeneration system for electricity and hydrogen around 2020 [XVI-5].

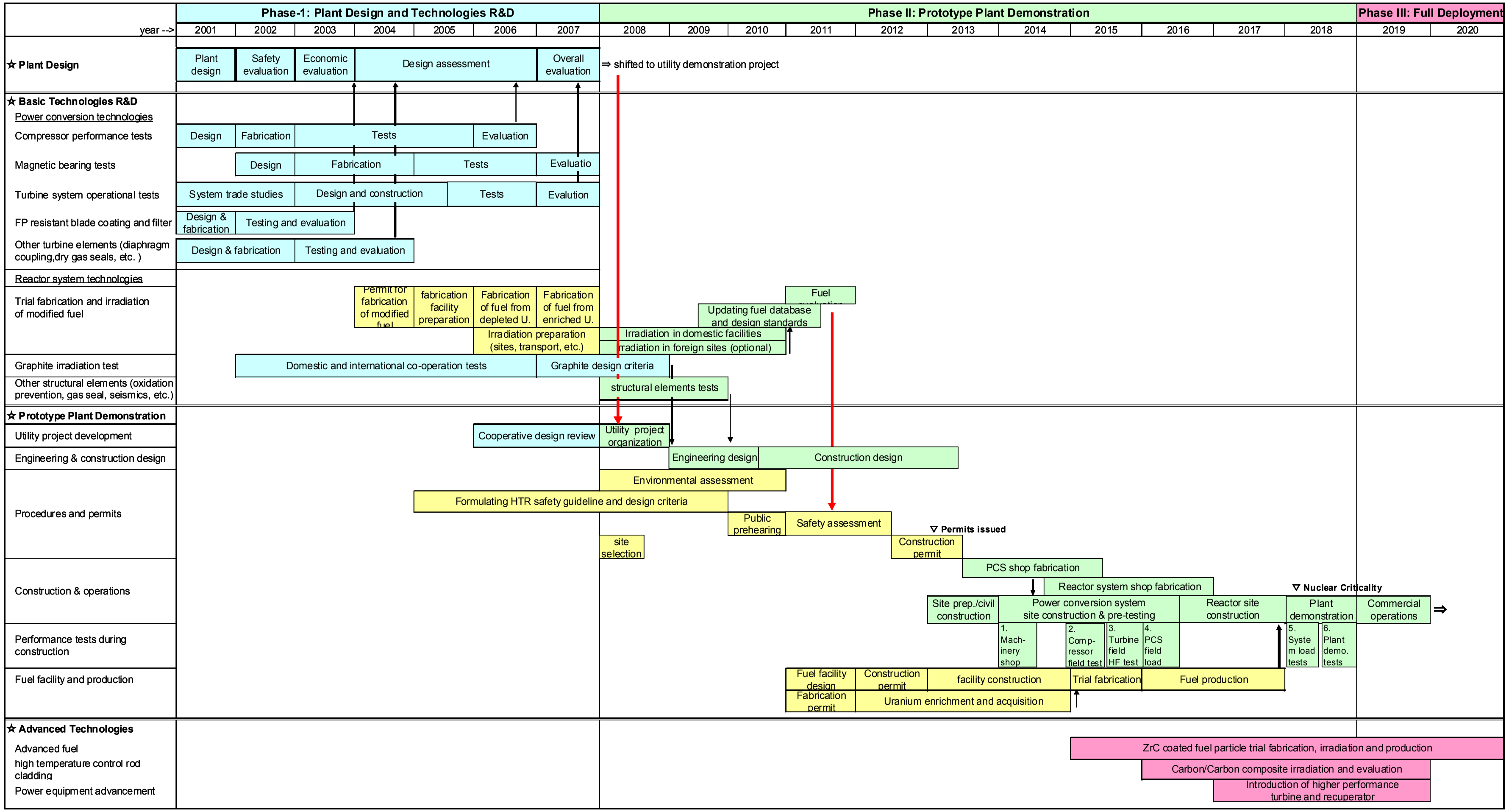


FIG. XVI-4. GTHTR300 deployment roadmap.

XVI-1.10. *Justification of why a demonstration prototype or a significant amount of demonstrations will be needed*

It is recognized that designing "right the first time" is not practicable for this or any other large-scale technological product. As discussed above, the GTHTR300 deployment roadmap comprises a series of evolutionary phases, including prototype demonstration, to develop the full system in stages. Such an evolutionary approach is intended to mitigate the risks associated with the simultaneous introduction of multiple innovative technologies.

The second phase of the GTHTR300 deployment roadmap calls for a utility prototype plant demonstration over a period of ten years (2008-2018) mainly in the private sector, with public funds added to cover only one-time costs. Full-scale system performance is to be validated through successive construction and tests of prototypical components, subsystems and finally, the full plant system. An increasing degree of confidence and system maturity is gained in stages for the power conversion system, the reactor and the full plant. Upon successful conclusion of demonstration tests, the prototype plant will be commissioned for commercial operations.

The prototype power conversion equipment, particularly the rotating components, will undergo shop acceptance tests to the fullest extent possible. These are followed by field tests of the partially to fully integrated power conversion system to verify the turbo-machinery aerodynamics and the system hot functions. To limit power input for cost saving, the field tests are planned at partial system pressures and partial to full turbine inlet temperatures, using either conventional or nuclear heaters. For example, the FSNL (i.e. full speed, no load) test, which has the main objective of validating the full-size compressor aerodynamics and efficiency, may be carried out at selected partial or rated conditions as indicated on the compressor performance map (Fig. XVI-5).

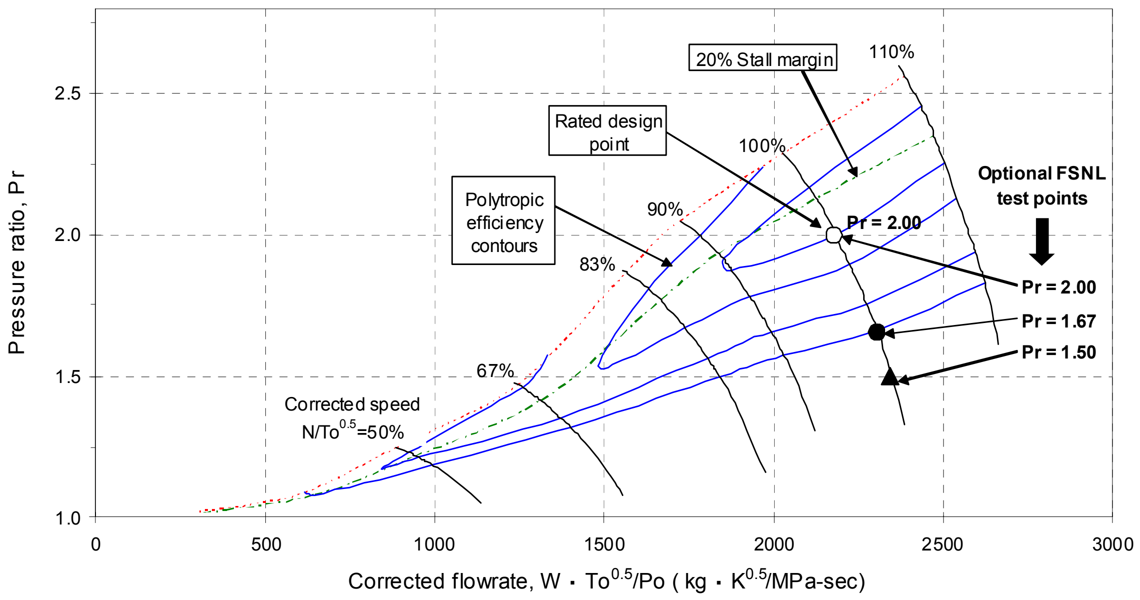


FIG. XVI-5. Optional FSNL test points marked on the compressor performance map.

The same range of power input to the FSNL test is suited to the power requirement of subsequent FSPL (full speed, part load) and the HF (hot functions) tests. The latter tests are conducted with partial compressor inlet pressures but full turbine inlet temperature. The essential tests on the power conversion system proceed to the reactor criticality and the demonstration tests of prototype plant operations at rated power.

XVI-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

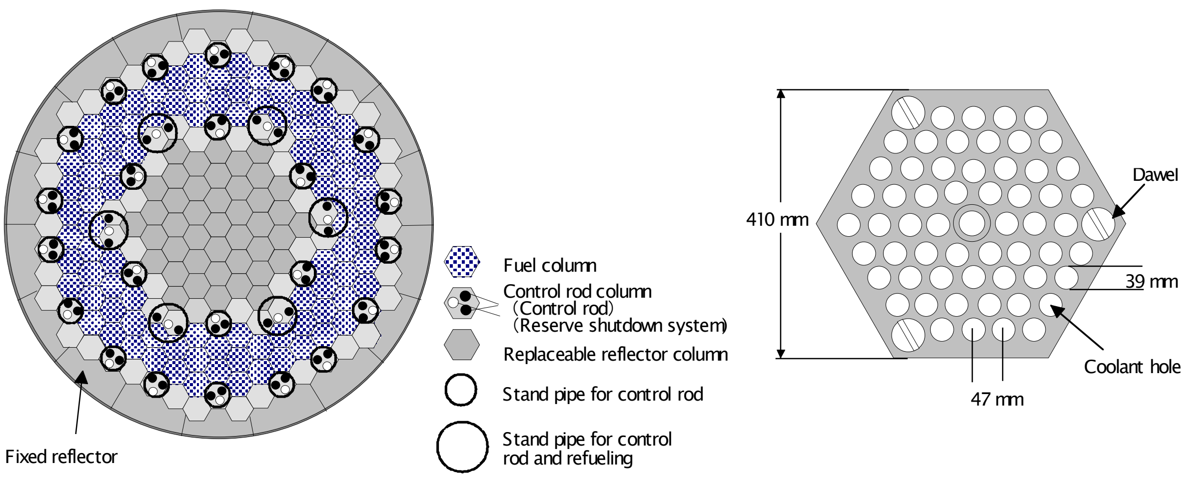
Development of the HTGR with gas turbine is underway in many countries. In specific, the commercial reactor project on Pebble Bed Modular Reactor (PBMR) is underway in South Africa [XVI-11]. The Gas Turbine Modular Helium Reactor (GT-MHR) concept is being developed in cooperation between the United States, Russia and other countries [XVI-12]. Various commercial systems have also been proposed in China [XVI-13] and the Netherlands [XVI-14].

XVI-2. Design description and data for GTHTR300

XVI-2.1. Description of the nuclear systems

Reactor core and fuel design

Figures XVI-6 and XVI-7 show the cross sections of the reactor core, fuel block, and coated particles.



(a) Reactor core (b) Fuel block

FIG. XVI-6. Cross sections of reactor core and fuel block.

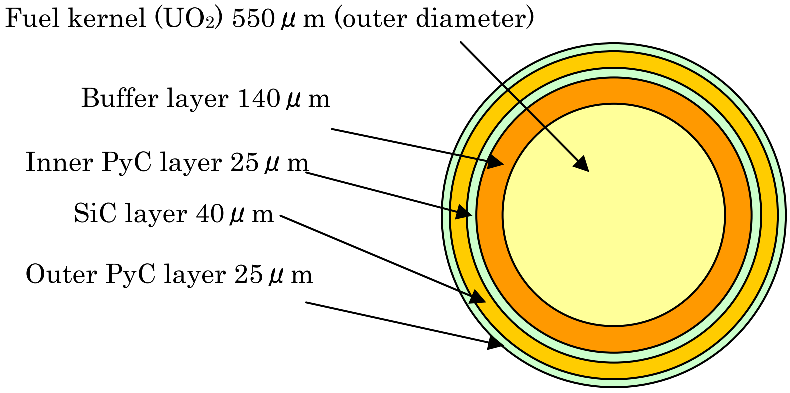


FIG. XVI-7. Cross cut of coated fuel particle.

Main heat transport system

A scheme of the GTHTR300 main heat transport system with specification of heat removal paths in normal operation and in accidents is shown in Fig. XVI-8.

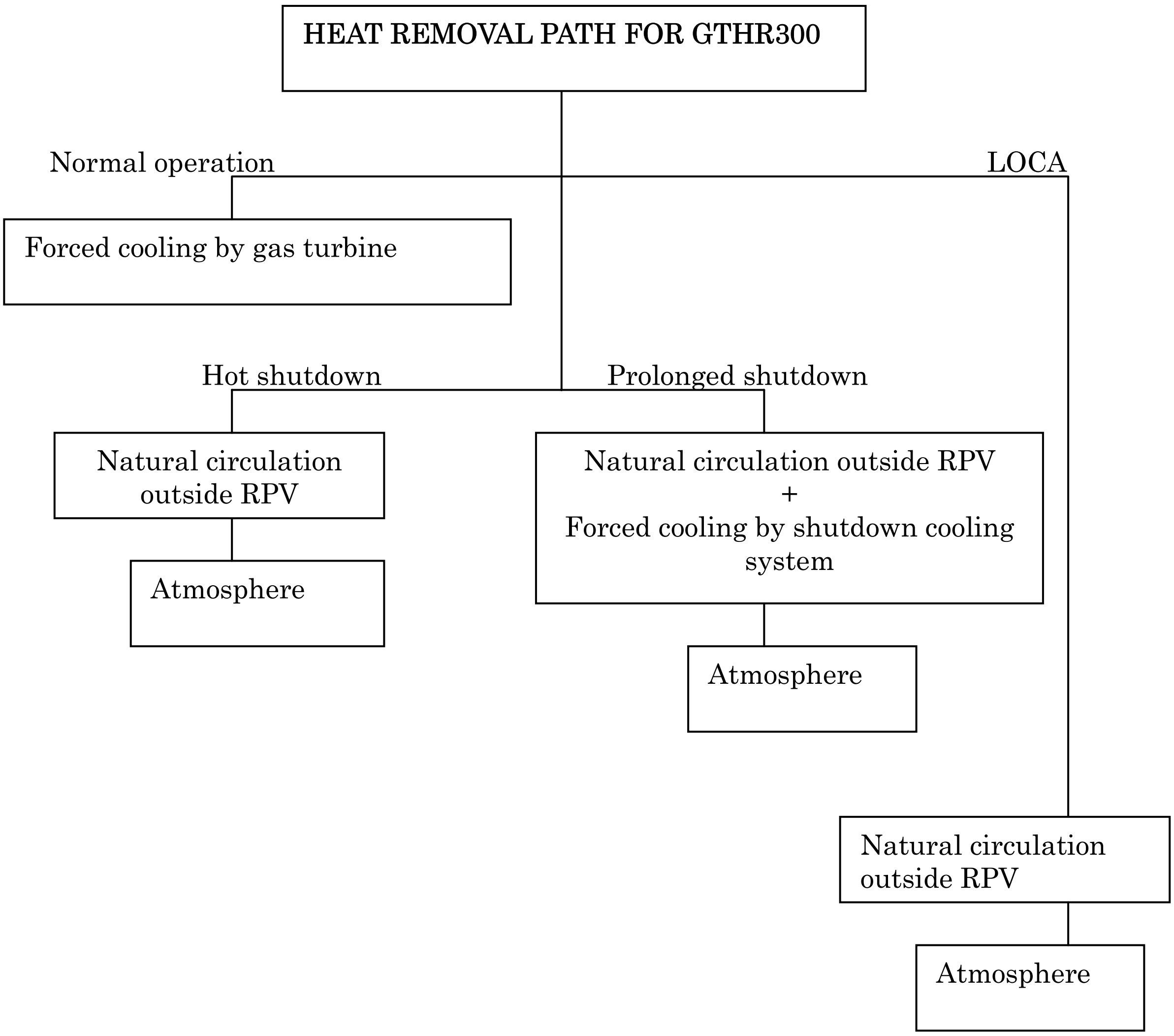


FIG. XVI-8. Heat removal paths for GTHTR300.

XVI-2.2. Description of the turbine generator plant and systems

The GTHTR300 uses a horizontally installed gas turbine, as shown in Fig. XVI-1. The system design utilizes the non-intercooled Brayton cycle because the study performed concluded that cycle intercooling results in substantial complexities in turbo-machinery and systems but provides no compelling advantage in overall plant economy. The non-intercooled helium turbo-machinery exhibits superior aerodynamic efficiency by fewer stages compared to that of the intercooled design and the rotor is lighter, shorter and more rigid, resulting in more robust vibration characteristics. The rotor is laid out horizontally to decrease load demands on bearings and to take advantage of the extensive field experience by industry in handling similar turbo-machines. The following validation and testing for the turbine unit and its separate components are in progress:

(1) Compressor aerodynamic performance test

The compressor aerodynamic performance test is aimed at verifying the aerodynamic performance and design method of the helium compressor. A 1/3-scale, 4-stage compressor test model and a helium gas loop, as shown in Figures XVI-9 and XVI-10 respectively, were designed and fabricated. Figure XVI-11 shows the view of the rotor installed on the lower-half casing of the test model. The main specification is shown in Table XVI-4. The model runs at 10800 rpm in helium gas at around 1 MPa with a design pressure ratio of 1.156. The model was designed to simulate the repeating stage flow and at the same time, have satisfactorily high machining precision, Reynolds number and measurement accuracy. Helium gas operating pressure is varied to investigate the effects of the Reynolds number on the efficiency and surge margin. Two sets of blades were fabricated to evaluate the effects of the end-wall over-camber angle. Test results will provide the basis for further improvement in the GTHTR300 compressor design.

TABLE XVI-4. MAIN SPECIFICATION OF THE COMPRESSOR MODEL

Working fluid	Helium gas
Flow rate	31450 m^3/min
Inlet temperature	30 ℃
Maximum inlet pressure	0.883 MPa
Pressure ratio	1.156
Reynoldes number (design point)	7.6×10^5
Base diameter	500 mm
Tip diameter (1st stage)	568 mm
Boss ratio (1st stage)	0.88
Number of stages	4
Rotational speed	10,800 rpm
Peripheral speed of rotor blade	321 m/s
Number of rotr/stator blades (1st stage)	72/94
Rotor/stator blade chord length (1st stage)	28.6/35 mm
Rotor/stator blade height (1st stage)	34/33.7 mm
Motor rated power	0.365 MW

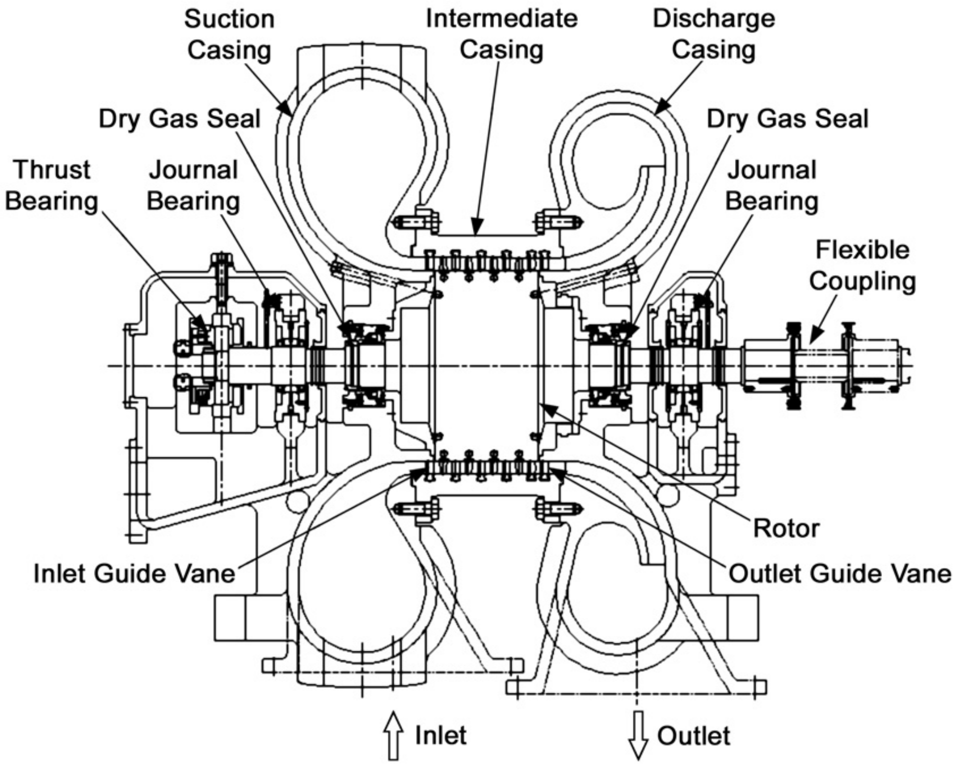


FIG. XVI-9. Longitudinal cross-section of compressor test model.

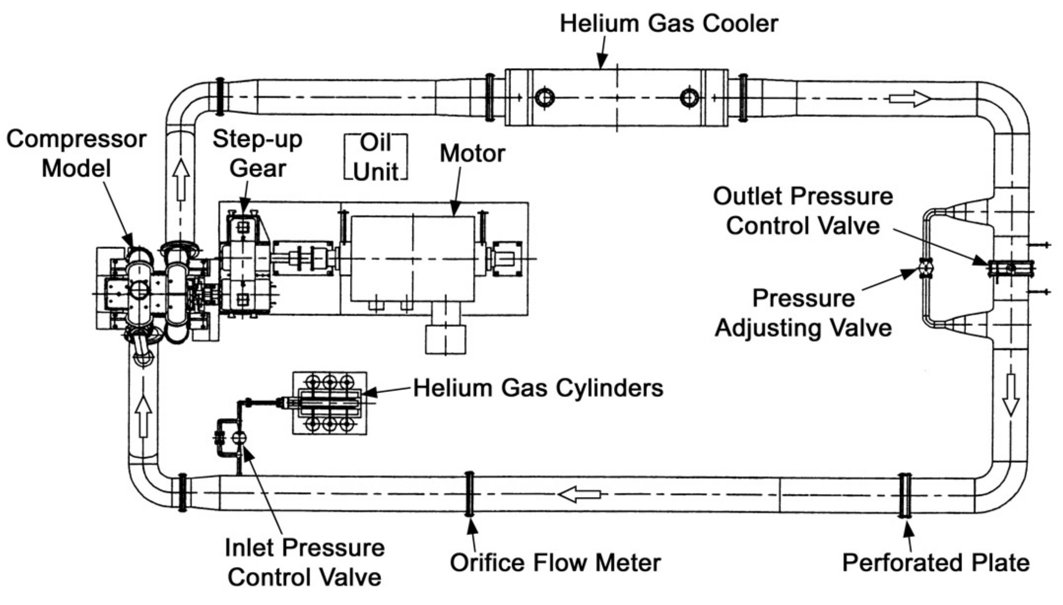


FIG. XVI-10. Layout of helium gas loop for compressor aerodynamic performance test.

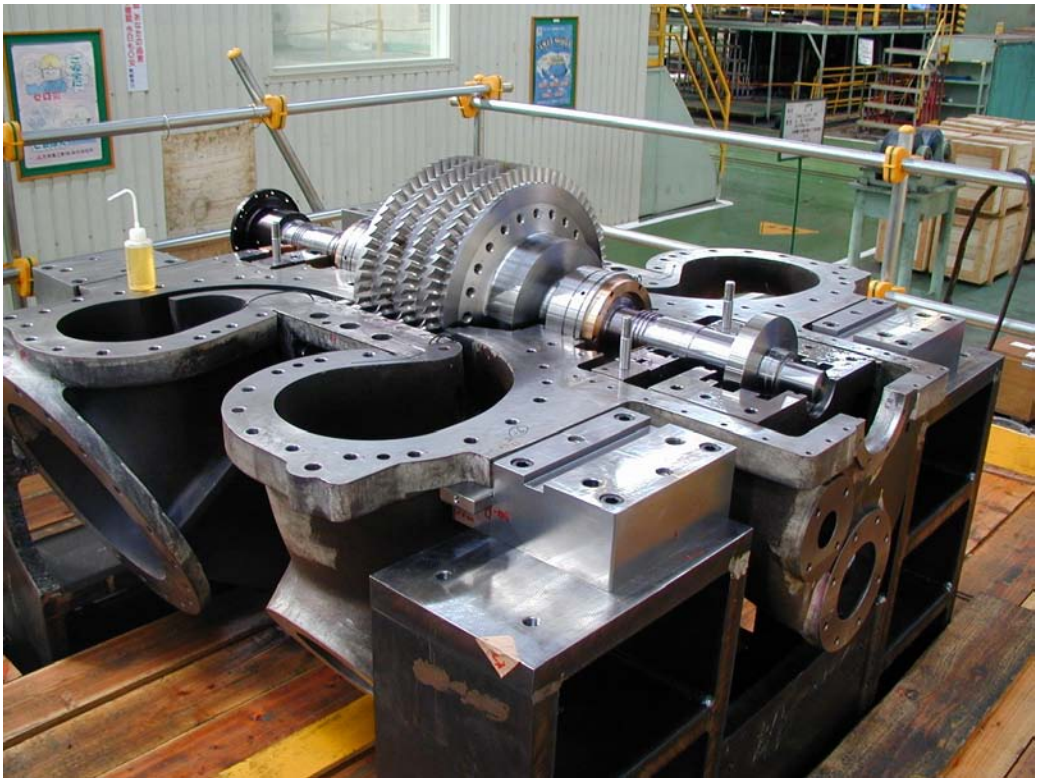


FIG. XVI-11. Rotor installed on lower casing of model compressor.

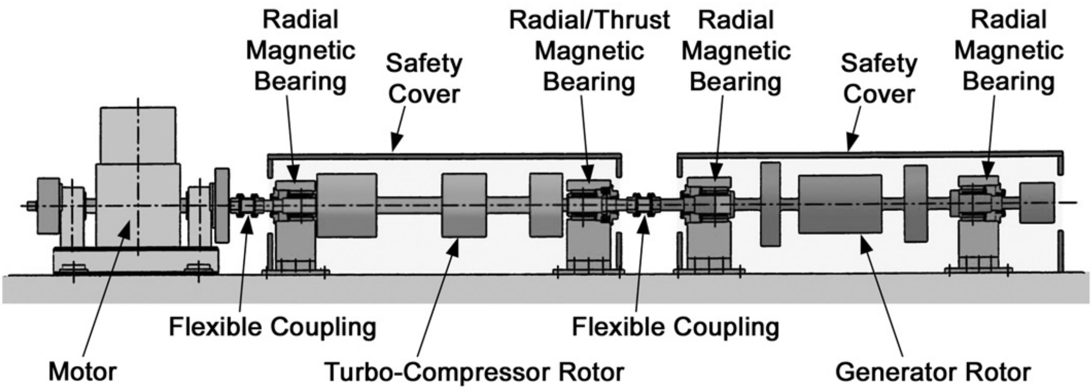


FIG. XVI-12. Layout of rotor test rig.

(2) Magnetic bearing development test

The magnetic bearing development test aims at developing the technology of the magnetic bearing supported rotor system. The test rig is composed of 1/3-scale turbo-compressor and generator rotor models connected by a flexible coupling as shown in Fig. XVI-12. Each rotor model is supported by two radial magnetic bearings with a high load capacity that is about 1/10 of the GTHTR300 design. Magnetic bearings supporting the rotor model have a load capacity of 27-35 kN. The rotor models were designed to match both the critical speeds and vibration modes of those of the actual rotors. A 1/3-scale test model of a turbo-machine rotor system and magnetic bearings were designed and fabrication of the test model was started. The main specification of the test model is shown in Table XVI-5. Testing of magnetic bearing performance, unbalance response, stability, and auxiliary bearing reliability will be carried out together with development of an advanced control method in the programme. The test will validate the design methods of the rotor system and magnetic bearing control system. The test will verify the rotor design of GTHTR300 and the results will identify technical issues in scaling-up magnetic bearings.

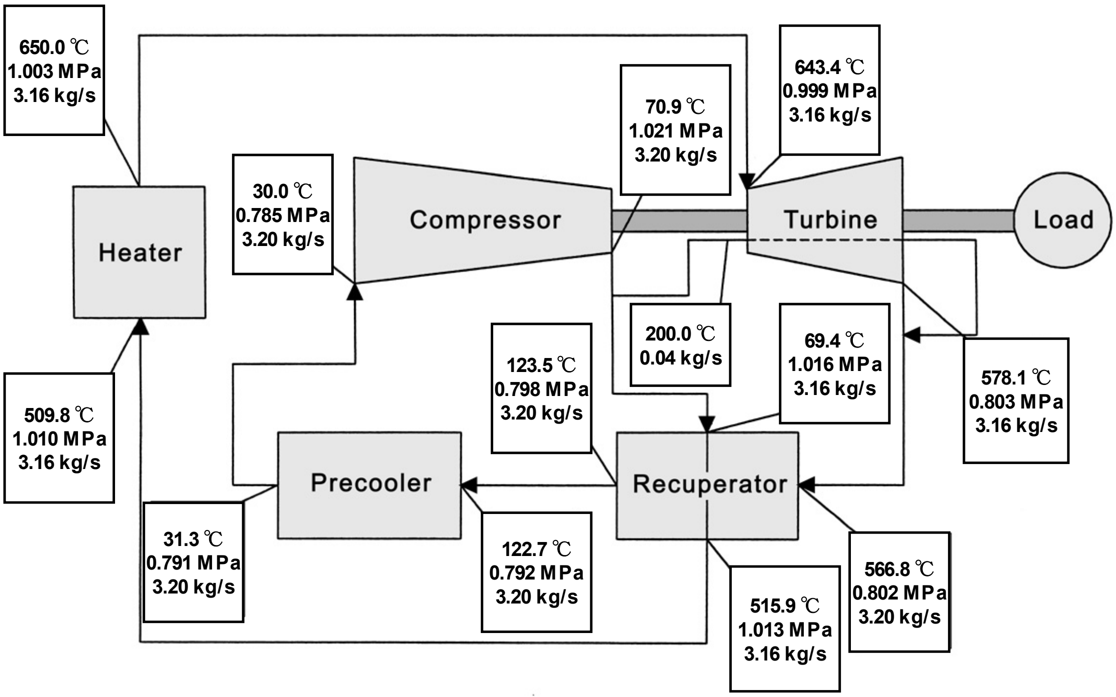


FIG. XVI-13. Heat and mass balance of test facility.

(3) Gas-turbine system operation and control test

The gas-turbine system operation and control test aim to demonstrate operability and control of the closed-cycle gas turbine system. The preliminary design for the test facility is an integrated scale-model of the GTHTR300 PCS. Pressurized helium gas at around 1 MPa is used as the working fluid and an electric heater simulates the reactor. The heat and mass balance of the test facility is shown in Fig. XVI-13. Planned test modes are normal operation, start-up, shutdown, load change, loss of load, and emergency shutdown.

Specifications of the system operation and control were defined to fulfil the test requirements as shown in Table XVI-6. The test will verify the design of the GTHTR300 control system and establish the operation and control method. Development and verification of a plant

dynamics analysis code will be made based on the test results. Major components of the facility, such as a turbine, a heater, a recuperator, and a pre-cooler, were designed to meet the specifications. Preliminary transient analyses were made for the test facility design. It was confirmed from the results of the analyses that the test facility can be operated to meet the requirements of the test plan and controlled satisfactorily. A revision of the design is underway to reduce the power demands and improve the control system.

TABLE XVI-5. MAIN SPECIFICATION OF THE TEST MODEL OF THE MAGNETIC BEARING SUSPENDED ROTOR DYNAMICS

		Turbo-compressor	Generator
Rotor mass		4800 kg	5500 kg
Rotor polar inertia		1.03×10^3 kg-m^2	1.23×10^3 kg-m^2
Total length		4085 mm	4280 mm
Bearing span		3025 mm	2730 mm
Critical speed	1st	557 rpm	515 rpm
	2nd	928 rpm	624 rpm
	3rd	2271 rpm	1043 rpm
	4th	6102 rpm	2162 rpm
	5th	-	6100 rpm
Coupling		Flexible diaphragm coupling	
Shaft orientation		Horizontal	
Rotor balance quality		ISO 1940 balance quality grade G2.5	
Response amplitude		$\leqq$75 μm peak to peak at rated speed	
		$\leqq$125 μm peak to peak at critical speed	
Rotational speed		3600 rpm	
Range of rotational speed		0-110 %	
Radial bearing			
Static load		21.90/26.30 kN	27.63/27.00 kN
Dynamic load		±5.50/±6.60 kN	±6.90/±6.75 kN
Maximum load		27/33 kN	35/34 kN
Journal diameter		248/248 mm	300/300 mm
Length		250/250 mm	274/237 mm
Thrust bearing			
Static load		20.00 kN	-
Dynamic load		±5.00 kN	-
Maximum load		25 kN	-

TABLE XVI-6. MAIN SPECIFICATIONS OF THE TEST FACILITY

Fluid	Helium gas
Flow rate	3.16 kg/s
Maximum/minimum temperature	650/30^{o}C
Maximum/minimum pressure	1.021/0.785 MPa
Turbine power	1.07 MW
Compressor power	0.68 MW
Heater power	2.30 MW

XVI-2.3. Systems for non-electric applications

No information was provided.

XVI-2.4. Plant layout

The GTHTR300 units are installed in an underground level of the reactor building as shown in Fig. XVI-14. This arrangement is effective in protecting the system from the crash impact of an aircraft.

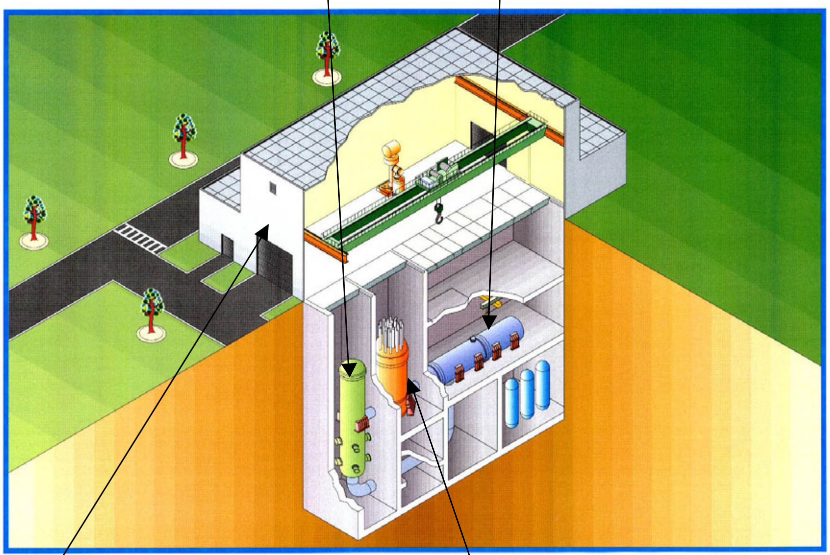


FIG. XVI-14. Arrangement of GTHTR300 unit in a reactor building.

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ANNEX XVII

HIGH TEMPERATURE GAS COOLED REACTOR – PEBBLE BED MODULE (HTR–PM)

INET, China

XVII-1. General information, technical features and operating characteristics

XVII-1.1. Introduction

The High Temperature Gas Cooled Reactor – Pebble Bed Module (HTR–PM) is a modular high temperature gas cooled reactor (HTGR) plant, which is designed by the Institute of Nuclear and New Energy Technology (INET), Tsinghua University of China. The HTR–PM design is now at the stage of design concept optimisation. The current HTR–PM design falls into the category of innovative small and medium sized reactors, featuring a 160 MW electrical output per module. HTR–PM is being promoted as an industrial demonstration plant.

The research and development work on high temperature gas cooled reactors in China started in 1970s [XVII-1-3]. Initially, the work was focused on gas cooled breeders using thorium fuel cycle. The R&D and design work was carried out for a helium cooled thorium breeder of 100·MW output, using spherical fuel elements. The activities included:

- Neutronic and thermo-hydraulic code development and reactor conceptual design.
- Tests and design for graphite reactor core internals, pressure vessels, reactor shutdown components, helium circulators, steam generators and other key components.
- Process system engineering.
- Helium technology.
- Coated particle and fuel element technology as well as spent fuel reprocessing.

Though the thorium reactor project was terminated late 1970s, the results obtained at that time formed a good basis for further technology development.

In the early 1980s, technology development for gas cooled reactors continued in China, and international cooperation was introduced into the R&D. The INET started to cooperate with the Nuclear Research Centre Juelich of Germany to perform design and safety studies of modular HTGR designs. Plenty of work was done for the development of codes for core physics, thermo-hydraulics and safety analysis. During that period, the State Science and Technology Commission of China supported the R&D on gas cooled reactors. The research projects included development of codes, safety analysis, fuel and fuel cycle technology development as well as analysis of HTGR applications. Conceptual designs of modular high temperature gas cooled reactors were developed for different fuelling schemes and for different application purposes. At that time, the conceptual design of a two-zone core modular reactor was proposed, with a central moving zone of graphite pebbles that are fuel-free. This two-zone reactor could generate thermal power of up to 500 MW and the maximum fuel temperature could still be limited to the allowable values in a loss of primary pressure accident, which is the most important safety feature of modular HTGR designs.

The research work on HTGR applications was performed in the early 1980s also. Investigations were performed with the oil and petrochemical industries and with the coal industry to define the application potential of a HTGR on power-heat co-generation basis. The goal was to use the HTGR as a substitute of fossil fuel plants to generate a large amount of process heat to be used in these industries, thus saving a lot of fossil fuels and resulting in less environmental pollution. Under the cooperation with German industries and research centres, feasibility studies were carried out together with the Shengli Oil Field and the Yanshan Petrochemical Complex to use HTGR process heat for the heavy oil recovery in the oil field and for the petrochemical plant.

In 1986, the HTGR R&D projects were included into a national high technology programme. From then on, research and development activities on modular HTGR technology were carried out systematically. The R&D projects covered the following areas, with several research and design institutions in China and international cooperation being involved:

- Modular HTGR design,
- Fuel manufacturing technology,
- Reprocessing technology for thorium fuel cycle,
- Graphite and other structural materials,
- Helium technology and components,
- Pressure vessel,
- Pebble bed flow and fuel handling technology.

The above mentioned R&D resulted in the following major achievements:

- Methodologies and capabilities for the designing of HTGR were established, and the conceptual design of a 10 MW pebble bed test reactor was developed.
- A remarkable progress was made in the manufacturing technology of coated particle pebble bed fuel, including the manufacturing of fuel kernels, coatings, matrix graphite, fuel elements, and in the measurement techniques. Equipment for fuel manufacturing was installed and sample coated particles were manufactured that proved to offer reliable performance.
- A number of test facilities were set up, for example helium test loops and fuel handling test rigs. The experiments were performed to test helium sealing and helium purification technologies and fuel handling components. The research was also performed on fuel burn-up measurement. Progress was achieved in the research on metallic alloy and insulation materials.
- The HTGR development strategies up to the year 2000 were formulated. It was recognized that modular high temperature gas cooled reactors possess the advanced inherent safety features and have a potential to be used for power generation with high efficiency and for process heat applications. Considering the conditions in China, it was proposed, as the first major strategic step of HTGR development, to build a 10 MW pebble bed test reactor by the year 2000. The purpose of building the test reactor is to acquire know-how in the design, construction and operation of modular helium cooled reactors.

In March 1992, the government approved the project for the construction of a 10 MW pebble bed high temperature gas cooled test reactor (referred to as HTR–10). The basic design of HTR–10 was completed in 1994 [XVII-4.], and the construction permit was issued after a licensing review of the preliminary safety analysis report and other relevant application

documents. The construction of HTR–10 formally started with the first concrete being poured in June 1995. In parallel to the engineering design and construction activities, a number of engineering tests were conducted, including tests for the fuel handling system, reactor shutdown system, once-through steam generator, etc. By the end of 1998, the key components such as pressure vessels, steam generator and helium circulator were manufactured, and the installation and commissioning work was started. In parallel with this, spherical fuel elements were manufactured and qualified through irradiation experiments. In December 2000, the HTR–10 was loaded with nuclear fuel and started critical operation.

In January 2003, the reactor reached its full power of 10 MW. HTR–10 is currently under operation, and used to perform tests related to safety. At the same time, technology development for a helium turbine is being performed. It is planned to couple a helium turbine power generation system to HTR–10.

With HTR– 10 being successfully constructed and operated, effort is being made to design an industrial modular HTGR demonstration plant. The project is currently referred to as HTR–PM: High Temperature Reactor – Pebble bed Module. The HTR–PM demonstration is strongly supported by the Chinese government. The electric power industries and nuclear industries are taking active part in the promotion of the demonstration project. The China Huaneng Group (one of the largest power companies in China), the China Nuclear Engineering Group Co. and the Tsinghua University have agreed to cooperate in the project. Other institutions are being involved for siting evaluation and plant design.

XVII-1.2. Applications

HTR–PM, as the first industrial HTGR demonstration plant in China, is currently designed for the purpose of electric power generation.

XVII-1.3. Special features

HTR–PM is a land-based modular nuclear power generating unit. When market requires, more modules can be constructed in series to form a larger nuclear power plant with an appropriate power output. The plant design is aimed at standardization and modularization.

XVII-1.4. Summary of major design and operating characteristics

The HTR–PM design intends to reflect as much as possible the past experience and lessons learned from HTGR development worldwide, and to use the proven methodologies and technologies of the HTR–10 test reactor. Now that there is a real project background, the mature steam turbine cycle has been chosen for power generation in order to avoid too much R&D items and to shorten the overall duration of a demonstration project.

The HTR–PM design has the following remarkable technical features:

- Spherical fuel elements with TRISO coated particles are used, which have a proven capability of fission product retention under 1600ºC in accidents.
- A two-zone core design is adopted, with one central movable column of graphite spheres surrounded by the pebble fuel elements. The purpose of using a two-zone core design is to increase the power output of a single reactor module while maintaining the passive decay heat removal capability.
- Ceramic materials, i.e. graphite and carbon bricks that are resistant to high temperatures surround the reactor core.

- The decay heat in fuel elements is assumed to be dissipated by means of heat conduction and radiation to the outside of the reactor pressure vessel, and then taken away to the ultimate heat sink by water cooling panels on the surface of the primary concrete cell. Therefore, no coolant flow through the reactor core would be necessary for the decay heat removal in loss of coolant flow or loss of pressure accidents. The maximum temperature of fuel in accidents shall be limited to 1600ºC.
- Spherical fuel elements are charged and discharged continuously in a so-called “multi-pass” mode, which means the fuel elements pass through a reactor core several times before reaching the discharge burn-up.
- Two independent reactor shutdown systems are foreseen. Both systems are assumed to be located in the graphite blocks of the side reflector. When called upon, neutron absorber elements are assumed to fall into the designated channels located in the side reflectors, driven by gravity.
- The reactor core and steam generator are housed in two steel pressure vessels, which are connected by a connecting vessel. Inside the connecting vessel, a hot gas duct is mounted. All pressure-retaining components, which comprise the primary pressure boundary, are in touch with the cold helium of the reactor inlet temperature.
- Under an accident with complete loss of pressure, the primary helium inventory is allowed to be released into the atmosphere. Then the helium release channel is assumed to be closed, and the reactor building is vented and serves as the last barrier to radioactivity release.
- Several HTR–PM modules could be built at one site to satisfy the power capacity demand of a utility. Some auxiliary systems and facilities could be shared among the modules.

The design and operating characteristics of HTR–PM are described in a more detailed way below.

Plant installed capacity

- Reactor thermal power: 380 MW.
- Generator power: 160 MW.

Mode of operation

The HTR–PM plant is designed primarily for base load operation, but it can also be operated in a load follow regime.

Targeted availability and load factors

- Availability factor target: 90%.
- Load factor target: 85%.

Summary of major design characteristics

- Type of fuel: UO_2 enriched up to 8.77% in coated particles dispersed in graphite matrix, spherical fuel elements of 6 cm diameter.
- Type of coolant/moderator: Helium/Graphite.
- Type of structural material: Graphite and carbon bricks as core structural materials, steel vessels.
- Core type/characteristic dimensions: Pebble bed core, 4.0 m diameter, and 9.43 m height.

- Vessel type/characteristic dimensions: Carbon steel reactor vessel, 6.7 m inner diameter and 23.84 m overall height.
- Cycle type: Indirect cycle with steam generator and re-heater.
- Number of circuits: Three circuits, with the first one being the primary system cooling the reactor, the second one being the steam turbine circuit and the third one being the condenser cooling water circuit.

Simplified schematic diagram of HTR–PM: see Figure XVII-1.

Neutron-physical characteristics

- The HTR–PM fuel and moderator have strong negative temperature reactivity feedbacks over the whole temperature range and for all operational states. The helium coolant has negligible effect on core neutron-physical characteristics. Since a continuous on-line refuelling is implemented, the equilibrium core exhibits a statistically constant burn-up distribution. To flatten the power distribution profile, a multi-pass refuelling scheme is envisaged.
- Average power density of fuel zone: 4.28·MW/m^3.
- Maximum power density of fuel zone: 9.75·MW/m^3.
- Average output power per fuel ball: 0.79 kW.
- Maximum power per fuel ball: 2.14 kW.
- Power peaking factor: 2.71.

Reactivity control mechanism

- Control rod and absorber ball systems are used for the reactor shutdown purpose. Because of the on-line refuelling capability of a pebble bed core, there is little excess reactivity to be compensated under normal operation, and a movement of control rods is the main reactivity control mechanism. To deal with the reactivity change from the initial critical state to equilibrium core, while keeping fuel operational parameters within certain limits, other additional measures are needed. These measures primarily include an appropriate core loading with fuel elements of different enrichment or with poisons.
- For a safe shutdown of HTR–PM, it is necessary to compensate the following types of reactivity: the reactivity required for power regulation; the reactivity required for the transfer from a hot full power operation to a cold state (27°C); the reactivity changes between the equilibrium xenon for a hot full power state and a zero-xenon state; the reactivity in a most severe accident; the reactivity margin for a safe shutdown. All these reactivity items are compensated by the control rod system and the absorber ball system. Table XVII-1 gives the details of a required reactivity and the reactivity control margin available from the control rod and absorber ball systems (10% of the uncertainty in reactivity calculation was assumed for the control rod and the absorber ball systems).

Cycle type and thermodynamic efficiency

- Indirect steam turbine cycle with reheating.
- Turbine generator thermodynamic efficiency: 42.1%.

Maximum/average discharge burn-up of fuel

- Maximum discharge burn-up: 80,857·MWd/t U.
- Average discharge burn-up: 80,000·MWd/t U.

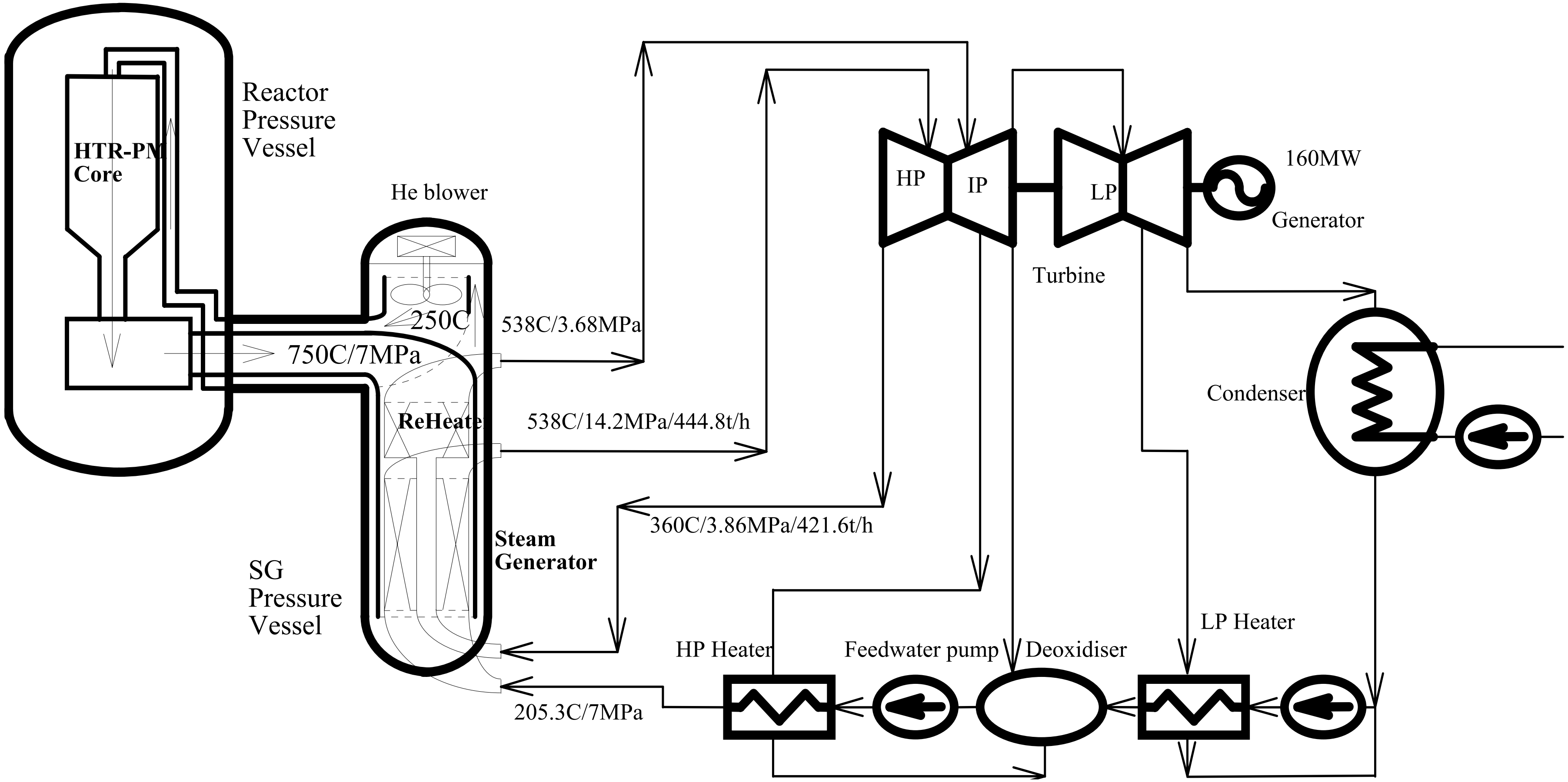


FIG. XVII-1. Simplified schematic diagram of HTR–PM plant.

TABLE XVII-1. REACTIVITY BALANCE FOR COLD SHUTDOWN STATE

ITEMS	REACTIVITY (%Δk/k)
Reactivity margin required for:	
Power regulation	1.00
Transfer from equilibrium xenon hot full power to 27°C	3.28
Transfer from equilibrium xenon to zero xenon under 27°C	4.55
Most severe accident	0.50
Sub-criticality margin	0.70
Total reactivity required	10.03
Available reactivity worth:	
Reactivity of 18 units of control rods	4.78
Reactivity of 18 units of absorber balls	8.77
Reactivity of 18 control rods and 18 absorber balls	10.93
Reactivity of 17 control rods and 18 absorber balls	10.81
Reactivity of 18 control rods and 17 absorber balls	10.44

Fuel lifetime in effective full power days (EFPD): 706.4

Thermo-hydraulic characteristics

Thermo-hydraulic parameters of the primary helium system and the secondary water-steam cycle are listed in Table XVII-2.

TABLE XVII-2. BASIC THERMO-HYDRAULIC PARAMETERS OF HTR–PM

Thermo-hydraulic parameters of the reactor core	
Coolant helium inlet temperature	250°C
Coolant helium outlet temperature	750°C
Coolant helium pressure	7.0 MPa
Coolant helium flow rate	145 kg/s
Temperature limit for fuel elements	1600ºC
Maximum fuel temperature under normal operation	1039ºC
Thermo-hydraulic characteristics of steam cycle	
Flow rate of superheated steam	444.8 t/h
Superheated steam pressure at steam generator outlet	14.2 MPa
Superheated steam temperature at steam generator outlet	538°C
Flow rate of reheated steam	421.6 t/h
Reheated steam pressure at re-heater outlet	3.68 MPa
Reheated steam temperature at re-heater outlet	538°C
Thermo-hydraulic parameters of the reactor core	
Reheated steam pressure at re-heater inlet	3.86 MPa
Feed water temperature at steam generator inlet	205.3°C
Turbine-generator output power	160 MW
Superheated steam pressure at turbine inlet	13.5 MPa
Superheated steam temperature at turbine inlet	535°C
Reheated steam pressure at turbine inlet	3.546 MPa
Reheated steam temperature at turbine inlet	535°C
Rated turbine backpressure	4.9 kPa

Mass balances/flows of fuel and non-fuel materials:

- Materials associated with the operation of HTR-PM primarily include nuclear fuel and graphite, which together form the fuel elements. For a full power operation, the HTR–PM plant discharges 679 spent fuel elements each day on average. This corresponds to the annual consumption of 41.5 tons of graphite and of 1474.6 kg of uranium enriched to 8.77% (roughly equivalent to 24668.6 kg of natural uranium), based on 85% load factor operation. These materials can be obtained domestically or on international markets. China has the capability of uranium enrichment and fuel manufacturing.
- The specific annual consumption is 305 kg/MW(e) for graphite and 181.39·kg/MW(e) for natural uranium.
- Weights of the core structural materials (graphite and carbon bricks) and the carbon steel vessels are given in the Table XVII-3.

TABLE XVII-3. WEIGHTS OF HTR–PM CORE INTERNALS AND VESSELS

Reflector graphite, total weight	352 t
Carbon bricks, total weight	162 t
Weight of core vessel	223.2 t
Weight of reactor pressure vessel	869 t
Weight of steam generator pressure vessel	410 t
Weight of hot gas duct pressure vessel	48.9 t

Design basis lifetime for reactor core, vessel and structures:

- The design basis lifetime for HTR–PM core, vessel and structures is 60 years.
- During the lifetime, side reflector graphite blocks need a one-time replacement.

Economics:

- The HTR–PM plant is designed for a construction period of 48 months including site preparation, construction, commissioning and trial operation. The specific construction cost for HTR–PM is projected not to exceed 1500 US$/kWe, and the generation cost is targeted to be about 4.5 cents/kW·h. The specific O&M costs and fuel costs are estimated at 0.76 cents/ kW·h and 1.09 cents/ kW·h respectively.

XVII-1.5. Outline of fuel cycle options

The current HTR–PM design uses low enriched uranium (LEU) as nuclear fuel. The uranium is enriched up to 8.77% of fissile material (^{235}U) for fresh fuel elements. The currently foreseen fuel cycle is a Once-Through-Then-Out (OTTO) cycle. In reactor operation, the spherical fuel elements go through the reactor core in a multi-pass mode before they reach the discharge burn-up and are discharged from the core into storage tanks. The average discharge burn-up is 80 000 MWd/t U.

As it was already noted, the currently foreseen fuel cycle is the once through cycle. The spent fuel can eventually be reprocessed in case fuel cycle strategies would change to recover the uranium and plutonium fissile material from the spent fuel. There has been much development for the reprocessing technology. The first steps would be to remove the matrix graphite in the fuel elements and the pyro-carbon and silicon carbon coating layers. The remaining spent fuel materials can be reprocessed using the existing technologies applied to fuel of present-day water cooled reactors.

A proposed option for spent fuel storage and disposal is to store spent fuel at the plant site for a certain period, then to store it at a centralized storage before moving it to a final disposal. For the moment, national strategies for the management of spent fuel elements with respect to storage and final disposal are not formulated yet. These strategies and their implementation will depend strongly on the scale of eventual HTR-PM deployment. For a first demonstration plant, the spent fuel elements corresponding to the whole plant lifetime are planned to be stored in an interim repository at the site.

XVII-1.6. Technical features and technological approaches that are definitive for HTR–PM performance in particular areas

XVII-1.6.1. Economics and maintainability

The Chinese economy has been developing very fast in recent decades. It is foreseen that the Chinese national economy will continue to grow rapidly in the coming 20-50 years. Meeting the demand on electric power will be a big challenge. China will also face strong pressure coming from the environmental constraints and the limitation of the greenhouse gas emissions. Therefore, large growth of nuclear capacities is expected in the coming decades in China. HTR–PM is foreseen to be a part of the overall nuclear capacity. One of the important features of HTR–PM is its capability to provide high temperature heat. Therefore, one particular application would be its introduction into the process heat market. One attractive application is to use the HTR–PM nuclear heat to produce hydrogen, when hydrogen would be used as an essential fuel. HTR–PM is also characterized by small and medium power output per module and by excellent inherent safety features. The initial capital investment for the construction of such modules in series is projected to be relatively low. These features could also be very attractive to some developing countries.

HTR–PM has the following characteristics which contribute to reducing the plant capital costs:

- Modular and standardized design, construction in modules and series, and short construction period,
- Simplified design of safety systems, broad implementation of inherent safety features,
- Adoption of the mature and conventional power conversion technologies,
- High coolant temperatures that result in high power conversion efficiency.

The HTR–PM technology foresees nuclear power plants with more modules on a single site. The plants will have a high degree of automation. The number of operation personnel is expected to be reasonably minimized. Maintenance staff could be shared between the modules. Some auxiliary systems and facilities could also be shared between the modules. HTR–PM will have a on-line refuelling, which provides for high availability factors. Also, the excellent safety features of HTR–PM could allow some simplifications in the plant safety management, such as reduction of the emergency panning measures. All these factors could contribute to reducing the plant operation and maintenance costs.

The currently targeted discharge burn-up of the LEU spherical fuel elements is 80 000·MWd/t U. Spherical fuel elements with coated particles have the potential of reaching a much higher burn-up, as it has been demonstrated in previous high temperature gas cooled reactors. It is expected that high fuel burn-up will contribute to the reduction of fuel cycle costs.

XVII-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

The currently foreseen HTR–PM fuel cycle is a LEU once-through cycle. In comparison to the existing water cooled reactor technologies, it uses the uranium resources more efficiently. If necessary, spent fuel can be reprocessed to recover the remaining fissile materials in order to ensure a better degree of fissile material utilization. It also could be noted that, as a helium cooled graphite moderated system, HTR–PM is flexible in the selection of fuel cycle technologies. It also could operate in a thorium-uranium fuel cycle. Considering a large amount of thorium resources in China, the HTR–PM technology has a good potential in terms of sustainability.

The HTR–PM fuel elements possess a good capability to confine radioactive materials generated from the fission process. The graphite-helium system does not have many activation problems. The primary helium coolant has low radioactivity level and is constantly purified. During normal operation, dose rates to the operational staff and radioactive releases to the environment are very low. The amount of liquid and solid radioactive wastes generated during plant operation is extremely small and these wastes will be duly managed. Therefore, the features of the HTR–PM technology with respect to adverse environmental impacts could be remarkably favourable.

XVII-1.6.3. Safety and reliability

Safety concept and design philosophy

The safety design philosophy of HTR–PM is to realize the required high level of safety and, at the same time, to simplify the design of the systems required only for safety purposes, to the extent possible. The emergency measures outside the plant boundary should be made technically not necessary or reduced to a minimum level. The HTR–PM safety design is to a large degree based on the inherent and/or passive safety features, while still adhering to the defence-in-depth principles. The following three features characterize the basic safety concept of HTR–PM:

- Radioactive materials are confined through the implementation of multiple barriers with a strong emphasis on fuel elements, especially in accidents. Fuel elements with coated particles serve as the first barrier. Every fuel kernel of about 0.5 mm diameter is coated with three layers of pyro-carbon and one silicon carbon (SiC) layer. A large number of coated particles are dispersed in the graphite matrix of 5 cm diameter to form the fuel-containing part of a fuel element, which in turn is protected by a 0.5 cm thick fuel-free graphite layer. The fuel elements used for HTR–PM were demonstrated to be capable of confining fission products within the coated particles under temperatures of ~1600ºC that are not expected for any plausible accident scenario. The second barrier is the primary pressure boundary, which consists of a pressure vessel that hosts units of the primary components. The third barrier is a reactor building and some additional auxiliary buildings, which house the primary helium-containing components.
- The decay heat is removed inherently under accident conditions. In case of an accident, the primary helium circulator is stopped. Because of a low power density and a large heat capacity of the graphite structures, the decay heat in fuel elements will dissipate to the outside of a reactor pressure vessel by means of heat conduction and radiation within the core internal structures, without leading to unacceptable temperatures of fuel.

- The overall negative temperature feedback is assured under all conditions. The reactor nuclear design assures that the temperature reactivity coefficients of fuel and moderator are always negative under all operating and accident conditions. Together with the protection action of stopping the primary helium blower, this will lead to a reactor automatic shutdown in accident cases.

Provisions for simplicity and robustness of the design

The HTR–PM design makes a good use of the above safety features. Because the reactor safety strongly relies on the inherent safety features, the engineered safety systems of the plant are kept at a minimum, so that the plant safety and safety management are simplified to a high extent. Also, the safety design concept is clear and transparent. When accidents occur, a very limited number of reactor protection actions will be called upon by the protection system. No or very limited actions through any systems or human interventions are required after the limited reactor protection actions are executed. The limited reactor protection includes actions to trip the reactor and the helium circulator, and to isolate the primary and secondary systems. When there is a large leak or rupture of steam generator heat exchange tubes, the design provides for a water-discharge system to minimize the amount of water ingress into the reactor core.

Another feature of the HTR–PM design is a long period of accident progression, which is due to a large heat capacity of the fuel elements and the reactor internal structures made of graphite. After the coolant is completely lost, it will take days for the fuel elements to reach their maximum allowed temperatures.

Structure of the defence-in-depth

The safety design and operation of HTR–PM follows the defence-in-depth principles, which are well formulated in the IAEA Safety Standards. The first three levels of the defence-in-depth apply in full to the HTR–PM design and operation, namely, Level 1 – Prevention of abnormal operation and failure, Level 2 – Control of abnormal operation and detection of failure, and Level 3 – Control of accidents within the design basis. For Level 4 of the defence-in-depth (mitigation of radiological consequences of significant release of radioactive materials), the mitigation measures are reduced to a minimum, as accident consequences are limited by an extremely high level of confidence at the first three levels of the defence-in-depth. For Level 5 of the defence-in-depth (off-site emergency responses), the HTR–PM design is aimed at making such off-site emergency response measures not necessary on a technical basis.

Design basis accidents and beyond design basis accidents

The accidents classification and safety analysis for the HTR–PM design will be performed according to the provisions of the safety standards, and the design, licensing and operation experience of the HTR–10 test reactor will be taken into account. For HTR–PM, the most significant design basis accident is a complete loss of primary helium coolant. This accident has already been analysed, and no additional core cooling systems are provided for by the design to cope with it. The decay heat coming from fuel elements will dissipate to the outside of the pressure vessel primarily by means of heat conduction and radiation through the reactor core internals. The maximum temperature of fuel during this accident is the highest among all accidents. Therefore, this scenario is most critical and challenging for the HTR–PM design. Figure XVII-2 shows the evolution of a maximum fuel element temperature throughout the scenario of such accident. It can be seen that the fuel temperature reaches its maximum of 1470°C about three days into the accident progression, and that the maximum temperature reached is still considerably below the fuel temperature limit of 1600°C. For modular HTGR designs, the analysis of accidents with loss of primary coolant is definitive for making a

judgement on the concept feasibility in terms of its reliance upon inherent safety features. Therefore, it should be performed even at a conceptual design stage. Other design basis accidents and the scenarios for beyond design accidents for HTR–PM have not been defined and analyzed so far.

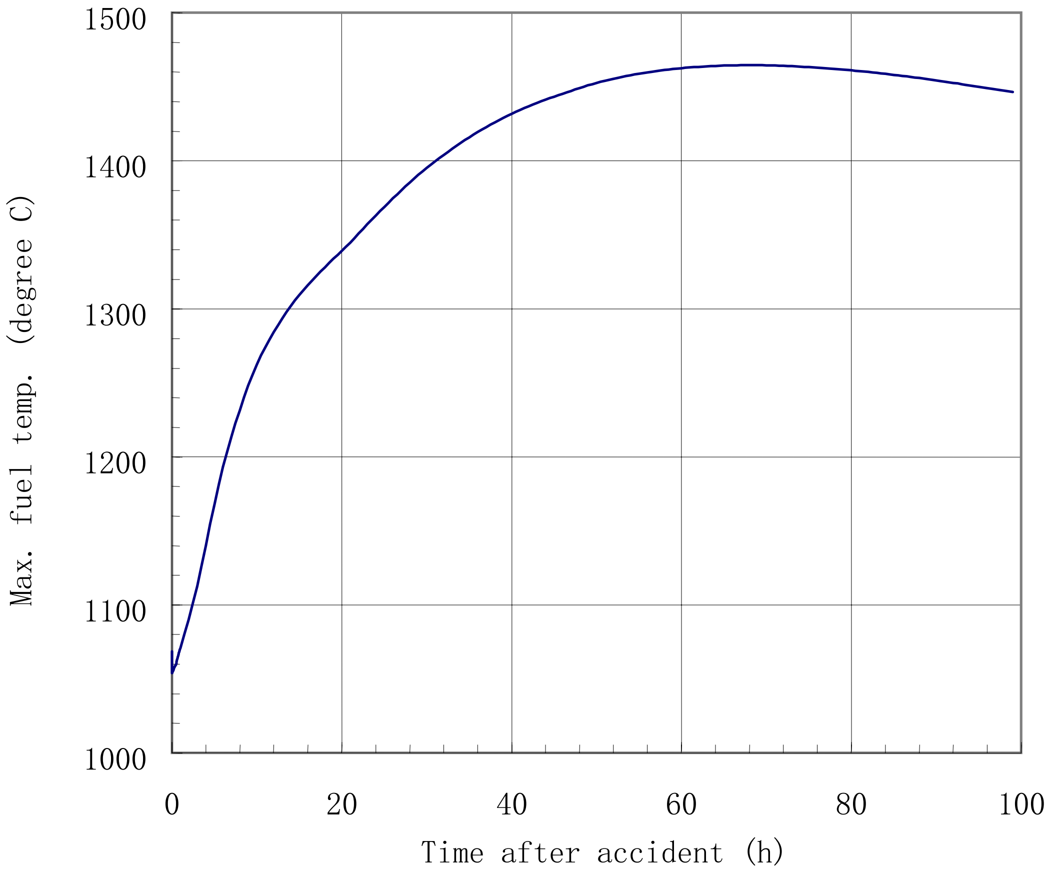


FIG. XVII-2. Maximum fuel temperature in loss of coolant accident.

XVII-1.6.4. Proliferation resistance

Some of the important technical features of HTR–PM, which may reduce the attractiveness of its spent nuclear fuel for clandestine nuclear weapon programmes, are as follows:

- The volume of fresh and spent fuel is rather large, and its inspection and verification could be rather easy.
- The burn-up of the discharged fuel is as high as 80 000·MWd/t U, which means that smaller amount of the fissile materials is present in spent fuel and that these materials are less attractive for weapon purposes.
- The currently adopted fuel cycle is once through and based on low enrichment uranium; no reprocessing of spent fuel is foreseen.
- Sophisticated and costly reprocessing techniques would be required to get any fissile materials from the HTR-PM spent fuel elements.

XVII-1.6.5. Technical features and technological approaches used to facilitate physical protection of HTR–PM

The design of HTR–PM plant is conceived in such a way that its nuclear safety does not rely strongly on the peripheral engineered safety systems. The radioactive materials in reactor core will remain confined in fuel elements as long as the reactor structures remain physically protected. The reactor itself is housed in a very thick concrete cavity located inside the reactor building. The reactor building and the primary concrete cavity form robust barriers against sabotage. The intrinsic safety features of the reactor also make it less sensitive to conceivable internal sabotage actions. Such robust technical features make the HTR–PM plant much less vulnerable to any sabotage. Notwithstanding this, the routine technical measures as prescribed by applicable standards are foreseen to secure physical protection of HTR–PM.

XVII-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of HTR–PM

Smaller power output of a single HTR–PM module and an option of incremental capacity increase could facilitate the attraction of investments, which could be smaller or spread in time for this case. Simple and transparent philosophy of the HTR–PM safety design could make it easier to address public acceptance issues.

XVII-1.8. Enabling technologies relevant to HTR–PM and their development status

HTR–PM is a graphite-moderated, helium-cooled high temperature reactor with spherical fuel elements. The innovative aspects of this technology are primarily related to the way in which the reactor as such is designed, e.g. the degree of reliance on inherent safety features and extent to which the release of a large amount of radioactivity due to fuel heating is ruled out. The feasibility of a high temperature gas cooled reactor has already been demonstrated. In the past, similar test and prototype reactors have been successfully constructed and operated in several countries around the world. Many enabling technologies for HTR–PM have been demonstrated in the past projects. The HTR–PM design incorporates the past experience and lessons learned from the HTGR development with respect to design methodology, safety analysis, component and system technologies, etc.

The basic enabling technologies of HTR–PM are:

- **Large scale manufacturing of qualified fuel elements.** China has successfully fabricated spherical fuel elements with coated particle fuel. Good performance of such fuel elements has been so far demonstrated in irradiation experiments and in the operation of the HTR–10 test reactor. The current fuel manufacturing capacity has been established for the needs of the HTR–10. To meet the fuel demand of HTR–PM, larger fuel manufacturing capacity has to be provided. The design of a larger-scale fuel plant is currently underway.
- **Safety design and licensing requirements.** Safety design criteria and licensing requirements need to be established for the implementation of the HTR–PM project. Such requirements could be established taking into consideration the experience of the HTR–10 test reactor. The discussions are being held with the regulatory body in this respect.
- **Test reactor facility.** In China, a 10 MW(th) test pebble bed reactor has been constructed and is under operation currently. Through this test reactor project China has gained an experience in the technological aspects of pebble bed reactors. Also, the technologies have been developed for fuel handling system, helium circulator, reactor

shut down systems, etc. The instrumentation and digital control and protection technologies were developed as well. An experience has been gained in design, licensing, construction, commissioning and operation of a pebble bed high temperature gas cooled reactor.

- **Inherent safety features.** HTR–PM incorporates inherent safety features, such as the intrinsic reactor shutdown under temperature increase and passive decay heat removal. Such features make it possible to simplify the design of safety systems, but need a demonstration to be rated as sound and convincing. Safety demonstration experiments are being conducted within the HTR–10 test reactor facility.
- **Components technology.** In a sense, HTR–PM is a large upscale of the HTR–10 test reactor. This is particularly true for some mechanical equipment, such as pressure vessels, steam generator and helium circulator. For some of these components, consultations with the manufacturing industries are necessary to investigate their commercial availability on the market. The steel pressure vessels used in the HTR–PM plant are rather large in size. Depending on site-specific options for transportation, some construction work at the site might be eventually needed for large sized vessels. The discussions and consultations with the construction companies are underway.
- **Power conversion system.** HTR–PM uses conventional steam turbine generator systems for power conversion. These are totally mature technologies.

XVII-1.9. Status of R&D and planned schedule

The HTR–10 test reactor is a comprehensive and successful research and development project, through which many technological achievements have been gained. This test reactor is operated currently, and the operational experience is being gained. Different from the HTR–10, HTR–PM is a big step forward in terms of power output. Some components, such as helium circulator or fuel handling equipment, are just scaled up. Certain research and development activities on some special technologies or equipment are needed. The R&D programmes on technology development for the HTR–PM systems and components are under planning currently. They will be carried out in the next couple of years, before the basic design is completed.

With the HTR–10 operation and experiments serving as a supporting technological test bed, the project of the HTR–PM commercial demonstration plant is being promoted. The utilities and nuclear industry partners have confirmed their intention to participate in the project. The government ministries are supporting this effort too. Currently, a siting evaluation is being performed for the first demonstration plant and for the follow-up units. The HTR–PM design is now at the conceptual stage, and the activities on design optimisation are under way. All participants of the project undertake great efforts, and the target is to have the HTR–PM demonstration plant constructed around 2010.

XVII-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

The HTR–PM design incorporates several important innovative approaches, most of them being related to a strong reliance upon the inherent safety features. No emergency core cooling systems and associated supporting systems are foreseen in the design. The reactor core is very high with respect to the core diameter, which is a special feature provided to optimise the power output versus safety. The primary systems are arranged in a so-called "side-by-side" way, which means that the reactor and the steam generator are housed in two

separate steel vessels that are connected by a horizontal connecting vessel. A demonstration plant is necessary to demonstrate the key technical and safety features mentioned above. The demonstration is in particular necessary to prove the long-term operability, maintainability and, last but not least, economic viability of the HTR–PM plant.

XVII-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

There is currently a lot of interest worldwide in the development and deployment of modular high temperature gas cooled reactors. Besides in China, similar design and technology development activities are underway in France, Japan, Russia, South Africa, the USA, etc. [XVII-5-7]. Main differences between the programmes are related to fuel technology (pebble fuel versus prismatic fuel, uranium versus plutonium) and to power conversion technology (steam turbine versus gas turbine).

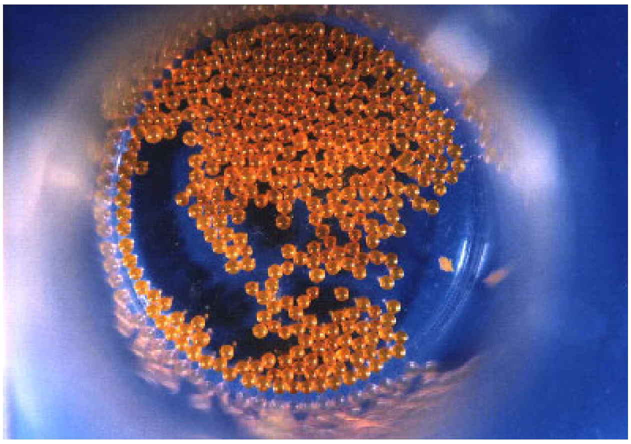
China has been participating actively in the international cooperation programmes carried out under the umbrella of the International Atomic Energy Agency, such as a series of coordinated research projects. China also maintains regular information exchange with other countries implementing HTGR development programmes.

XVII-2. Design description and data for HTR–PM

XVII-2.1. Description of the nuclear systems

Reactor core and fuel design

The fuel elements are of spherical shape. Each fuel element contains 7 g of heavy metal. The enrichment by 235U is 8.77%. The uranium kernels of about 0.5 mm diameter are coated by three layers of pyro-carbon and by a single layer of silicon carbon. Coated fuel particles are dispersed in a graphite matrix of 5 cm diameter. Surrounding the fuel-containing graphite matrix is a 5 mm thick graphite layer. Figure XVII-3 shows the cross-section of fuel.



(a) UO_2 Kernels

(b) Spherical fuel elements

FIG. XVII-3. HTR–PM fuel elements.

The key design parameters of a fuel element are given below:

Fuel Element:

Diameter of a ball	6.0 cm
Diameter of fuel zone	5.0 cm
Density of graphite in matrix and in outer shell	1.73 g/cm^3
Heavy metal (uranium) load per ball	7.0 g
Enrichment by ^{235}U (weight %)	8.77

Coated particle:

Fuel kernel	
Radius (mm)	0.25
UO_2 density (g/cm^3)	10.4
Coatings	
Coating layer materials (starting from kernel)	PyC/PyC/SiC/PyC
Coating layer thickness (mm)	0.09/0.04/0.035/0.04
Coating layer density (g/cm^3)	1.1/1.9/3.18/1.9

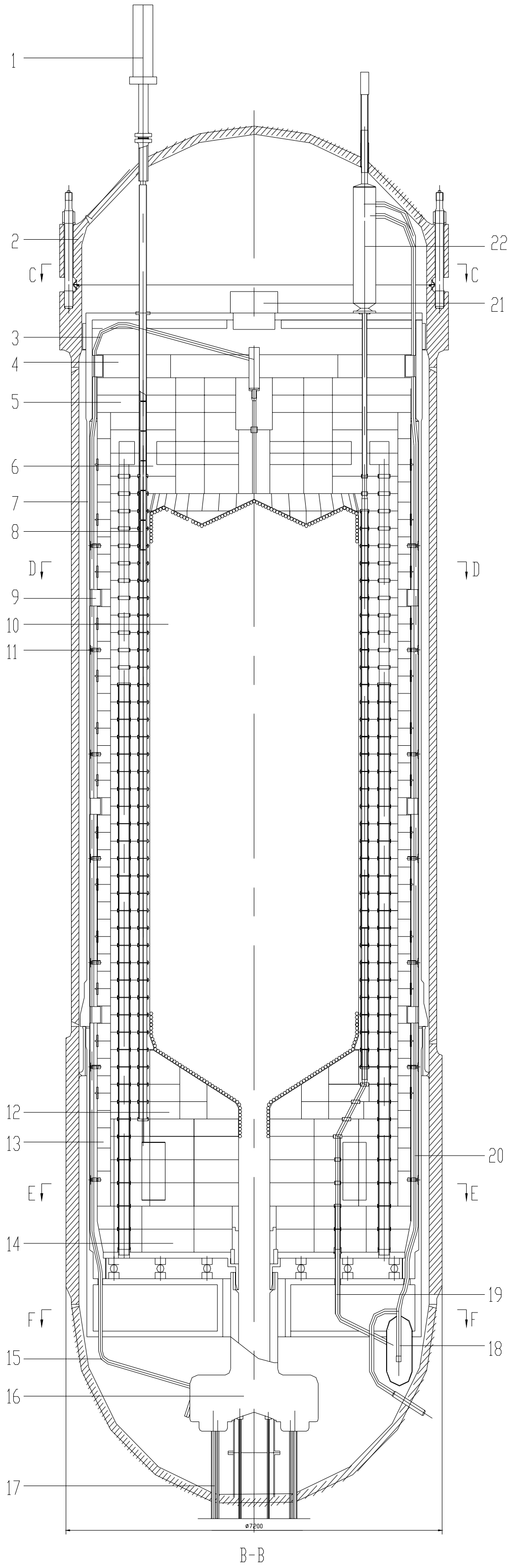
In its current design, the HTR–PM reactor has a pebble bed core consisting of two zones. The central zone is a moveable column of the graphite balls, which have the same dimensions as fuel element balls. The diameter of this central graphite ball column is 2.0 m. The graphite balls are surrounded by the zone of fuel element balls, which has the outer diameter of 4.0 m. The effective height of active core part is 9.43 m. The effective core volume is 118.5 m^3. In equilibrium state, the reactor core contains 479 358 fuel elements and 159 786 dummy graphite balls.

There are top, side and bottom reflectors that are composed of the graphite blocks, which are arranged in layers. In the circumferential direction, every layer of a graphite reflector consists of 36 graphite blocks. The graphite blocks of the side reflectors include a number of channels for the reactor shutdown systems and for helium flow. The bottom reflector upper surface has a conical shape to facilitate the pebble flow. Inside the bottom reflector, channels are designed for the flow of hot helium. The hot helium chamber is designed in the bottom reflector area. In this chamber hot helium of different outlet temperatures is agitated and then directed to the hot gas duct, from which it flows to the steam generator. A fuel discharge tube of 600 mm diameter is located in the centre of the bottom reflector.

Two reactor shutdown systems are provided, namely the control rod system and the small absorber ball system. Both shutdown systems use boron carbide as neutron absorbing material. The current design assumes 18 control rods and 18 small absorber ball units. The absorbers of both systems fall into the side reflector channels when called-upon.

Figures XVII-4 and XVII-5 give cross-sectional views of the reactor core.

The primary helium coolant operates at 7.0 MPa. The mass flow rate is 145 kg/s. Helium coolant enters the reactor from the bottom area inside the pressure vessel with an inlet temperature of 250ºC. Then it flows upward through the cannels in side reflector and reaches the top reflector level where the flow direction gets reversed, after which helium flow enters the pebble bed from the top. Bypass flows are organized in the fuel discharge tubes to cool fuel elements that are there, and in the control rod channels to cool control rods. Helium is heated up in the active reactor core and then mixed reaching the average outlet temperature of 750ºC, and then it flows to the steam generator.



1. Control rod drives
2. Reactor pressure vessel
3. Fuel charging tubes
4. Steel plates
5. Thermal insulation
6. Top reflector
7. Core vessel
8. Control rod channels
9. Supports
10. Reactor core
11. Supports
12. Bottom reflector
13. Side reflector
14. Thermal insulation
15.Fuel loading tube
16. Reducer
17. Discharging tubes
18. Bottom storage tank for absorber balls
19. Transport tubes for absorber balls
20. Gas carrier tube for absorber balls
21. Shielding plug
22. Top storage tank for absorber balls

FIG. XVII-4. Vertical cross-section of HTR–PM reactor.

Main heat transport system

Reactor cooling systems are the main cooling system and the decay heat removal system (also called cavity cooling system). Steam generator and helium circulator are the basic components of the main cooling system; they are used during reactor operation and for a

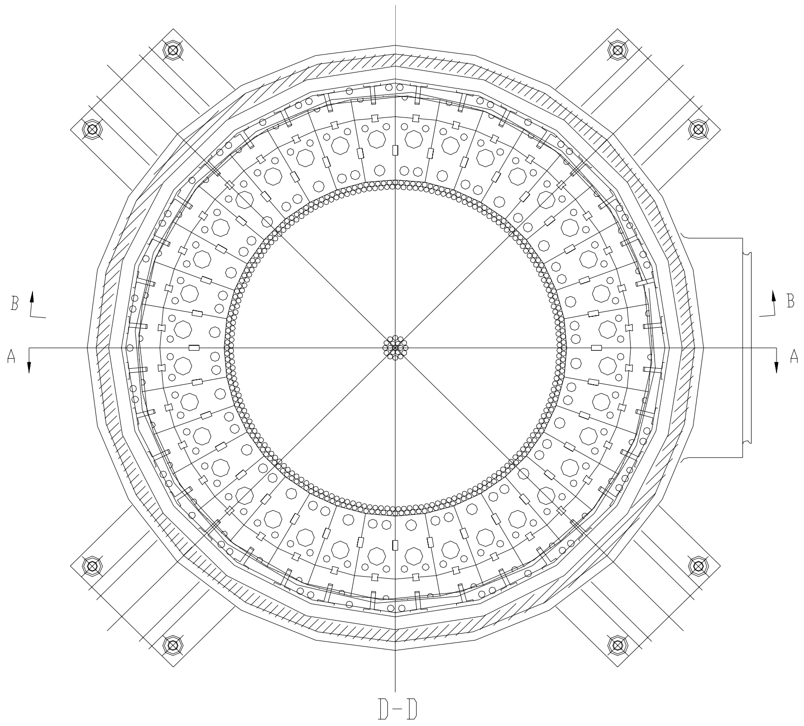


FIG. XVII-5. Horizontal cross-section of HTR–PM reactor.

normal reactor shutdown. The decay heat removal system consists of a set of water cooled panels mounted on a concrete wall that surrounds the reactor pressure vessel. The system is designed with 2×100% capacity. Figure XVII-6 shows the heat removal paths of the HTR–PM plant under normal operation and in accidents.

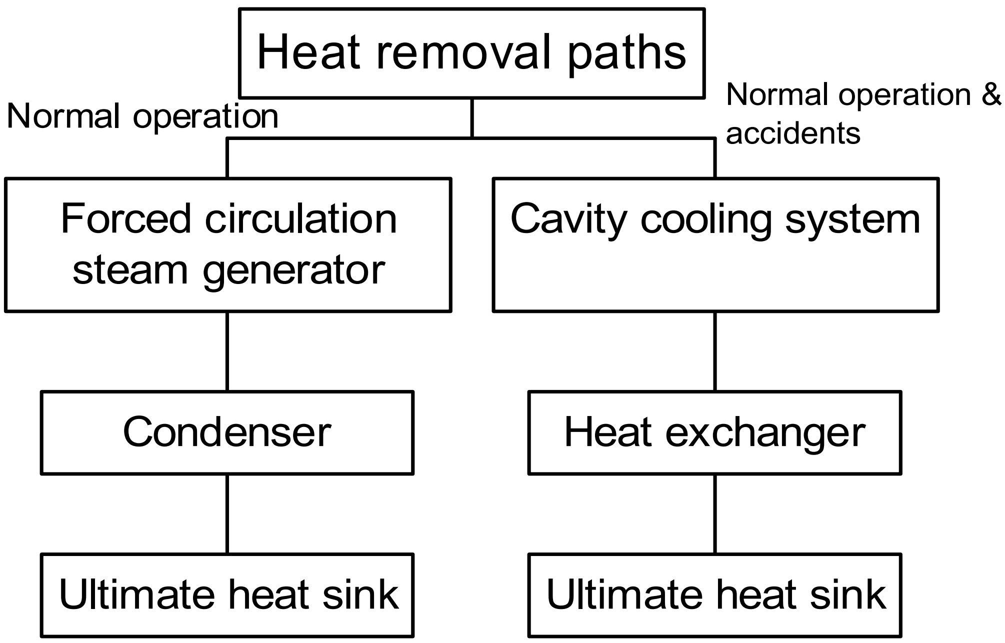


FIG. XVII-6. Simplified scheme of heat removal paths in normal operation and in accidents.

Figure XVII-7 shows the arrangement of the reactor building and the turbine generator building. Figure XVII-8 presents the plant layout.

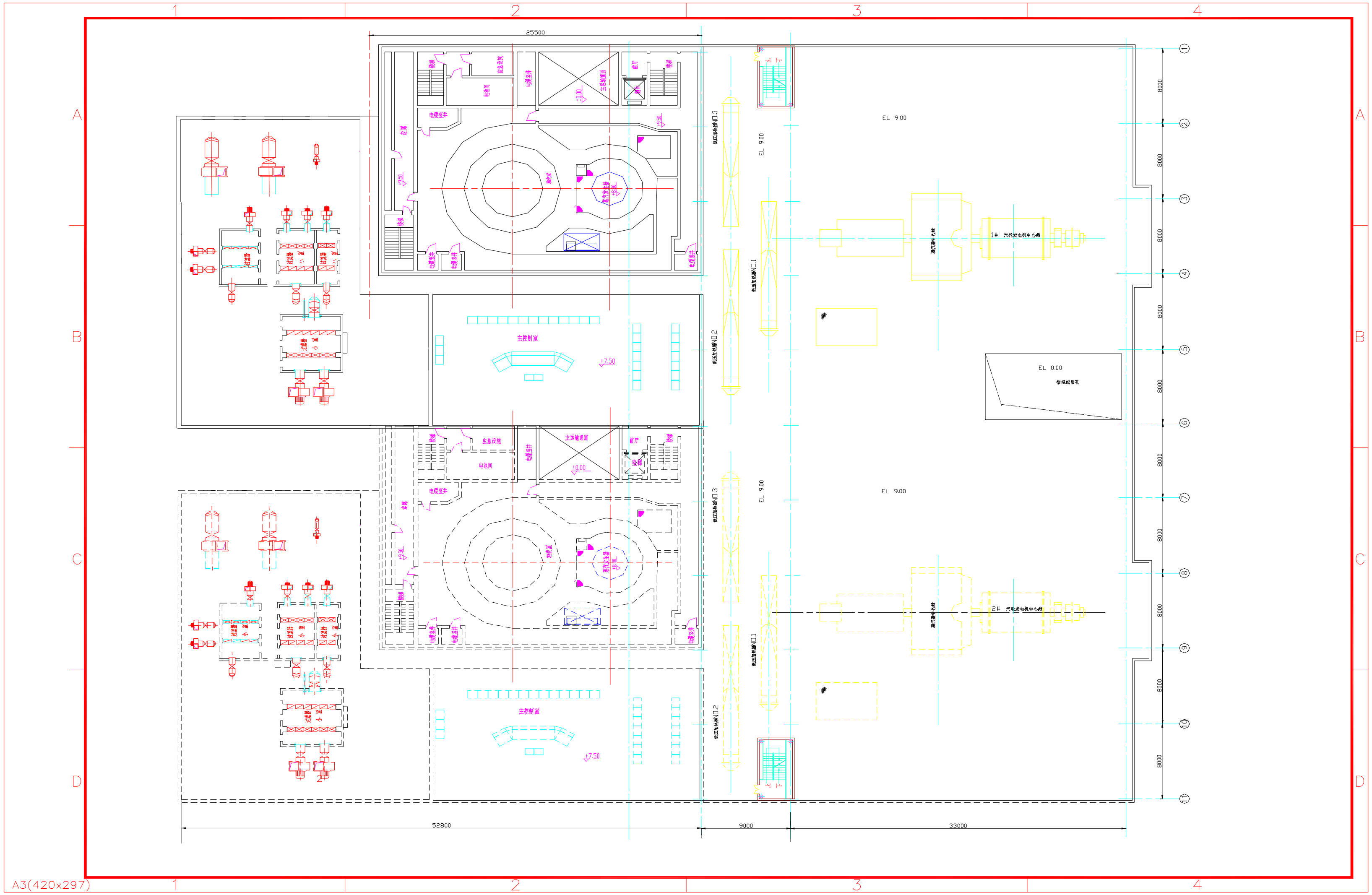


FIG. XVII-7. HTR–PM reactor building and turbine generator building (two modules).

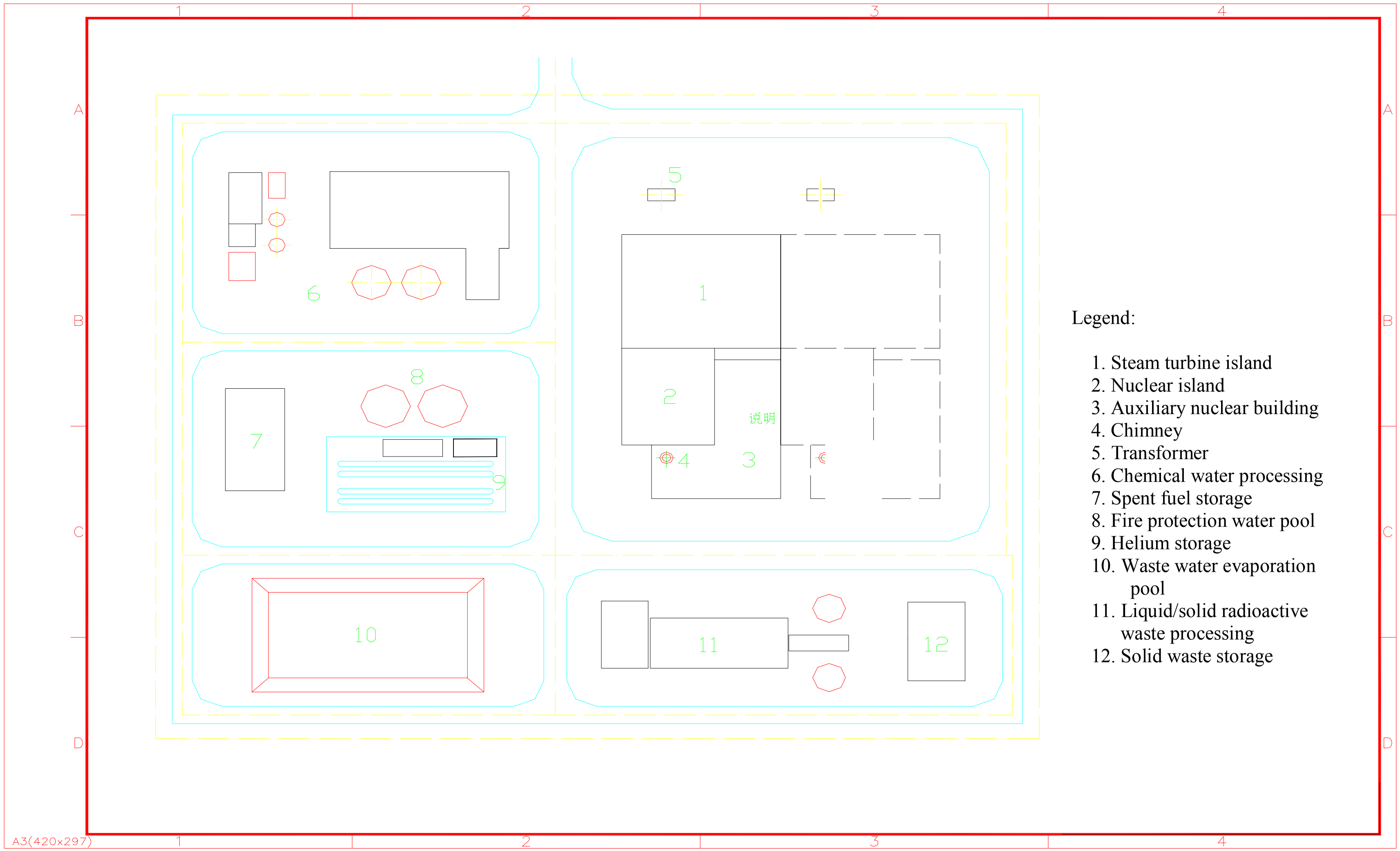


FIG. XVII-8. Layout of the HTR–PM plant (two modules).

XVII-2.2. Description of the turbine generator plant and systems

Commercially available steam turbine generator equipment is used in the turbine generator plant.

XVII-2.3. Systems for non-electric applications

The current HTR-PM design makes no provision for non-electrical applications.

XVII-2.4. Plant layout

The philosophy governing plant layout is as follows:

- At the moment, the detailed plant layout is being developed for a single-module, i.e., demonstration plant. This layout reserves an area for the second module.
- The buildings include: reactor building, turbine-generator building, helium storage and supply building, spent fuel storage building, waste treatment building, waste storage building and evaporation pool, fire protection water supply tank.
- Auxiliary buildings are considered for two HTR–PM modular reactors.

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ANNEX XVIII

FAPIG HIGH TEMPERATURE GAS COOLED REACTOR, (FAPIG-HTGR)

FUJI Electric, Japan

SHORT DESCRIPTION

XVIII-1. Basic summary

A basic summary of the FAPIG-HTGR concept is given in Table XVIII-1.

TABLE XVIII-1. BASIC SUMMARY OF FAPIG-HTGR CONCEPT

Full and abbreviated name	FAPIG-HTGR: First Atomic Power Industry Group - High Temperature Gas Cooled Reactor
Principal stakeholders	Leading companies of the First Atomic Power Industry Group (FAPIG), Japan: Fuji Electric Systems, Kawasaki Plant Systems, LTD., Shimizu Corporation
Reactor type	Pebble bed fuel; helium coolant; graphite moderator
NPP applications	Electricity generation; applicable to other high temperature heat applications with some system modifications
Capacity	100 MW(e) / 220 MW(th)
Reactor style; Maximum temperature/ Pressure	Modular; 900°C / 6MPa
Sources of more detailed information	[XVIII-1]

XVIII-2. Major design and operating characteristics

Table XVIII-2 summarizes major design and operating characteristics of the FAPIG-HTGR. The layout of a FAPIG-HTGR plant is shown in Fig. XVIII-1.

TABLE XVIII-2. MAJOR DESIGN AND OPERATING CHARACTERISTICS OF FAPIG-HTGR

Design targets	1) Electric power 100 MW(e), 2) Construction cost ～1200 US$/kW(e) 3) No evacuation in hypothetical accidents without leak-tight reactor containment vessel

TABLE XVIII-2 (continued)

Design features	1) Core power increase: (1.1) 2-region pebble bed core to decrease maximum temperature; (1.2) High performance reactor cavity cooling system. 2) Cost reduction: (2.1) Power conversion system: low cost, compact and with high thermal efficiency - High speed gas turbine system with vertical single shaft rotor; (2.2) Mn-Mo steel reactor pressure vessel. 3) Enhanced safety: (3.1) Simple 3-vessel system; (3.2) Decay heat removal by passive system utilizing natural circulation of atmospheric air.
Plant layout	The plant layout is shown in Fig. XVII-1.
Operation mode	Basic load and load following
Availability	95% (target)
Fuel	1) Coated particle fuel (TRISO) with about 1mm diameter; 2) 8 to 10 weight % enriched UO_2 3) Burn-up of 80 – 100 GW·d/t U 4) Allowable temperature limit is 1600°C
Core	1) 2-region pebble bed core consists of graphite fuel balls with a diameter of 6 cm (fuel balls are circulated in the outer region first, then in the inner region.) 2) Diameter and height of the core is 3 m and 11 m, respectively 3) Inlet and outlet temperature of the core is 500°C and 900°C.
Reactor pressure vessel	Mn-Mo steel vessel with the inner diameter of 5.9 m
Reactivity control system	1) 6 control rods used for hot shutdown; 2) 18 KLAK systems used for cold shutdown and as a secondary shutdown system 3) All systems are inserted in reflector region. 4) No burnable poisons.
Thermal cycle	Direct gas turbine cycle (recuperated Brayton cycle).
Heat removal systems	1) Gas-turbine system is used for normal core heat removal and transient decay heat removal. 2) Cavity cooling system is the only safety-related system; it removes reactor heat through reactor pressure vessel passively, by naturally circulating atmospheric air. 3) Auxiliary cooling system is a non-safety grade system used to shorten cooling time under normal and accident transient conditions.

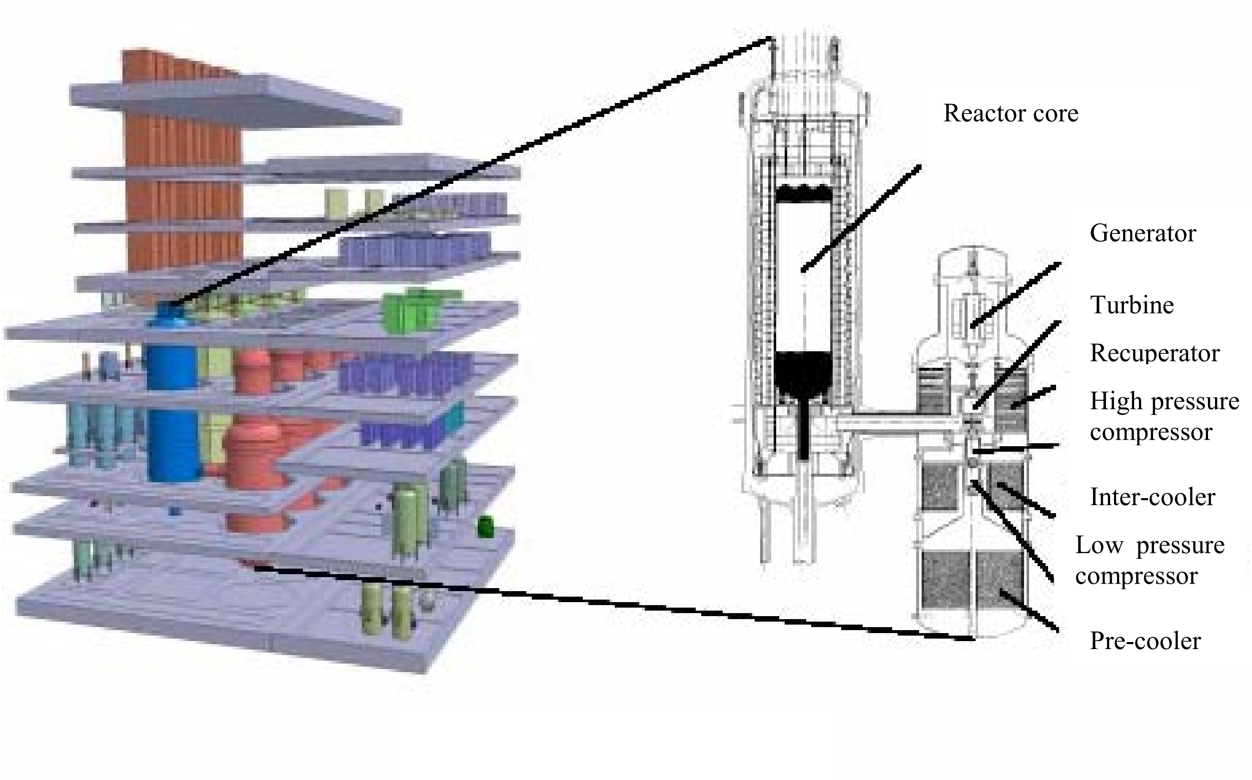

Fig. XVIII-1. Schematic view of the FAPIG-HTGR 4-module plant.

XVIII-3. List of enabling technologies and status of their development

The FAPIG-HTGR is at a pre-conceptual design stage currently. A list of the enabling technologies relevant for the FAPIG-HTGR is presented in Table XVIII-3.

TABLE XVIII.3. LIST OF ENABLING TECHNOLOGIES FOR FAPIG-HTGR

DESIGN OBJECTIVE	ENABLING TECHNOLOGY	STATUS OF DEVELOPMENT
Core power increase	2-Region pebble bed core:	
	-Coated particle fuel (CPF	CPF integrity has been verified in Germany and Japan. Additional burn-up irradiation experiment is required over 100 GW d/t.
	-Spherical fuel elements	Integrity has been verified in Germany.
	-2-Region fuel circulation	Cold experiment was already conducted at several institutes to investigate mixing of fuel balls between 2 regions.
	High performance reactor cavity cooling system (RCCS)	Small-scale experiment has been conducted.

DESIGN OBJECTIVE	ENABLING TECHNOLOGY	STATUS OF DEVELOPMENT
Compact power conversion system with low cost and high thermal efficiency	High-speed gas turbine system with vertical single shaft rotor	Design feasibility study has been conducted. R&D is required.
	Low cost and compact frequency converter	R&D is required. R&D plan has been compiled.
	High-speed large power generator	R&D is required. R&D plan has been compiled.
Low cost reactor pressure vessel	Mn-Mo steel reactor pressure vessel (RPV)	No R&D required. Based on LWR technology.
Enhanced safety	Passive decay heat removal with natural circulation of atmospheric air	Small-scale experiment has been conducted.
	3-vessel system	No R&D is required.

REFERENCE

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ANNEX XIX

ADVANCED ATOMIC COGENERATOR FOR INDUSTRIAL APPLICATIONS (ACACIA)

NRG, Netherlands

XIX-1. General information, technical features and operating characteristics

XIX-1.1. Introduction

Until now, nuclear power has been successful in the market of large-scale electricity generation. Other markets, like heat and power cogeneration or distributed electricity generation still await penetration by the uranium-based energy source.

For these applications, the power level required per location is much smaller than for the existing nuclear plants however, traditionally, nuclear plants need economy of scale to become economically viable. Therefore, small nuclear plants must use other mechanisms to compete in a market. One way is modularization: multiple small, identical units are built on a site instead of one large unit. This has several advantages: capital cost per unit (and therefore financial risk) is low and therefore decisions to build further units are less fraught with risk and complication. Series production of the components brings cost reduction. Maintenance outages can be spread over the year and will not fully interrupt power production on the site. However, if the power demand on the site is small, only one small power plant will be needed. In that case, series production is still possible through application of identical units on many different sites but economic power production should not be fully dependent on this. Also, the first few plants should be attractive enough to apply in market circumstances, i.e. the first-of-a-kind capital costs and risks should be acceptable for a non-government-owned company in a deregulated market. To reach this goal, the plants must not only have as few components as possible but these components should carry a minimal technical risk. The small power level (compared to traditional nuclear plants) offers opportunities for this.

First, the pebble bed high temperature reactor technology has been selected for its inherent safety features: a design without emergency core cooling and shutdown systems is possible. Secondly, the on-line fuelling and defuelling system characteristic of existing pebble bed high-temperature reactor designs is omitted and the core remains unchanged for three years, after which it is replaced as a whole, like a power cartridge.

Several advanced nuclear power plant designs with long-life cores already exist for deployment for the long term, requiring a large development effort for fuel and plant technology, e.g. the lead cooled encapsulated nuclear heat source [XIX-1].

For the short to medium term however, the use of existing technology is to be maximized and additional development limited. For instance, an energy conversion system with a direct Brayton cycle is often proposed to couple with a pebble bed high-temperature reactor to increase thermal efficiency and reduce energy cost. To avoid the necessity of a helium turbine, a component still under development and therefore carrying a high technical risk, an indirect cycle with nitrogen as a secondary medium was selected. In this way, a commercially-available expander turbine (usually running with air as its medium) can be applied, at the same time eliminating the possibility of air ingress in a hot pebble bed core, in the scenario of a heat exchanger leak.

A similar philosophy is followed with regard to the fuel; the development, licensing and the creation of an industrial production facility is not only costly but too time-consuming for application on a relatively near term. As high temperature reactor technology is most suitable with regard to the desired inherent safety characteristics, it is proposed to use the pebble fuel as planned for the PBMR (Pebble Bed Modular Reactor) plant [XIX-2]. Thus, both fuel development in the framework of the German HTR programme of the 1980s and the industrial infrastructure dedicated to the PBMR project, is used.

The considerations above resulted in the design of the ACACIA (Advanced Atomic Cogenerator for Industrial Applications) concept, a 60 MW(th), 23 MW(e) (maximum) nuclear plant design with an indirect Brayton cycle [XIX-3, XIX-4]. The principal stakeholder for ACACIA is Nuclear Research and consultancy Group (NRG), the Netherlands.

XIX-1.2. Applications

Two applications are analyzed [XIX-5]: one for cogeneration of electricity and process steam, and one for electricity generation only. Potable water production was considered within the cogeneration option.

XIX-1.3. Special features

This concept is well suited as an autonomous energy source (nuclear cell). ACACIA also allows for an incremental capacity increase through a modular approach in its design. ACACIA may be viewed as a small reactor with simplified on-site refuelling. The lifetime of the cartridge core is 3 years and the intention is to exchange the core several times during the plant lifetime. This will be done for the whole core in one step, instead of the traditional method of element-by-element.

XIX-1.4. Summary of major design and operating characteristics

For cogeneration, a 60 MW(th) helium cooled pebble bed reactor is coupled with a secondary nitrogen cycle through a He/N_2 heat exchanger. If the application is electricity production only, a combined cycle of a gas turbine and a steam turbine is used. Major design and operating characteristics of the ACACIA plant are summarized in Tables XIX-1 and XIX-2.

TABLE XIX-1. MAJOR DESIGN AND OPERATING CHARACTERISTICS OF ACACIA INDIRECT CYCLE PLANT, IN COGENERATION MODE AND IN ELECTRICITY-ONLY MODE

CHARACTERISTIC	COGENERATION	COMBINED CYCLE
Reactor power, MW(th)	60	
Core inlet/outlet temperatures, ºC / K	352/900 (625/1173)	
Helium inlet pressure, MPa	4.1	
Helium mass flow, kg/s	21	
Net electrical power output, MW(e)	18.1	23.2
Gas turbine output, MW(e)	18.8	18.8
Steam turbine output MW(e)	-	5.2
Process steam production, t/h at 425ºC / 4.14 MPa	27.8	-
Net power generation efficiency, % (maximal)	30.1	38.7
Net total thermal efficiency, %	70.0	38.7

TABLE XIX-2. STANDARD DESIGN PARAMETERS FOR THE CORE MODEL

Core outer radius, m	1.45
Core height, m	7.5
Inner reflector radius, m	0.65
Thickness of burnable poison layer, m	0.34 (inner reflector)
Height of gas plenum, m	0.5
Thickness of outer reflector, m	1.0
Thickness of top/bottom reflector, m	2.0
Number of fuel elements	213500
Packing fraction	0.61
Fuel enrichment, %	8.1
Heavy metal mass per fuel element, g	9.0
Average discharge burn-up, MW·d/kg U	34
Average power density, MW/m^3	1.4
Outlet temperature, °C	900
Burnable poison concentration, ppm	100 (upper zone) 200 (middle zone) 100 (lower zone)
Cartridge lifetime, full power days	1180

Simplified schematic diagram of the ACACIA plant, thermal-hydraulic characteristics

Figure XIX-1 presents the component arrangement for the cogeneration plant. The reactor and energy conversion components are placed into four modules, one nuclear and three non-nuclear. The reactor pack houses the reactor, the helical tube He/N_2 heat exchanger (nitrogen heater) and the helium blower. The hot nitrogen is transferred to the gas turbine pack, where it drives the gas turbine. The turbine also drives the two compressors of the inter-cooled cycle and a generator delivering 18.8 MW(e). After leaving the turbine, the gas, now cooled down to 516ºC, flows to the adjacent heat cogeneration unit. Here it is directed through four heat exchangers in a row: the heat recovery steam generator, the recuperator, the feed water heater and the pre-cooler. By now, the gas has been cooled down to 28ºC, and is sent back to the gas turbine pack for inter-cooled compression. Before being sent back to the nitrogen heater, it is preheated in the recuperator to 299ºC. Figures XIX-2 and XIX-3 give the temperature, pressure and gas flow on the main locations of the energy conversion system and the reactor. In this way, net power of 18.1 MW(e) is generated simultaneously with high quality process steam at 27.8 t/h, at 425ºC and 4.14 MPa. All gas-water heat exchangers are helical coil, once-through types, and the recuperator is of a plate offset-fin type.

For the combined cycle electric plant, hot steam leaving the heat recovery steam generator is expanded in an additional steam turbine coupled with a second generator, giving an additional 5.2 MW(e). After leaving the turbine at 40ºC, the steam is condensed and fed back into the feedwater heater.

The core outlet temperature is set to 900ºC, which is below what has been, or will be demonstrated, by the fuels in AVR (Arbeitsgemeinschaft Versuchs Reaktor), HTR-10, HTTR (High Temperature Test Reactor), etc. No safety hazard of water or steam ingress into the primary system exists, since all water and steam circulation is remotely located in the third loop. The nitrogen heater is essentially pressure balanced with a slightly higher secondary pressure to ensure that no fission products enter the secondary system in case of leaking tubes.

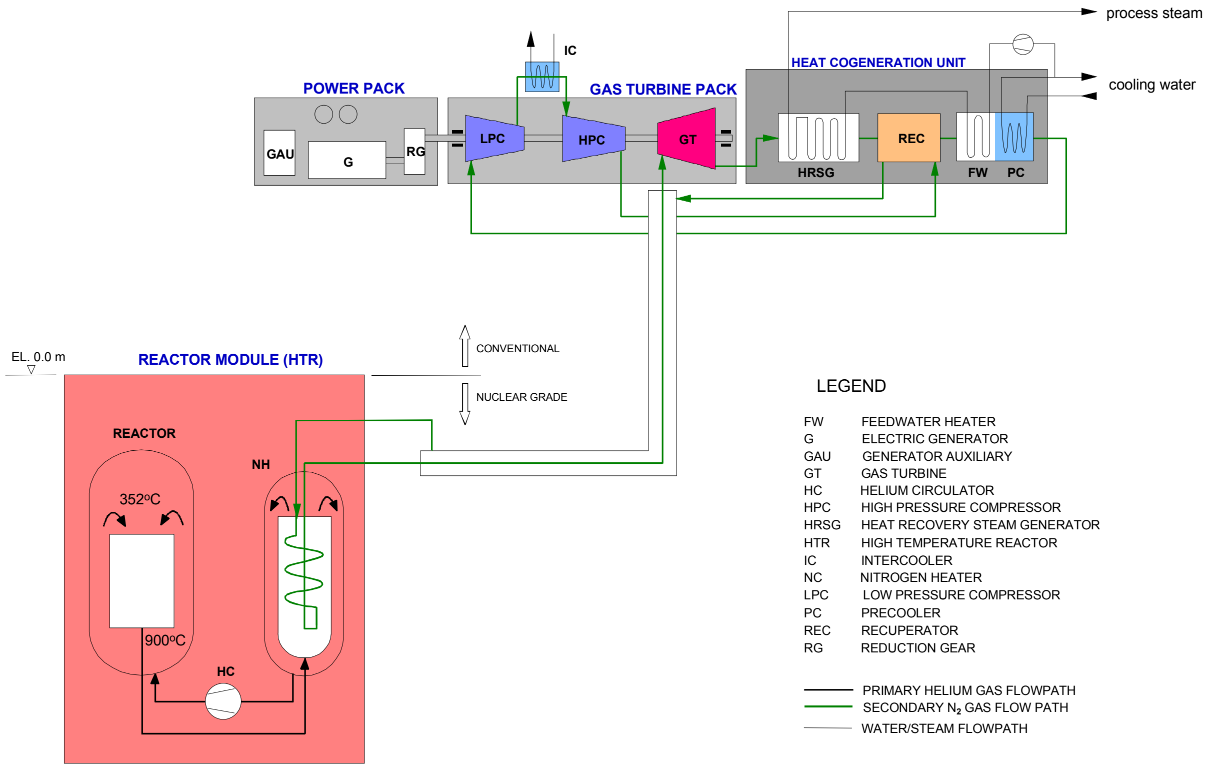

FIG. XIX-1. ACACIA cycle design for cogeneration of electrical power and process steam.

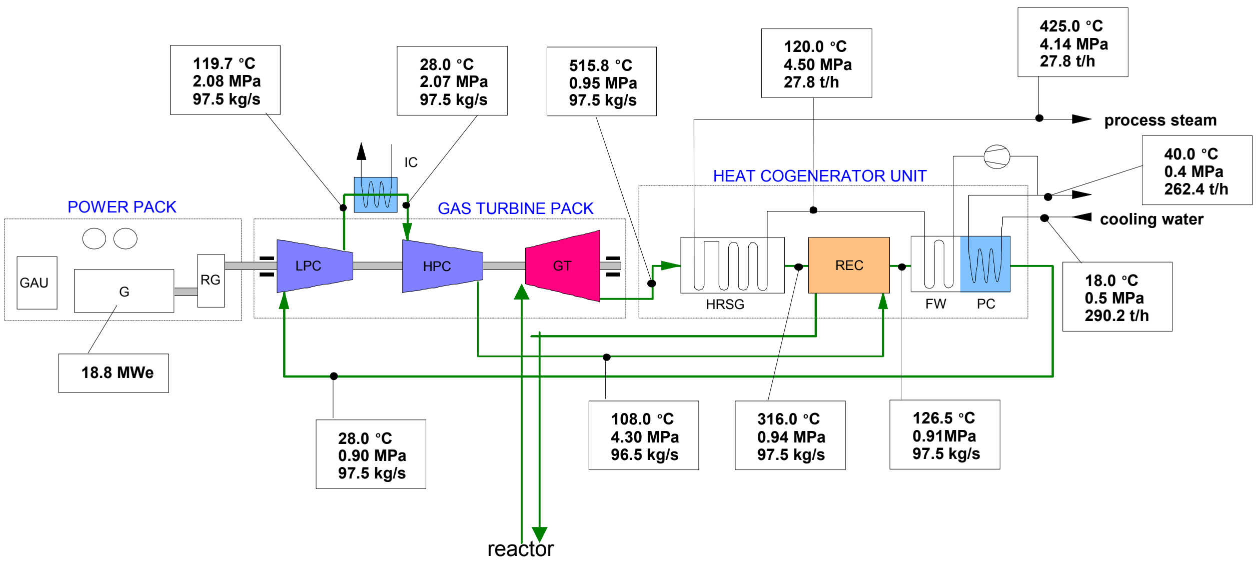

FIG. XIX-2. ACACIA cogeneration cycle main parameters for the energy conversion system.

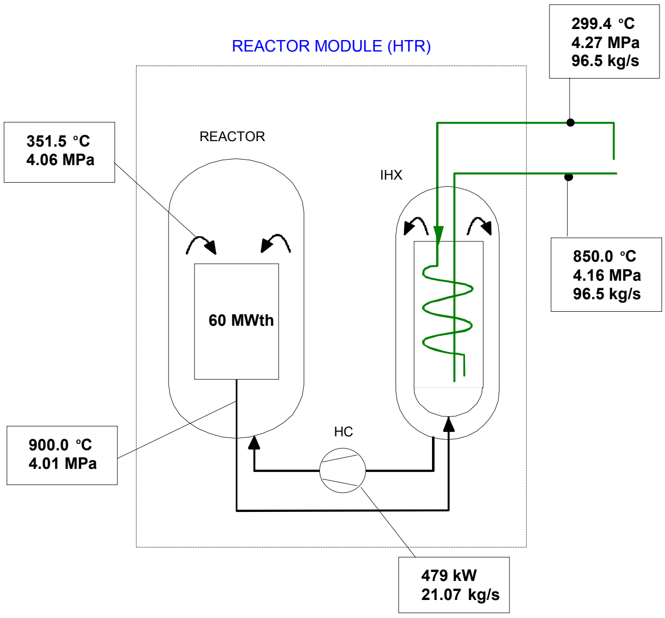

FIG. XIX-3. ACACIA reactor main thermal-hydraulic parameters.

The reactor vessel will be kept below 370ºC, so normal SA533 steel can be used. The nitrogen heater can be designed as compact, as a modular HTR steam generator [XIX-6], by selecting:

- A large temperature difference of 50ºC between the primary and secondary inlet and outlet fluids.
- An optimal pressure balance inside and outside of the tubes of 4.2 vs. 4.0 MPa.
- A minimal tube wall thickness of 2.5 mm.

Main heat transport system

The scheme of the main heat transport system with specification of the heat removal path in normal operation and in accidents is presented in Fig. XIX-4.

Reactivity control mechanism

To compensate for the strong core over-reactivity, burnable poison will be used. The burnable poison will also improve core power distribution and reduce power peaking. The remaining excess reactivity should be limited to values that can be compensated with a total control rod worth comparable to values used in the HTR-Module design. In this way, the use of active control elements is reduced and accident scenarios involving control rod ejection will have only limited consequences. As original PBMR fuel elements are foreseen, the burnable poison must be located in the reflector. Core geometry will be annular, with a fixed central reflector column consisting of graphite blocks (Fig. XIX-5). The outer ring of the central reflector blocks carries boron carbide sticks as burnable poison. The boron concentration has been zoned axially; i.e. the boron concentration in the axial centre zone was increased with respect to the bottom and top zones. In this way, the effect of the axial flux distribution was counteracted and reactivity stayed almost constant over 1150 days, after which it decreases rapidly to subcritical.

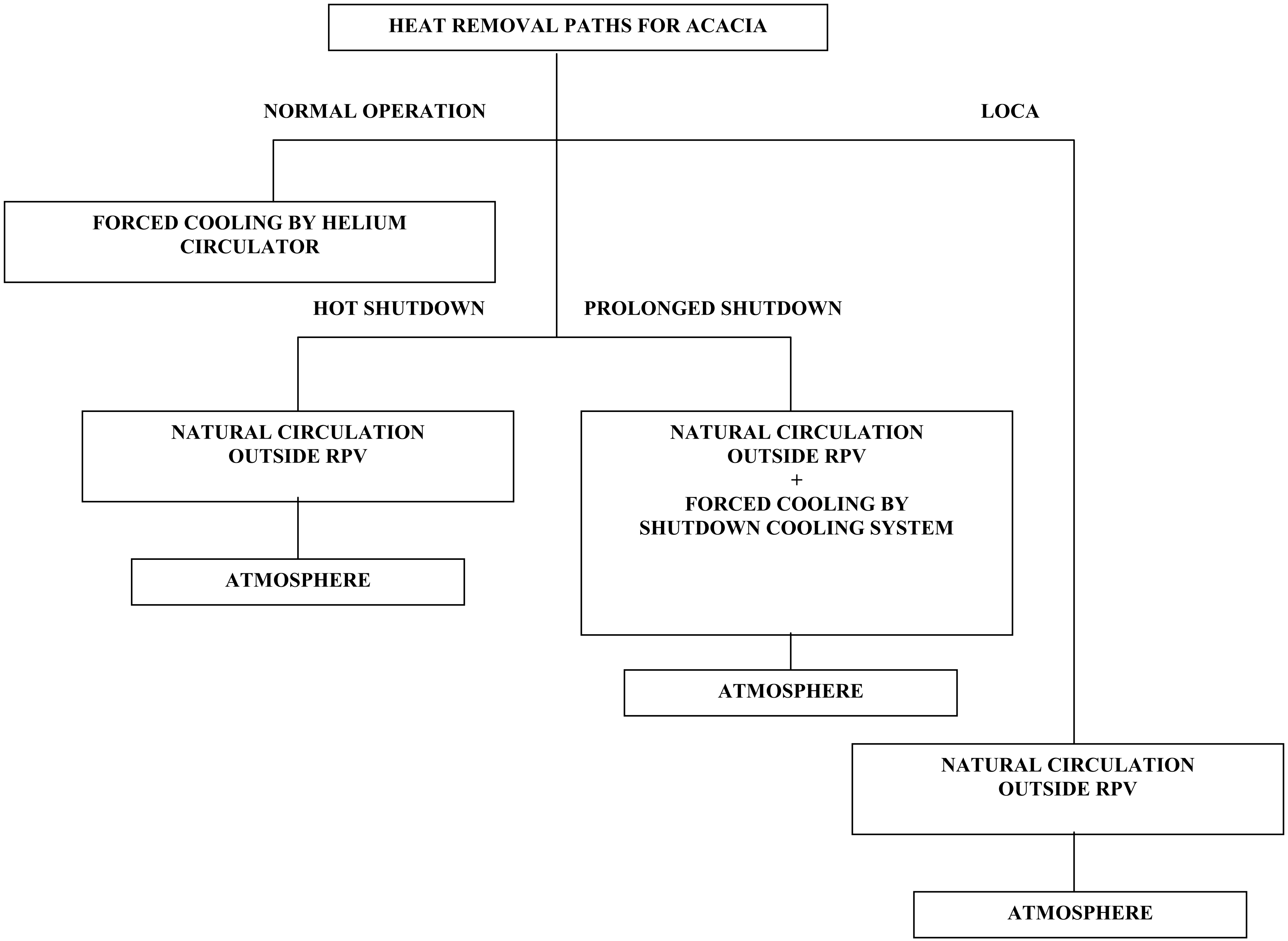

FIG. XIX-4. Main heat transport system of the ACACIA.

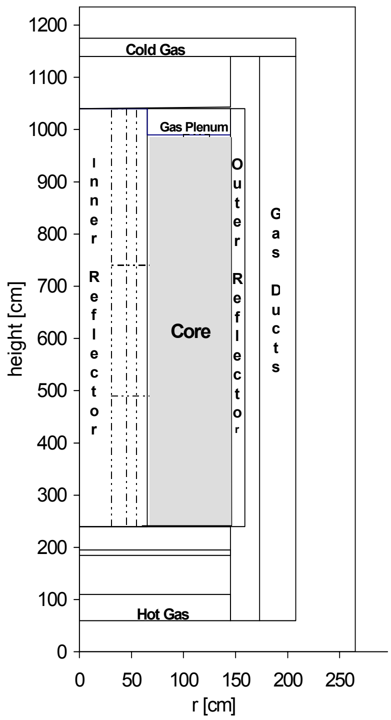

FIG. XIX-5. ACACIA cartridge model for an annular core.

Design and operating characteristics of systems for non-electric applications

A multi-stage flash (MSF) desalination plant was considered for the ACACIA. About 60% of the world's desalting capacity comes from MSF plants. Figure XIX-6 shows a schematic diagram of the MSF system [XIX-7]. An MSF system consists of a main heat exchanger and a number of evaporation stages, in this case three, each operating at different pressures and temperatures. In the first stage, seawater is evaporated and then condensed on the tubes of a stage condenser. In the following stages, the resulting brine of the previous stage is treated the same way. The energy gained from condensation is used to pre-heat the seawater before it enters the main heat exchanger (brine heater). A main water pump is located at the inlet of the system. Additionally, each step is equipped with its own small pump, capable of compensating for the stage pressure drop.

Superheated steam from the ACACIA steam supply system enters the desalination system at a pressure of about 4.0 MPa, a temperature of 417ºC and a mass flow of about 7.7 kg/s. The salt water inlet flow is about 50 kg/s. This water is pressurized to about 4.2 MPa and then heated up to a temperature of about 251ºC in the heat exchangers. The hot water enters stage 1, where the pressure is equal to 0.88 MPa. At this pressure the water is superheated and flashing occurs. The evaporated water (about 9.5 kg/s) flows up to the heat exchanger compartment where it is condensed and collected. From here it flows to stage 2. The water that did not evaporate in stage 1 flows to the evaporation sections of stage 2 through the level control valve. The pressure in stage 2 is 0.22 MPa, so further flashing occurs. The evaporation rate in stage 2 is about 4.3 kg/s. Finally, the process continues in stage 3, where pressure is already close to atmospheric. About

35.3 kg/s of waste brine, with a salinity of about 7.3 weight %, is discharged from stage 3. The amount of fresh water obtained from the system is about 15.5 kg/s, which is about 1340 t/day. This would be a relatively small-scale desalination plant, typically meant for smaller islands or remote industrial locations.

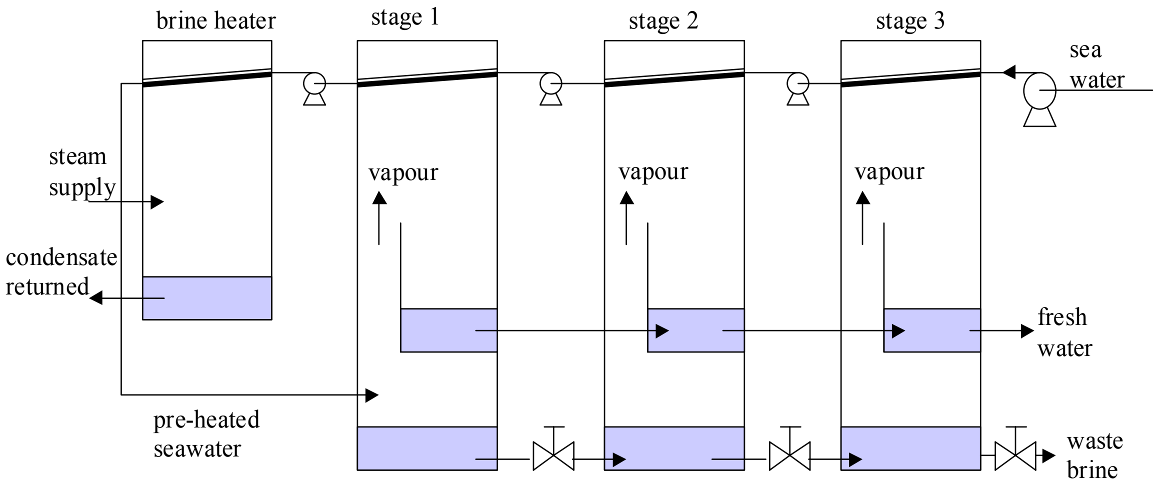

FIG.XIX-6. Schematic diagram of the ACACIA multi-stage flash (MSF) desalination system.

Economics

The specific overnight cost for ACACIA ranges between 2700 and 2200 US \$/kWe for first-of-a-kind and serial plant respectively.
The series plant cost of electricity is ~3 cents/kWh (US\$) in a cogeneration mode and ~6 cents/kWh in a 'power generation only' mode.

XIX-1.5. Outline of fuel cycle options

No information was provided.

XIX-1.6. Technical features and technological approaches that are definitive for ACACIA performance in particular areas

XIX-1.6.1. Economics and maintainability

Generally, for small sized power plants below 100 MW(e), capital investment costs and the cost of electricity are higher than for larger plants because of the economy of scale laws. However, the absolute amount of capital is considerably lower.

A first-of-a-kind production cost of US \$63 million (overnight cost, US \$ as of 1997) has been calculated for the cogeneration plant. With traditional learning curves and series production advantage factors, a series plant production cost of US \$52 million has been derived. This must be compared with the US \$160 million needed for a single PBMR plant, the next "cheapest" plant with respect to capital investment. It is therefore advantageous to companies with not only a smaller demand per site but also those in a lesser position to raise large amounts of capital.

Because of the strategy of applying commercially available equipment in the secondary circuit, the heat transport system and the power conversion system together make up no more than 10% of the total plant investment cost. For a direct cycle system this would be about 20%, and for an indirect cycle system with a secondary helium circuit even 30%. The omission of the on-line refuelling system not only lowers the capital costs (this is only a few percent), but also influences operation and maintenance costs and plant availability in the right direction. The latter two parameters are difficult to quantify, but experience with the operated first-generation pebble bed reactors AVR and THTR showed that this component caused a significant plant down-time [XIX-8, XIX-9], and Schwarz, 1988). Only a worse fuel utilization (34 MWd/t·U, on average) would raise the cost of electricity and heat compared to a plant with a recirculation pebble bed high temperature reactor, if the fuel cartridge was disposed of after use in an ACACIA plant. However, allowing the pebbles from the used cartridge a 'second life' may be economically advantageous. The used cartridge could be 'sold' at no cost to the owner of a recirculation pebble bed reactor, offering the advantage of free partly-used fuel and the advantage of avoiding final disposal costs.

The series plant cost of electricity has been calculated as 3.05 cents/kWh (US $), with rather traditional financial parameters except for accounting for cogeneration and the construction period. A cogeneration credit has been accounted for, mainly consisting of fuel costs avoided by not using a dedicated conventional gas-fired boiler for separate process heat production. This is an important factor because of the high quality of the ACACIA process steam. Also, a short construction time of 24 months was assumed as a result of the small primary system and the highly-modularized energy conversion system. Finally, the after-use of the cartridge fuel in a recirculation pebble bed reactor as proposed above has been assumed. The cost of electricity for ACACIA in power generation only mode would then be 6.09 cents/kWh (US $).

XIX-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

No information provided.

XIX-1.6.3. Safety and reliability

Safety concept and design philosophy, provisions for simplicity and robustness of the design

In the design of the reactor core, the reduction of the number of subcomponents received priority over the use of designs with maximum operating experience. Unlike previous pebble bed reactor designs, ACACIA lacks the on-line refuelling system. Refuelling will be a short off-line operation, in which a mobile refuelling system is mounted on top of the reactor vessel and the whole core is replaced at once.

Active and passive systems and inherent safety features

These are similar to other pebble bed modular reactors.

Design basis accidents and beyond design basis accidents

In this section, the two most important safety-related transient scenarios for pebble bed reactors are analyzed for the ACACIA: Pressurized and depressurized loss of forced cooling, without scram (LOFC and DLOFC, respectively).

DLOFC is associated with a large primary helium leak, whereas LOFC means a failed circulator. In both cases, it is assumed that control rod insertion fails as well. For the DLOFC, it is considered that the initial helium pressure of 4 MPa is lost within an interval of 30 seconds, whereas for the LOFC the helium flow is ramped down to zero within 60 seconds and the system pressure is kept unchanged. The transient analyses have been done for a fresh core with equilibrium xenon concentration (BOL) and with the control rods adjusted for criticality (≅45% insertion). Moreover, the core is supposed to be in operation long enough that the decay heat contribution at the start of the incident can be considered in equilibrium. Calculations have also been done for other time moments during the core lifetime, but the situation described here is the worst case.

For both scenarios a number of parameters have been calculated as a function of time after the start of the incident (Figures XIX-7 and XIX-8) with PANTHERMIX code [XIX-10, XIX-11]: core reactivity, maximum fuel temperature, xenon concentration and reactor power; the latter are subdivided into decay power and fission power. In both cases the reactor shuts itself down autonomously because of the negative temperature coefficient. Xenon increases for the first seven hours after the incident, whereas the average core temperature decreases after an initial increase of about 35 K. As long as the negative xenon effect is dominant over the reactor temperature effect, the reactor stays subcritical. But at a certain point in time the xenon decays, the reactivity starts to increase, and re-criticality occurs. At first, a series of power oscillations sets in but because of the negative temperature coefficient of reactivity, these oscillations are damped out within about 5 hours. The difference in oscillation frequency between the LOFC and DLOFC can be explained by the difference in the thermal time constant between the two scenarios. The pressurized core has a larger heat transfer coefficient (but about the same heat capacity as for the depressurized core) and therefore thermally, it responds faster to changes in power. In a depressurized core, the point in time of first re-criticality is later than in the pressurized core because of this difference in thermal properties: 12 hours versus 5 hours after the start of the incident. The core temperature for the pressurized core decreases at a higher rate so that the re-criticality occurs before the xenon concentration reaches its maximum.

The maximum reactor power of 1.5 MW for the pressurized core (1.0 MW for the depressurized core) is reached about 29 hours after re-criticality (21 hours for the depressurized core). The pressurized core has a greater external heat leakage. Therefore, larger power is necessary to increase the temperature sufficiently to compensate for the positive reactivity effect associated with the decay in xenon concentration. At this moment, the decay heat contribution for the depressurized core amounts to 20% (33% for the pressurized core) of the total reactor power. After that, the total power decreases at a much faster rate for the pressurized core than for the depressurized core and reaches a level of 900 kW (650 kW for the depressurized core) 80 hours after the incident. At that moment the decay heat contribution is about 250 kW for both incidents.

The maximum fuel temperature increases monotonically after criticality, to slightly below 1900 K. This maximum temperature is reached only after 80 hours (3.3 days) and for only a very small percentage of the volume of the core during a short period of time. Therefore, it can be safely stated that no fuel degradation takes place. Later points in time during the cartridge cycle have been investigated as well, and they all give lower maximum fuel temperatures.

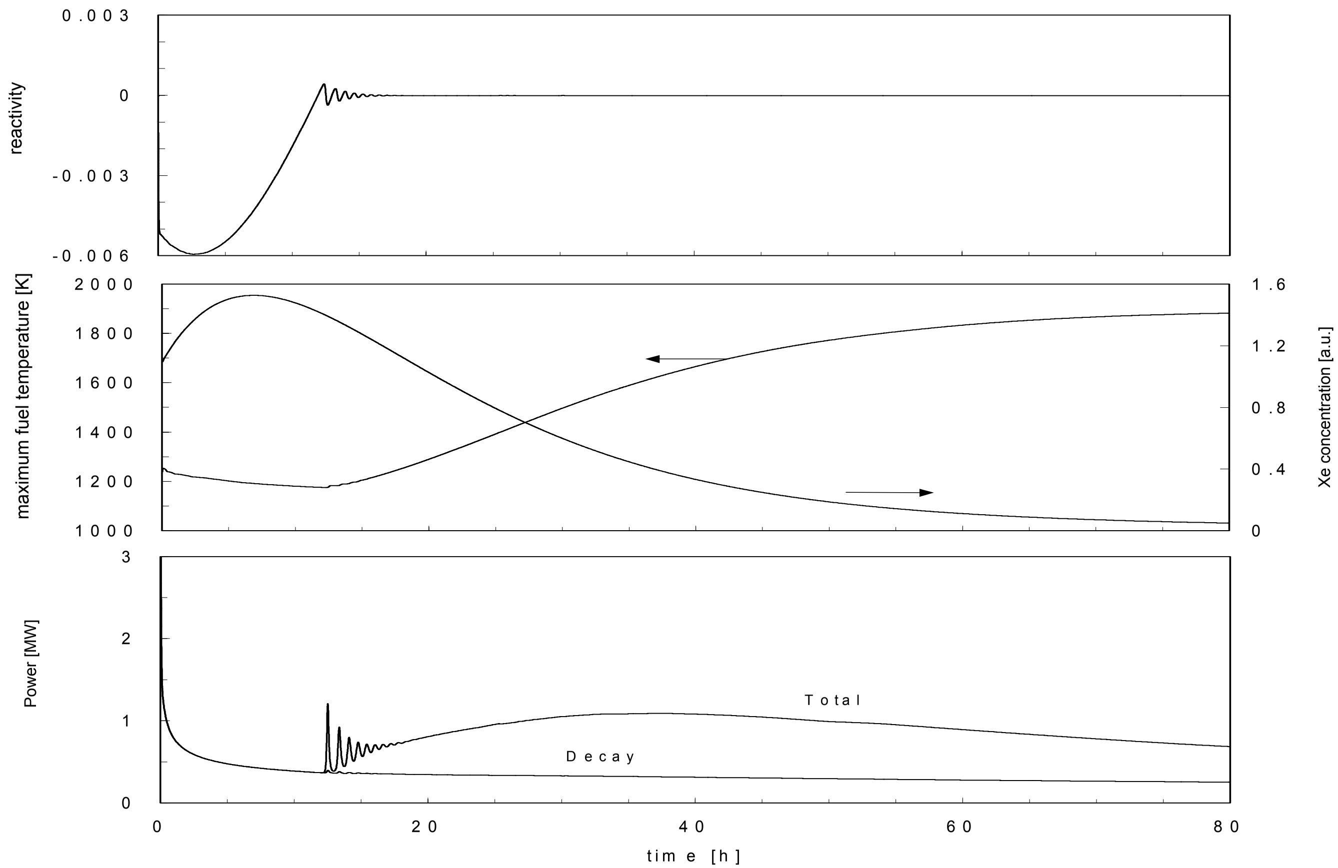

FIG.XIX-7. Reactivity, maximum fuel temperature, xenon concentration and thermal power of the reactor (BOL) after a depressurized loss of forced cooling without scram.

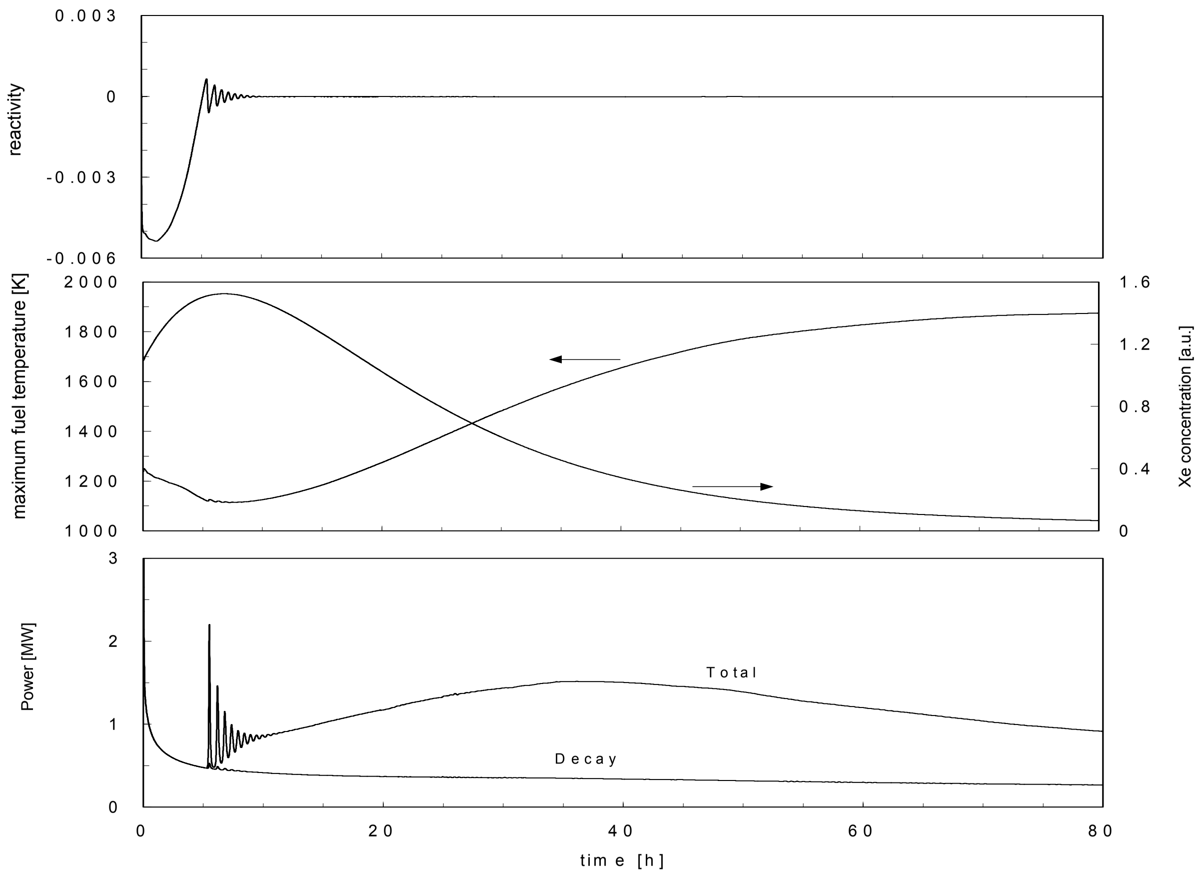

FIG.XIX-8. Reactivity, maximum fuel temperature, xenon concentration and thermal power of the reactor (BOL) after a pressurised loss of forced cooling without scram.

XIX-1.6.4. Proliferation resistance

The cartridge character of the core, with full core replacement once every three years, could facilitate nuclear material accounting and verification. This especially includes a scenario in which the ACACIA plant location is in a developing country or country with limited nuclear infrastructure, and a foreign fuel company is exchanging the core every three years.

XIX-1.6.5. Technical features and technological approaches used to facilitate physical protection of ACACIA

No information provided.

XIX.1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of ACACIA

Due to simplicity and a low absolute level of capital costs, ACACIA may suit the needs of developing countries with small electricity grids and limited turnover of capital in the energy-producing sector. With core replacement as a whole within a single batch, the reactor may be attractive for fuel or NPP leasing and full-scope fuel cycle service agreements but these issues have not been elaborated in detail so far.

XIX-1.8. List of enabling technologies relevant to ACACIA and status of their development

Among past and current system and component technologies supporting the present N_2 based closed thermodynamic cycle are:

- Experience: a number of closed cycle air gas turbine generators were built in the past for up to 30 MW(e) in Europe, the U.S. and Japan; some were operated for 100 000–150 000 hrs;
- More modern gas turbine technologies developed in conventional gas turbines such as dry gas shaft seal, high-temperature blade materials, aerodynamic and rotor dynamic modelling;
- Conventional gas-to-water coolers and heat recovery steam generators;
- Conventional or retrofitted steam turbines and auxiliaries.

The performed design study has identified little or no R&D requirements for the energy conversion system of the ACACIA indirect cycle plant noting that all major equipment can be obtained based on available experience or from off-the-shelf products.

ACACIA is also based on certain PBMR technologies, e.g. TRISO type coated particle fuel and spherical fuel elements, and the HTR-Module technologies, e.g. control rods.

XIX-1.9. Status of R&D and planned schedule

No further R&D is planned at the moment, but the design team intends to pursue further development of the ACACIA design.

XIX-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

Though the amount of R&D needed to qualify systems and components of the ACACIA seems limited, the overall plant configuration differs from other known HTGR designs. Therefore, construction of a demonstration prototype may be required before licensing the plant in series.

XIX-1.11. List of other similar or relevant SMRs for which the design activities are on-going

Other SMR designs of relevance to ACACIA are:

- PBMR [XIX-2];
- HTR-PM [XIX-12];
- ENHS: Encapsulated Nuclear Heat Source [XIX-1].

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DESIGN DESCRIPTIONS OF LIQUID METAL COOLED SMRs

ANNEX XX

KOREA ADVANCED LIQUID METAL REACTOR (KALIMER)

KAERI, Republic of Korea

XX-1. General information, technical features and operating characteristics

XX-1.1. Introduction

The LMR design technology development project was approved as a national long-term R&D programme in 1992 by the Korea Atomic Energy Commission (KAEC), which decided to develop and construct a liquid metal cooled reactor (LMR). Based upon the KAEC decision, the Korea Atomic Energy Research Institute (KAERI) has been developing KALIMER (Korea Advanced Liquid Metal Reactor) [XX-1, XX-2].

The goal of the LMR design technology development project is to develop LMR design technologies necessary for the efficient utilization of uranium resources and reduction of high level wastes. The design objectives of the KALIMER are enhanced safety, competitive economics, proliferation resistance and environmental friendliness.

Limited number of basic design methods, computer codes and sodium technologies had been developed before 1997, and the initial design concept was proposed through a feasibility study of various innovative design concepts. Through the revision of the long-term nuclear R&D plan, the scope of the LMR design technology development project was modified in January 1999 to focus on the development of LMR design technologies rather than to emphasize development of designs for construction.

The LMR design technology development project has been carried out phase by phase as follows:

- Phase 1 (July 1997 – March 2000) — Development of basic technology and preliminary conceptual design

  In phase 1, the basic computer codes and methods necessary for the design and analyses have been developed or updated, and maximum effort has been spent to establish a self-consistent conceptual design of the system configuration, arrangement and key features to satisfy the design requirements. Efforts have also been made to develop the basic sodium technologies, such as measurement or detection techniques, thermal-hydraulics and sodium fires.

- Phase 2 (April 2000–March 2002) — Development of advanced basic technologies and conceptual design

  During phase 2, the conceptual design of the KALIMER has been completed. The basic computer codes and methodologies developed during phase 1 have been improved. These codes and methodologies have been applied for the development of the conceptual design based on a preliminary conceptual design developed during phase 1.

Currently, the LMR design technology development project is in the last year of phase 3, and focuses on the development of the basic key technologies and advanced concepts.

The design and development of the KALIMER has been supported by the Ministry of Science and Technology of the Republic of Korea.

XX-1.2. Applications

KALIMER is an NPP designed to produce 150 MW(e)(net) of electricity.

XX-1.3. Special features

KALIMER is a land-based nuclear power station.

XX-1.4. Summary of major design and operating characteristics

Installed capacity

The reactor is designed to produce 392.2 MW(th), generating 150 MW(e)(net).

Mode of operation:

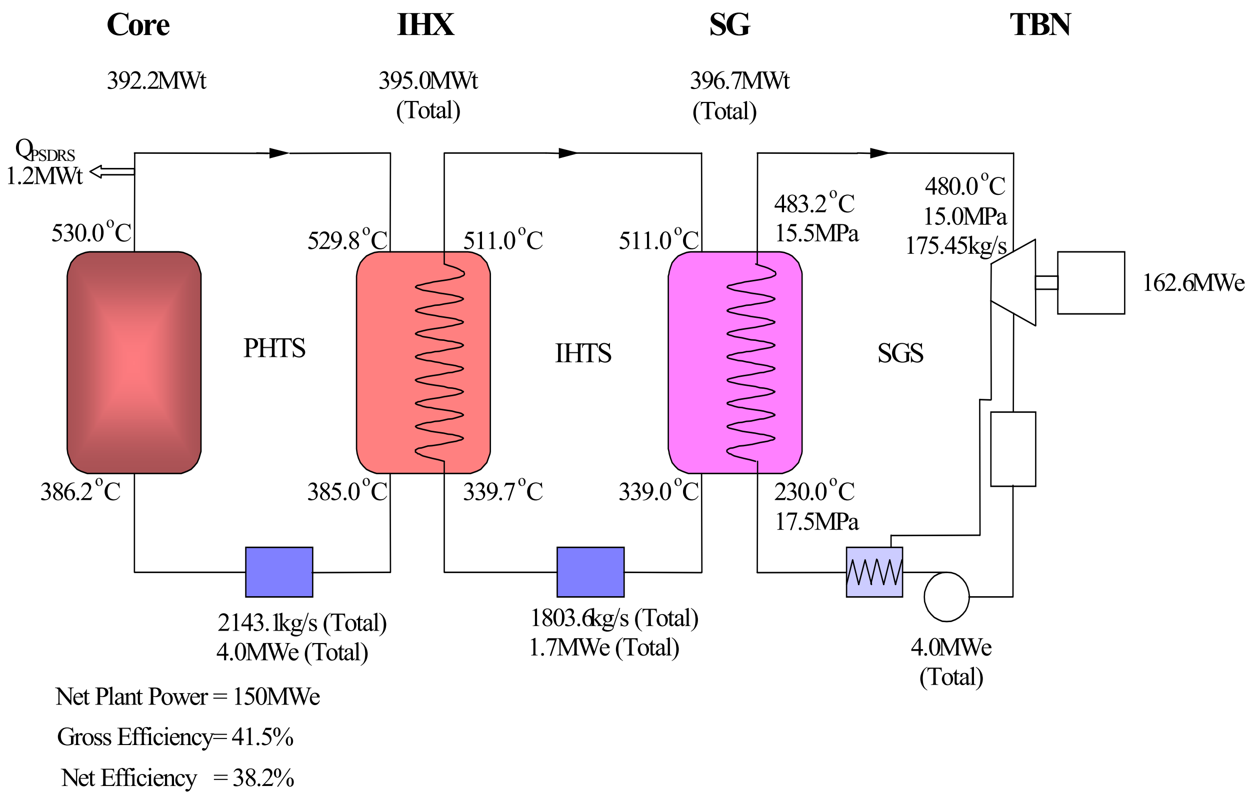
Base load.

Load factor/ Availability

The target lifetime availability factor for KALIMER is $\geq 89\%$.

Simplified schematic diagram

A simplified schematic diagram of the KALIMER NPP is presented in Fig. XX-1. This figure also specifies the major thermal-hydraulic parameters in circuits.



The parameters correspond to operation at 100 % power.

FIG. XX-1. Simplified schematic diagram of KALIMER plant.

The major design and operating characteristics of the KALIMER are given in Table XX-1.

TABLE XX-1. MAJOR DESIGN AND OPERATING CHARACTERISTICS OF KALIMER

CHARACTERISTIC	VALUE OR TYPE
Overall	
Gross plant efficiency, %	41.5
Net plant efficiency, %	38.2
Reactor style	Pool type
Number of circuits	3 (PHTS*, IHTS**, steam system)
Number of IHTS loops	4
Seismic design	Seismic isolation system
Reactor	
Core configuration	Heterogeneous
Core height, mm	1000
Maximum core diameter, mm	3373
Axial blanket thickness, mm	0
Fuel form	U–Pu–10% Zr alloy
Feed driver fuel TRU enrichment for equilibrium core, %	30.0
Assembly pitch, mm	161.0
Fuel/blanket pins per assembly	271/127
Cladding material	HT9
Refuelling interval, months	18
PHTS	
Reactor core inlet/outlet temperature, °C	386.2/530.0
Total PHTS flow rate, kg/s	2143.1
Primary pump type	Electromagnetic
Number of primary pumps	4
IHTS	
IHX inlet/outlet temperature, °C	339.7/511.0
IHTS total flow rate, kg/s	1803.6
IHTS pump type	Electromagnetic
Number of IHXs	4
Number of SGs	2
Steam system	
Steam flow rate, kg/s	175.45
Steam temperature, °C	483.2
Steam pressure, MPa	15.5

* PHTS stands for primary heat transport system

** IHTS stands for intermediate heat transport system

Nuclear design criteria

The nuclear design of the KALIMER core is based on the core design criteria discussed below and on the constraints given in Table XX-2, derived from the currently available metal fuel database. Altogether, they define the important performance parameters of the reactor that assure proper performance and safety of fuel and core.

TABLE XX-2. DESIGN BASIS AND CONSTRAINTS USED IN THE NEUTRONIC DESIGN

Reactor power	392.2 MW(th)
Peak linear power limit	<440 W/cm
Local peak burn-up limit for fuel	<150 MW·d/kg
Peak fast neutron fluence	$<4.0\times10^{23}$ n/cm^2.
Capacity factor	85%
Refuelling interval	18 month

The fuel form for the core is U–Pu–Zr ternary. For a startup core, the fresh fuel is composed of recovered LWR transuranics and depleted uranium. In subsequent cycles, where the discharged TRU become available for manufacturing new fuel feeds, the fissile makeup of a core load will be based on recycled Pu. In addition, minor actinides are assumed to be recycled in a proportion in which they are present in the spent fuel, consistent with the assumed reprocessing technique.

The core design provides for an inherent control of power by reactivity feedback within the core, such that acceptable fuel design limits are not exceeded for the defined beyond design basis accidents.

The breeding ratio is to be near 1.05, assuming 0.1% loss of TRU during the reprocessing. In addition, 5% of the rare earth fission products will be recycled.

The burn-up reactivity swing should be kept at a minimum during cyclic operation, because it has a direct impact on the available shutdown worth of the control system and on the control of several reactivity induced transients. In order to ensure proper reactivity control, the allowable burn-up reactivity swing should be around 1000 pcm, and limited by 1500 pcm (~5$).

The power distribution should be relatively flat to minimize the peak linear power, peak burn-up, and peak fast neutron fluence. The raw average linear power in the equilibrium cycle should be around 7 kW/ft (~230 W/cm). If possible, it is preferable to have the power peak in the outer core region, in order to minimize the sodium voiding potential which might bring about a positive reactivity addition.

The enrichment of the U–Pu–Zr ternary fuel in equilibrium cycle (with all TRUs recycled) is recommended to be less than 30 weight % to fall within the currently established metallic fuel database. The design of a fuel assembly should support the metal fuel reliability requirements, including the limits for peak fuel burn-up and fast neutron fluence.

Neutron-physical characteristics

The nuclear performance of the KALIMER is summarized in Table XX-3.

TABLE XX-3. SUMMARY OF NUCLEAR PERFORMANCE

Feed driver fuel enrichment(weight %)	
Fissile Pu in U− TRU	21.15
Total Pu in U− TRU	29.12
Burn-up reactivity swing (pcm)	896.4
Average breeding ratio	1.051
Peak fuel discharge burn up (MW·d/kg)	120.7
Peak linear power(W/cm)	287.1
Power peaking factor for driver fuel	
BOEC	1427
EOEC	1448
Peak neutron flux (10^{15} n/cm^2 sec)	
Driver fuel	3.01
Internal blanket	3.00
Radial blanket	1.88
Peak fast neutron fluence at discharge (10^{23} n/cm^2)	
Driver fuel	2.41
Internal blanket	2.36
Radial blanket	3.07

Reactivity feedbacks and control system worth

The summary of reactivity feedbacks resulting from Doppler effect, uniform radial expansion, and various sodium voidings in the equilibrium core and during the startup cycle are given in Table XX-4. This table also includes the reactivity worths of the control rod system, the GEM (gas expansion module), and the USS (ultimate shutdown system). The gas expansion module is a device that introduces negative reactivity through gas expansion under temperature increase. Its position in the core is shown in Fig. XX-7; no additional details are provided. The USS is explained further in this section.

The control and shutdown system must have a sufficient worth to shut down the core from any operating condition at any time during the reactor cycle and to maintain subcriticality over the full range of temperatures expected during a shutdown. The overall reactivity could be categorized into several components: temperature defect, burn-up reactivity swing, axial fuel growth, overpower margin, shutdown margin and, last but not least, uncertainties. Temperature defect is the reactivity change from a hot full-power critical state to a zero power state at the refulleing temperature. This positive reactivity comprises the Doppler effect, radial and axial core contraction, and sodium density change. Burn-up reactivity swing is the excess reactivity built for the compensation of reactivity loss in fuel burn-up. The axial fuel growth is pertinent to the metallic fuel, which expands with fuel burn-up because of the accumulation of fission products. For this assessment, a 5 % axial growth was assumed. The overpower margin is allocated to permit the reactor to operate at overrated power. The shutdown margin is required for the assurance of subcriticality.

The fuel temperature (Doppler) coefficients due to the Doppler effect vary as about $1/T^{1.42}$, which shows that the present plutonium core has a hard neutron spectrum typical of a small, metallic fuelled fast reactor. The fuel temperature coefficients do not show any substantial change with fuel burn-up. In sodium-void case, the fuel temperature coefficient becomes less negative as compared to the nominal case.

TABLE XX-4. SUMMARY OF REACTIVITY PERFORMANCE

PARAMETER	BOC 1	EOC 1	BOEC	EOEC
Fuel temperature (Doppler) coefficient (Δk/k/K)				
Nominal	$-0.0869T^{-1.44}$	$-0.0819T^{-1.42}$	$-0.0865T^{-1.42}$	$-0.0837T^{-1.41}$
Voided	$-0.0879T^{-1.47}$	$-0.0866T^{-1.46}$	$-0.0910T^{-1.46}$	$-0.0815T^{-1.45}$
Uniform radial expansion coefficient				
(dk/k)/ (dR/R) (pcm / %)	–572	–564	–564	–555
dk/dT ($\times 10^{-6}$)(K^{-1})	–8.4434	–8.0292	–8.1726	–7.8932
Sodium void effect (pcm)				
Driver fuel (DF)	561	760	826	930
Inner blanket (IB)	639	699	717	751
Radial blanket (RB)	–201	–173	–160	–137
DF + IB	1240	1499	1582	1717
Total (DF + IB + RB)	1031	1320	1418	1557
DF + IB + RB + GEM	–108	311	457	722
Worth of control rods (pcm)	7688	8051	8071	8264
GEM (pcm)	1073	948	907	802
USS (pcm)	1442	1847	1797	2182
Effective delayed neutron fraction	0.00397	0.00392	0.00355	0.00358
Neutron generation time (μsec)	0.27	0.29	0.29	0.30

When voided independently, all core zones except the radial blanket result in a positive reactivity insertion. Even the activation of the GEMs is not sufficient to bring the core to a subcritical state. The sodium void reactivity increases with burn-up due to fission product buildup and Pu quality deterioration.

The uniform core radial expansion due to coolant temperature increase is one of the major negative reactivity insertion mechanisms in a metallic fuelled reactor. The radial expansion coefficients are insensitive to fuel burn-up and the degree of radial expansion.

The total control rod worths are 8071 and 8264 pcm at BOEC and EOEC, respectively. They show a weak dependence on fuel enrichment variation and spectrum changes during burn-up. The total control rod worth implies a sufficient shutdown potential to bring the core to a subcritical state even in the sodium-void cases. The negative reactivity induced by GEM

activation only is not enough to bring the core to a subcritical state in any sodium-void condition. However, even the maximum positive sodium void reactivity can be compensated by the operation of a passive shutdown system, USS. This fact indicates that passive shutdown could be achieved in the event of a complete failure of the normal scram system and after the inherent reactivity feedbacks have brought the core to a safe but critical state at an elevated temperature.

Due to insufficient knowledge of core characteristics and data at the conceptual design stage, the uncertainties in reactivity components could be large, especially as comes to calculated core parameters. In setting a shutdown reactivity requirement, it is prudent to include these large uncertainties to assure that the absorbers will have an ample worth. The total uncertainty is obtained by statistically combining all uncertainties.

The reactivity requirement calculated for the shutdown system must be satisfied with the highest worth control element pulled out of the core. That is, each control rod unit should have a sufficient worth such that the remaining rods can shut down the reactor to a cold subcritical state.

Reactivity control mechanism

There are two independent and divesrse reactivity control and shutdown systems. One is an active, motor-driven control rod system, and the other is a passive reactor shutdown system, driven by gravity.

The KALIMER active control rod system consists of drive mechanisms, drivelines, the absorber bundles, and the absorber channels.

The active reactivity control and shutdown system consists of six control rods that are used for power control, burn-up compensation, and reactor shutdown. Each control rod unit consists of an array of B_4C-containing tubes. The absorber material moves within a hexagonal duct. Two control rod clusters of the same shape are located in the annular ring of the internal blankets incorporated in the driver fuel region, and each of them contains three control rods, Fig. XX-7. The control rod clusters are designed so that each cluster has a rapid reactor shutdown capability in case of a reactor protection signal origination. Two control clusters consisting of six control rods are operated simultaneously to provide a normal operation control. The control rod design satisfies both the one-rod-stuck condition and the unit control rod worth condition against an unprotected transient overpower (UTOP) event.

The KALIMER adopts the ultimate shutdown system (USS) as an alternative means of shutting down the reactor. The design concept is a self-actuated shutdown system (SASS) located in the centre of the core.

The SASS is a passive reactor shutdown system actuated in an emergency by the natural physical phenomena without any external control signals or any actuating power. The Curie point electromagnet (CPEM) is to be used as a key component of the USS, with its saturated magnetic flux density being remarkably reduced at the curie point of a temperature-sensitive material used in the CPEM. When the temperature of the primary sodium goes up to a Curie point, the CPEM loses its electromagnetic force and exerts the shut-off rod. Then the shut-off rod with the CPEM drops into the core, driven by gravity. The shut-off rods are designed to be of an articulate type for their easy insertion into the core even when the guide tubes are deformed due to an earthquake.

Thermodynamic cycle type

The KALIMER uses a Rankine cycle to supply superheated steam to a turbine.

Fuel lifetime/period between refuellings

The reactor has a provision for on-line refuelling. On average, 12 driver fuel assemblies about 11 blanket assemblies will be replaced assuming 310 effective full power days (EFPD) of operation per year, with an average residence time of the driver fuel assemblies and the internal blanket assemblies of the core being approximately 4.5 years, while it will be 9 years for the outer blanket assemblies. The average discharge burn-up of the driver fuel (without reconstitution) is 87.6 MW·d/kg U, and the maximum one is 120.7 MW·d/kg U.

Mass balances/flows of materials

The annual material requirements for the KALIMER fuel have are given in Table XX-5, assuming 310 EFPD of operation per year in the equilibrium closed fuel cycle.

TABLE XX-5. ESTIMATED ANNUAL FUEL MATERIAL REQUIREMENTS FOR KALIMER

MATERIAL	REQUIREMENT/GW(e) INSTALLED	REMARKS
^{238}U	59 582 kg/yr	recycled
^{235}U	91 kg/yr	recycled
Total U	59 680 kg/yr	recycled
^{239}Pu	4844 kg/yr	recycled
^{241}Pu	148 kg/yr	recycled
Total Pu	6754 kg/yr	recycled
MA	190 kg/yr	recycled

Design basis lifetime for reactor core, vessel and structures

The design life of all non-replaceable structures (including containment vessel, internal structures, and major piping and concrete structures) is 30 years.

Economics

The overnight capital cost of the first module of a 150 MW(e) KALIMER plant is estimated to be US $2300 /kW(e). However, cost saving of 30% to 40% is possible through the standardization and construction in series, multiple unit construction, design improvement, and improvement of the construction methods. By these cost saving measures, the overnight capital cost for KALIMER could be between US $1400 and US $1600/kW(e).

XX-1.5. Outline of fuel cycle options

A closed nuclear fuel cycle is to be used for the KALIMER. U, TRUs, and some fission products will be recovered from the spent fuel and recycled. A schematic of the KALIMER fuel cycle is shown in Fig. XX-2.

Pyro-processing is the most appropriate method for reprocessing of metal fuels. The process is relatively simple compared to a conventional aqueous process and has fewer steps, which could ensure economic benefits. In pyro-processing, plutonium is not separated from uranium, and most of the minor actinides are recovered [XX-3].

The fuel cycle facilities (fabrication and reprocessing) for KALIMER will be co-located with the reactor at the same site.

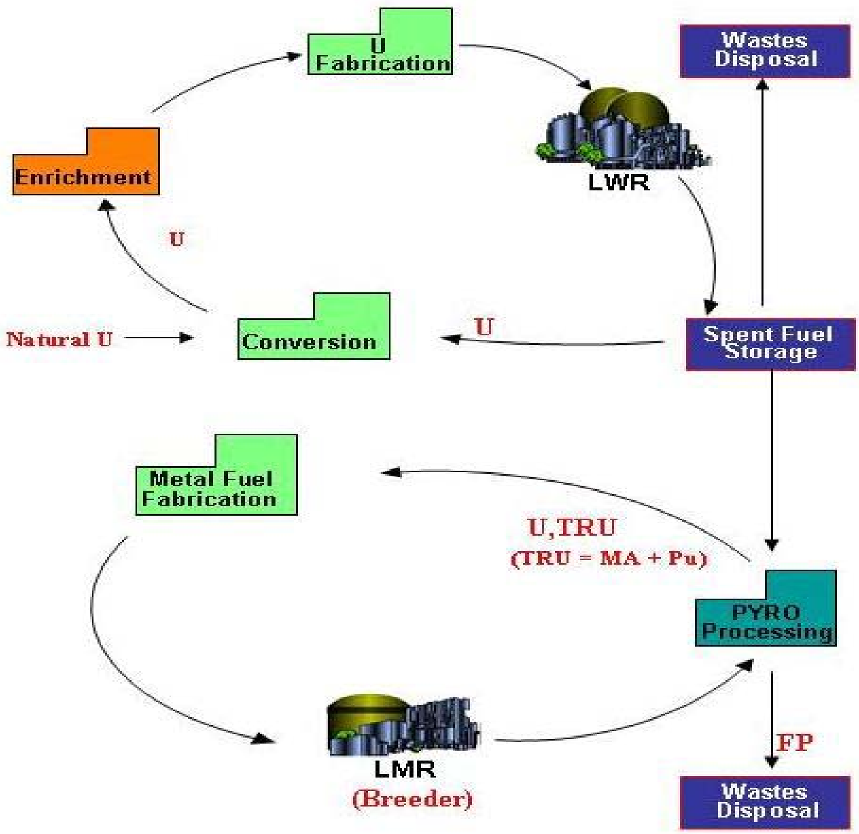


FIG. XX-2. Fuel cycle for KALIMER.

XX-1.6. Technical features and technological approaches that are definitive for KALIMER performance in particular areas

XX-1.6.1. Economics and maintainability

When standardized and deployed in series, the KALIMER could benefit from the following cost savings (relative to the overnight capital cost of the first module):

- Improved construction methods: maximum 5% reduction by modularization and reduced construction schedule.
- Design improvement : maximum 5% reduction by fully passive decay heat removal systems and simplification of design.
- Standardization and construction in series : maximum 15% reduction by standardizing the design, manufacturing, construction, licensing and operation approaches.
- Multiple unit construction : maximum 10% reduction by sharing common facilities.

XX-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

The KALIMER core configuration, which has an average breeding ratio of 1.05, is developed with the pursuit of no need for an external source of fissile material in an equilibrium fuel

cycle. The reprocessing and recycling of both fissile and fertile materials would facilitate low consumption of natural uranium.

The present KALIMER core design employs an integral fuel cycle strategy in which 99.9% of the actinides is recycled and 0.1% of the TRU is lost in waste streams during the metal fuel reprocessing. In addition, 5% of the rare-earth fission products are recycled but all other fission products are moved to the waste stream.

Compared to a conventional aqueous process, the pyro-process has fewer process steps, and the facility and equipment are much more compact, which contributes to the reduction of waste.

XX-1.6.3. Safety and reliability

Safety concept and design philosophy

The ultimate objectives for the KALIMER design are to make it safer, more economical, more resistant to nuclear proliferation, and producing less impact on the environment. The specific objectives for safety design include high accident resistance, core damage prevention, prevention of large radioactivity releases, and mitigation of accidents.

Active and passive systems and inherent safety features

Incorporation of several inherent and passive safety features, good thermal characteristics of the liquid metal coolant and metal fuel, and large coolant inventory are some of the important provisions for the simplicity and robustness of the KALIMER design.

The *active systems* include:

- Reactivity control and shutdown system.

The *passive systems* include:

- Passive safety-grade decay heat removal system (PSDRS).
- Gas expansion modules (GEMs).
- Ultimate shutdown system (USS).

The *inherent safety features* include:

- Negative power coefficient of reactivity and minimized burn-up reactivity swing.

Structure of the defence-in-depth

The KALIMER is designed in accordance with a defence-in-depth safety philosophy that provides for multiple fission product barriers to prevent the release of radioactivity to the environment, and for multiple levels of safety to protect these barriers and to reduce the consequences of their failure. The KALIMER has three barriers that physically prevent a release of the fission products to the outside environment. These are the fuel pin cladding, the primary vessel boundary, and the containment.

The first level of safety is based on inherent safety features and basic design characteristics. This level focuses on a reliable normal operation and on accident prevention through the implementation of certain features of the plant design, construction, operation and maintenance. It also includes reliability enhancement through the redundancy, vigorous quality assurance, testability, inspectability, and simplified fail-safe system designs. The second level limits the challenges to the above barriers recognizing that external challenges,

e.g. earthquakes, and component failures may occur during the plant design life. The third level is concerned with protection against anticipated and unlikely events. The fourth level limits the core damage at a given failure or late response of the reactor shutdown system. The fifth level focuses primarily on the extremely unlikely events, which may lead to core meltdown. This level ensures that the released fission products and molten fuel remain confined within the reactor vessel. The sixth level provides further protection to the public by ensuring that the off-site doses are sufficiently low to require any evacuation even under extreme assumptions that are beyond the design basis. The seventh level provides the overall protection that ensures that even if the previous levels of safety have failed, the risk to the public health and environment would remain acceptably low.

Design basis accidents and beyond design basis accidents

All conceivable challenges to safety systems that could prevent them from carrying on their safety functions are considered as candidates for the design basis events (DBEs). Specifically, the initial events of interest are those postulated sequences that challenge and have a definable impact on the design of the safety-related components or systems, which are associated with reactor shutdown, heat removal in a shutdown, and control of the radiological releases. Among these initial events, one or more may clearly dominate or envelope others, while some events could be eliminated because of their small probability of occurrence. A set of event categories corresponding to the events that must be considered in system design was defined. Each identified event is put into one of the four DBE categories or to the BDBE (beyond design basis event) category, using its nominal frequency as a criterion. The dividing line between DBEs and BDBEs is the frequency of 10^{-6} per reactor-year. The DBE categories defined for the purposes of safety analysis are:

- Normal operation or moderate frequency event (ME) − Any condition of system start-up, design range operations, hot standby, or shutdown (frequency ≥ 10^{-1} per reactor-year).
- Infrequent event (IE) − Off-normal condition that is expected to occur once or more during the plant lifetime (10^{-1}> frequency ≥10^{-2} per reactor-year).
- Unlikely event (UE) − Off-normal condition that is not expected to occur during plant life; however, when integrated over all components, these events may be expected to occur a number of times during the plant life (10^{-2}> frequency ≥10^{-4} per reactor-year).
- Extremely unlikely event (XU) − Off-normal conditions of such extremely low probability that no events in the category are identified as design basis (10^{-4} > frequency ≥10^{-7} per reactor-year).

The final category – XU – corresponds to those severe events that beyond the traditional DBE envelope. However, since reliabilities and performances of safety components are not well defined in all cases at present, such events are still considered for the purpose of conservatism. The identification and use of such event category is consistent with the NRC's severe accident policy statement. The XU category includes the events with a frequency down to 10^{-7} per plant-year. In selecting the events to be categorized as XU, the KALIMER design has to be specifically reviewed to identify those events that have a potential for large radioactivity release, core melting, or reactivity excursion, to ensure that an adequate prevention or protection is furnished for these events.

Some traditional BDBEs with the probability of less than 10^{-6}/yr are normally excluded from

direct consideration in conjunction with the design requirements. However, the KALIMER top-level requirements demand a subset of BDBEs, namely the bounding events (BEs), to be accommodated without a loss of the reactor integrity and without a radiological release. This requirement complies with the 1986' advanced reactor policy statement of the U.S. NRC. Although the selected bounding events are not rigorously quantified in terms of probability, a judgment is made that their probability could reasonably be in the lower range of the XU ($\sim 10^{-7}$/yr). There are several reasons to include a set of bounding events in the XU category for the KALIMER design. The uncertainties incorporated in the design are not defined. Also, there is a difficulty in identifying the reliability for all failure modes of a system or a component, particularly at the conceptual design stage. The uncertainty that affects the event selection is mainly due to the limited data on the performance and reliability of the critical systems, such as the PSDRS or a negative reactivity feedback mechanism.

The only BDBEs considered for the KALIMER are hypothetical core disruptive accidents (HCDAs). The probability of such accidents is less than 10^{-7} plant-year, which is so low that such events fall into the residual risk classification. However, these events may have the potential consequences that would justify their consideration in the design.

Figure XX-3 summarizes the categorization of DBEs and BDBEs for the KALIMER. This figure also gives the lists of events for each category. Figures XX-4 and XX-5 show the system response for two typical unprotected events: the UTOP, and the unprotected loss of flow (ULOF). During a UTOP transient, the radial expansion adds a crucial amount of the negative reactivity that eventually limits the power increase to 1.48 times of the rated power and contributes to the subsequent power reduction. Figure XX-5 presents the power performance during a ULOF. The power drops first and then reaches the decay heat level in about 60 seconds, since there is enough negative reactivity due to the operation of the GEMs. The analysis of these results indicate that the KALIMER response to the anticipated transient without scram (ATWS) is benign because of the adequate reactivity feedback characteristics.

Figure XX-6 presents the release of the mechanical work energy per unit of fuel mass arising from sodium expansion during the super-prompt critical power excursion in the KALIMER. The analyses were performed with the finite heat transfer rate model as realized in the SOCOOL− II code [XX-4]. The figure shows the work energy densities for the fuel particle diameters of 0.1 cm, 0.5 cm, and 1.0 cm, respectively, as a function of sodium mass fraction during a thermal interaction of the liquid fuel at 3430°C with the sodium at 530°C. It can be seen that, as the fuel diameter gets larger, the work energy potentials rapidly decrease and are saturated with a lesser amount of sodium per unit mass of fuel. For a typical size of the fuel particle (1.0 cm in diameter), the work energy reaches its maximum, 10.7 J/g of fuel, when the mass of sodium per unit mass of fuel is 0.06. Since the total mass of the reference core is about 8.4 MT, the total energy release would amount to approximately 90 MJ. This value is far less than the structural design criterium, which is 500 MJ for the KALIMER reactor system.

Safety under seismic conditions

With its safety-related structures, systems, and components, the KALIMER is designed to accommodate seismic loadings produced by the Operating Basis Earthquake (OBE) with a zero-period acceleration (ZPA) of 0.15 g and the safe shutdown earthquake (SSE) having a ZPA of 0.30 g. The design response spectra described in the guide of the Korea Institute of Nuclear Safety (KINS) or, equivalently, in the US NRC Regulatory Guide 1.60, and scaled to the appropriate ZPA values were used in the KALIMER design.

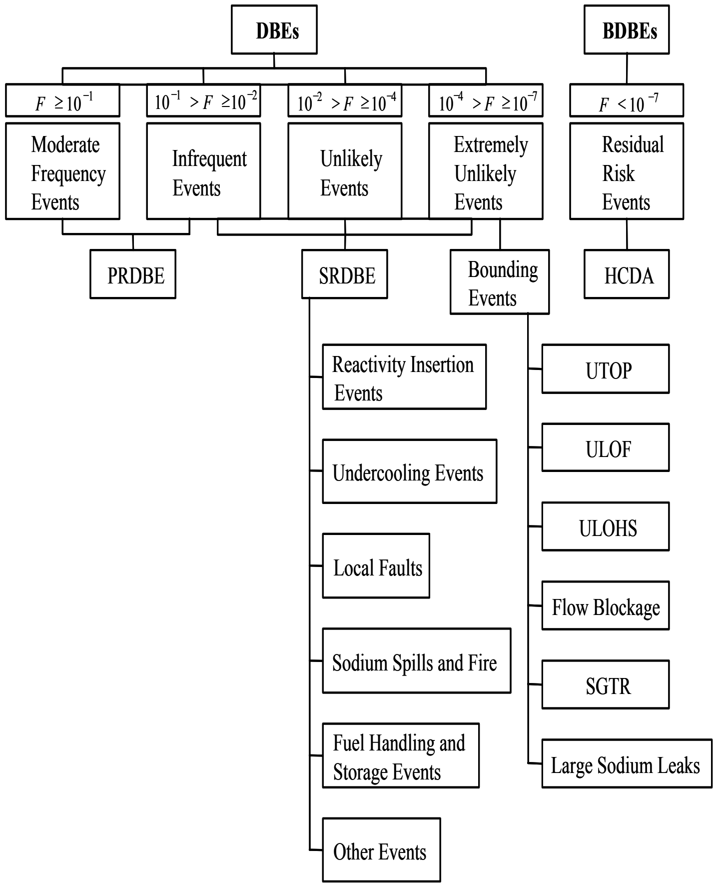


FIG. XX-3. KALIMER event categorization.

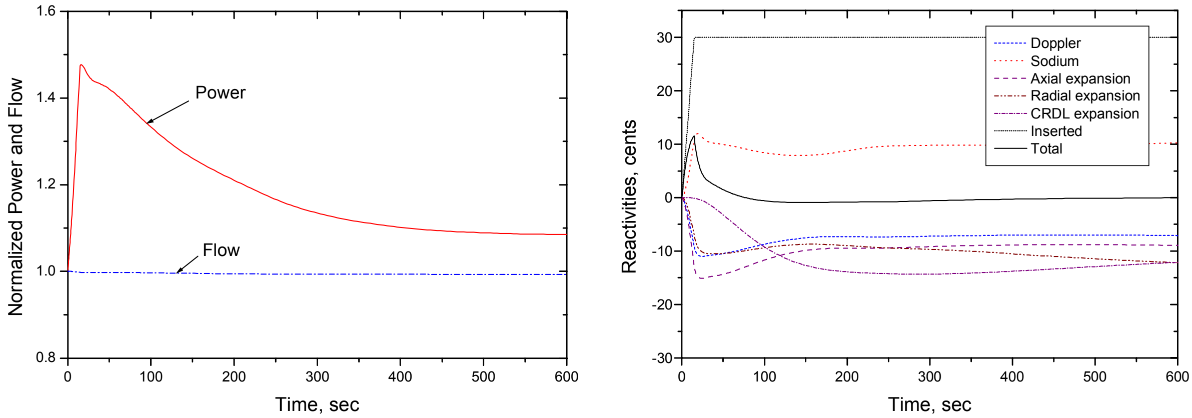


FIG. XX-4. System performance during a UTOP in equilibrium core.

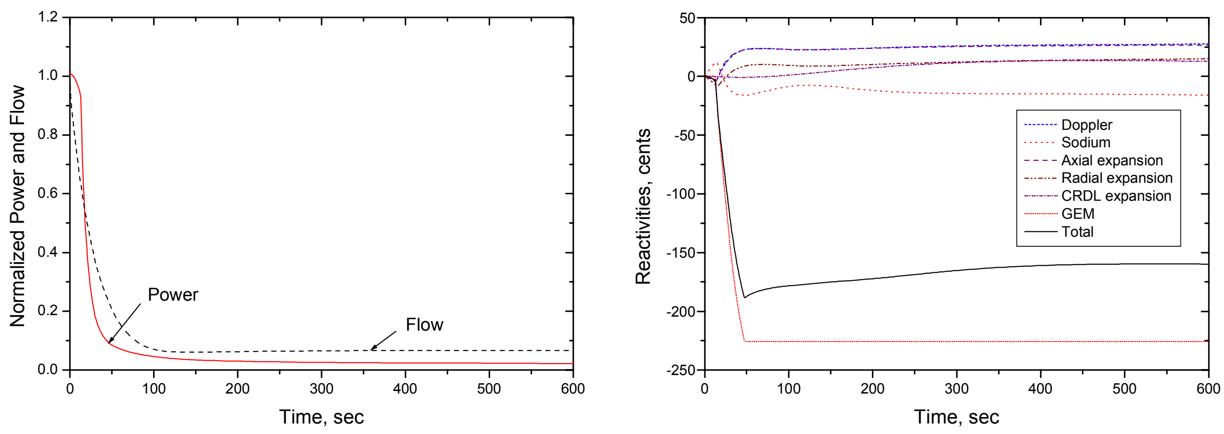


FIG. XX-5. System performance during a ULOF in equilibrium core.

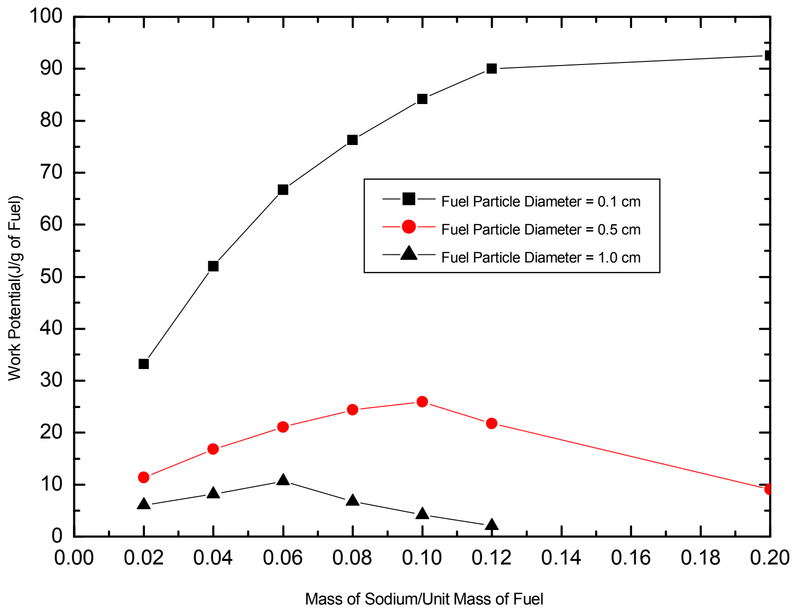


FIG. XX-6. Work energy release from sodium expansion.

Probability of unacceptable radioactivity release beyond the plant boundaries

It is expected that the probability of unacceptable radioactivity release beyond the plant boundary would be less than 1×10^{-7} per reactor-year.

Measures planned in response to severe accidents

An important design objective for the KALIMER is to enhance the level of safety to eliminate the need for any intervention in the public domain beyond the plant boundary as a consequence of any hypothetical core disruption accident within the plant. To reduce the core damage probability, a sufficient margin in the fuel and core design is provided. The negative power reactivity coefficient is also crucial in preventing the core damage.

XX-1.6.4. Proliferation resistance

The important technical features that reduce the attractiveness of the KALIMER spent nuclear fuel use in any clandestine nuclear weapons programme are as follows:

- Inherently low decontamination factor of the product, which makes it highly radioactive, and
- No separation of Pu at any stage of the fuel cycle.

The two factors indicated above also contribute to the prevention and discouragement of the diversion of the KALIMER nuclear materials. Any attempts to alter the KALIMER fuel cycle to produce plutonium with an isotopic content suitable for weapons material would require frequent refuelling and the use of fuel loadings that differ substantially from the normal fuel loadings, which would result in noticeable and readily detectable changes in the neutronic properties and operation schedule. Safety and operational requirements of the KALIMER will require strict compliance with the requirements for fuel handling and placement in the core to ensure that core power and temperature distributions are within the limiting conditions assumed in the safety analysis. The fuel accountability and inspection requirements would preclude the major alterations in the fuel cycle aiming to produce the desired plutonium isotopic content, and would also preclude the diversion of spent fuel for clandestine reprocessing.

XX-1.6.5. Technical features and technological approaches used to facilitate physical protection of KALIMER

Locating the containment vessel underground provides a significant barrier to the reactor damage that could arise out of the external impacts.

XX-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of KALIMER

Future nuclear power plants face several challenges related to the provision of effective resource utilization, minimized waste and reduced environmental impacts, economic competitiveness, enhanced safety and reliability, and adequate proliferation resistance and physical protection.

In order to meet these challenges the Ministry of Science and Technology (MOST) established the comprehensive nuclear energy promotion plan of 2001 which sets the basic framework of R&D programmes in the nuclear industry. According to the plan, it is suggested to develop liquid metal reactor technologies for the efficient utilization of uranium resources with an emphasis on basic key technologies. It is also suggested to participate in international collaborations including the Generation IV programme. Furthermore, as a result of the nuclear technology roadmap activities in the Republic of Korea, sodium-cooled fast reactor was chosen as one of the two future reactor options deployable by 2030.

XX-1.8. List of enabling technologies relevant to KALIMER and status of their development

The enabling technologies for the KALIMER plant are summarized in Table XX-6.

TABLE XX-6. ENABLING TECHNOLOGIES FOR KALIMER BASED NUCLEAR POWER PLANT

MAIN OBJECTIVES	ENABLING TECHNOLOGIES	STATUS OF DEVELOPMENT
Use of Metal Fuel.	Remotized fuel fabrication technologies	R&D in progress.
	Dry reprocessing of fuel.	Laboratory level process development in progress.
Negative power coefficient.	Use of metal fuel.	Neutron-physical experiments performed in experimental facility.
Optimum use of passive systems for core heat removal.	Passive decay heat removal system.	R&D in progress.
Increased burn-up of fuel	Duplex cladding.	No R&D needed within the KALIMER programme..
	Provision for on-line refuelling.	Laboratory level process development done.
Enhanced safety following ATWS.	Gas expansion modules.	Experiment for the response time required.
	Use of metallic fuel.	Ongoing experimental programme.
Additional features to achieve low core damage frequency	Negative reactivity coefficients.	R&D for detailed reactivity analysis models ongoing.

XX-1.9. Status of R&D and planned schedule

The R&D for KALIMER are fully supported by the Government of the Republic of Korea. The design and development of this reactor are performed by the KAERI.

During phases 1 and 2 (1997–2001) of the national fast reactor programme, the basic technologies and conceptual design of a 150 MW(e) KALIMER–150 have been developed. Since there is no plan for the construction of a prototype or a demonstration reactor, the basic key technologies and the advanced concept of a 600 MW(e) KALIMER–600 have been developed during phase 3, in 2002–2004. The discussions on future activities of the programme were on-going and the final plan had to be established by December, 2004.

The conceptual design of the KALIMER–150 has been completed in 2001. The key design technologies are being developed currently.

XX-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

The principal innovations of the KALIMER nuclear energy system are related to the front-end and back-end of its fuel cycle and to several passive systems proposed for the reactor design. While the design is being qualified through theoretical and experimental works, these innovations must be demonstrated for the first time in the first-of-a-kind prototype plant.

XX-1.11. List of other similar or relevant SMRs for which the design activities are on-going

Based on the success of the previous DOE sponsored advanced liquid metal reactor (ALMR) programme, the General Electric has continued to develop and assess the technical viability

and economic potential of a follow-on fast reactor called Super PRISM (S–PRISM) [XX-5]. The S–PRISM is an advanced fast reactor plant design that utilizes compact modular pool-type reactors sized to enable factory fabrication and an affordable prototype test of a single nuclear steam supply system (NSSS).

The JNC sodium cooled fast reactor (JSFR) [XX-5] is a sodium cooled, MOX (or metal) fuelled advanced loop type design that evolves from the Japanese fast reactor technologies and experience. The plant size is ranging from a modular system composed of medium sized reactors to a large monolithic design. The large-scale sodium cooled reactor benefits from economies of scale by setting the electricity output at 1500 MW(e). On the other hand, the medium sized modular reactor (modular type JNC sodium fast reactor – M–JSFR) [XX-6] offers an advantage of flexibility in meeting the varying levels of power requirements from the utility companies, and ensures the reduction of development risk as compared to large-scale reactors. The JSFR employs several advanced technologies to reduce the construction cost. These are the ‘compact design of reactor structures’, the ‘shortened piping layout’, the ‘reduction of the number of loops’, the ‘integration of components’, and the ‘simplification of decay heat removal system through the enhancement of its natural circulation capability’.

Major specifications of the large-scale JSFR and S-PRISM are given in Table XX-7.

TABLE XX-7. MAJOR SPECIFICATIONS OF JSFR & S-PRISM

Items	Specifications	
	JSFR	S-PRISM
Electricity output, MW(e)	1500	760
Thermal output, MW(th)	3570	1000×2
Primary sodium temperature, °C	550/395	510/371
Secondary sodium temperature, °C	520/335	496/321
Main steam temperature / pressure, °C /MPa	495/16.67	468/17.3
Plant efficiency, %	~42	38
Fuel type	Oxide/Metal	Oxide/Metal

XX-2. Design description and data for KALIMER

XX-2.1. Description of the nuclear systems

Reactor core and fuel design

The KALIMER core utilizes a radially heterogeneous configuration that comprises annular rings of internal blankets and driver fuel assemblies as shown in Fig. XX-7. This heterogeneous core configuration reduces positive sodium void worth by utilizing (in)elastic scattering and capture reactions on ^{238}U contained in the internal blankets. There are no upper or lower axial blankets surrounding the core, which also contributes to the sodium void worth reduction by increasing the axial leakage. The driver fuel assembly includes 271 pins. The pins contain columns of U-Pu-Zr alloy. Figure XX-8 shows an overview of the fuel rod design used in the KALIMER. Figure XX-9 shows the KALIMER fuel assembly.

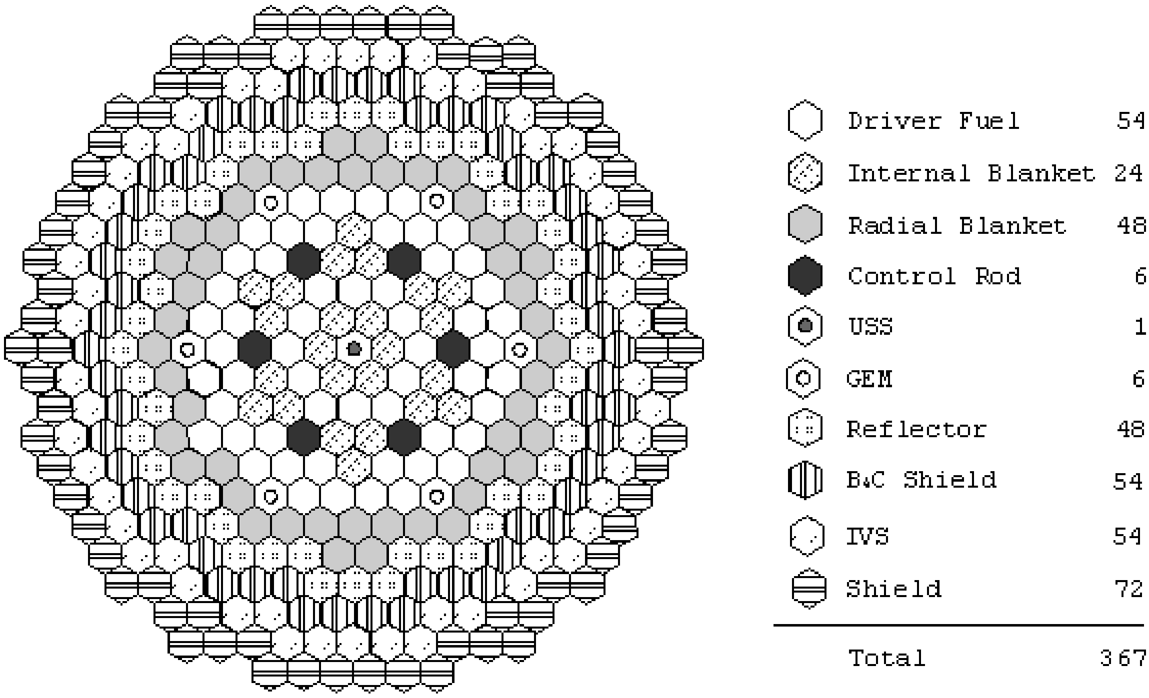


FIG. XX-7. KALIMER core map.

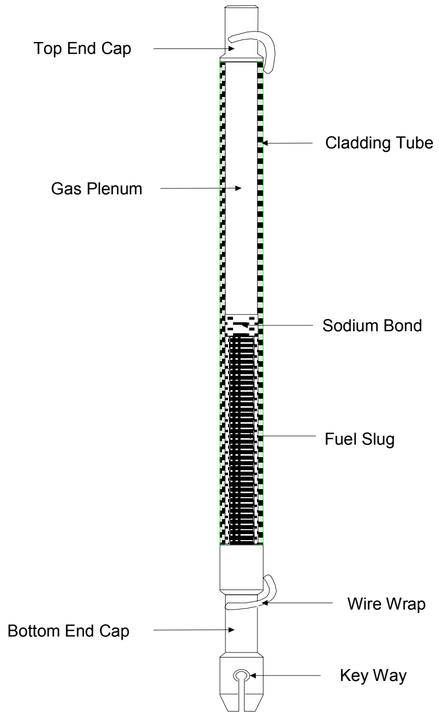


FIG. XX-8. Schematic of the KALIMER driver fuel pin.

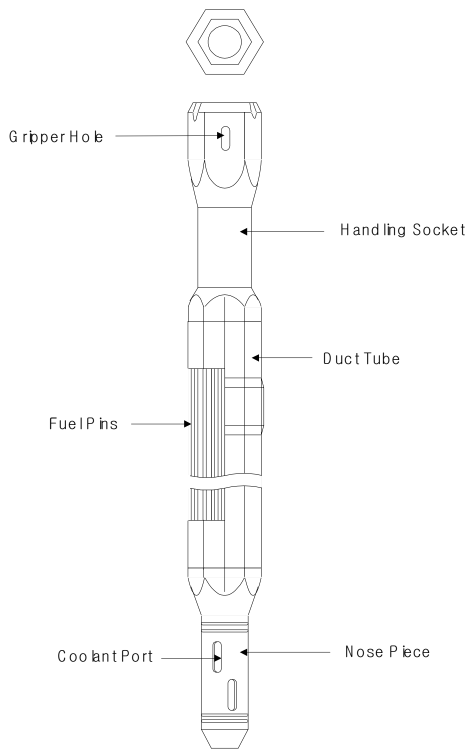


FIG. XX-9. KALIMER fuel assembly.

Main heat transport system

The KALIMER heat transport system is composed of a primary heat transport system (PHTS), an intermediate heat transport system (IHTS) and a steam generation system (SGS). The PHTS is a pool type system and this feature provides a large thermal inertia of the primary system, which enhances the plant safety. The function of the PHTS is to deliver core heat to

the SGS through the IHTS via the intermediate heat exchanger (IHX) submerged in a reactor pool. In order to enhance plant safety, a passive decay heat removal system, PSDRS, is adopted for the cases of accidents. The PSDRS is operated without active components and any operator action within 72 hours after reactor shutdown.

During power operation, the heat generated in the core is removed by the normal heat transport path, which is PHTS–IHTS–SGS. When the reactor is shutdown, the decay heat removal is normally accomplished through a path that is the same as that for normal operation, except that the steam bypasses the turbine and is dumped to the condenser or atmosphere until the system temperature is reduced down to a refuelling temperature. After reaching the refuelling temperature, the steam generator auxiliary cooling system (SGACS) removes the decay heat and the PSDRS just supplements the heat removal. When the normal means of decay heat removal are not available during an abnormal plant condition, then the safety grade PSDRS is used for heat removal. Figure XX-10 shows a simplified diagram of the main heat removal paths for the KALIMER in normal operation and in accident conditions.

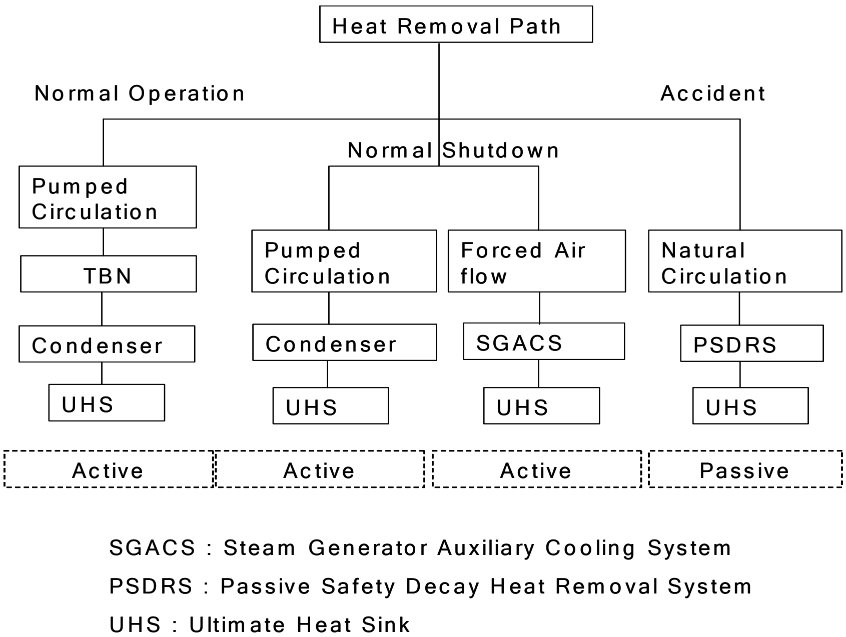


FIG. XX-10. Diagram of heat removal paths for KALIMER.

Intermediate circuit

The IHTS consists of two loops, and each loop has its own steam generator and related systems. This enhances the plant operation flexibility and safety. A focus has been made on the prevention and mitigation of possible sodium-water reactions by specially adjusting the IHTS piping routes and the SG design. The system reliability is improved by using the electromagnetic (EM) pumps, which have no moving parts, for both primary and intermediate coolant pumping. The low momentum of inertia of a EM pump is compensated by a special flow inertia device. This device stores the kinetic energy of rotation when the EM pump runs normally. In case of a pump power supply failure, the device converts the stored kinetic energy of rotation to electricity and supplies it to the EM pump.

Steam generation system

In the SGS, a superheated steam cycle is implemented to ensure high plant efficiency. It should be noted that higher thermal efficiency reduces the heat discharge from the plant, resulting in less impact to the environment.

The operating temperature and component dimensions were optimized to achieve the plant net thermal efficiency of 38.2%. The feedwater pumps are shared by the two identical steam generation systems, while each one is equipped with its own main and auxiliary flow control valves, as shown in Fig. XX-11. This feature secures an optimum use of plant resources and introduces better plant economy and operational flexibility.

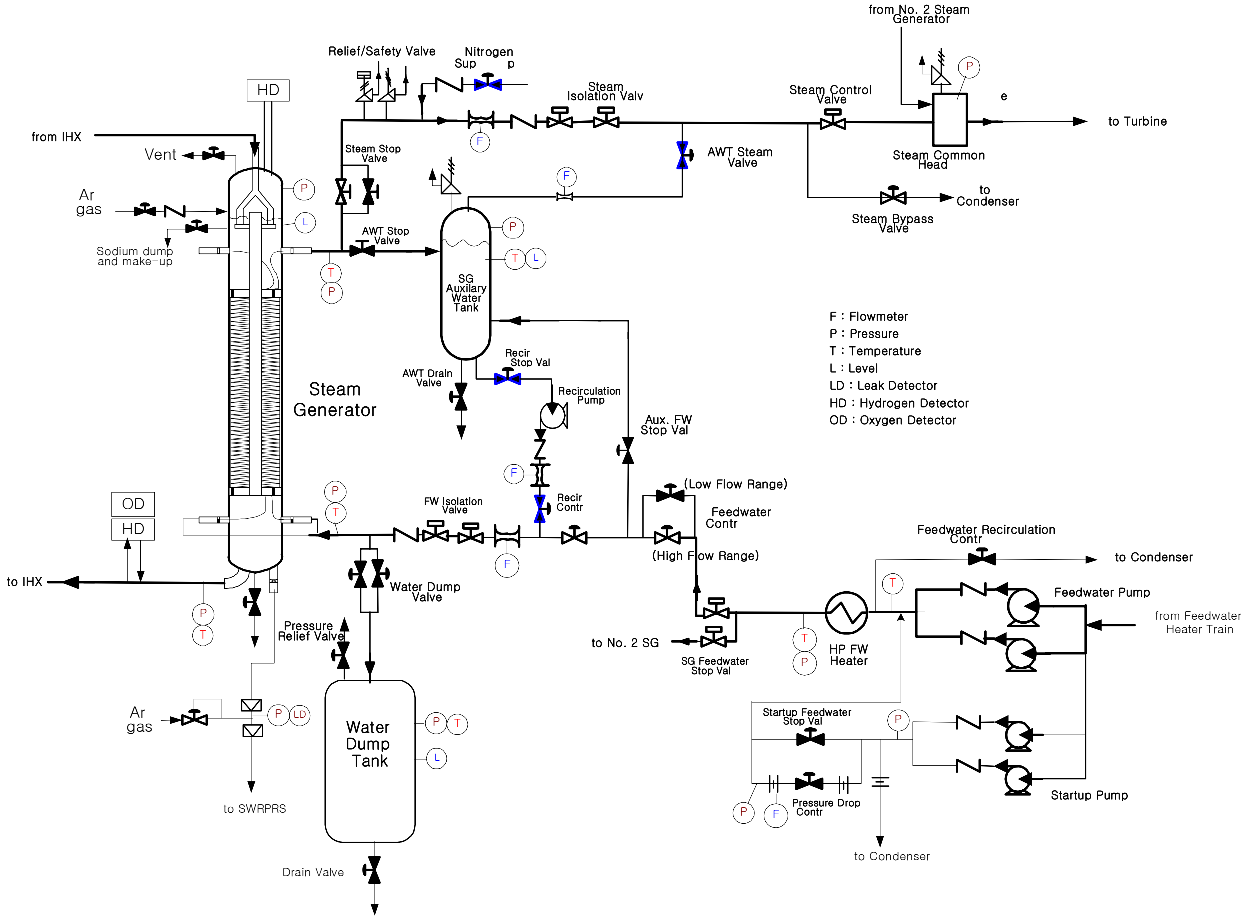


FIG. XX-11. Steam generation system arrangements of KALIMER.

XX-2.2. Description of the turbine generator plant and systems

Commercially available equipment will be used in the turbine generator plant.

XX-2.3. Systems for non-electric applications

Non-electrical applications of the KALIMER are not foreseen at the moment.

XX-2.4. Plant layout

The KALIMER plant layout is optimized to meet various functional needs related to the safety, radiation zoning, piping and cabling requirements, the erection and construction requirements, and the access and security considerations.

Floor plans of various buildings have been optimized based on the equipment and systems within, functional needs, space utilization, radiation zoning, accessibility, serviceability, maintenance, transportation and ventilation aspects, and the access philosophy.

Reactor building and containment layout

The reactor structures, shown in Fig. XX-12, include the containment vessel, the reactor vessel, the reactor head, intermediate heat exchangers (IHXs), electromagnetic pumps (EMPs), control rod drives, the reactor internal structures, the reactor support structure, and the IHTS piping. The reactor vessel provides a boundary for the PHTS and also contains and supports it during all temperature, pressure and load variations that occur during the operation lifetime. The KALIMER internal structures include the core support structure, the inlet plenum, the support barrel, the reactor baffle, the reactor baffle plate, the separation plate, the flow guide, the EMP nozzle, the inlet pipe, and the radiation shield structures. The reactor head supports IHXs, the EMPs, the reactor vessel, the thermal shield structures, and the refuelling equipment.

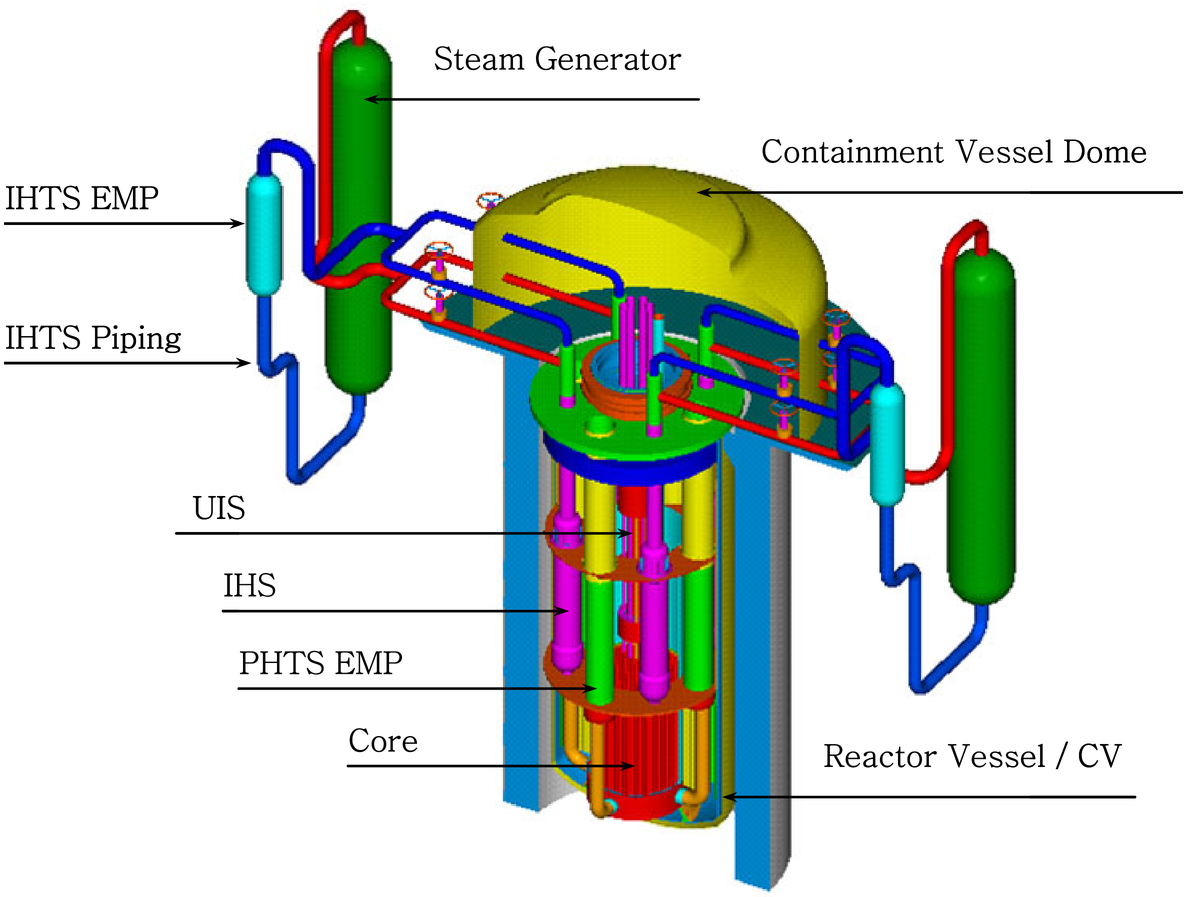


FIG. XX-12. KALIMER reactor structures (conceptual design).

The internal structures have 3 main functions, which are to provide (1) core support, (2) primary coolant flow path, and (3) component support. The internal structures are designed to meet these functional requirements, Fig. XX-12.

The containment vessel provides a low leakage, pressure retaining boundary which completely surrounds the primary system boundary. The containment includes a containment vessel (lower part) (to contain leaks from the reactor vessel) and a containment dome (upper part), which is to mitigate the releases due to postulated severe accidents. The upper and lower parts of the containment are connected to each other by a containment ring located at the same elevation as the top of the reactor head.

An overview of the KALIMER plant is shown in Fig. XX-13. The reactor building, which adopts a seismic base isolation system to improve its structural safety as well as the economics, is separated from both the fuel handling & storage building and the turbine generator building. For a refuelling, the shielded fuel transfer cask moves 6 core assemblies between the reactor and the fuel handling & storage building. Other facilities include control building, maintenance building, and service building. During conceptual design stage, the focus was made on the design of a nuclear steam supply system (NSSS) located inside the reactor building and on selected features of the balance of plant (BOP), essential for the reactor safety.

The reactor building arrangement provides for the NSSS interfaces with the reactor refuelling system, the PSDRS, the IHTS, and the seismic isolation system. The key feature of the reactor building arrangement is that the reactor structures related to all major systems, structures, and components are installed in a single building supported by the horizontal seismic base isolation system at 9 m below the grade level. The building serves to the PSDRS by having four airway stacks that protrude from the building top to the reactor bottom. The arrangement makes provision for an optimum layout of the IHTS piping to accommodate stresses that could occur in an unlikely sodium-water reaction and in other operating transients. The dimensions of the reactor building are 39 m(W) × 52 m(L) × 56.8 m(H). The SGs are located at both sides of the reactor and the SG dump tanks are just below the SGs. Under the dump tank, there is an insulated catch pan with the suppression deck that collects spilled sodium. A conceptual drawing of the reactor building is given in Fig. XX-13.

The buildings and structures comprising the KALIMER plant as shown in Fig. XX-14 are divided into two categories: safety-related and BOP. The safety related buildings and structures include the reactor building, the fuel handling and storage building, the radioactive materials handling building, and the health physics building. They are located within the high security boundary. The BOP buildings and structures include the control building, the turbine generator building, the maintenance building, a warehouse, a pump house, condensate storage tanks, transformers, switchyard, etc.

The fuel handling and storage building is designed to provide a refuelling path for the core assemblies, Fig. XX-14. The turbine generator building houses the turbine generator system, the feedwater system, the steam and condensate system, the main steam system and related systems. The section and plan views of the fuel handling and storage building, reactor building, and turbine generator building are given in Fig. XX-15 and XX-16.

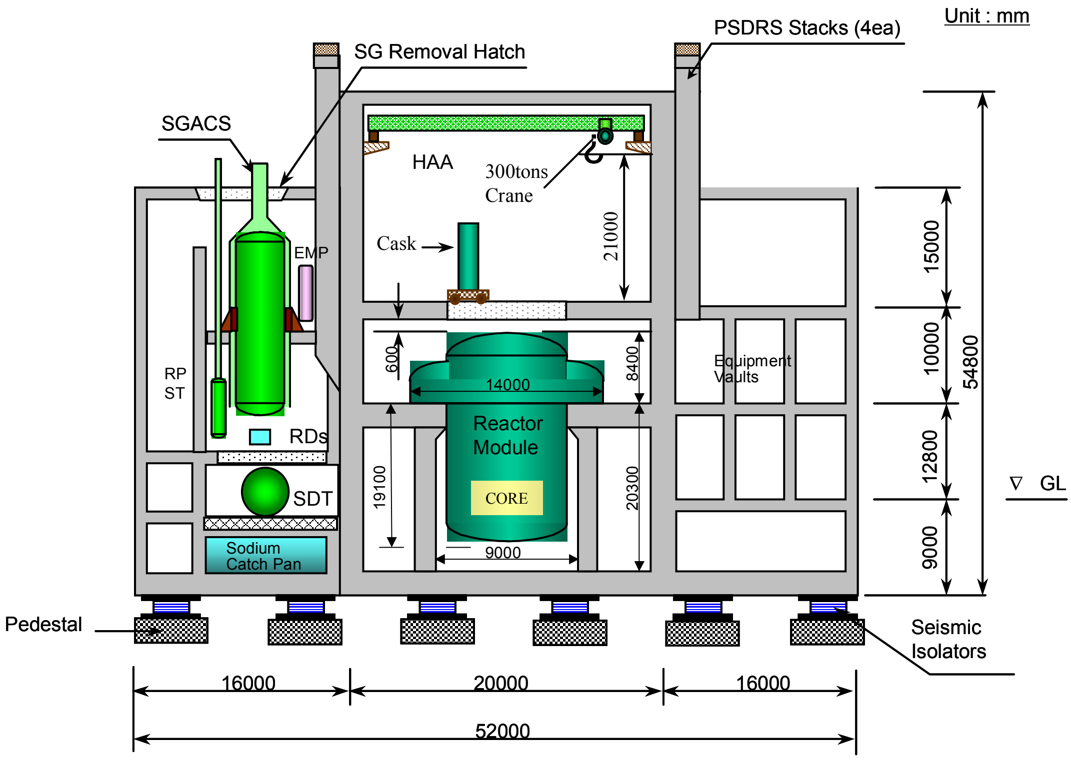
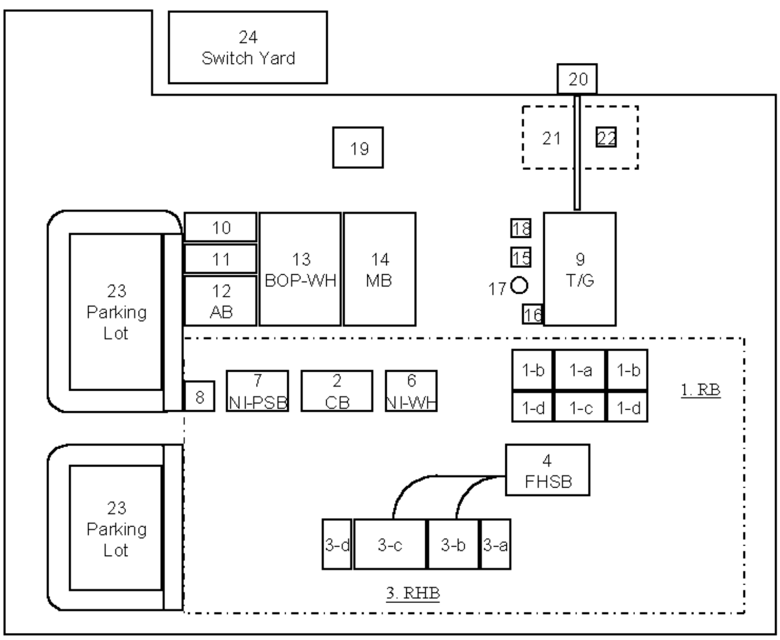


FIG. XX-13. Conceptual drawing of KALIMER building.



1. Reactor Building:
   a. Reactor Facility
   b. Steam Generator Facility
   c. Electrical Equipment Vault
   d. Sodium Purification Vault/
      Inert Gas Storage
2. Control Building
3. Radioactive Handling Building
   a. Utility Facility
   b. Radioactive Waste Facility
   c. Reactor Maintance Facility
   d. Remote Shutdown Facility
4. Fuel Handling and Storage Building
9. Turbine Generator
13. BOP Warehouse
14. Maintenance Building
16. Circulating Water Pump House
17. Condensate Storage Tanks
18. Transformers
20,21,22 Condenser Cooling Water
Related Facility
24. Switch Yard

FIG. XX-14. Overview of KALIMER plant.

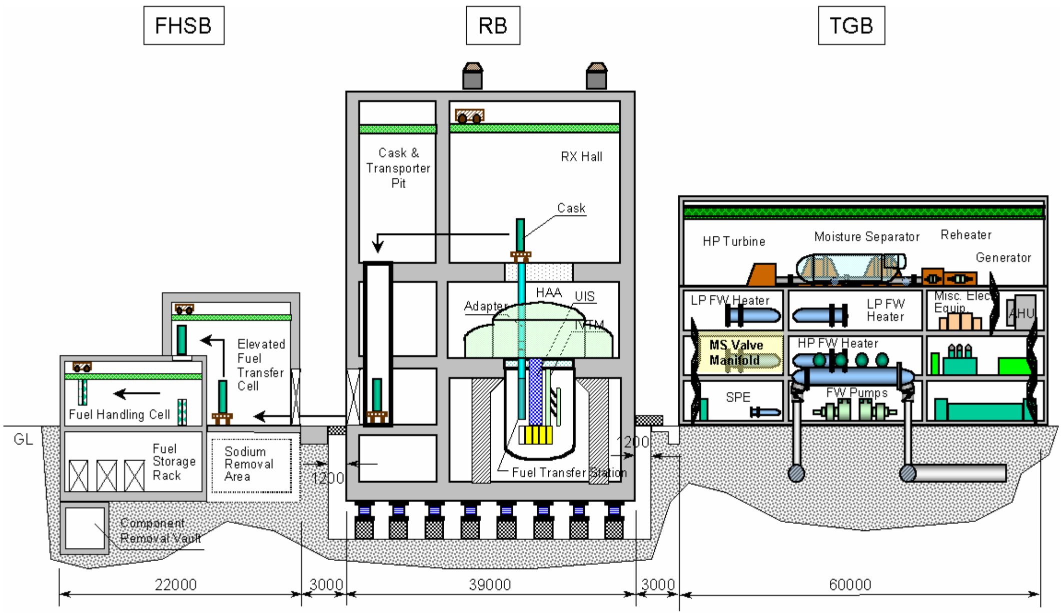


FIG. XX-15. Section view of the fuel handling and storage building, reactor building, and turbine generator building.

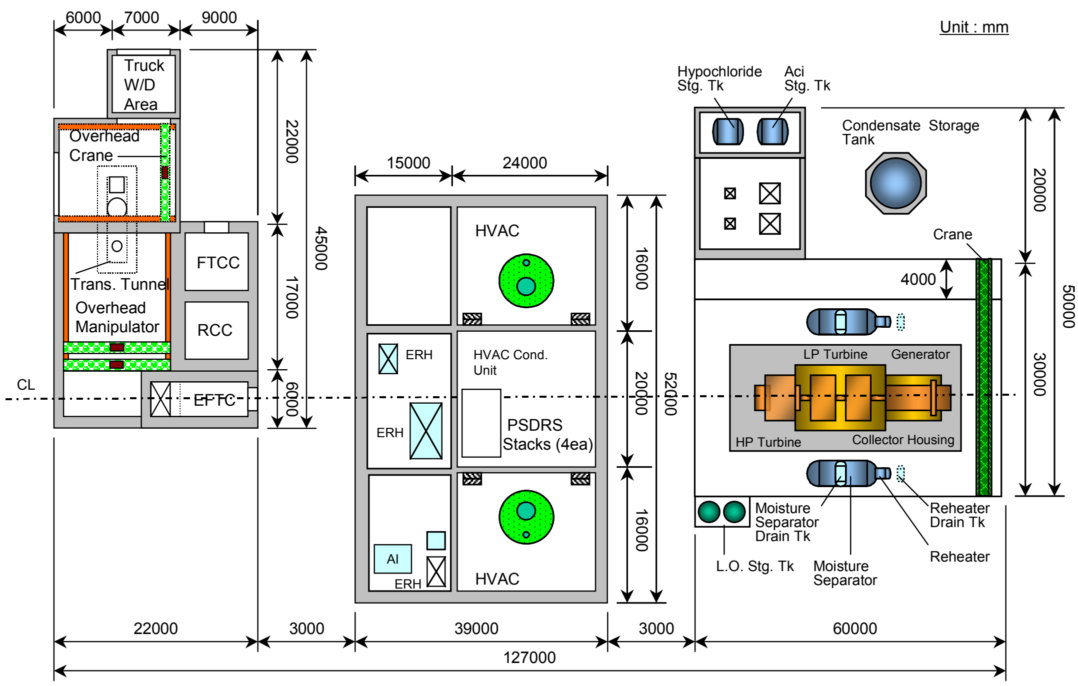


FIG. XX-16. Plan view of the fuel handling and storage building, reactor building, and turbine generator building.

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ANNEX XXI

MODULAR PLANT WITH SODIUM COOLED FAST REACTOR (BMN-170)

OKBM, Russian Federation

XXI-1. General information, technical features and operating characteristics

XXI-1.1. Introduction

The BMN-170 is a modular nuclear power plant (NPP) with a sodium cooled fast reactor. The name reflects the main engineering bases of the concept: a fast spectrum of neutrons, metallic (sodium) coolant, NPP mounting from factory-built equipment modules and a nominal value of electric power.

The BMN-170 preliminary design is based upon documents of a modular NPP conceptual design produced using the national experience in design and operation of sodium cooled fast reactors and documented accounts of the world tendencies in nuclear power development.

The BMN-170 preliminary design was developed in the 1990s by Russian design and scientific organizations, such as OKB Mechanical Engineering (OKBM), Sankt Peterburg (SPb), Atomenergoproekt (AEP) and Institute of Physics and Power Engineering (IPPE); the concept was shaped up based on experience in the development and operation of BOR-60, BN-350, BN-600, and BN-800 NPPs [XXI-1, XXI-2, XXI-3, and XXI-4].

The main goal was to establish a design for multi-purpose NPPs based on a modular sodium cooled fast reactor in support of economic, reliable and safe solutions to the issue of supply of power and heat to consumers.

The BMN-170 design is based on the following technical solutions experimentally validated and proven during the operation of NPPs with fast sodium cooled reactors BOR-60, BN-350, BN-600 and during development of the NPP design for the BN-800 reactor:

- Coolant technology and structural materials developed for fast sodium cooled reactors.
- Integral design of the primary circuit.
- Three-circuit thermal diagram of the NPP.
- Design of fuel elements, fuel assemblies and control rods.
- Passive means of reactor emergency shutdown.
- Refuelling technology.

At the stage of BMN-170 conceptual design development, several options of core arrangement and reactor module design were considered to enhance safety and economic effectiveness and to search for optimum solutions. One of main factors addressed during development was the use of reactor vessel and main NPP equipment transport by water and trailers.

A number of measures were considered to minimize void reactivity effects, including the insertion of sodium interlayers between the core and the upper axial blanket.

This document specifies data for the modular concept of an NPP (defined as BMN-170) for which some proven methods of optimizing safety effectiveness of fast sodium cooled reactors are implemented [XXI-3].

XXI-1.2. Applications

It is supposed that NPP with BMN-170 along with electricity generation in the basic mode can produce nuclear fuel and produce industrial heat for desalination or district heating, e.g.:

- To generate power of 170 MW(e).
- To generate power of 75 MW(e) and steam of 630 t/h.
- To generate power of 100 MW(e) and heat of 260 GCal/h.

XXI-1.3. Special features

The BMN-170 design is characterized by the following main innovative solutions aimed at obtaining high economic indices and providing high safety levels:

- Application of a modular design of the main NPP equipment.
- The possibility of using both oxide, nitride and metallic fuel.
- Application of passive residual heat removal systems.
- Application of passive reactor shutdown systems.

The BMN-170 concept opens the possibility of using these plants in multi-component structures suitable for nuclear power. The main tasks for a fast reactor are to provide effective closing of the fuel cycle for U and Pu and to expand breeding of nuclear fuel.

The direct purpose of the BMN-170 is the economically effective generation of electricity or co-generation of heat and power in autonomous power systems.

XXI-1.4. Summary of major design and operating characteristics

A NPP with the BMN-170 (Fig. XXI- 1) includes one integral modular reactor, and two-loop intermediate and steam-water circuits. Each loop of the intermediate circuit includes an intermediate heat exchanger, a steam generator (SG), a circulation pump of the secondary circuit and pipelines.

The reactor is located in the cavity. Its concrete walls are covered with a leak-tight metallic liner. A cavity liner above the reactor cover becomes a leak-tight shell under which control rod drives, drives of the primary circuit pumps and in-reactor refuelling mechanisms are located. Secondary circuit steam generators and pipelines of each loop are arranged in individual rooms.

Ex-reactor refuelling is carried out by the loading-discharging machine, which moves by rails above the reactor leak-tight shell and can be connected with the refuelling penetration in the reactor cover.

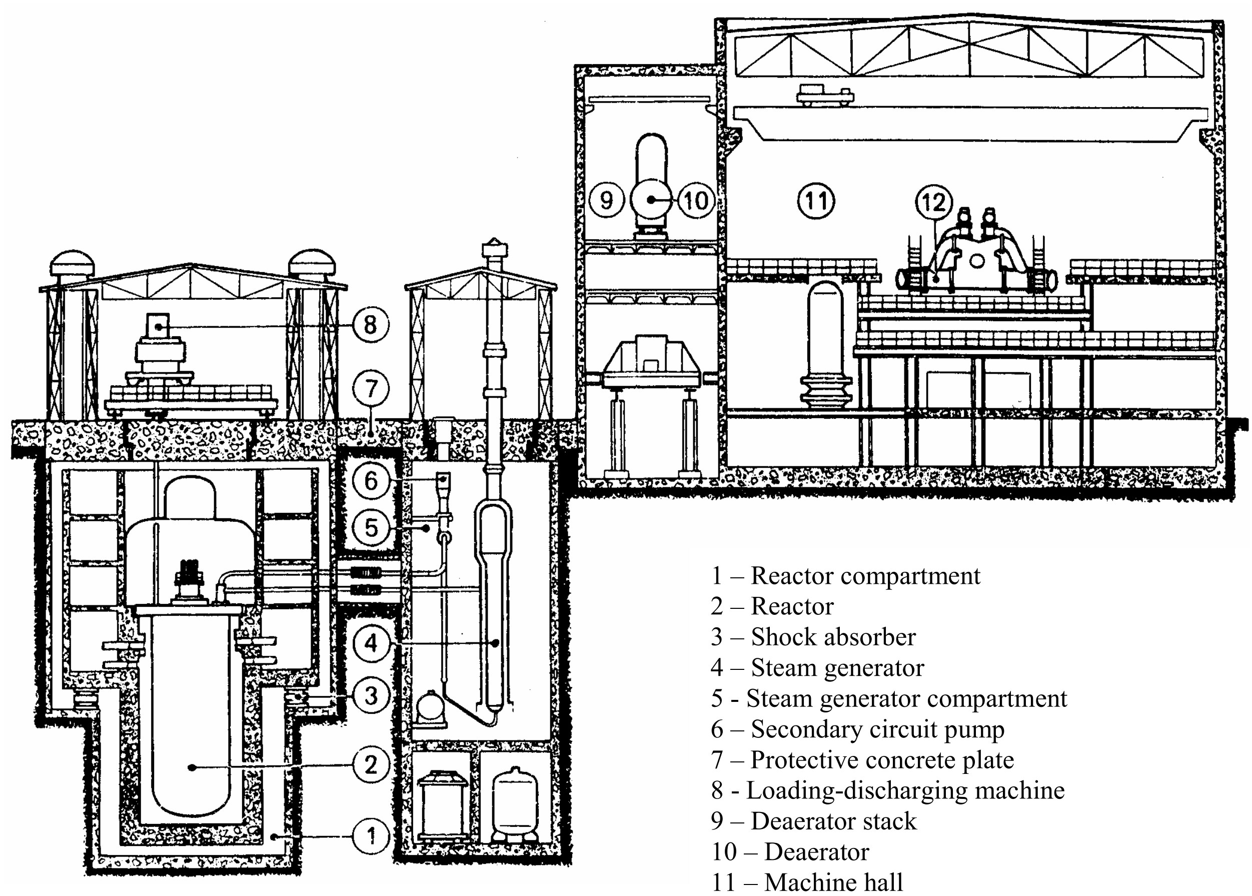
The plant emergency cooldown system is intended to remove residual heat released from the core to the ultimate sink (air), in case of such initiating events as NPP blackout or loss of feedwater supply to the steam generators. The emergency cooldown system is a safety system and consists of passive elements eliminating action by operating personnel or actuation of automatic machinery to put them into operation.

Residual heat is removed from the reactor through two channels:

- Through the reactor vessel by atmospheric air in natural draught, which passes through the reactor cavity cooling channel (air at the downcomer sector passes through the gap between the cavity liner and the separating shell in the reactor cavity, thus preventing the cavity concrete from overheating; air in the riser sector

passes between the separating shell and reactor safety vessel, thus removing heat, and is rejected into the atmosphere through exhaust tubes);

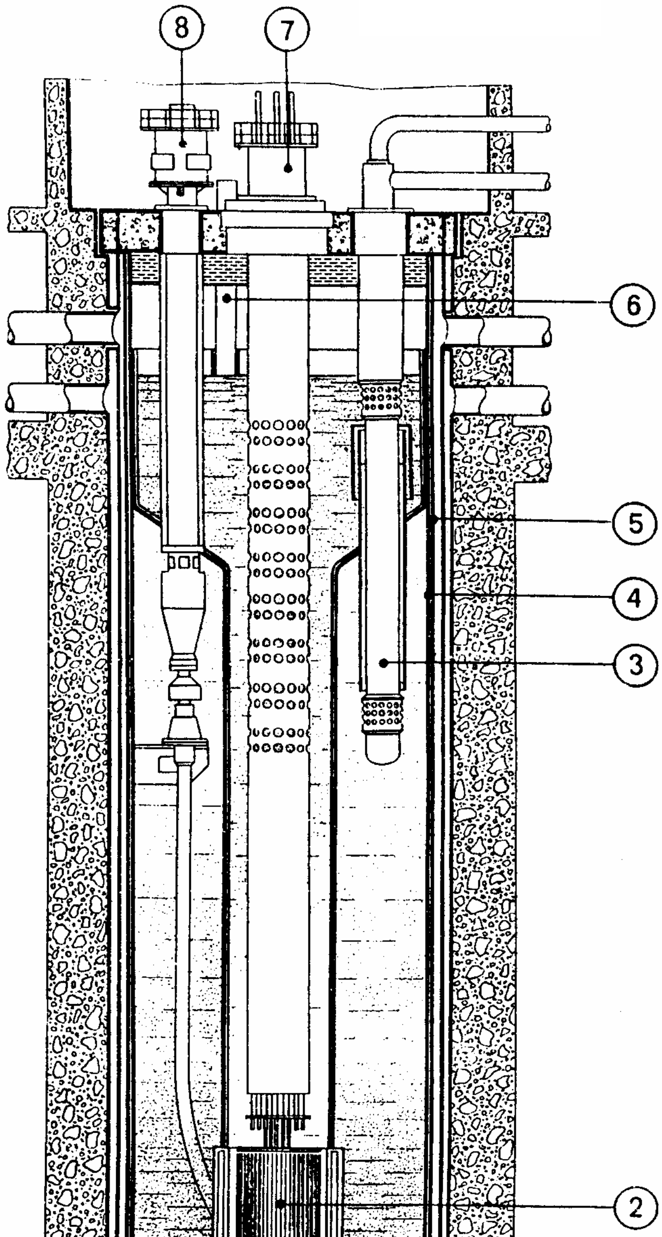
- Through casings of the steam generators by atmospheric air in natural draught and natural circulation of sodium in the primary and secondary circuits.



1 – Reactor compartment
2 – Reactor
3 – Shock absorber
4 – Steam generator
5 - Steam generator compartment
6 – Secondary circuit pump
7 – Protective concrete plate
8 - Loading-discharging machine
9 – Deaerator stack
10 – Deaerator
11 – Machine hall
12 - Turbogenerator

FIG. XXI-1. NPP with BMN-170.

For modular reactors (Fig. XXI-2), the integral arrangement was selected when primary circuit equipment is in a single vertical vessel-tank. The reactor vessel is enclosed in a full strength safety vessel and has an upper support unit. The reactor cover is in the form of a flat ceiling and provides support for the primary circuit equipment. The safety vessel is not provided with thermal insulation needed to improve residual heat removal to the ultimate sink (air) in natural draught. The reactor vessel accommodates two intermediate heat exchangers and the four main circulation pumps of the primary circuit. The core is installed on the vessel bottom with a pressure chamber and a device to isolate core fragments and remove heat in case of beyond design basis accidents. A multi-layer "hot box” on the pressure chamber is intended to divide areas of hot and cold coolant in the reactor. In-reactor refuelling is performed by two refuelling mechanisms, which carry out operations on the installation and withdrawal of fuel assemblies from the core, and one vertical elevator to deliver assemblies from the core to the refuelling penetration in the reactor cover and back. Data summarizing BMN-170 characteristics are specified in Table XXI-1.



1 – Pressure chamber
2 – Core
3 – Intermediate heat exchanger
4 – Vessel
5 – Safety vessel
6 – Cold trap
7 – Core protection system column
8 – Primary circuit pump

FIG. XXI-2. Reactor general view.

TABLE XXI-1. MAJOR DESIGN AND OPERATING CHARACTERISTICS OF BMN-170

MAIN DESIGN CHARACTERISTICS	
Installed power: - Thermal - Electric	 400 MW(th) 170 MW(e) without heat takeoff 100 MW(e) at simultaneous cogeneration based district heating 75 MW(e) at simultaneous water desalination
Fuel type	Cylindrical fuel elements with fuel pellets made of uranium-plutonium oxide, nitride or metallic fuel in steel claddings *
Enrichment	Plutonium content in oxide fuel – 34%
Coolant	Sodium
Moderator	No
Structural materials	Claddings of fuel elements – ChS-68hd steel (06Cr16Ni15Mo2Mn2TiB). Reactor structures – Cr18Ni9 austenitic stainless steel.
Core	Cylindrical, made of hexagonal fuel assemblies (FA) with a pitch of 100 mm, effective diameter – 1.5-2 m, height of FA active part – 1.0 m, heterogeneous. Can be surrounded by side and end breeding screens.
Reactor vessel	Cylindrical vessel with elliptical bottom and flat covering in the safety vessel. Dimensions of the main vessel (diameter x wall thickness x height, m) – 6.5×0.03×18.
Number of circuits, type of thermal cycle	Three-circuit arrangement with sodium in the primary and secondary circuits, and water-steam in the third circuit. Primary circuit has no loops. The secondary circuit is provided with two loops of equal power.
NPP operation	In the mode of electricity generation (basic mode)
Efficiency	43%
Design capacity factor	0.82
Design service life, year	60

NEUTRONIC CHARACTERISTICS	
Average specific power, MW/m^3	225
Burn up, % heavy atoms	15
Maximum linear power in FA, KW/m: - BOC - BOE	 26.4 26.4
Reactivity void effect, % ΔK/K	-0.15
Reactivity variation within the interval, % ΔK/K	2.3
Reactivity variation within the interval with account of neptunium effect, % ΔK/K	2.4

REACTIVITY CONTROL AND PROTECTION SYSTEM		
Types of reactivity control systems	Two independent protection systems with mechanical drives of reactivity control rods	Each system can shut down the reactor and maintain it in subcritical state
Effectiveness of systems, % Δk/k	First system (5 safety rods (SR)) Second system (7 regulating rods (RR))	4.5 5.15
Subcriticality created by the systems, %Δk/k	First system without one of more effective rod (4 SR)	2.0
	Second system (7 RR)	1.45

TABLE XXI-1. (Continued -1)

THERMAL-HYDRAULIC CHARACTERISTICS		
Circulation	Forced circulation at power levels, natural circulation at residual heat removal	
Circulation system parameters	Primary circuit	
	Sodium temperature at the core inlet, °C	395
	Sodium temperature at the inlet to intermediate heat exchanger, °C	550
	Sodium flow rate, t/h	7300
	Sodium pressure at the core inlet, MPa	0.8
	Pressure in the gas cavity, MPa	0.04
	Secondary circuit	
	Sodium temperature at the inlet to intermediate heat exchanger, °C	350
	Sodium temperature at the outlet from intermediate heat exchanger, °C	530
	Sodium flow rate, t/h	6280
	Sodium pressure at the pump discharge nozzle, MPa	0.5
	Third circuit	
	Feedwater temperature at the steam generator inlet, °C	274
	Capacity of live steam, t/h	546
	Live steam pressure, MPa	24
	Superheated steam pressure, MPa	3.6
	Live steam temperature, °C	505
	Superheated steam temperature, °C	505
Operation limits on temperature parameters	Temperature of fuel element cladding with account of uncertainty of parameters, °C	700
	Fuel temperature with account of uncertainty of parameters, °C	2500
	Reactor vessel temperature, °C	450
	Temperature of in-reactor metallic structures, °C	600

BALANCES OF FUEL		
Residence time, effective day	1200	Selected on the basis of analysed behaviour of fuel elements during operation at nominal power level
Refuelling interval, effective day	300	Selected to minimize reactivity swing between refuellings
Portion of refuelled FA:		
- Core	1/4	
- Internal blanket	1/3	
- Side blanket	1/6	
Loaded fuel, kg		
$(U\text{-}Pu)O_2$	4670	
Pu	1400	
Fissile materials, kg/MW(th) per effective year	1.07	Consumption of materials related to one year of reactor operation at nominal power
Natural uranium, kg/MW(th) per effective year	16.1	

TABLE XXI-1. (Continued - 2)

Economic estimates		
Method for evaluating specific capital outlays for construction	Through specific steel intensity of NPP equipment, t/MW(e)	
Specific steel intensity of reactor equipment	3.95 t/MW(e)	Better than a similar index for BN-600 (6.9), BN-800 (5.6) reactors, and more than that for new generation reactors of high power BN-1600M (1.5), EFR (0.87)
Specific steel intensity of NPP equipment	6.7 t/MW(e)	Several times more than that for new generation reactors of high power
Other cost estimates	Not performed at this work stage	

* All characteristics are presented for the option of uranium-plutonium oxide fuel.

XXI-.5. Outline of fuel cycle options

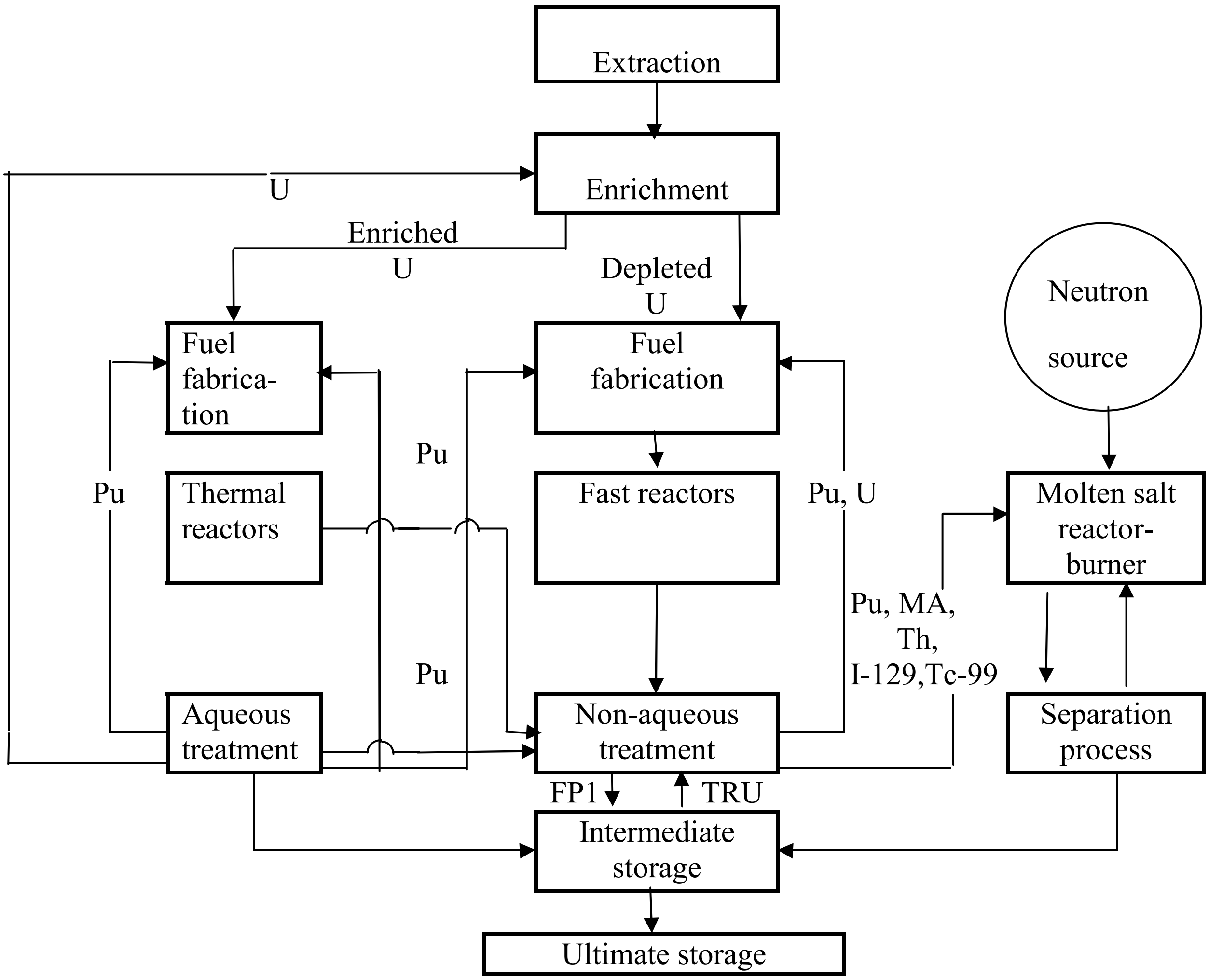


FIG. XXI-3. Schematics of a three-component nuclear power system with a closed fuel cycle for all actinides including Pu and hazardous long-lived fission products.

To develop nuclear power as a stable, large-scale power technology, it must meet requirements for effectiveness, utilization of resources and safety.

This technology must not only solve problems related to waste produced but from the beginning of development, deal with the potential for accumulated radioactive waste and spent fuel. For a structure meeting these requirements, the concept of a three-component nuclear power system with closed U-Pu (Th-U) fuel cycle is currently being developed [XXI-5, XXI-6, and XXI-7], Fig. XXI-3.

Within this concept, the future nuclear power should: implement closed U-Pu and Th-U fuel cycles with optimum neutron and nuclide balances; provide for the production of nuclear fuel and multiple recycling of fuel; minimize the amount of radioactive waste; and have the possibility of applying useful fuel materials.

The BMN-170 concept provides for use of this fast reactor as an element in a multi-component structure, with optimized nuclide flows between elements. Therefore, it is possible to consider BMN-170 operation in the nuclear power system in different modes, in particular:

- Both fissile materials from the spent fuel of light water reactors (uranium or mixed) and plutonium produced in other breeder reactors can be used for the reactor makeup. In the distant future when cheap uranium reserves are exhausted, the reactor can be provided with an option of ^{233}U fuel breeding to change thermal reactors to the Th-U cycle.
- Closed plant-level fuel cycles with external makeup using depleted uranium can be considered. Excessive plutonium is used in other reactors or to start up reactors of the same type.

To analyze the reactor characteristics in both cases requires selection of an operational pattern and connections of all structural elements of the nuclear power system. In this case, the technologies of fuel recycling with superficial purification of fission products can be used. Nuclide flows supplied to the reactor depend on the amount, purpose and characteristics of all structural elements. Analyses show that the effectiveness of fuel utilization depends on the organization of nuclide flows in the nuclear power system to a greater extent than on the breeding ratio level of a fast reactor.

First evaluations have already been made of nuclide flows in the simulated multi-component nuclear power system with a closed fuel cycle for all actinides, including Pu and most hazardous long-lived fission products. Thus, it is already possible to consider BMN-170 operation within a closed fuel cycle because of the role it would play in the proposed nuclear power system.

At present, only the most probable options of BMN-170 operation were considered in detail, at the beginning of its implementation in the structure of the nuclear power system when there are considerable reserves of spent fuel from light water reactors after decay in storage.

XXI-1.6. Technical features and approaches that are definitive for BMN-170 performance in particular areas

XXI-1.6.1. Economics and maintainability

Due to specifics of the expected BMN-170 application, concept development did not address special market requirements; in particular, requirements of developing countries and specific features of their nuclear power options were not considered.

The NPP modular concept allows the consecutive initiation of facilities, which spreads the investments over time and reduces investment risk. Improvement of the economic indices of the BMN-170 at a consistently high safety level is achieved due to the following concept characteristics:

- Reduction of dimensions and simplification of equipment design leading to reduction of repair and maintenance activities.
- The NPP modular structure provides constant operation of the NPP during repair and maintenance activities or refuelling of one of the modular units.
- The possibility of removing a module for disposal after ceasing operation and of installing a new module in the free space, allowing buildings and auxiliary systems to be used for a longer than usual time.
- Maximum prefabrication with the possibility of delivery of the assembled object, thus reducing the scope and cost of mounting activities at the NPP site.
- Large-scale manufacture since modular design allows construction of NPPs of practically any power based on the standard reactor module.
- Reduction in the nominal number of servicing for the NPP systems, which are common for groups of modules (coolant purification system, refuelling system, etc.).
- Simplified control systems since the reactor has two main states, operation at 100% power level and residual heat removal. In this case a group of modules can be controlled by one team of operators.
- Simplified emergency power supply system since emergency cooldown of modules is performed only by passive systems based operation, sodium natural circulation in the reactor and self-regulated natural circulation of atmospheric air, removing heat from the reactor vessel.

During the design process, no special measures were taken to provide low capital costs for the fuel cycle. It is presumed to use standard approaches to the solution of these problems, including operation of the BMN-170 in the integral system of a large-scale nuclear power system.

XXI-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

Fast reactors including the BMN-170 solve the problem of providing sustainable nuclear power due to transmuting uranium-238 to plutonium-239. The reactor is characterized by high levels of fuel utilization achieved by the introduction of internal, side and lower axial blankets for breeding the secondary nuclear fuel.

For the BMN-170 concept, such notions as providing sustainable development and waste handling are inseparably linked with NPP operation in the integral system of a large-scale nuclear power system, where each element plays its own role.

Specifically, in a system where reactors like the BMN-170 are mainly intended to maintain neutron balance, specialized means for burning minor actinides and transmutation of some fission products (reactors-burners) can be provided along with traditional thermal and fast power reactors. In spite of the possibility of using the BMN-170 for these purposes, the option of reactor loading with fuel of low minor actinide content has not yet been considered.

Since the BMN-170 must meet requirements for the integrity of the fuel elements in normal operation, protective barriers prevent radioactive releases to the environment and a negligible level of irradiation is expected to personnel and population.

Evaluations based on operating experience in fast sodium cooled reactors show that for the BMN-170, the radiation impact on the environment consists of the following:

- During normal operation, the irradiation dose beyond the 5 km exclusion area will not exceed 0.008 mSv/year at an allowable average annual dose of 1 mSv/year.
- For maximum design basis accidents it will not exceed 0.16 mSv per accident. This does not exceed the limit of 5 mSv for the first year of accidents, established in [XXI-8].

More detailed evaluations of the radiological consequences of radioactivity release from the BMN-170 were not performed.

XXI-1.6.3. Safety and reliability

Safety concept and design philosophy

The approach to BMN-170 plant safety is based on the concept of retaining radionuclides in the fuel during normal operation and accidents, so that radiation impact on personnel and population in the NPP area lies within allowable limits. Barriers for fission products release are mainly claddings of fuel elements and reliable operation is provided under the specified operating conditions.

Provisions for simplicity and robustness of the design

During BMN-170 development, the design principles to provide plant safety was aimed at an optimum combination of reliance on intrinsic safety features and application of engineered (active and passive) systems. To be specific, the protection system being developed employs hydraulically suspended reactivity control rods that effectively influence the reactivity and convert the reactor to a sub critical state when flow rate is reduced through the core [XXI-9]. Use of a passive emergency cool down system allows complete removal of residual heat.

Structure of the defence-in-depth

The in-depth protection entails multiple barriers preventing radioactivity release from the fuel as well as measures to maintain the integrity of those barriers. These barriers rely mainly on prevention of radionuclide release to the coolant by cladding of the fuel elements. Additional barriers preventing radioactivity release into environment are the reactor vessel itself, safety vessel, secondary circuit and NPP building.

Design basis accidents and beyond design basis accidents

At this stage of development, the list of accidents adopted was based on operating experience at existing NPPs and is not final. Probabilistic analysis of the entire spectrum of potential events has yet to be performed.

Currently, the most severe beyond design basis accident related to NPP blackout and non-actuation of all reactivity control rods and additional failure of heat removal to the secondary circuit has been considered.

The accident proceeds as in the following scenario:

1. NPP operation at nominal power
2. NPP blackout
3. Failure of emergency protection signal generation
4. Non-actuation of all control and protection system (CPS) rods
5. Coastdown of primary and secondary circuit pumps with shutdown in ~100 s

6. Failure to put into operation the system for heat removal through the surface of SG casings by atmospheric air
7. Heat removal from the core is performed only through the reactor vessel surfaces, to atmospheric air

Variation of sodium and reactor vessel temperatures during the accident is shown in Fig. XXI-3.

The figure shows that in this beyond design basis accident the reactor is cooled down without sodium boiling. The temperature of the main reactor vessel does not exceed 670 °C.

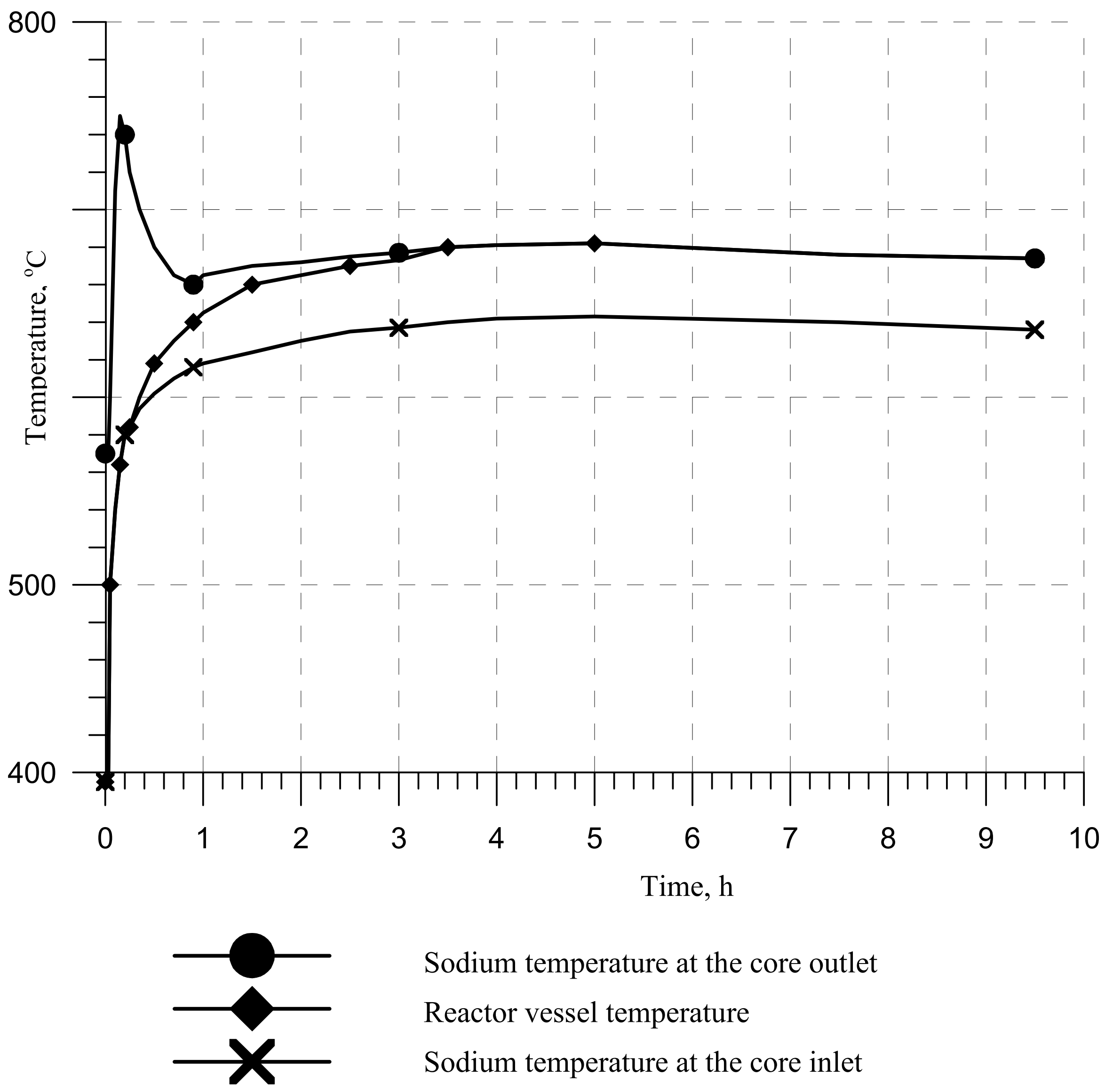


FIG. XXI-3. Variation of temperatures during the accident.

Thus, it is shown that a high level of BMN-170 safety is provided by intrinsic properties of the core, thermal-physical properties of the sodium coolant, use of natural circulation to organize core emergency cooldown, use of passive safety systems along with traditional active systems and maintained by the possibility of high quality fabrication of the equipment in workshops.

It is assumed to provide safety from seismic impacts according to the usual procedures prescribed by regulatory documents.

It is noteworthy that a strength analysis was performed for the BN-600 plant, which is an analogue of BMN-170 and which, in terms of mass and size characteristics, considerably exceeds the design modifications proposed for the BMN-170. It confirmed plant seismic stability at a maximum design earthquake of 7 points on the MSK scale.

XXI-1.6.4. Proliferation resistance

The most evident problems related to the potential proliferation of nuclear materials for fast reactors (which also include the BMN-170) are conditioned by the possibility of carrying out extended fuel breeding, first of all in blankets. Besides, the fuel assemblies of fast reactors can include breeding inserts, which do not noticeably worsen the working characteristics of the reactor.

The components of proliferation resistance are considered in two large groups:

(1) Intrinsic barriers – the integral barriers of the nuclear power system (including plants, materials, applied technologies) in the location in which the reactor technology operates;
(2) External measures – organizational and verification measures.

During BMN-170 conceptual development, major attention was drawn to the first group of questions. The following should be listed among the intrinsic properties of proliferation resistance in the BMN-170 design:

(1) It is assumed to use non-aqueous methods of fuel reprocessing.
Non-aqueous technologies developed in Russia and related to the processing of fuel from fast reactors, produce incomplete purification of fission products (about 1% of the fission fragments remains in refabricated fuel) and release only curium from the fuel (neptunium and americium, and 1% of the curium remain in the fuel), resulting in the production of fresh fuel for fast reactors which can not be directly used to create a nuclear weapon [XXI-10].

(2) Regarding systems analysis of the proliferation issue, it is noteworthy that Russian experts performed a study of different concepts on development of the nuclear power [XXI-6, XXI-7, and XXI-11], with the following conclusions:

- If the approach is used to evaluate the properties of barriers to the proliferation of nuclear materials at all stages of the fuel cycle, it is possible to show that when excessive plutonium is used for the fuel makeup of fast reactors (which are gradually being substituted for thermal reactors running on uranium), the proliferation risk in the system can be reduced considerably since the extent of uranium mining and separation activities decreases.
- The concept of a multi-component system with a limited number of fast reactors (including the BMN-170) that meet the demands of other types of reactors for fissionable materials appears rather attractive in terms of non-proliferation.

It is understood that these preliminary results that are generally favourable to the selected pattern of BMN-170 operation in a large-scale nuclear power system require further confirmation.

Obviously, the least intrinsically protected nuclear fuel cycle stages of the nuclear system with the BMN-170, namely, factories for fuel assembly fabrication, transportation and storage of fresh fuel assemblies, require intensified measures of accounting, monitoring, physical protection and national and international verifications, even with an account of the properties of the ‘dirty’ fuel used.

XXI-1.6.5. Technical features and technological approaches used to facilitate physical protection of BMN-170

Due to the specific features of the BMN-170 for application in remote areas, the option of underground installation was considered a possibility to enhance protection from external impacts and penetration to the nuclear island.

Transients started by initiating events during NPP operation run rather slowly with considerable heat capacity of the primary circuit and passive heat removal from the reactor vessel; this offers the possibility of taking timely measures toward accident management.

The operation of passive safety systems is based on physical laws, substantially reducing the possibility of failure by malevolent personnel actions.

XXI-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of BMN-170

At this stage of the BMN-170 design, specific requirements and market needs (especially in respect to developing countries) were not considered. It is assumed to render full-scale service of the fuel cycle within the framework of the multi-component structure of the nuclear power system.

XXI-1.8. List of enabling technologies relevant to BMN-170 and status of their development

Creation of the BMN-170 mainly using technical solutions confirmed by the operating experience of fast sodium cooled reactors will require little time for development since:

- The qualified personnel for development of the plant design is available.
- The industrial base for fabrication of the entire complex of NPP equipment and systems has been established.
- The infrastructure necessary to perform R&D in support of innovative solutions already exists.

The main problems requiring solution when implementing nuclear plant design are:

- Validation of the passive cooldown concept.

  Currently there is no experimental validation of the proposed concept. Therefore, it is necessary to perform complex experiments to investigate modes of sodium flow in the primary circuit and air circulation outside the reactor module, as well as sodium and air hydrodynamics under conditions typical for normal and emergency operation of the BMN-170.

  Special attention shall be paid to experimental investigations of thermal regimes with an irregular distribution of airflow on the reactor vessel perimeter.

- Operational validation of a core employing fuel elements with innovative nitride and metallic fuels.

  It is necessary to refine the database on properties of nitride and metallic fuel incorporated in support of the BMN-170 concept.

  Fabrication of fuel and fuel elements should be mastered.

  It is necessary to obtain reliable experimental data on operability of fuel elements with nitride and metallic fuel under reactor conditions.

Economic viability of the concept depends very much on the capability of the fuel composition (nitride or metallic) to provide rather high burn-up. Currently, oxide fuel is considered the main option and actually achievable parameters were adopted, which so far do not provide radical improvements in fuel utilization compared with available experience. This requires a search for new solutions.

XXI-1.9. Status of R&D and planned schedule

Currently, conceptual investigations of the BMN-170 project are performed on the initiative of OKBM specialists (Nizhny Novgorod, Russia).

During recent years (1997-2003), activities for the BMN-170 have also been stimulated by an exchange of scientific and technical information with companies in Russia and abroad; they are currently developing concepts of sodium cooled fast reactors and interested in the possibility of implementing the project. Among them it is possible to indicate the interest of the Ministry of Atomic Industry of Kazakhstan.

At this preliminary design stage of the BMN-170, companies and institutions, which can be involved in R&D are not specifically defined, nor is the time frame of activities.

The most probable scenario for further development of the BMN-170 is to join efforts of the originators of the initial (conceptual) BMN-170 designs, namely, such Russian enterprises as SPb AEP, OKBM and IPPE. Previously, full-scale designs of power units using fast sodium reactors BN-600, BN-800 were developed through such cooperative efforts.

Evaluation of the time interval, when R&D in support of the project can be implemented and when the project itself can be implemented in the Russian Federation, currently depends on funding.

XXI-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

The BMN-170 concept incorporates many proven technical solutions used for its predecessors (BOR-60, BN-350, BN-600 and BN-800). These concepts were first developed for a breeder component of the multi-component structure for the nuclear power system.

The BMN-170 concept is considered innovative since approaches very different from those presently used to optimize traditional reactor systems are applied during its development, in contrast to the predecessor NPPs.

The main conceptual solutions requiring factual confirmation at the prototype plant are the following:

- Validation of operability under the concept of emergency residual heat removal only by passive means; and
- Validation of operability for a core based on fuel elements with innovative nitride and metallic fuel.

Clearly, it is necessary to perform additional experimental studies of individual phenomena and processes to validate solutions for the small power NPP. However, it is known from the history of development of fast sodium cooled reactors that problems related to fuel and coolant were solved by large-scale experimental investigations and by making account of the operational experience of a number of operating plants consequently created.

The full scope of the BMN-170 innovative features of the concept can be demonstrated only using a prototype, with final solutions on parameters and operation modes being selected on the basis of such demonstration.

XXI-2. Design descriptions and data for BMN-170

XXI-2.1. Description of the nuclear systems

Core and fuel design

Direct prototypes of the BMN-170 reactor are designs of cores for fast reactors BOR-60, BN-350, BN-600 and BN-800.

By the time of BMN-170 development, the scope of R&D performed and experience in the operation of fast sodium reactors gained allowed use of the following solutions on the core:

- Mixed oxide fuel is considered the main option and nitride and metallic fuel as advanced options.
- A core arrangement with annular heterogeneity is used based on the application of internal breeding blankets with depleted uranium dioxide.

The heterogeneous core arrangement (Fig. XXI-4) provides the following advantages:

1) Fuel of a single plutonium enrichment is used.
2) The core breeding ratio increases due to internal breeding in the core. This also reduces the rate of reactivity loss for burn-up.
3) The sodium void reactivity effect (SVRE) changes toward negative values.

A sodium plenum is introduced above the core to additionally reduce the SVRE and transmit it to the area of negative values. Zero SVRE is provided by separation of the upper axial blanket from the core by the sodium plenum, while replacement of uranium dioxide in the upper axial blanket for boron carbide if there is a sodium plenum, reduces the SVRE to minus 0.15% $\Delta k/k$ (for the reactor as a whole).

For the core fuel assemblies, a four-batch refuelling is selected with an interval of 1 year. For fuel assemblies of the inner blanket, three-batch refuelling is selected; fuel assemblies of the side blanket have an average life of 500 effective days.

The core includes fuel assemblies with MOX fuel (Fig. XXI-5), fuel assemblies of the internal blanket, fuel assemblies of the side blanket, shielding assemblies containing natural boron carbide, and cells for in-reactor storage. Design characteristics of the BMN-170 core and fuel assemblies are presented in Table XXI-2.

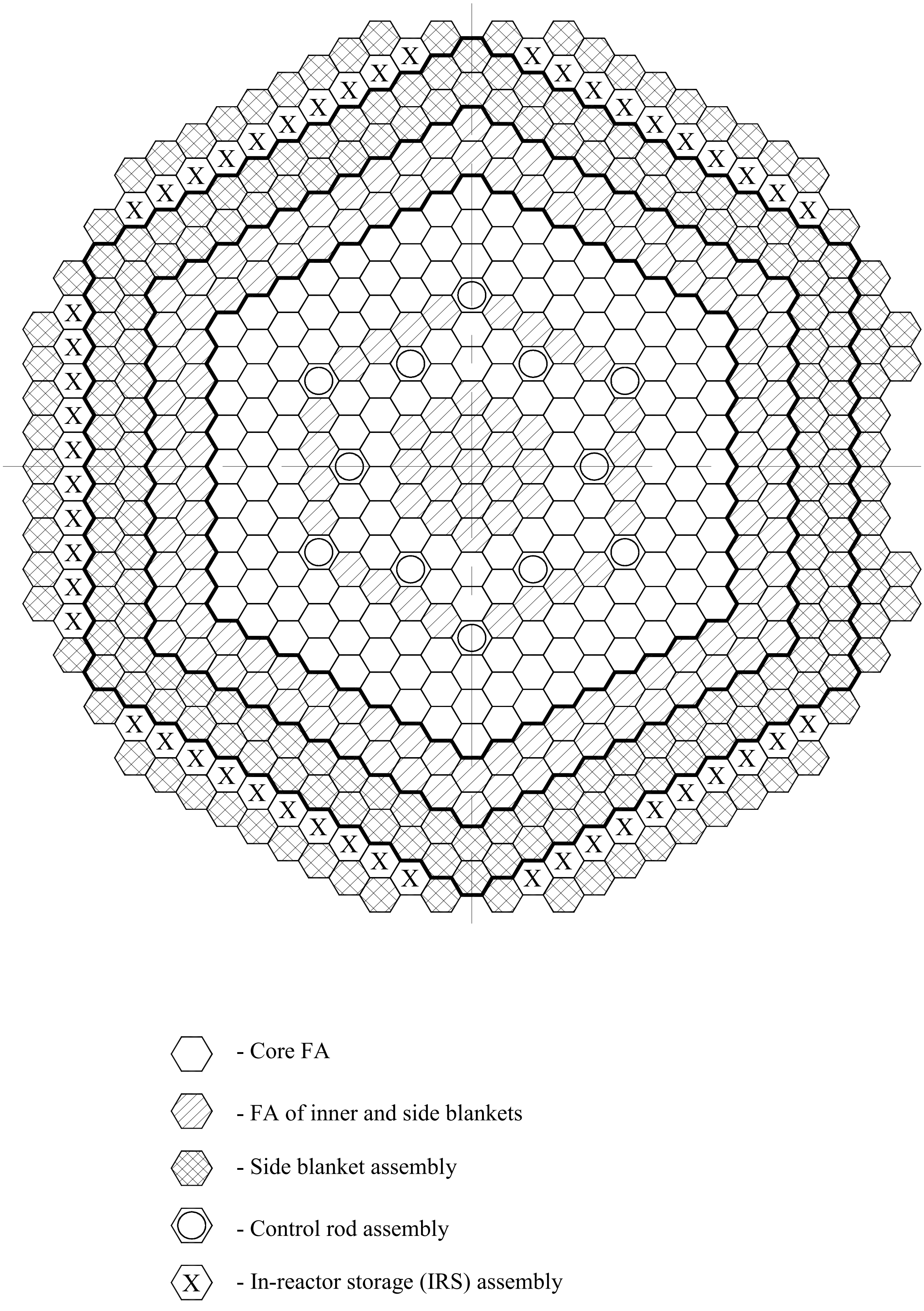


FIG. XXI-4. Core cross section.

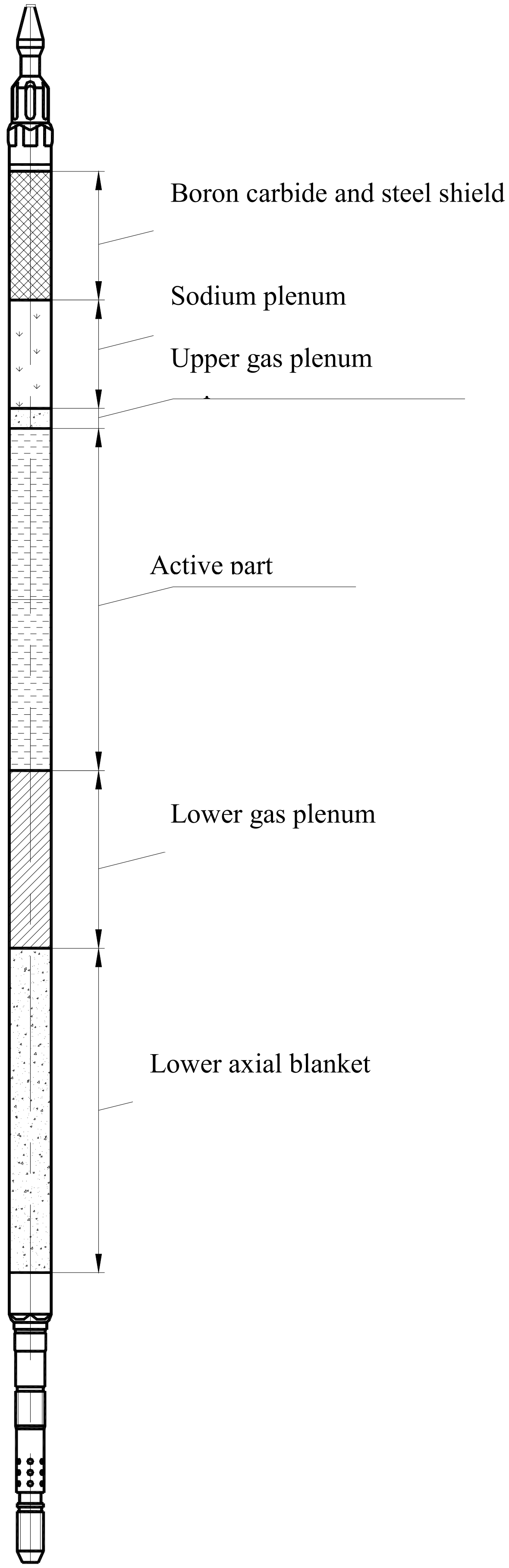


FIG. XXI-5. Core fuel assembly.

TABLE XXI-2. DESIGN CHARACTERISTICS OF CORE AND FUEL ASSEMBLIES

CHARACTERISTIC	VALUE
Number of FA: Core Internal blanket Side blanket	 162 43 114
Number of boron carbide assemblies	209
Number of CPS rods	12
Number of in-reactor storage cells	50
Width across flats of the FA wrapper, mm	96
Number of fuel elements in FA: Core Internal blanket	 127 37
Fuel element diameter, mm:	
Core	6.8
Internal and side blankets	14.0
Height of active part, mm	1000
Height of lower axial blanket, mm	300
Height of sodium plenum, mm	200
Height of boron carbide shielding, mm	300

Systems for reactivity monitoring, control and protection

According to the Russian regulatory documents, two independent reactor shutdown systems must be provided. The reactor uses two groups of reactivity control rods with mechanical drives of different types. Each of these systems can shut down the reactor and maintain it in a subcritical state from any nominal or emergency state provided that one most effective control rod does not actuate.

Absorber rods in the shutdown systems are standardized in terms of absorber element diameter and number in the shroud tube and also in terms of the dimensions of the shroud tube with rods developed for the BN-800 reactor. The effective density of the absorber material, boron carbide, is also standardized. Five rods are intended for reactor emergency protection, five rods for compensation of the reactivity effects (all rods with B^{10} enrichment of 60%) and two rods with natural boron carbide for power control. The reactivity balance during reactor refuelling is given in Table XXI-3.

TABLE XXI-3. REACTIVITY BALANCE DURING REACTOR REFUELLING

BALANCE COMPONENT	VALUE, %ΔK/K
Maximum reactivity margin	3.7
Total effectiveness of CPS rods	9.7
Subcriticality level of shutdown reactor	6.0

Systems for core power control by neutron flux measurement are traditional for BN reactors.

Primary circuit systems and reactor module

The reactor module has an integrated arrangement. All primary circuit systems, including the core, intermediate heat exchanger and circulation pumps of the primary circuit are arranged in double cylindrical vessel. Dimensions of the main and safety vessels are ∅6500×30 mm and ∅6300×30 mm, respectively. The vessel height is about 18 m. The gap between the main and safety vessels is filled with gas in which pressure is regulated and limited. The gas gap is also intended to heat up the reactor module prior to start-up by forced circulation of hot gas.

Besides, the safety vessel allows isolation of radioactive coolant in case of main vessel depressurization. The reactor module does not contain external pipelines with primary coolant.

Intermediate and steam turbine circuits and other systems

Apart from the reactor module the BMN-170 plant includes an intermediate sodium circuit with steam generators, control and monitoring system, refuelling system, system of gas heating for the reactor module vessel, primary coolant filling and drainage system, system for monitoring the integrity of fuel element claddings, primary coolant purification system and others systems for NPP operation.

The option of arranging the reactor module and main equipment of the intermediate sodium circuit in separate cavities located below the ground surface was considered. The design data for intermediate and steam turbine circuits is presented in Table XXI-4.

TABLE XXI-4. SUMMARY OF DESIGN CHARACTERISTICS FOR SECONDARY AND THIRD CIRCUITS

CHARACTERISTIC	VALUE
Secondary (intermediate) circuit	
Number of loops	2
Sodium flow rate, t/h	6280
Temperature at the inlet of steam generator and reheater, °C	530
Temperature at the inlet of intermediate heat exchanger, °C	350
Third (steam-water) circuit	
Type of reheating	Sodium
Number of loops	2
Number of steam generators + number of reheaters	2+2

Main heat transport system

Figure XXI-6 shows the heat transport path from the core to the ultimate sink during normal operation and under emergency conditions. The normal heat removal system is based on a three-circuit design and includes a loopless (pool type) primary circuit in the reactor module, two equivalent loops of the intermediate sodium circuit, two loops of the steam-water circuit and a turbogenerator facility. During normal operation, heat released in the core, including residual heat release of the shutdown reactor is transferred to the steam-water circuit. Steam can be taken off from the third (steam-water) circuit for industrial applications and district heating. The steam-water circuit is designed to supply steam to the turbine generators of 500 or 800 MW power, taking the heat from three or five BMN-170 reactors.

XXI-2.2. Description of the turbine generator plant and systems: No data provided.

XXI-2.3. Systems for non-electric applications

At this stage of development, systems for non-electric application in the BMN-170 plant were not considered in detail. Experience in the design of appropriate systems for steam-water circuits with close parameters, e.g., the experience of BN-350 indicates that BMN-170 could be easily adjusted for potable water production and district heating.

XXI-2.4. General plant layout

At the current design stage the general plan of an NPP with the BMN-170 was not developed.

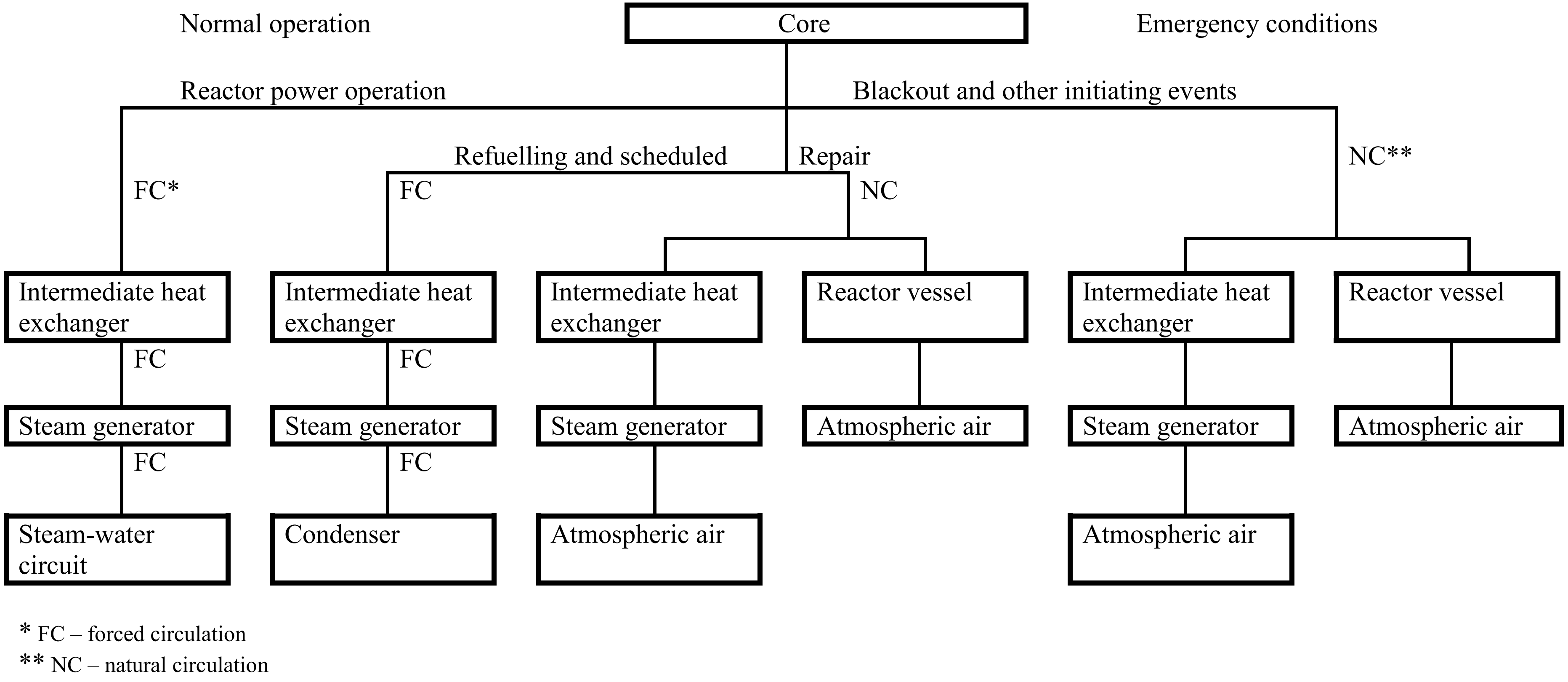


FIG. XXI-6. Heat removal path during normal operation and in accidents.

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ANNEX XXII

MODULAR DOUBLE POOL LIQUID METAL COOLED FAST BREEDER REACTOR (MDP)

CRIEPI, Japan

XXII-1. General information, technical features and operating characteristics

XXII-1.1. Introduction

The modular double pool liquid metal cooled fast breeder reactor (MDP) has been designed to reduce construction costs and improve reliability through factory production of most components [XXII-1, XXII-2]

For practical deployment of fast breeder reactors (FBRs) it is necessary to reduce construction costs. The construction cost of an FBR is high in comparison with an LWR due to the use of sodium as the coolant and the adoption of intermediate heat transport system. Therefore, the reduction and limitation of the sodium handling area and the simplification of intermediate heat transport system are effective measures for cost reduction. The objective of the MDP study is to establish a modular reactor concept that can compete with an LWR on construction costs.

The double pool design is intended to reduce the distances in the intermediate heat transport system by installing steam generators and secondary pumps in the sodium filled annular space formed between the primary and secondary vessel. The reduction in the secondary piping system allows the reduction of piping support structures, sodium leak monitoring systems, pre-heating systems, etc. The reduction and limitation of the sodium handling area allows a reduction in size and volume of the nuclear building, reduction of lining against a sodium leak and limitation of the sodium fire area. On the other hand, because of this compact arrangement, with steam generators being located adjacent to the primary reactor vessel, the structural integrity of the primary reactor vessel should be assured against sodium-water reaction accidents.

In addition to simplification of the reactor module by the double pool system, the design standardization by seismic isolation, utilization of common equipment, reduction of construction work at a plant site and a shorter construction period should compensate for the scale-related disadvantages of a small or medium sized reactor.

The development of the MDP reactor concept has been performed and funded by the Central Research Institute of Electric Power Industry (CRIEPI), Japan.

XXII-1.2. Applications

The MDP is designed to generate electricity.

XXII-1.3. Special features

The MDP is designed as a land-based nuclear power station, allowing for incremental capacity increase through modular approach. With the electric output of 325 MW(e) per module, a power plant with 4 MDP modules could generate1300 MW of electricity. This electrical output is almost equal to that produced at the state-of-the-art large-scale LWRs in Japan.

XXII-1.4. Summary of major design and operating characteristics

TABLE XXII-1. SUMMARY OF MAJOR DESIGN AND OPERATING CHARACTERISTICS

Electrical output	325 MW
Thermal output	840 MW
Primary sodium temperature	530°C / 375°C
Secondary sodium temperature	480°C / 315°C
Steam parameters	450°C / 12.7MPa/ 1 440 000 kg/h
Feed water temperature	200°C
Core type Equivalent diameter	Quasi-homogeneous 2.05 m
Fuel	U-Pu-Zr ternary alloy
Pu composition: ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu (weight %)	Pu discharged from fast breeder reactors 63:32:3:2
Pu enrichment: Inner core (weight %) Outer core (weight %)	 10 16.5
Breeding ratio	1.16
Burn-up reactivity swing ($)	1.5 [2.5] (1)
Sodium void reactivity ($)	4.8
Average fuel burn-up (GW d/t)	140
Peak fuel burn-up (GW d/t)	170
Peak linear power (W/cm)	390
IHX	Shell and tube type, 4 units
SG	Helical coil type, 4 units
Seismic design	3-dimensional seismic isolation system

(1) including the effect of fuel swelling

Table XXII-1 shows the main specifications of the MDP plant. Figures XXII-1 and XXII-2 show a schematic diagram of the reactor module. An electric power output of 325 MW(e) has been set, considering the size of the primary vessel, secondary vessel and cavity wall combination that could be factory produced and transported to a site. The principal components of the reactor system are:

- The primary vessel including the core and support structures, 4 intermediate heat exchangers (IHXs) and 4 primary circulating electromagnetic pumps (EMPs) integrated with cold traps.
- The secondary vessel including 4 steam generators (SG), 4 secondary circulating electro-magnetic pumps and 2 cold traps.
- The upper internal structure.
- The fuel handling mechanisms.
- The roof slab.
- The short secondary sodium piping.

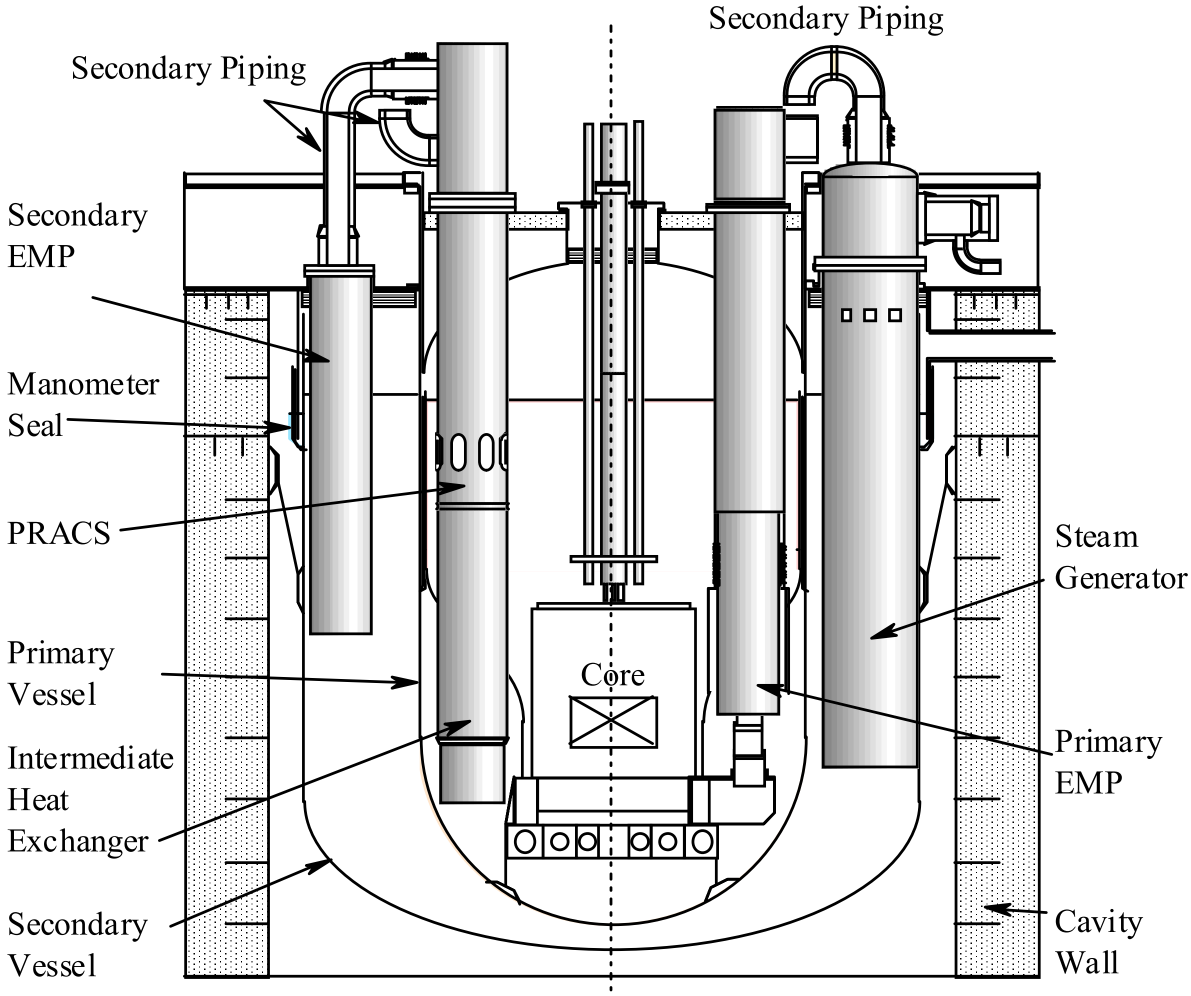


FIG. XXIII-1. Vertical view of MDP.

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 electromagnetic pump; conversely, the secondary sodium pressurized by the secondary electromagnetic pump enters the intermediate heat exchanger through the secondary piping. The sodium flows upward in the shell side of the IHX and enters the steam generator through the secondary piping. After heat transfer to water/steam, the sodium flows to the secondary electromagnetic pump.

The core design aims at a passive shutdown capability based on features of the metallic fuel and small-sized core. A quasi-homogeneous core layout is used to achieve the compact radial core size. The reactivity control in normal operation is accomplished by six control rods constituting two individual shutdown systems (each comprising 3 control rods), each of which can shut down the reactor independently.

The diameter of the primary vessel is 9 m; the diameter of the secondary vessel is 14.4 m. The primary vessel is supported from the roof-slab. At the same time, the secondary vessel is supported from the cavity wall and retains the secondary coolant. This secondary vessel also serves as a safety vessel if a leakage occurs in the primary vessel. It functions to prevent lowering the sodium level in the primary vessel. To cope with the thermal expansion of the secondary vessel and to form the cover gas boundary between the secondary cover gas space and the cavity space, a manometer-seal of a low melting point alloy is used. This manometer-seal also works to release the pressure in case of a beyond design basis accident with sodium-

water reaction. As for the pressure rise after a design basis sodium-water reaction accident, 4 piping systems with rupture disks are used in common with 4 steam generators to release the pressure.

A seismic isolation system for the reactor building is adopted to standardize the design. The base size of the building is 40×40 m, and the height is 55 m.

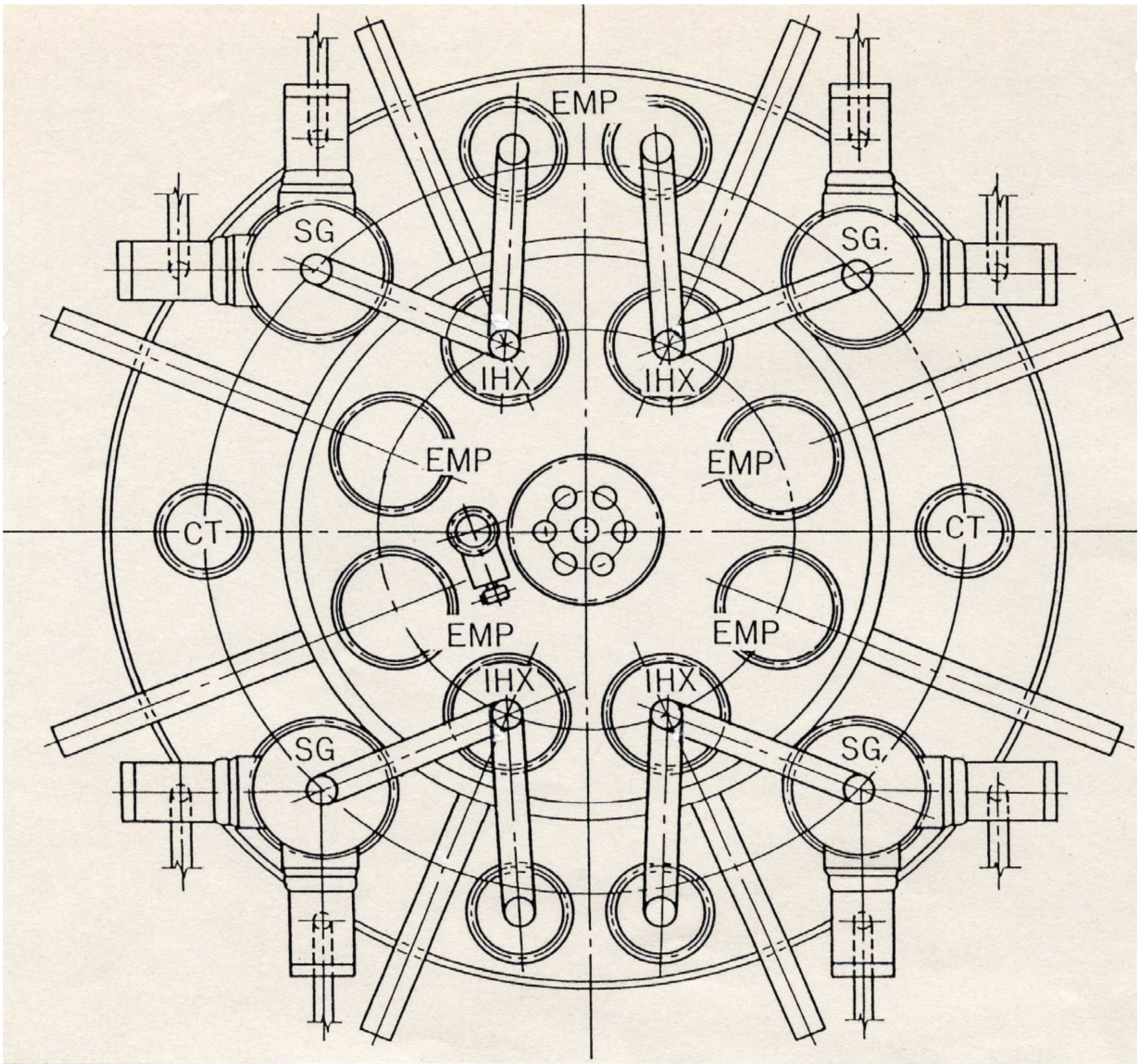


FIG. XXII-2. Plane view of MDP.

XXII-1.5. Outline of fuel cycle options

A closed U-Pu fuel cycle is adopted for the MDP. A pyro-reprocessing for the spent nuclear fuel and an injection casting for fuel fabrication are to be adopted for the MDP.

XXII-1.6. Technical features and technological approaches that are definitive for MDP performance in particular areas

XXII-1.6.1. Economics and maintainability

The double pool reactor has a high potential to shorten the construction schedule because it is assembled and tested at a factory, reduces the sodium piping and sodium handling area, and provides compactness for the nuclear steam supply system (NSSS). A construction period of thirty-one months is confirmed; this short period contributes to the reduction of interest on the investments in construction.

In the construction cost for one module, 25% is for equipment in common with the other modules such as fuel handling machines, and 30% of the cost is for design and analyses, etc. In the 4-module power plant, about a 40% reduction of the cost could be achieved as shown in Fig. XXII-3. Figure XXII-4 illustrates the cost reduction approach of the MDP. The construction costs of a 2[nd] 4-module plant would be competitive with the construction cost of a large-capacity LWR.

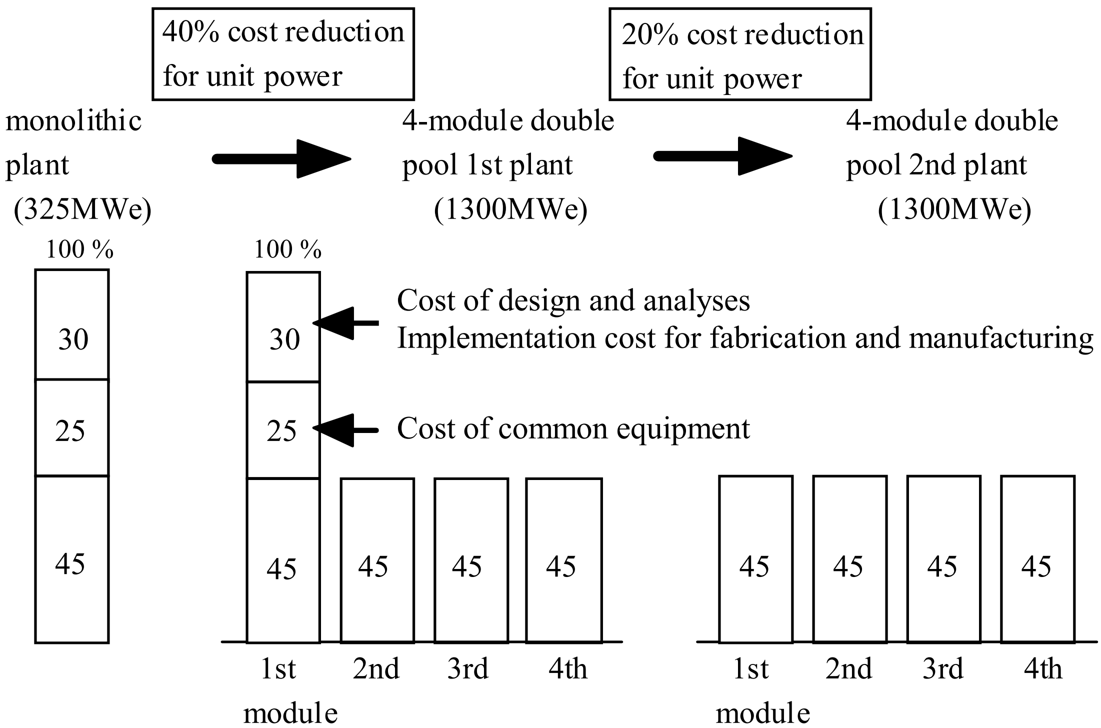


FIG. XXII-3. Cost reduction through standardization and common use of equipment.

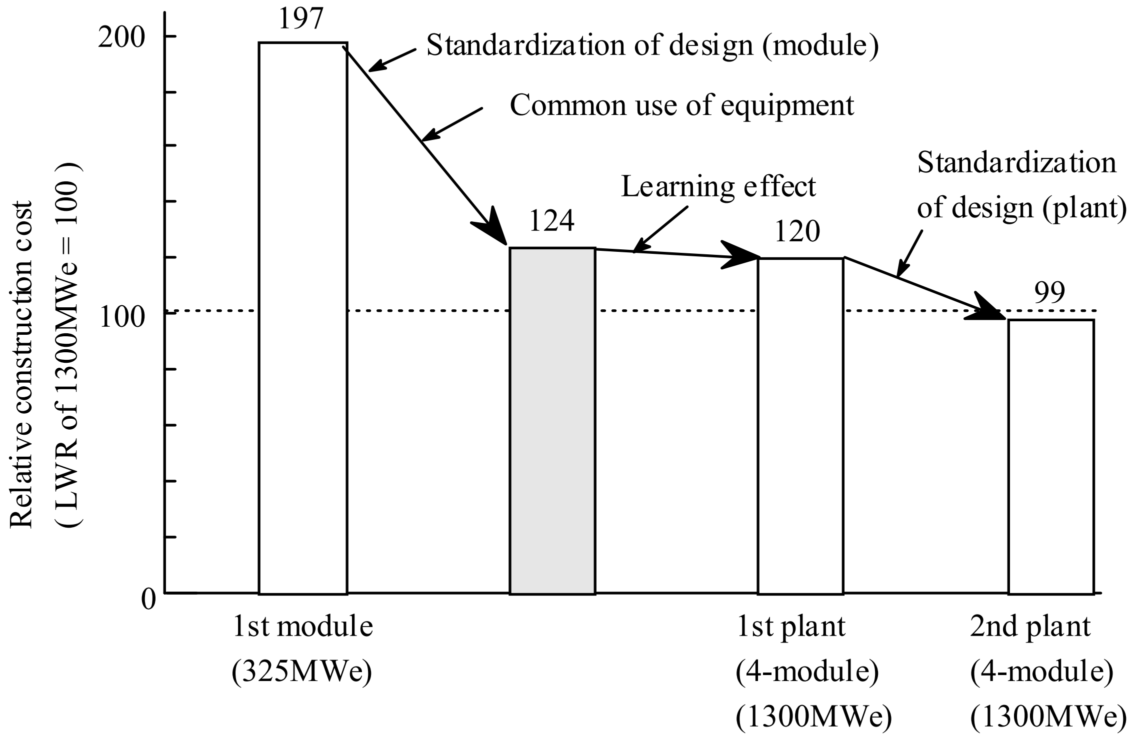


FIG. XXII-4. Evolution of MDP construction costs.

XXII-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

As mentioned above, a closed U-Pu fuel cycle is for the MDP with a breeding ratio of 1.16, which contributes to an effective utilization of natural uranium resources.

XXII-1.6.3. Safety and reliability

Safety concept and design philosophy

The MDP is being designed to ensure a reasonable balance between inherent safety features and the redundant active and passive safety systems, in a cost-effective way.

Active and passive systems and inherent safety features

An enhancement of inherent safety features is expected by the neutronic and thermal characteristics of the metallic fuel. A redundancy and diversity of the shutdown systems is provided and a sufficient grace period is expected due to the increased thermal capacity ensured by the large amount of sodium in both primary and secondary vessels.

To prevent an anticipated transient without scram (ATWS), the independent shutdown systems, a low burn-up swing core and the primary flow coastdown control system are adopted. Furthermore, features of the metallic fuel, thermal expansion of the control rod driveline, radial thermal expansion of the core and large heat capacity of the primary and secondary sodium are expected to mitigate an ATWS. In the double pool reactor structure, SGs are located adjacent to the primary vessel, which is the primary coolant boundary housing the core support structures. If a water leak should occur in the SG tubes, the primary vessel is exposed to an external buckling load. Therefore, an effective mitigation system against sodium-water reaction events has been designed for the MDP.

The primary vessel of the double pool is in the secondary coolant, so that heat of the primary coolant can be transferred to the secondary coolant through the primary vessel. The combination of a vertical and liner structure is considered for thermal protection of the primary vessel in normal operation. It would work as a heat transfer structure under transient conditions because the sodium level in the redan rises when the primary pump trips.

Design basis accidents and beyond design basis accidents

Total blackout

To evaluate the effect of the secondary sodium surrounding the primary vessel, a preliminary analysis was performed for a transient event of a total blackout. As shown in Figure XXII-5, the temperature transition analysis for the primary plenum indicates that the hot plenum temperature is lowered by the effect of heat transfer. Therefore, it should again be emphasized that the large heat capacity of the cold secondary coolant works as a heat sink.

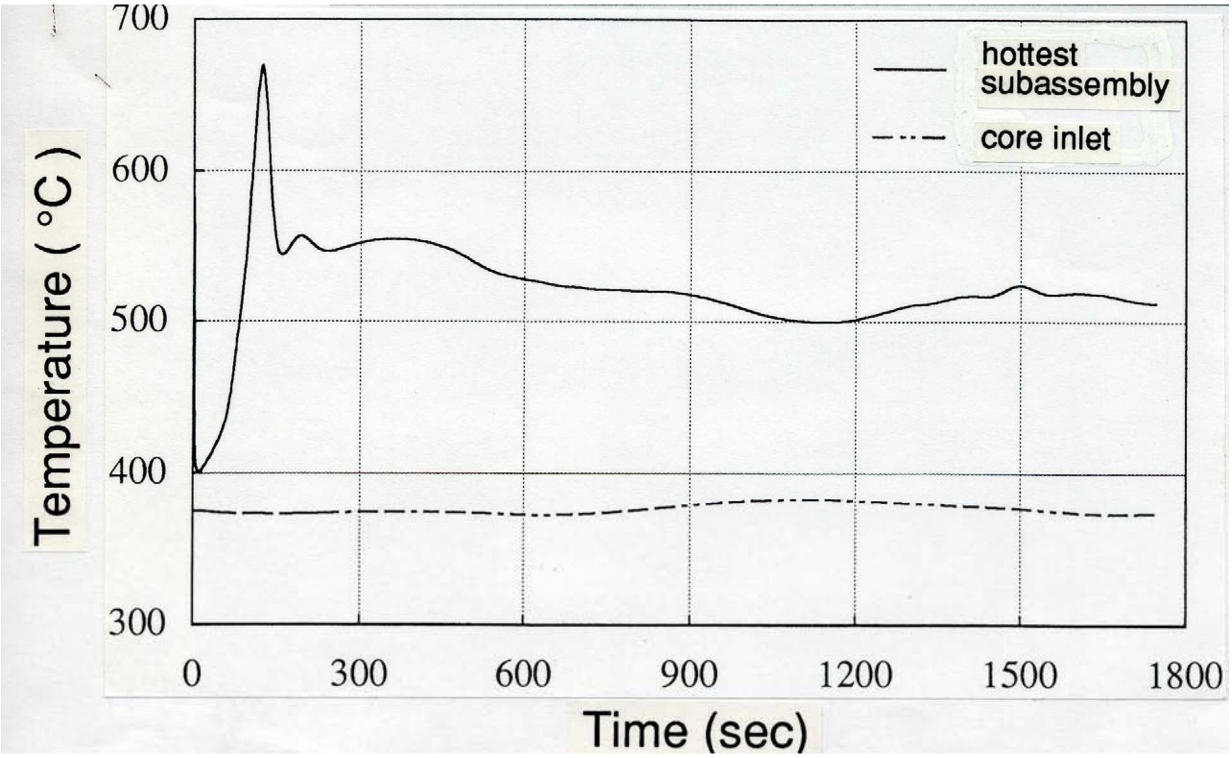


FIG. XXII-5. Temperature transition in case of total blackout.

Unprotected loss of flow

Figures XXII-6 and XXII-7 show the analytical results on unprotected loss of flow (ULOF) as an example of an ATWS. Figure XXII-6 shows the corresponding reactivity feedback and Figure XXII-7 shows the temperature transient of the coolant. The assumptions used are as follows:

- The flow halving time for the primary pump is 10 seconds; for the secondary pump it is 5 seconds.
- Low flow rate operation of the primary and the secondary pump (15% of nominal flow rate).
- Nominal values were used for reactivity feedbacks.

- The relative displacement between the core and control rods is not considered.
- PRACS operation is not considered.
- Heat removal by water in the SG is not considered.

As shown in Figure XXII-7, the maximum coolant temperature at the highest temperature fuel assembly is 820°C. The settled temperature of the coolant is less than the eutectic temperature.

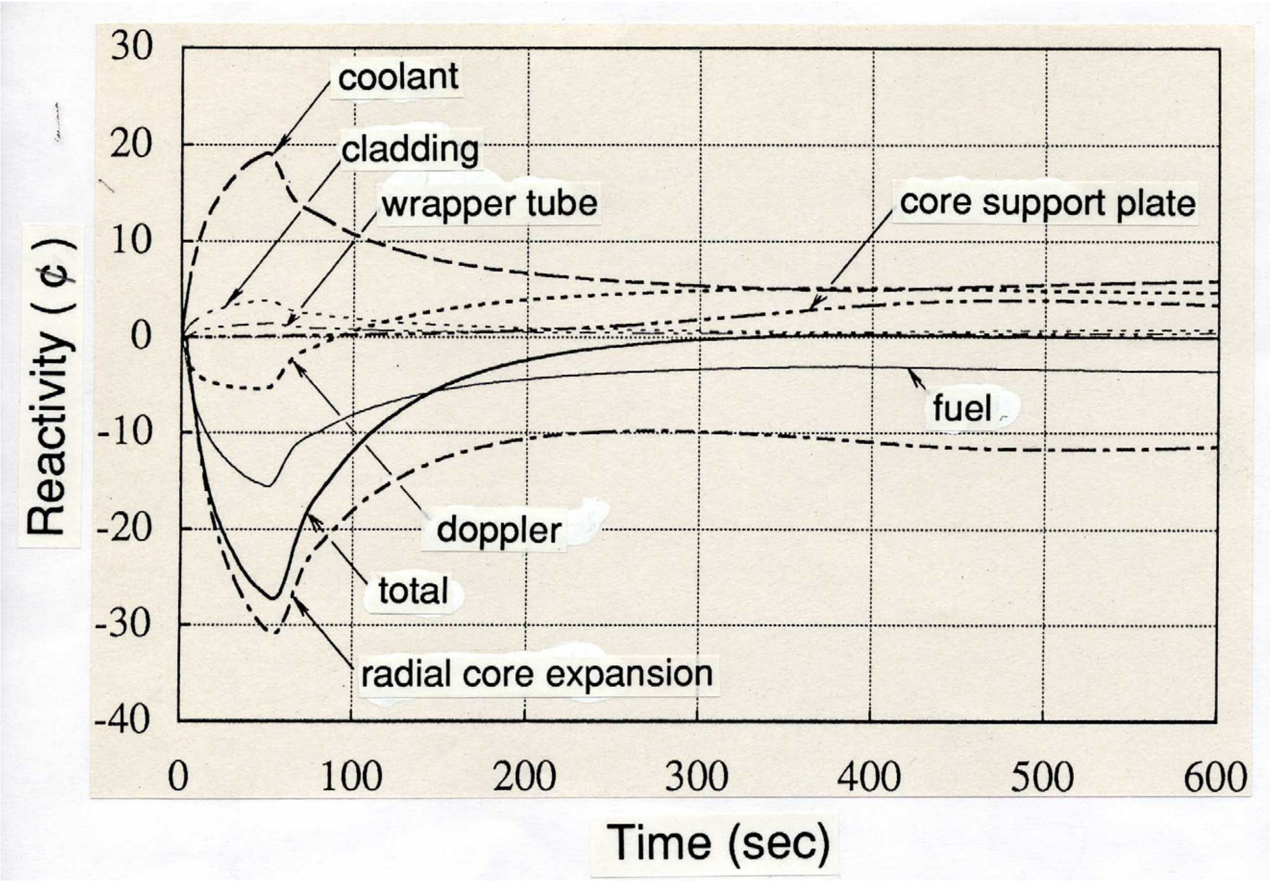


FIG. XXII-6. Reactivity feedback in case of ULOF.

Sodium-water reaction evaluation

The SGs are located adjacent to the primary vessel and immersed in the secondary sodium. If a water leak occurs in the SG tubes, the primary vessel is exposed to the external buckling load and the secondary sodium becomes contaminated by the sodium-water reaction products.

The following preventive measures against a sodium-water reaction are considered for the MDP:

- Quality control.
- Rapid leak detection through a hydrogen meter, acoustic detector, cover gas pressure gauge, rupture sensor and manometer-seal level meter.
- Protective action to ensure the isolation and rapid blow-down of the water/steam system.
- Fast pressure release by a sodium-water reaction pressure release system for a DBL (design basis leak) event and a BDBL (beyond design basis leak).
- Purification of the secondary sodium succeeding a sodium-water reaction.

The safety design criteria for a DBL, 4×DEG (double ended guillotine rupture of the heat

transfer tubes), are specified based on the results of leak propagation analysis using conservative assumptions.

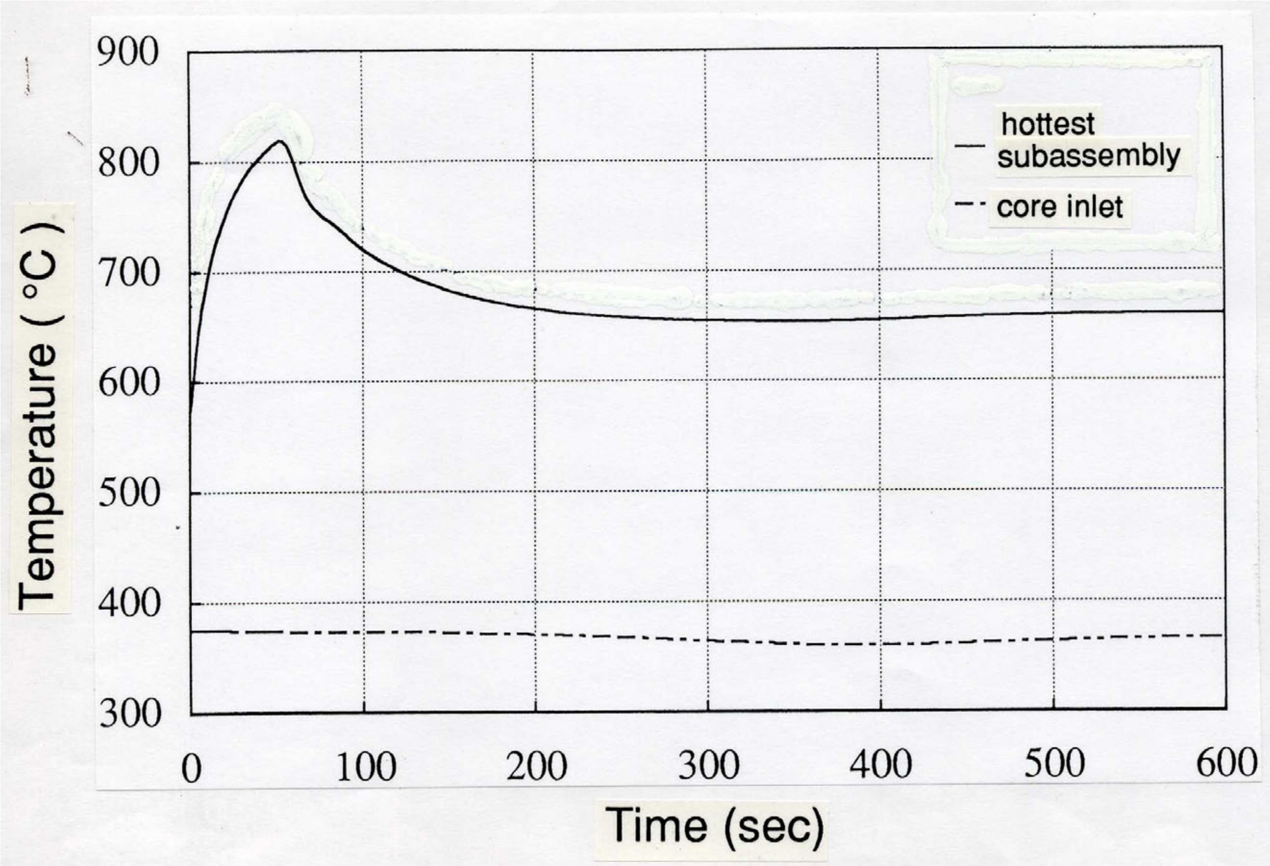


FIG. XXII-7. Temperature transition in case of ULOF.

The following equipment is provided to control the leak rate:

(1) Hydrogen meters are provided at the SG outlet for small leakages, less than 10 g/s.
(2) Acoustic detectors and cover gas pressure gauges are provided for medium leakages, less than 1 kg/s.
(3) Rupture sensors are provided for large leakages, less than DBL.
(4) Manometer-seal level meters are included as a back-up leak detection system for a BDBL.

A sodium-water reaction product / pressure release system consisting of 30B×4 piping and rupture disks are provided at the secondary vessel for a DBL event.

A manometer seal is provided at the circumference of the secondary vessel in addition to the pressure release system for a BDBL event.

From preliminary analysis, the increase of the quasi-steady state pressure of the cover gas in the secondary vessel is about 0.12 MPa for a DBL and 0.22 MPa for a BDBL. These values are much less than the external buckling pressure of the primary vessel and thus, the structural integrity of the primary vessel can be secured. The analytical results are summarized in Table XXII-2.

TABLE XXII-2. PRESSURE INCREASE IN CASE OF DBL AND BDBL

	Maximum water leak rate	Secondary cover gas	Reactor cavity
		Pressure rise / Allowable pressure	Pressure rise / Allowable pressure
Design basis leakage (DBL)	50 kg/s	0.12 MPa / 0.44 MPa	-
Beyond design basis leakage (BDBL)	125 kg/s	0.22 MPa / 0.44 MPa	0.22 MPa / 0.30 MPa

XXII-1.6.4. Proliferation resistance

A detailed examination has not been performed.

XXII-1.6.5. Technical features and technological approaches used to facilitate physical protection of MDP

A detailed examination has not been performed.

XXII-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of MDP

A detailed examination has not been performed.

XXII-1.8. List of enabling technologies relevant to MDP and status of their development

The enabling technologies for the MDP are as follows:

(1) Technologies to prevent sodium-water reaction or to mitigate its consequences. The scope of necessary R&D includes:
- Development of a water leak detection method.
- Evaluation of pressure propagation in the secondary vessel.
- Mock-up tests for a large leak rate.
- Establishment of post-accident maintenance and repair methods.

(2) In-service inspection (ISI) technology and guidelines. Required are:
- Development of ISI methods.
- Establishment of guidelines for ISI.

(3) Reactor and fuel cycle technology for metallic fuel. The necessary R&D include:
- Acquisition of irradiation data for fuel licensing.
- Development of cladding materials.
- Demonstration of safety and fuel integrity.
- Demonstration of the technical feasibility of fuel reprocessing and fabrication technologies.

(4) Immersion type large-scale electromagnetic pumps. The required R&D include:
- Evaluation of high temperature and in-sodium characteristics of the EMP coil.
- Evaluation of irradiation characteristics of the EMP coil.

- Evaluation of flow characteristics in the EMPs.
- Demonstration of structural integrity and performance by a large-scale model.

(5) Seismic isolation design. The R&D is necessary for:
- Development and evaluation of seismic isolation technology.
- Development of a reliability evaluation procedure.
- Establishment of guidelines for seismic isolation technology.

(6) The design of upper internal structure (UIS) and variable arm type in-vessel transfer machine (IVTM). Required are:
- Design and element tests.
- Mock-up tests.

XXII-1.9. Status of R&D and planned schedule

A preliminary conceptual design has been completed but, at the moment, there is no financial support for further R&D.

XXII-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

The double pool (pool-in-pool) design represents a radical conceptual change in system configuration for sodium cooled fast reactors; its qualification would require substantial R&D, feasibility tests and a prototype or demonstration plant to be implemented. The major innovations in the engineering design of the MDP are related to incorporation of metallic fuel, the EMPs and the seismic isolation. These would also require substantial R&D, feasibility tests and demonstrations, as outlined in previous sections.

XXII-1.11. List of other similar or relevant SMRs for which the design activities are on-going

No similar SMRs are under design elsewhere.

XXII-2 Design description and data for MDP

XXII-2.1. Description of the nuclear systems

Tables XXII-3, and XXII-4 show specifications of the MDP core and fuel.

TABLE XXII-3. NUMBER OF ASSEMBLIES

ASSEMBLY TYPE	NUMBER OF ASSEMBLIES
Core fuel assemblies	115
Blanket fuel assemblies	42
Shielding assemblies	174
Control rods	6
In-vessel storage capacity	42

Figures XXII-8 and XXII-9 show the core map and a cross section of the MDP fuel; the heat removal paths of the MDP are shown in Figure XXII-10.

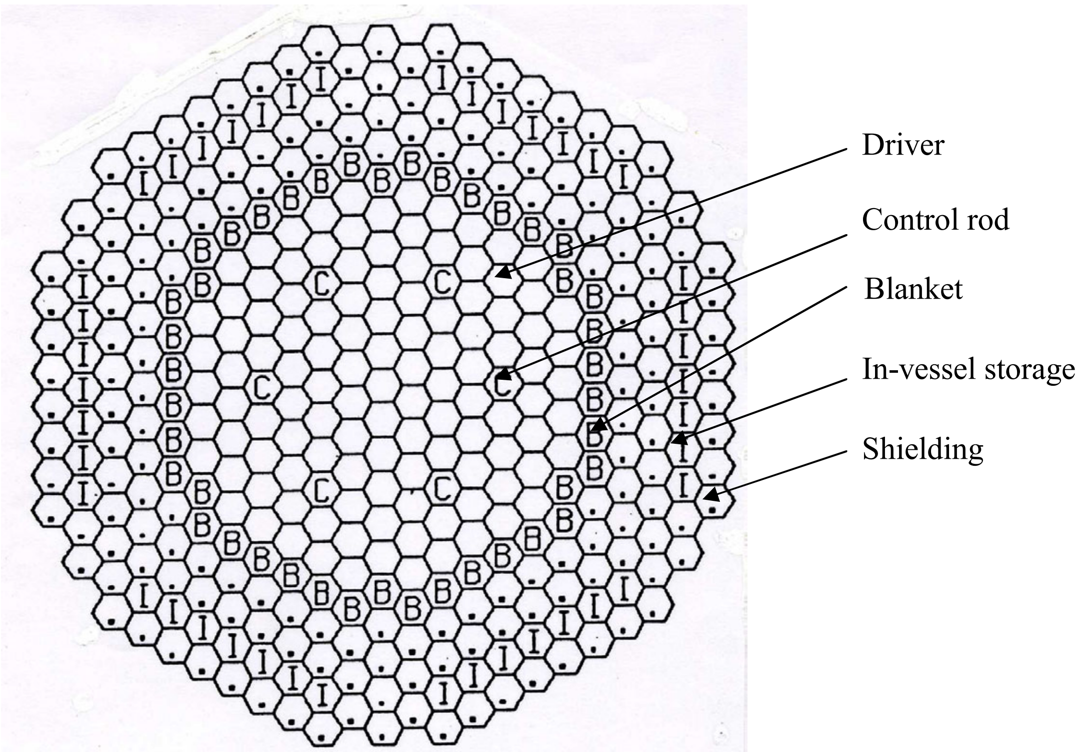


FIG. XXII-8. Core map.

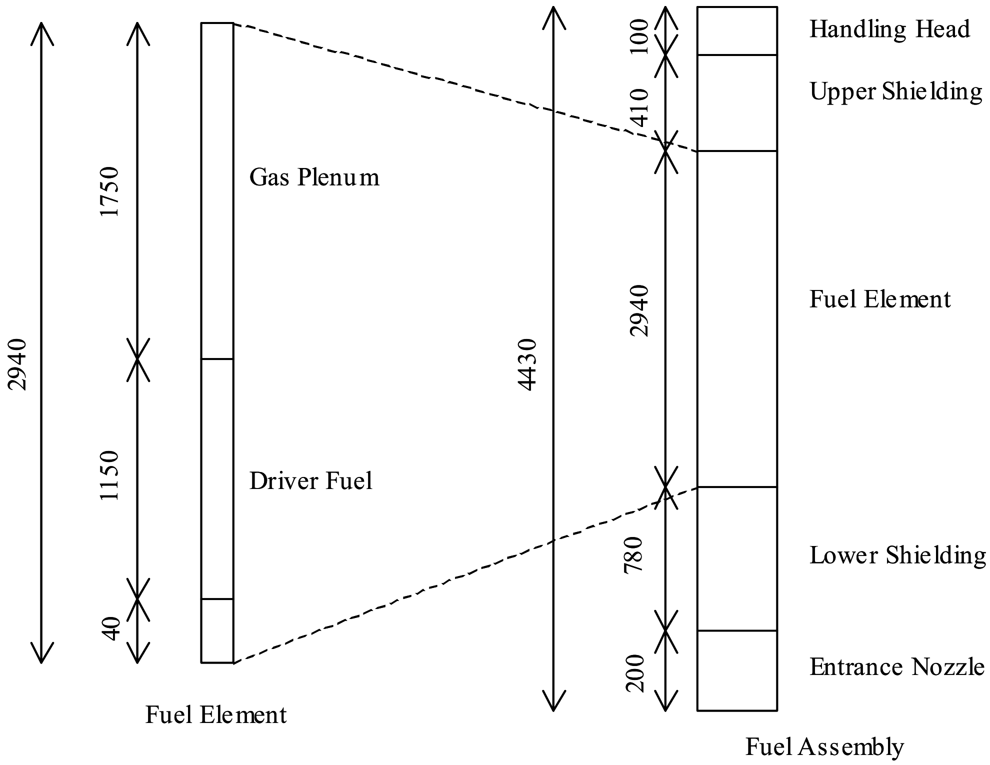


FIG. XXII-9. Cross section of fuel.

TABLE XXII-4. FUEL SPECIFICATIONS

PARAMETER	CORE	BLANKET
Number of pins per assembly	271	127
Smear density (% theoretical density)	80	85
Cladding thickness (mm)	0.42	0.4
Pin diameter (mm)	8.0	12.5
Active length (cm)	115	115
Plenum length (cm)	175	175
Pin pitch (mm)	9.36	13.6
Duct inside flat-to-flat (mm)	157	157
Duct wall thickness (mm)	4.0	4.0
Gap distance between ducts (mm)	4.0	4.0
Assembly lattice pitch (mm)	169	169
Assembly length (cm)	450	450

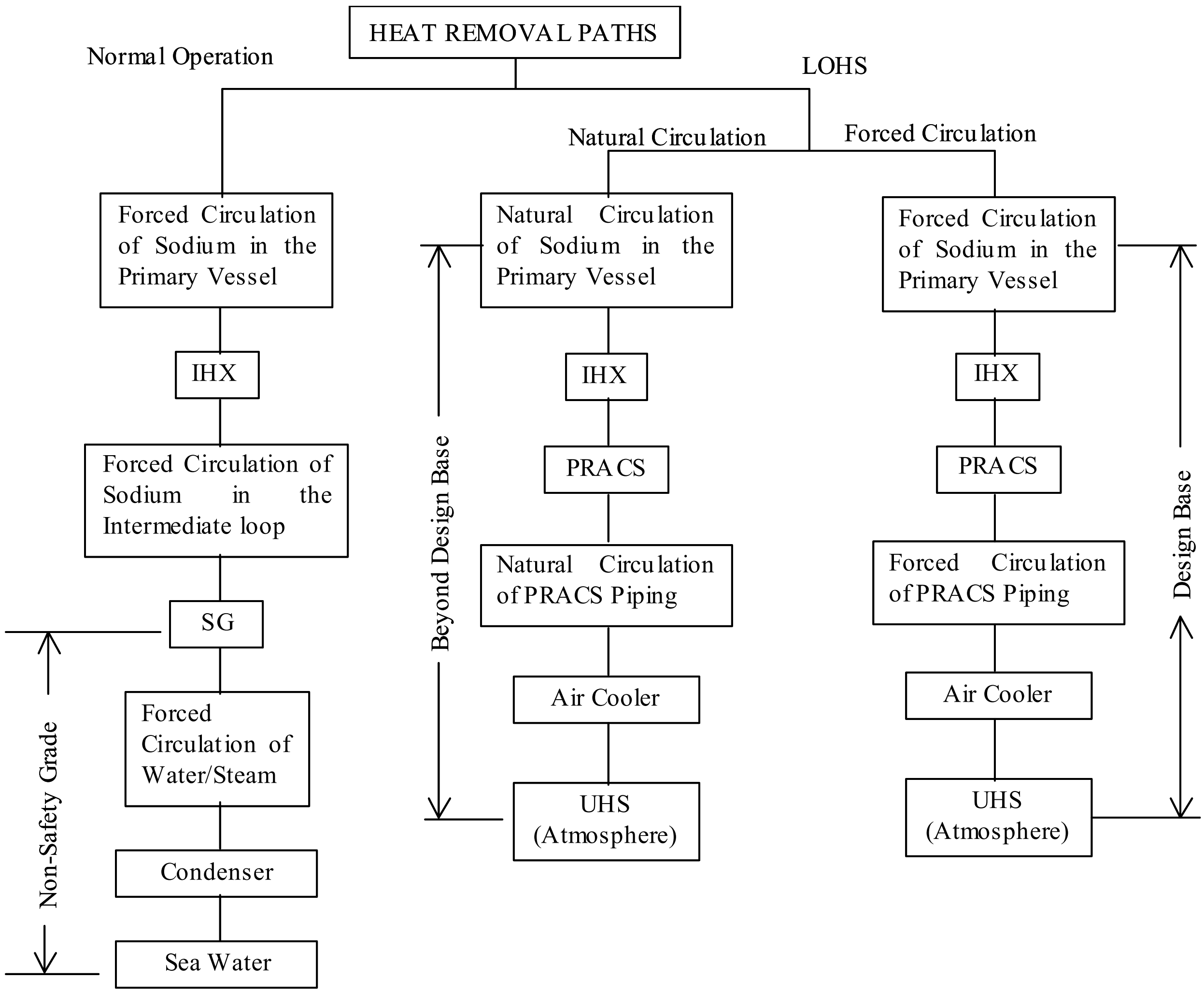


FIG. XXII-10 Heat removal paths of MDP.

XXII-2.2. Description of the turbine generator plant and systems

A superheated steam cycle is used in the MDP, based on the TC6F-23 turbine plant equipment.

XXII-2.3. Systems for non-electric applications

The MDP incorporates no systems for non-electric applications.

XXII-2.4. Plant layout

Figure XXII-11 shows the plant layout for a 4-module MDP plant producing 1300 MW(e). Each reactor has its own turbine generator because considerations are given to the first operation of each module to assure the independence of each module. The control building, the fuel transport equipment, etc., are commonly used.

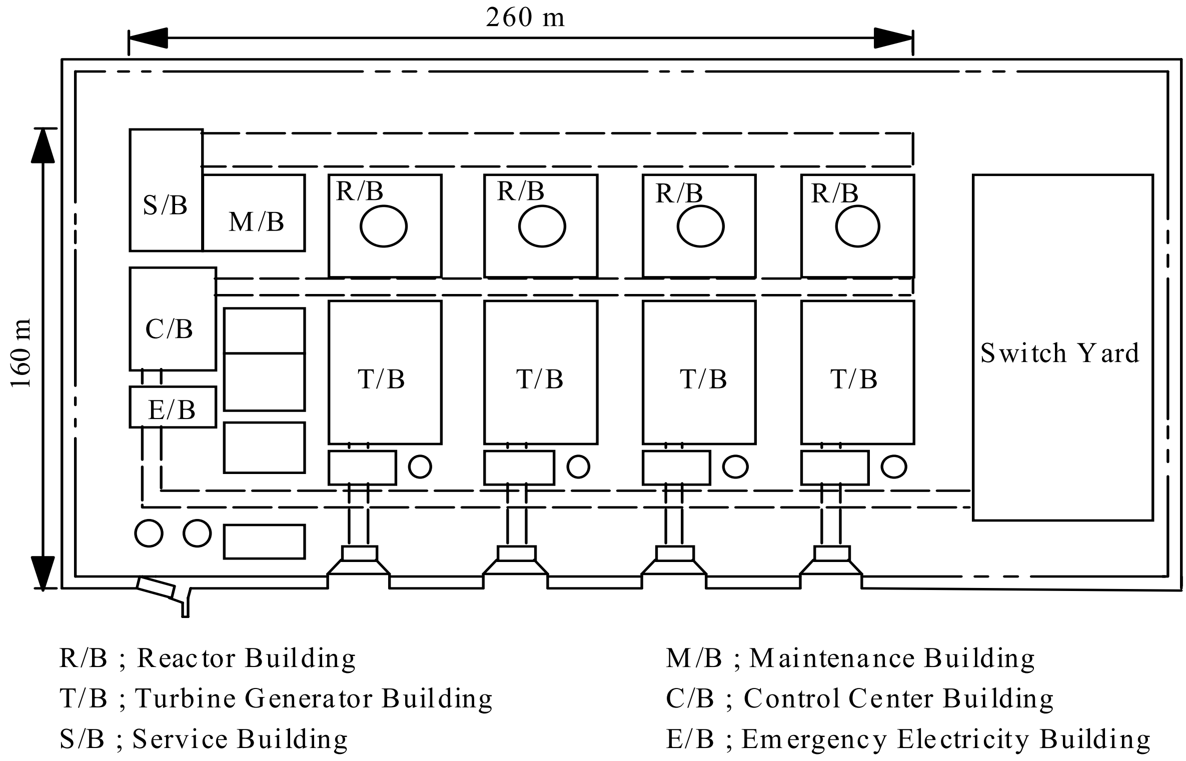


FIG. XXII-11. Layout of a 1300 MW(e) MDP based plant.

REFERENCES

[XXII-1] HATTORI, S., An innovative LMFBR concept –Double pool type LMFBR, International Symposium on LMFBR Development (Proc. of the Institute of Applied Energy (IAE) Tokyo 1984), p.233.

[XXII-2] KINOSHITA, I., et al., Development of small modular double pool reactor for early realization of FBR practical application, Fast Reactors and Related Fuel Cycles (Proc. of Int. Conf. Kyoto, November 1991), Vol.2, p.14.3–1.

ANNEX XXIII

LEAD-BISMUTH COOLED REACTOR WITH A HIGH LEVEL OF NATURAL CIRCULATION (RBEC-M)

RRC "Kurchatov Institute", Russian Federation

XXIII-1. General information, technical features and operating characteristics

XXIII-1.1. Introduction

The RBEC-M is a lead-bismuth cooled fast reactor with a high level of primary coolant natural circulation and a gas lift system in the primary circuit to ensure a supply of inert gas (argon) in the coolant under the core.

The name reflects the basic technology of the concept: a fast neutron spectrum, heavy metal lead-bismuth coolant, a high level of natural circulation with a nominal operation of inert gas blowers and safe cooldown of the core after the trip of gas supply blowers.

The RBEC-M reactor is a conceptual development based on the preliminary design of the RBEC reactor, hereafter referred to as a "basic project".

The direct predecessor of the RBEC-M is the design named RBEC [XXIII-1, XXIII-2, XXIII-3, and XXIII-4], one of the Russian-developed designs of fast reactors with heavy metal coolants.

The preliminary design of the RBEC reactor of 900 MW(th) and 340 MW(e) was completed in the 1990s by Russian design and scientific institutions: OKB "Gidropress", Russian Research Centre (RRC) "Kurchatov Institute" and IPPE, with the participation of VNIINM and RIAR.

The main objective of the development of the RBEC lead-bismuth cooled fast reactor was to provide a reliable solution for nuclear fuel breeding, while using an approach alternative to sodium cooled fast reactors. It was assumed that design development of a nuclear power plant (NPP) with such reactor could be completed in a rather short period, with modest expenditures for additional testing and qualification of separate equipment units.

The following technological solutions, validated in practice in previous reactor developments, became the focus of the RBEC reactor concept:

- Technology for quality control of the liquid metal coolant and methods for the protection of structural materials against corrosion/erosion, as well as structural materials themselves, developed for marine nuclear reactors [XXIII-5].
- Wide (or "open") fuel rod lattices, application of spacer grids to fix fuel rod bundles, fuel assemblies without shrouds (ducts), a cluster-type control and protection system, developed for light water reactors [XXIII-6].
- A three-circuit nuclear steam supply system (NSSS), a system of rotating plugs and the design of fuel rods, developed for fast reactors with sodium cooling [XXIII-7].

Already at the stage of the conceptual development, several options of the RBEC plant and core layouts had been studied to find optimal solutions for safety enhancement and economy improvement. A principle approach to the development of the RBEC modifications was to use the same reactor vessel, main NPP equipment and operational parameters as developed earlier for the basic RBEC project.

In particular, measures to minimize the void reactivity effect were considered, including reduced core dimensions or abandoned lateral and axial blankets, reduced fractions of the structural materials and coolant in the core and a decrease in the effective coolant density. Use of the gas lift system to reduce the effective coolant density was found to be one of the most promising approaches to enhance reactor safety and improve economics.

The present description gives an overview of a modified option of the RBEC reactor concept named RBEC-M [XXIII-8, XXIII-9]. The RBEC-M is characterized by the following major innovations aimed at the enhancement of safety parameters and economic efficiency:

- Application of a gas lift system (a supply of inert gas under the reactor core) to improve the neutron-physical parameters, safety and plant economy.
- Use of mixed U-Pu nitride fuel based on ^{15}N, to improve breeding parameters and optimize reactivity effects.
- Use of a two-circuit nuclear steam supply system (NSSS) with an integral layout of equipment to improve plant economy.
- Application of passive systems for reactor shutdown and auxiliary cooling.

The main task to be addressed at present for development of the RBEC-M concept is the transition from the initial stages, which consisted of the demonstration of the possibility of combining the advantages of different reactor technologies in one nuclear power plant, to the next stage of optimizing the concept. This must include new specific requirements imposed on fast reactors considered within the nuclear energy system.

The conceptual studies for the RBEC-M are performed at the RRC “Kurchatov Institute” (Moscow, Russia).

XXIII-1.2. Applications

It is assumed that fast reactors of the RBEC type, while producing electricity in a base load mode, would also provide the breeding of nuclear fuel and maintain necessary neutron balances in the whole nuclear energy system. In the future, such reactors could help eliminate the necessity to supply enriched uranium to the entire large-scale nuclear energy system and thus, would minimize ecological and proliferation risks related to natural uranium mining and enrichment.

Other applications of nuclear power plants with the RBEC-M have not been considered to date.

XXIII-1.3. Special features

The RBEC concept provides for this reactor to be used as an element of the multi-component nuclear power structure with optimized nuclide flows between elements. The main functions of the RBEC-M within such a system are to provide effective closure of the nuclear fuel cycle with respect to U and Pu, extended breeding of nuclear fuel and economically effective base-load electricity generation.

The RBEC-M is a land based nuclear power station.

XXIII-1.4. Summary of major design and operating characteristics

The simplified principal scheme of a NPP with the RBEC-M is shown in Fig. XXIII-1.

To create a RBEC-M cylindrical core with an effective diameter of 4.24 m, hexagonal fuel assemblies without shrouds are used, with a 1.0 m height of the active fuel assembly part.

The RBEC-M module is about 10 m in height, has a double-walled cylindrical vessel and includes: core; gas lift system; control and protection system; automatic control systems; and emergency depressurization system (to be used in case of steam generator failure). Besides, the reactor module contains 12 steam generators installed around the circumference of ∅6.0 m. Each steam generator has a vessel of ∅1.3 m; the height of the tubing coils is 4.5 m.

Twelve sections of the passive reactor auxiliary cooling system (PRACS) are installed in the reactor module around the circumference of ∅7.0 m. The diameter and height of the PRACS vessel is 0.47 m and about 6 m, respectively. The nominal Pb-Bi coolant flow rate through 12 PRACSs makes 15% of the whole reactor flow rate. The PRACS working fluid is naturally circulated environmental air. About 1.4 % of the generated power in nominal conditions is transferred to the environmental air.

The RBEC-M concept applies such innovative design features as abandoning main circulation pumps and use of a gas lift system. The latter bubbles an inert gas (argon) into the Pb-Bi coolant under the reactor core and promotes coolant circulation in the primary circuit.

The RBEC-M reactor includes: a refuelling system; a system for gas heating of the reactor module vessel; a system for filling and draining the primary coolant; a clad failure detection system; a system of the primary coolant chemistry, etc.

The plant is placed in a hermetic reinforced concrete containment that may be partly or fully underground to increase seismic stability of the equipment and to create the best conditions for localizing and eliminating the consequences of hypothetical accidents.

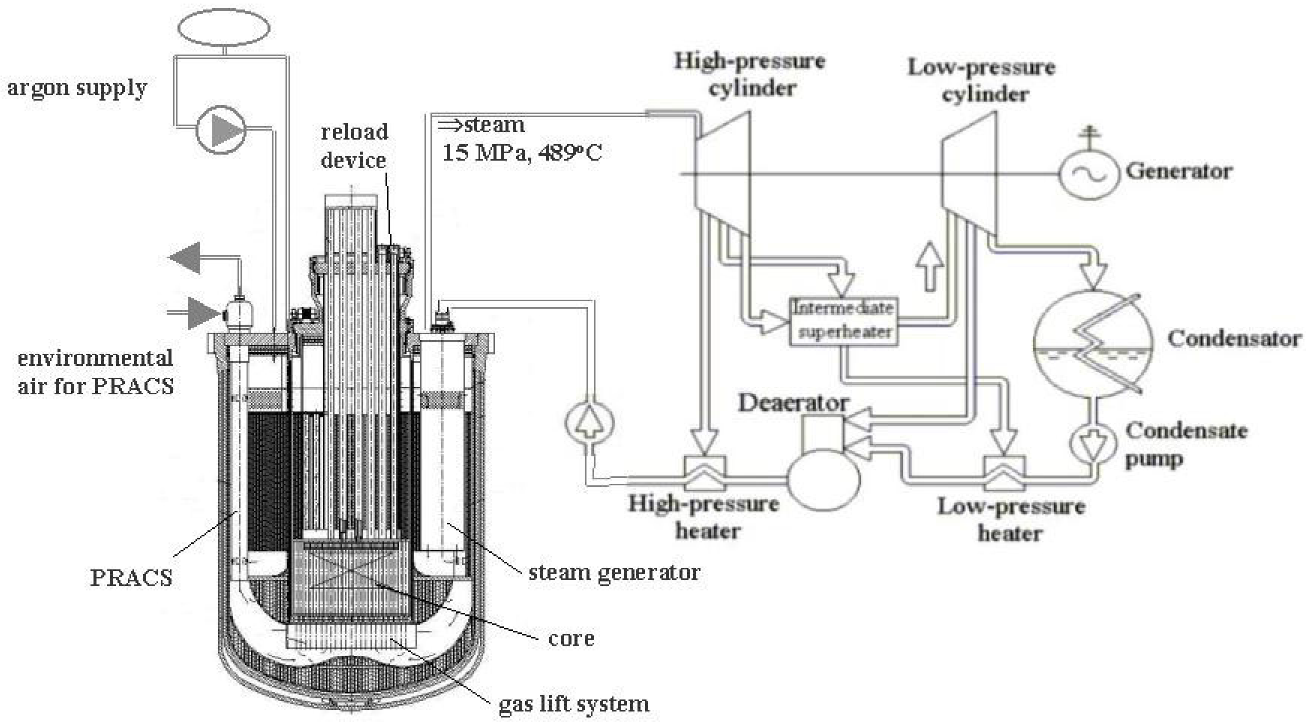


FIG. XXIII-1. Principal scheme of RBEC-M plant.

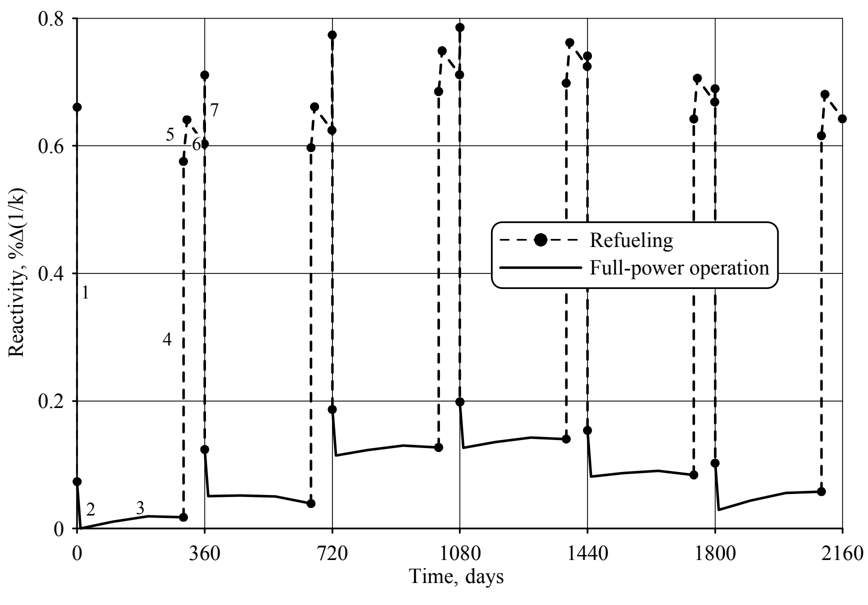
Table XXIII-1 summarizes the major design and operating characteristics of the RBEC-M.

TABLE XXIII-1. SUMMARY OF MAJOR DESIGN AND OPERATING CHARACTERISTICS

MAJOR DESIGN CHARACTERISTICS	
Installed capacity - Thermal - Electric	 900 MW(th) 340 MW(e)
Fuel type	Cylindrical container-type fuel rods with fuel pellets of mixed uranium-plutonium nitride fuel of 13.3 g/cm^3 density; steel claddings
Fuel enrichment	Total plutonium content in the mixed fuel is 13.6 weight %, fissile plutonium content– 9.4 weight %
Coolant	Lead-bismuth eutectics with argon; average gas content– 31% vol.
Moderator	No
Structural materials	Fuel rod clads and steam generator tubes: – ferritic-martensitic steel EP-823 (12%Cr-Si); Reactor module – steel; In-reactor displacers - gray cast iron.
Core	Cylindrical, arranged from hexagonal subassemblies without shrouds; effective diameter is 4.24 m, active core height is 1.0 m. Surrounded by fertile side blankets (one row of subassemblies) and fertile axial blankets 100 mm thick
Reactor vessel	Double-wall monoblock-type cylindrical vessel with built-in primary systems. Dimensions of main and guard vessels are ∅8400×80 mm and ∅8000×80 mm, respectively. Vessel height is about 10 m.
Number of circuits; thermodynamic cycle type	Two-circuit nuclear steam supply system with generation of superheated steam (15 MPa, 489ºC) in the secondary circuit and feedwater supply at 288ºC
NPP operation mode	Electricity generation in the base load
Thermodynamic cycle efficiency	38%
NPP style	Modular, integral type
Design load factor	0.82
Design service lifetime, years	40

TABLE XXIII-1. (Continued - 1)

NEUTRON-PHYSICAL CHARACTERISTICS		
Temperature reactivity coefficients [1]	Doppler at nominal temperature, $\Delta(1/k)/K$	-3.38×10^{-6}
	Doppler constant	-3.81×10^{-3}
	Core axial expansion, $\Delta(1/k)/K$	-1.63×10^{-6}
	Core radial expansion, $\Delta(1/k)/K$	-9.12×10^{-6}
	Reactor power reactivity coefficient at N_{nom}, $\Delta(1/k)/\%N$, where N is power	-1.61×10^{-5}
Reactivity coefficient on coolant effective density, $\Delta(1/k)/K$	In the whole core	2.83×10^{-6}
	In lateral blanket	-3.72×10^{-7}
	In lateral Pb-Bi reflector	-2.95×10^{-7}
	In upper blanket and chimney	-1.46×10^{-6}
	In lower blanket and plenum	-1.12×10^{-6}
	Total (in the whole reactor)	-2.37×10^{-7}
Reactivity effects,[2] $\Delta(1/k)$	Temperature isothermal effect while heating from refuelling temperature up to core inlet temperature	-2.68×10^{-3}
	Reactor power effect when transferring from zero to nominal power	-3.19×10^{-3}
	Neptunium effect	-7.6×10^{-4}
	Change of reactivity with burn-up between refuelings (burn-up reactivity swing)	$-1.2\times10^{-4} \div 2.8\times10^{-4}$
	Change of reactivity with loading of a fresh fuel assembly	$1.6\times10^{-4} \div 1.5\times10^{-3}$
	Void reactivity effect	-7.02×10^{-2}
Time history of reactivity changes in the RBEC-M equilibrium fuel cycle		



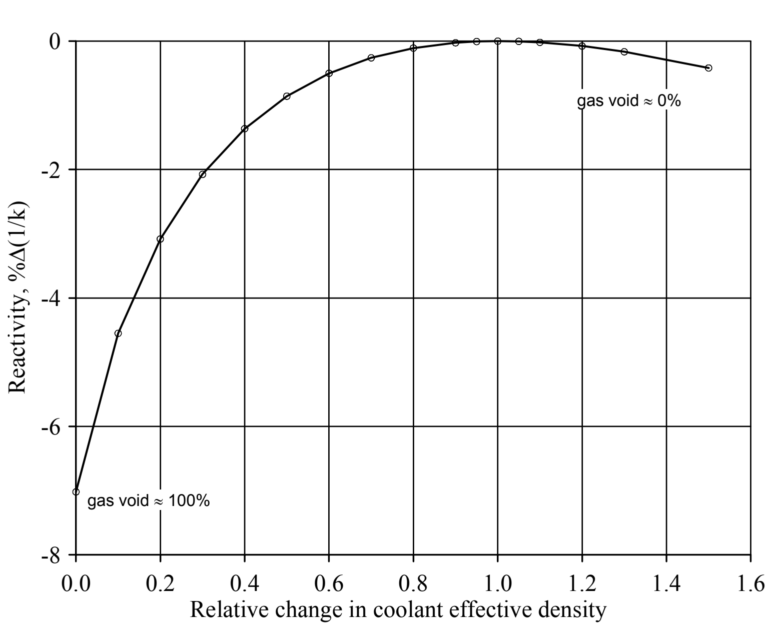
[1] The reactor state, for which all reactivity coefficients were determined, was assumed to be full-power operation with nominal parameters of the coolant (temperature and flow rate at the core inlet) and gas lift flow.

[2] The effective fraction of delayed neutrons is 0.0037; the lifetime of prompt neutrons is $4.5\cdot10^{-7}$ s.

TABLE XXIII-1. (Continued - 2)

<table>
<tr><td colspan="4">NEUTRON-PHYSICAL CHARACTERISTICS</td></tr>
<tr><td>Power peaking factors in the core

(changes of these values between refuelings are insignificant)</td><td colspan="2">K_q (fuel assembly)
K_z (core axial)
K_V (core volume)

Small variation of power peaking factors is stipulated by the chosen approach to power flattening by the amount of fuel (use of three core zones with different fuel rod diameters and the same Pu content), and by ensuring the value of the active core breeding ratio close to 1.</td><td>1.26
1.20
1.51</td></tr>
<tr><td colspan="4">REACTIVITY CONTROL</td></tr>
<tr><td>Control and protection systems (CPSs)</td><td>- Two independent mechanical systems (active and passive) consisting of absorber elements; rods of the active system are grouped in clusters of three rods each.
- Gas lift system</td><td colspan="2">Each of these systems can shut down the reactor and keep it subcritical from any nominal or accident state even in case of failure of the most effective cluster</td></tr>
<tr><td rowspan="2">Efficiency of mechanical systems, Δ(1/k), %</td><td>- Active CPS consisting of absorber rods located in the central tubes of 72 fuel assemblies of the 1st core zone, combined in 24 clusters; each of them controls three absorber rods</td><td colspan="2">- One cluster of active CPS in average – 0.19
- Total efficiency – 4.45</td></tr>
<tr><td>- Passive CPS consisting of 48 absorbers in the periphery of the 1st core zone and in the 2nd core zone</td><td colspan="2">- One rod of passive CPS in average – 0.05
- Total efficiency – 2.24</td></tr>
<tr><td>In normal operation at rated power, the insertion of negative reactivity and decrease of reactor power is provided when for some reasons the gas void in the core is uniformly changed (both increased and decreased)[3]</td><td colspan="3">
</td></tr>
</table>

[3] Stability of the reactor operation is provided by negative reactivity feedbacks and by the coolant flow rate feedback: possible small deviations from a nominal gas void cause coolant flow rate change, which compensates for this deviation.

TABLE XXIII-1. (Continued - 3)

THERMAL-HYDRAULIC CHARACTERISTICS		
Circulation type	A feature of the primary thermal-hydraulic scheme of the RBEC-M reactor module is an absence of reactor circulation pumps and application of the gas lift system for organization of Pb-Bi coolant circulation by supply of inert gas (argon) under the core	
Parameters of circulation system	Mass coolant flow rate through the core, kg/s Mass argon flow rate through the core, kg/s Average gas void in the core, % Coolant pressure at fuel assembly inlet, MPa Hydraulic resistance of the primary circuit, MPa	44 527 2.3 31 0.7 0.2
Distribution of coolant temperature in the RBEC-M reactor module, K	Core, inlet/outlet Steam generator, inlet/outlet Passive reactor auxiliary cooling system, Inlet/ Outlet Lower plenum; Inlet/ Outlet	650 / 792 792 / 620 792 / 767 642 / 650
Velocities of Pb-Bi coolant in the RBEC-M reactor module, m/s	Core Lateral blanket Chimney Steam generator PRACS Downcomer	1.2 – 1.6 -0.06 1.6 – 2.1 0.5 0.5 1.3
DESIGN LIMITS		
Design temperature limits ensuring working stability of “fuel-coolant” composition	Coolant (non-freesing) Fuel rod clads (corrosion stability) Fuel (absence of changes in nitride fuel structural state due to production of Pu metal phase and evaporation of Pu)	Higher than 396 K Lower than 900 K Lower than 2000 K
Design limit on mechanical interaction	Existence of fuel-clad gap - absence of pellet-clad mechanical interaction (PCMI) - during the whole fuel lifetime to reduce the probability of clad failure	
Design limit on burnup	Maximum burn-up is 14% FIMA (fissile materials)	

The figures XXIII-2 and XXIII-3 show peak values of fuel and cladding temperatures for each fuel assembly during the fuel cycle (without accounting for uncertainty factors). The peak fuel temperature is reached in a fresh fuel assembly and makes 1375 K (Fig. XXIII-2); the peak cladding temperature is 888 K (Fig. XXIII-3).

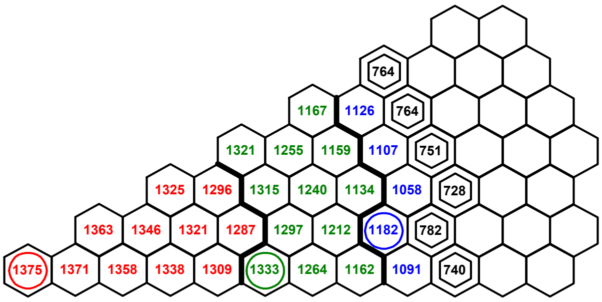


FIG. XXIII-2. Peak fuel temperatures of RBEC-M core during irradiation, K.

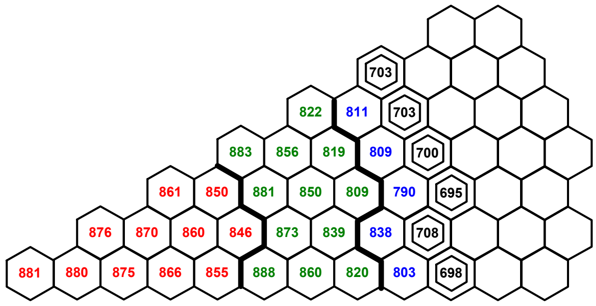


FIG. XXIII-3. Peak cladding temperatures of RBEC-M core during irradiation, K.

Thermal-mechanical analysis of the basic irradiation regime was conducted for the fuel rods with maximum linear power in each of three core zones. Fuel assemblies containing these fuel rods, are indicated by circles.

It was determined that the peak volume fuel swelling is 14%, 8% and 7%, for the first, second and third core zones, respectively. Thus, the design requirement to exclude PCMI during the fuel lifetime in the RBEC-M core is fulfilled. Decrease of the fuel-clad gap width causes a fuel temperature decrease during the entire fuel lifetime. The peak fuel temperature decrease during irradiation is from 1375 K to 1095 K in the first core zone, from 1333 K to 1101 K in the second core zone and from 1182 K to 1058 K in the third core zone. The irradiation cycle and breeding ratio data are given in Table XXIII-2.

TABLE XXIII-2. DATA ON BURN-UP CYCLE AND BREEDING RATIO

CHARACTERISTIC	VALUE	COMMENT
Total fuel lifetime, effective full power Days (EFPD)	1800	- chosen on the basis of fuel rod performance analysis in basic irradiation conditions.
Interval between refuellings, EFPD	300	- chosen to minimize the burn-up reactivity swing between refuelings.
Fraction of the core reloaded at once	1/6	- uniformly in all three power profiling core zones.
Fraction of reactor zones in fuel breeding,	Core Axial blanket Lateral blanket	1.037 / 0.998 0.137 / 0.137 0.152 / 0.150
After / before refuelling	Reactor	1.326 / 1.282
Average / maximum fuel burn-up, % FIMA	Core -1 Core -2 Core -3	8.8 / 13.2 b 6.4 / 9.6 4.0 / 6.0

Detailed characteristics of the material balances for fuel isotopes are given in Tables XXIII-3 and XXIII-4.

TABLE XXIII-3. INVENTORIES OF FISSILE ISOTOPES IN ONE FUEL ASSEMBLY (FA) OF ACTIVE CORE ZONES AND ONE SUBASSEMBLY (SA) OF FERTILE BLANKETS, KG

Isotope	FA Core-1	FA Core-2	FA Core-3	Axial blankets FA Core-1	Axial blankets FA Core-2	Axial blankets FA Core-3	SA of lateral blanket
^{235}U	0.069	0.082	0.114	0.016	0.019	0.026	0.136
^{238}U	69.363	82.066	113.770	16.075	19.019	26.366	135.764
^{238}Pu	0.146	0.172	0.239	-	-	-	-
^{239}Pu	6.651	7.869	10.909	-	-	-	-
^{240}Pu	2.676	3.166	4.389	-	-	-	-
^{241}Pu	0.919	1.087	1.507	-	-	-	-
^{242}Pu	0.546	0.646	0.895	-	-	-	-
^{241}Am	0.089	0.106	0.146	-	-	-	-
Total heavy atoms	80.459	95.194	131.969	16.091	19.038	26.392	135.900

TABLE XXIII-4. CHANGE (IN KG) OF ISOTOPIC COMPOSITION OF THE DISCHARGED FUEL (FUEL LIFETIME 1800 EFPD OR SIX CALENDAR YEARS) WITH RESPECT TO THE FUEL OF A FRESH FA OR BLANKET SA, AT AVERAGE FUEL BURN-UP

Isotope	FA core-1	FA core-2	FA core-3	Axial blankets FA core-1	Axial blankets FA core-2	Axial blankets FA core-3	SA of lateral blanket
^{234}U	0.005	0.006	0.009	0.000	0.000	0.000	0.000
^{235}U	-0.040	-0.038	-0.037	-0.007	-0.006	-0.005	-0.028
^{236}U	0.009	0.009	0.010	0.001	0.001	0.001	0.006
^{238}U	-7.648	-6.731	-5.817	-1.041	-0.865	-0.723	-3.733
^{237}Np	0.019	0.020	0.020	0.002	0.002	0.002	0.008
^{238}Pu	-0.029	-0.028	-0.027	0.000	0.000	0.000	0.001
^{239}Pu	0.656	0.634	0.644	0.761	0.679	0.610	3.191
^{240}Pu	0.202	0.143	0.113	0.044	0.027	0.016	0.095
^{241}Pu	-0.399	-0.446	-0.547	0.002	0.001	0.000	0.003
^{242}Pu	-0.029	-0.025	-0.018	0.000	0.000	0.000	0.000
^{241}Am	0.082	0.129	0.238	0.000	0.000	0.000	0.000
^{242m}Am	0.007	0.007	0.008	0.000	0.000	0.000	0.000
^{243}Am	0.065	0.059	0.054	0.000	0.000	0.000	0.000
^{242}Cm	0.008	0.008	0.008	0.000	0.000	0.000	0.000
^{243}Cm	0.001	0.000	0.000	0.000	0.000	0.000	0.000
^{244}Cm	0.010	0.007	0.004	0.000	0.000	0.000	0.000
Total heavy atoms	73.376	88.949	126.631	15.850	18.874	26.290	135.428
Fission products	7.083	6.245	5.338	0.241	0.164	0.102	0.472
Burn-up, % FIMA	8.8	6.6	4.0	1.5	0.9	0.4	0.3

Economic characteristics of the RBEC-M plant are presented in Table XXIII-5.

TABLE XXIII-5. ECONOMIC CHARACTERISTICS OF RBEC-M PLANT

How specific capital cost was estimated	Via metal demands and costs per kg of nuclear steam supply system (NSSS) equipment (US $/kg)	- Fraction of NSSS equipment cost in the total NPP cost was assumed equal to 25% in calculations. - For estimating metal mass and cost parameters of the elements and systems of RBEC-M NSSS, which are not comprised in the reactor module, the data for Russian VVER-1000 and BN-600 reactors were used. - Specific cost of the RBEC-M reactor module was assumed equal to that of BN-600.
Specific cost of RBEC-M NSSS equipment	Serial module US $418-438 /kW(e)	- There are no estimates for a prototype module due to the lack of reliable data on capability of the present-day Russian industry to produce equipment for the RBEC-M NSSS.
Specific capital cost for a NPP with RBEC-M	US $1670-1750 /kW(e)	- Similar values obtained under the same assumptions are US $1600 /kW(e) for VVER-1000 and US $2560 /kW(e) for BN-600.
Specific cost of coolant for RBEC-M	US $79-106 /kW(e)	- With account for coolant, total specific cost of a RBEC-M NSSS with the coolant exceeds the similar value for VVER-1000 by 25-35%.
Estimates of other costs	Have not been performed at this (conceptual) design stage	

XXIII-1.5. Outline of fuel cycle options

To sustain the development of nuclear power as a large-scale energy technology, the structure of a nuclear energy system should meet certain requirements for efficiency, resource utilization, and safety. In addition, the structure of the future nuclear energy system should also be capable of providing solutions to the disposal of radioactive wastes and spent nuclear fuel accumulated during the previous stages of nuclear power evolution.

The RRC “Kurchatov Institute”, in cooperation with other Russian organizations, is currently developing the concept of a three-component nuclear energy system with closed U-Pu and Th-U nuclear fuel cycles [XXIII-10, XXIII-11], Fig. XXIII-4. According to this concept, the structure of a future nuclear energy system should be based on closed U-Pu and Th-U fuel cycles with optimal neutron and isotopic balances, should provide necessary nuclear fuel breeding and multiple fuel recycling, and should be able to minimize the quantities of radioactive wastes. In addition to this, purposeful use of by-products, e.g. certain radioactive isotopes, could be foreseen.

As it was already mentioned, the RBEC concept assumes that reactors of this type could be used as elements of the multi-component NP structure with optimized nuclide flows between elements.

In principal, there could be different modes of RBEC operation within the nuclear energy system:

- Multiple recycling to ensure maximum use of the power potential of uranium, plutonium and probably, of some quantities of minor actinides (MAs). The RBEC could be fuelled by fissile materials from the spent fuel of light-water reactors (uranium and MOX) and/or by plutonium generated in the blankets of breeder reactors. Thus, different fuel cycle chains could be used for the RBEC. In the distant future when cheap uranium is exhausted, the reactor can be equipped with thorium blankets to produce ^{233}U for thermal reactors.
- An on-site closed fuel cycle with only depleted uranium being consumed could be considered. In this cycle, spent fuel assemblies are reprocessed after cooling, the discharged fuel is mixed and separated from fission products with a specified grade of purification, and depleted uranium is added to provide the specified content of plutonium and MAs in the "fresh" fuel. The excess plutonium of RBEC-M could then be used in other reactors or for the deployment of new reactors of RBEC-M type.

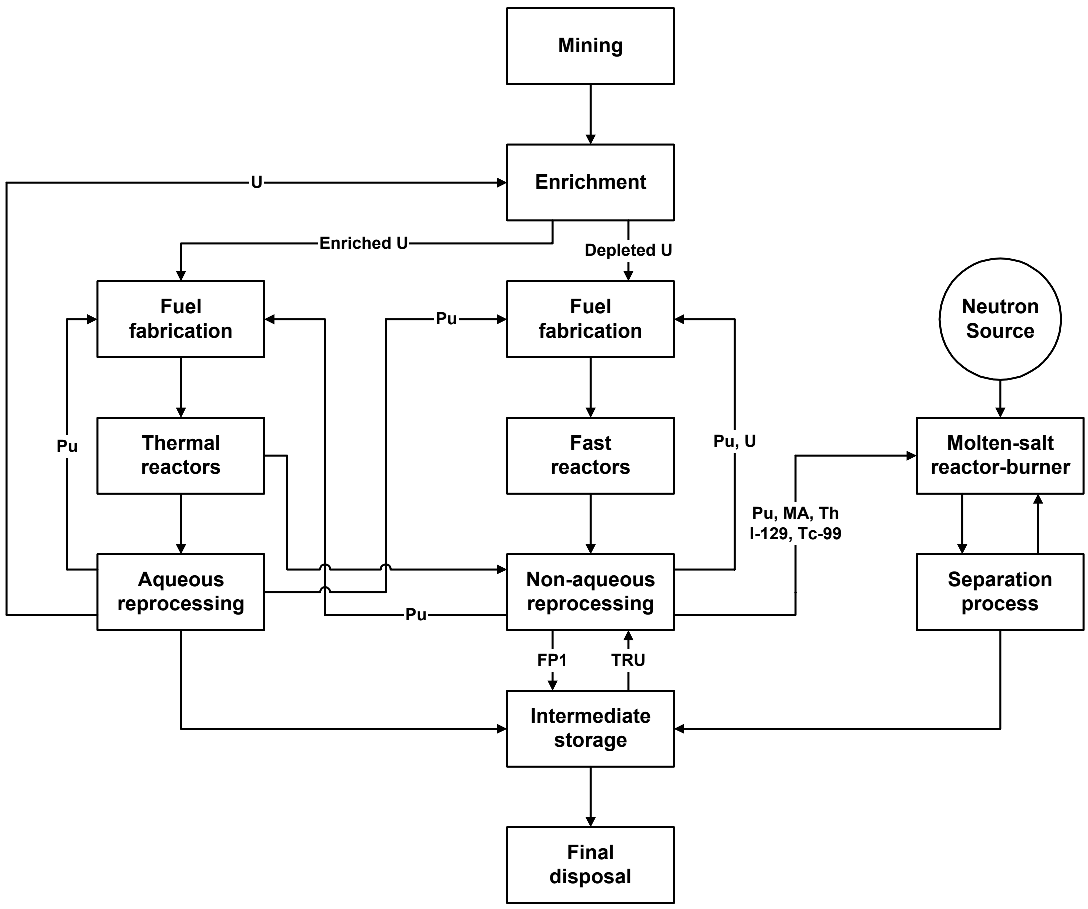


FIG. XXIII-4. Possible schematics of the three-component nuclear power system with a closed fuel cycle for all actinides, including Pu and hazardous long-lived fission products.

To analyze the RBEC fuel cycle concept in both cases, a scheme of operation for all elements of the NP structure as well as all functional links between these elements, should be considered. Optionally, technologies of fuel reprocessing without deep purification of fuel and/or without separation of pure Pu could be considered. For this reason, nuclide flows directed to RBEC depend on the amounts, functions and parameters of other elements of the structure of a nuclear energy system. It should be noted that the efficiency of fuel utilization could depend to a greater extent on the organization of nuclide flows in the nuclear power structure than on the breeding ratio of fast reactors themselves.

First estimates have already been obtained in the RRC "Kurchatov Institute" for nuclide flows in the simulated multi-component system of nuclear power with a closed fuel cycle for all actinides, including Pu and hazardous long-lived fission products [XXIII-12].

The RBEC-M studies performed so far focus on the most realistic options of reactor operation. This corresponds to the initial introduction of this reactor type in the structure of a emerging large-scale nuclear power, when significant amounts of cooled spent fuel from light water reactors would available and the Pu from this spent fuel could be used to feed the RBEC-M.

Nevertheless, preliminary studies show there are no major limitations on the RBEC-M reactor operation in any closed fuel cycle, including the on-site fuel cycle. Reactor parameters necessary for safety and other aspects can be provided by a sufficiently flexible set of reactor core variables.

XXIII-1.6. Technical features and technological approaches that are definitive for RBEC-M performance in particular areas

XXIII-1.6.1. Economics and maintainability

Due to a selected distinctive RBEC-M application, special market requirements and needs, specifically, the possible requirements of developing countries and peculiarities of their power sectors were not considered in the technical development.

On the other hand, during the concept development, special attention was paid to reducing capital and construction costs. The main objective of the initial RBEC concept optimization was to improve economic efficiency while preserving a high safety level.

Improvement of economic efficiency in the RBEC variant is achieved by: abandoning of intermediate circuits and reduction of the reactor module height through the integral layout of equipment; abandoning of main circulation pumps and use of the gas lift system to ensure the primary coolant circulation; and minimization of the coolant volume in the reactor module.

A moderate reactor module height (10 m) with adjustments for the change of coolant volume and dimensions of the in-vessel equipment has significant effects on the economics. The cost of the RBEC-M NSSS (with coolant) is estimated to be 47% lower than that of the basic RBEC variant and close to the specific cost of NSSS equipment for operating light-water reactors. Moreover, reasonable dimensions of the RBEC-M equipment could make it possible to fabricate certain equipment items at a factory and deliver them to a site in an assembled form.

To date, there has been no special consideration for low O&M and fuel reloading costs in the RBEC-M design development.

XXIII-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

For the RBEC-M type reactors, the perception of sustainability and waste management is related to their function in the large-scale nuclear power system, where each element plays its own special role.

In the proposed nuclear power structure, reactors of the RBEC-M type are basically designated for electricity generation and maintenance of the neutron balance, along with "traditional" fast and thermal power reactors. Dedicated facilities designed for burning minor actinides and transmutation of some fission products ("reactors-burners") could be foreseen. Therefore, in spite of the principal possibility to use the RBEC-M for the purposes of transmutation, core loading with a fuel with high minor actinide content has not yet been considered in detail.

For environmental impacts, the RBEC-M concept has as a design goal to maintain the integrity of the fuel element under all normal and accident conditions. It is suggested that the occupational radioactive exposure of personnel and public is kept below the prescribed limits. Detailed estimates of the radiological consequences of the release of radionuclides from fuel elements have not yet been performed.

XXIII-1.6.3. Safety and reliability

Safety concept and design philosophy

The RBEC-M safety approach and design philosophy are based on the idea of retention of radionuclides in the fuel elements under all normal and postulated accident conditions to the degree that personnel doses and doses at the NPP boundary fall within the existing guidelines.

Defence in depth is assured by redundant and diverse barriers to the release of radionuclides from the fuel, in combination with provisions to ensure the integrity of the barriers. The barrier structure is based on the properties of the fuel to retain a significant fraction of each of the radionuclides deposited in the fuel pellet, and mainly, on the prevention of radioactivity release to the primary coolant by fuel rod claddings. For this purpose, the absence of pellet-clad mechanical interaction is ensured during the fuel lifetime and in accidents, to reduce the probability of clad failure. An additional barrier preventing the release of radionuclides is the two-circuit NSSS scheme.

Active and passive safety systems and inherent safety features

The RBEC-M design was optimized to achieve the optimum combination of inherent safety features and safety systems (active and passive).

Structure of the defence-in-depth

Some major highlights of the design, structured to meet the three first levels of defence in depth are given below:

Level 1: Prevention of abnormal operation and failures

- Negative and high magnitude values of the power and temperature reactivity effects and coefficients.
- Burn-up and fuel breeding in the core are balanced; therefore, a positive reactivity which can be inserted in the core, is minimized.
- Use of the gas-lift system as a part of the coolant chemistry system for transporting and introducing the necessary gas components into the coolant directly under the core.

- Pb-Bi coolant is fire and explosion-proof; Pb-Bi has a high boiling temperature, exceeding the melting temperature of structural materials, eliminating the possibility of significant pressure increases and positive reactivity insertion caused by the void effect, as well as the possibility of a departure from nucleate boiling.
- Quality-assured factory fabrication of equipment, including the mono-block.

Level 2:Control of abnormal operation and detection of failures

- The high level of Pb-Bi natural circulation in the primary circuit providing inertia of the physical processes and stability.
- Insertion of negative reactivity and decrease of reactor power when the gas void in the core is uniformly changed (both increased and decreased).
- Automatic control of coolant chemistry and other processes.
- Application of a double-wall cylindrical vessel of the mono-block with a controlled gas gap between walls.

Level 3: Control of accidents within the design basis

- Use of two independent systems, active and passive, affecting the reactivity; each system can bring the reactor to a subcritical state and maintain it under normal and emergency conditions, provided that one of the most effective control rods fails to operate.
- The gas lift system in RBEC-M allows for practically prompt reactivity control and if necessary, for transition of the reactor to a reduced power level or to a subcritical state.
- Use of the coolant with natural circulation for emergency core cooling.
- A passive reactor auxiliary cooling system inside the reactor module allows the total removal of decay heat in case of a steam generator trip at temperatures below 1000 K.

Design basis accidents and beyond design basis accidents

At the current stage of development, a tentative list of design basis accidents was elaborated based on the experience of operating plants. This list cannot be considered final or complete. A probabilistic analysis of the entire range of possible initiating events has not yet been performed.

The following transients without scram were considered as design basis accidents:

- Insertion of $1 positive reactivity over a second.
- A trip of the gas supply to the gas lift system.
- A two-fold increase in the rate of gas supply to the gas lift system.
- A decrease in feedwater temperature at the steam generator inlet to 300 K.
- Total NPP blackout (a trip of gas supply to the gas lift system and feedwater supply to the steam generator).

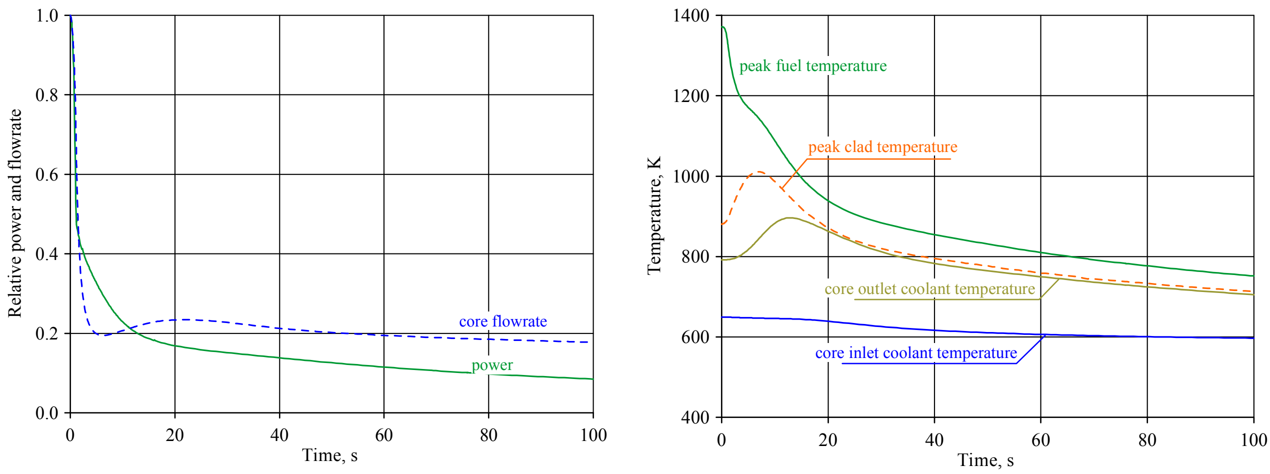
The estimated peak fuel and cladding temperatures in accidents without scram are given in Table XXIII-6.

At the trip of the gas supply to the gas lift system, the scenario leads to a fast decrease in the coolant flow rate down to natural circulation levels of about 18% of the nominal reactor flow rate (Fig. XXIII-5).

TABLE XXIII-6. PEAK FUEL AND CLADDING TEMPERATURES IN ACCIDENTS WITHOUT SCRAM

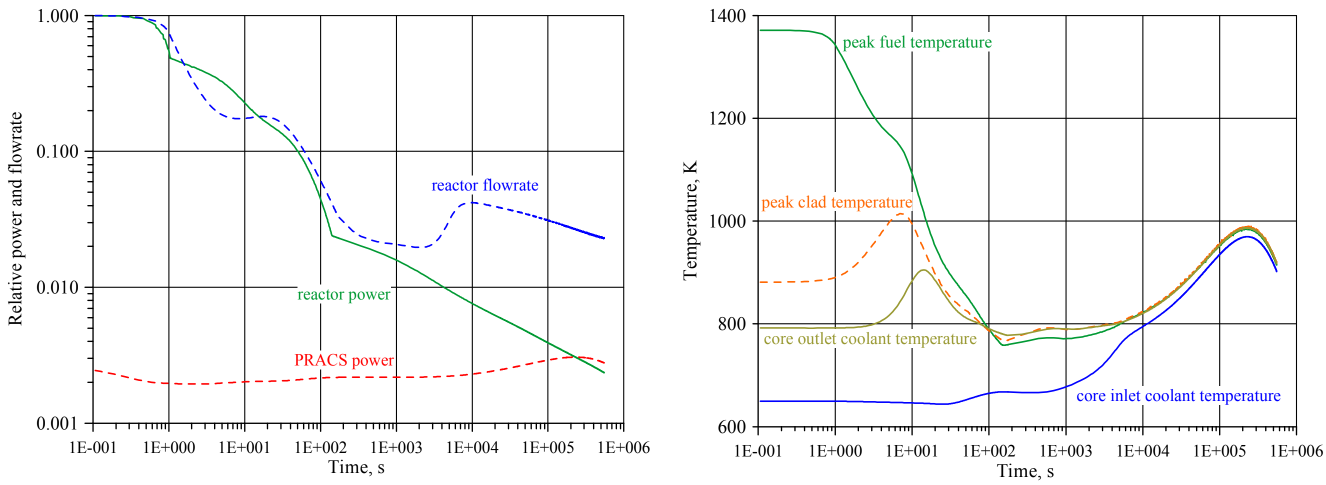
ACCIDENT PROCESS	PEAK TEMPERATURE, K	
	Fuel	Cladding
Transient overpower	2382	1303
Loss of gas flow	1375	1010
Increase of gas flow	1375	881
Overcooling of primary circuit	1536	893
NPP blackout	1375	1015

An NPP blackout accident (Fig. XXIII-6) is considered to be the most dangerous event for cladding integrity because in this accident the cladding remains at temperatures of above 900 K for a long time, if no accident management actions are undertaken.



a) Relative power and coolant flow rate　　b) Fuel, cladding and coolant temperatures

FIG. XXIII-5. Calculation results for loss of gas flow.



a) Relative power and coolant flow rate　　c) Fuel, cladding and coolant temperatures

FIG. XXIII-6. Calculation results for NPP blackout.

At present, the list of beyond design basis accidents for the RBEC-M is not complete. Such a list could contain e.g., the leak of Pb-Bi coolant from the reactor module vessel at its failure or massive depressurization of the steam generator tubes, leading to the ingress of significant quantities of water into the primary coolant.

Provisions for safety under seismic conditions

To ensure the seismic stability of the RBEC-M plant, it is suggested that common design procedures prescribed by regulatory documents should be followed.

It is necessary to note that for a reactor module of the basic variant (RBEC reactor), which considerably exceeds by mass-dimensional parameters the design suggested for the RBEC-M, mono-block strength analysis proved the seismic resistance of the plant to earthquakes up to magnitude 8 on the MSK-64 scale.

XXIII-1.6.4. Proliferation resistance

The most obvious problems related to the possible proliferation of nuclear materials for fast reactors to which the RBEC-M belongs, stem from an opportunity for an expanded breeding of fuel, first of all, in fertile blankets surrounding the reactor core. Besides, design of the fuel subassemblies of fast reactors can include fertile inserts, which do not necessarily noticeably affect the reactor performance.

Basically, there are two major groups of measures to secure proliferation resistance, which are: (1) intrinsic features - those features resulting from the technical design of nuclear energy systems, including facilities, materials and technologies; and (2) extrinsic measures - those measures resulting from legislative decisions and undertakings related to nuclear energy systems, e.g. safeguards.

At the conceptual design stage of the RBEC-M, focus was made on the intrinsic features. The following design solutions could be classified as intrinsic features contributing to an enhanced proliferation resistance of the RBEC-M:

(1) The RBEC-M fuel assembly design differs essentially from designs of the traditional fast reactor subassemblies (e.g., sodium cooled reactors of the Russian BN-type):

- The presence of a channel in the center of a fuel assembly allows measurements of nuclear materials (to realize IAEA safeguards) not only from outside of the assembly (i.e. at the external rows of fuel rods) but also from within it.
- The use of wide fuel rod lattice and spacer grids allows the manufacture of a RBEC-M fuel assembly as a dismountable component, which could also facilitate implementation of the IAEA safeguards. In this case, there is an opportunity to secure all the fuel rods by measurements and to exclude the possibility of an undeclared replacement of intermediate fuel rod rows by pins with fertile materials. In the case of a non-dismountable fuel assembly design, the sensitivity of contemporary methods of measurement for nuclear materials allows measurements only of the external fuel rod rows and up to two internal fuel rod rows (if a guide tube for control in the fuel assembly center is available).
- In a dismountable fuel assembly, the registration unit is a separate fuel rod, not the whole fuel assembly.

(2) The use of non-aqueous methods is assumed for reprocessing of spent RBEC-M fuel:

- New non-aqueous technologies for reprocessing the spent fuel of fast reactors have been developed in Russia. These are characterized by incomplete removal of fission

products (about 1 % of them remains in reprocessed fuel) and separation of only curium from the fuel (neptunium and americium, and 1 % of the curium remain), allowing the production of "fresh" fuel for fast reactors but preventing the use of such fuel for weapon programmes [XXIII-12].

(3) A dedicated study conducted in the RRC "Kurchatov Institute" provided a systematic analysis of different concepts of nuclear power development [XXIII-13]. The main outcomes of this study are the following:

- Using the approach of estimating the protective properties of barriers preventing the diversion of nuclear materials at all stages of the fuel cycle, it is possible to show that when the breeding of fuel is expanded and excess plutonium is used to feed thermal reactors (thus gradually replacing uranium), the proliferation risk in nuclear power can be reduced, since the volumes of uranium mining and separation are reduced.
- The concept of a multi-component nuclear energy system with a limited number of fast reactors (e.g. of the RBEC-M type) producing the fuel for other types of reactors could be rather attractive from the viewpoint of non-proliferation.

These conclusions are based on a judgment that uranium mining and enrichment are elements of the nuclear fuel cycle with the greatest contribution to the proliferation risk. Certainly such results, which on the whole are favourable for accepting the suggested RBEC-M application in a system of large-scale nuclear power, need further confirmation.

Obviously, the least intrinsically protected stages of the RBEC-M fuel cycle, namely, fabrication, transport and storage of fresh fuel assemblies, would demand strengthened measures of accounting and verification, even in view of fuel being contaminated by fission products.

XXIII-1.6.5. Technical features and technological approaches used to facilitate physical protection of RBEC-M

For physical protection of a NPP of the RBEC-M, the development of specific protection measures differing from those foreseen for reactors of other types was not assumed.

Transients initiated by postulated events in the operation of a NPP with RBEC-M proceed sufficiently slowly due to the significant thermal capacity of the first circuit and passive heat removal from the reactor vessel, so that the initiation of accident management is possible in due time.

The function of passive safety systems is largely based on physical laws, appreciably reducing opportunities to impair operation of these systems by the intentional actions of the personnel.

XXIII-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of RBEC-M

Specific market requirements and needs (particularly, those of developing countries) were not considered at the present design stage of the RBEC-M.

Providing full-scale fuel cycle service is assumed within the framework of the described multi-component structure of nuclear power.

XXIII-1.8. List of enabling technologies relevant to RBEC-M and status of their development

The basic enabling technologies requiring a substantial amount of R&D are the following:

(1) Gas lift concept:

- Undeniably, the experimental qualification now available of the proposed concept of the gas lift application is insufficient.
- Experiments are necessary to study flow regimes, hydrodynamics, gas void and processes for separating the mixtures of lead-bismuth and inert gas under the conditions typical for normal and abnormal regimes of the RBEC-M operation.
- Special attention should be focused on an experimental study of the thermal regimes of fuel rods in a two-phase flow of lead-bismuth and inert gas.

(2) Core arrangement based on fuel rods with innovative nitride fuel and claddings of a steel of the ferritic-martensitic grade.

- Deeper knowledge is required on the properties of the fuel composition, the fuel pellet-cladding interaction (FPCI), which is the basis for the RBEC-M core performance.
- Accident analysis shows that at the NPP blackout accident, the ferritic-martensitic steel cladding remains at temperatures above 900 K for a rather long time. Therefore, reliable test data are necessary on the strength and corrosion properties of this steel at high temperatures to make final conclusions about fuel rod failure probability in this accident.
- Additional tests are required to study the thermal stability of (U-Pu) N and to obtain the precise ultimately allowable temperature for mixed nitride fuel.
- In the future it would be useful to develop cost estimates for nitrogen enrichment by the isotope ^{15}N.

(3) The economic viability of the RBEC-M substantially depends on the fuel composition to provide sufficiently high fuel burn-ups. Currently, moderate achievable parameters are considered which do not ensure radical improvements in fuel use compared with the available experience, and therefore, a search for new solutions in this field is required.

XXIII-1.9. Status of R&D and planned schedule

The conceptual studies for the RBEC-M are performed at the RRC “Kurchatov Institute” (Moscow, Russia).

Recently, work toward elaboration of the RBEC-M concept was also motivated by an exchange of scientific and technical information with the organizations in Russia and abroad, especially those involved earlier in the development and/or those developing new reactors cooled by heavy metals. In particular, the French Commissariat à l’Energie Atomique (CEA) and the Japan Nuclear Cycle Development Institute (JNC) should be mentioned. In 2001, the RBEC concept description in the format for the Generation IV programme was submitted to the US DOE and included in the final document on concepts of reactors cooled with liquid metals [XXIII-14].

At the current stage of the RBEC-M development - conceptual design – the developers have not contemplated companies or institutions that may be involved in the project research, design and demonstration (RD&D) nor the associated time frame, though they have preferences in this respect.

Advancement of the RBEC-M concept could most probably be realized by an association of developers of the initial (basic) project, namely, the Russian design and scientific institutions

OKB Gidropress, RRC “Kurchatov Institute” and IPPE, with the participation of VNIINM and RIAR. In such cooperation in the 1980-1990s, the preliminary design of the basic RBEC plant was developed.

Estimations of the time frame within which the RD&D to substantiate the RBEC-M project could be finalized and later, the design could be implemented, in the current conditions of Russia, depends on financial resources and political decisions. However, without prejudice, the time intervals for RD&D will be similar to those foreseen in the Generation IV Roadmap [XXIII-15] for reactors of such types - about 20 years, but current expenses in Russia could be lower than the estimate of US $990 million given in [XXIII-15].

XXIII-10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

Different from the basic RBEC, the RBEC-M will require a significant amount of R&D, detailed studies and experimental validations of the innovations. Consequently, the proposed reactor concept should be associated with installations of the 4-th generation of nuclear power reactors.

The RBEC-M reactor concept incorporates many known technical features initially used for its predecessor, the RBEC reactor that in turn was developed as a nearest alternative for sodium breeders.

Contrary to its predecessor, the RBEC-M concept is innovative since it incorporates radical conceptual changes in design approaches and system configuration in comparison with existing practice. The basic conceptual features which require validation, testing and demonstration through prototype / pilot facilities are outlined in Section XXIII-1.8.

From the history of the development of marine reactors cooled by heavy metals, it is known that problems of fuel and coolant compatibility were basically solved by large-scale experimental testing and by observing the experience of several operating installations consequently constructed.

The combination of innovative qualities of the RBEC-M can be demonstrated in sufficient volume only on a prototype reactor. The construction of a full-scale pilot facility may also be required to elaborate on the selection of parameters for reactor operation.

XXIII-2. Design description and data for RBEC-M

XXIII-2.1 Description of the nuclear systems

Reactor core and fuel design

Three zones with different fuel rod diameters are used in the RBEC-M core to flatten power distribution, coolant temperature and velocity in the core (Fig. XXIII-7).

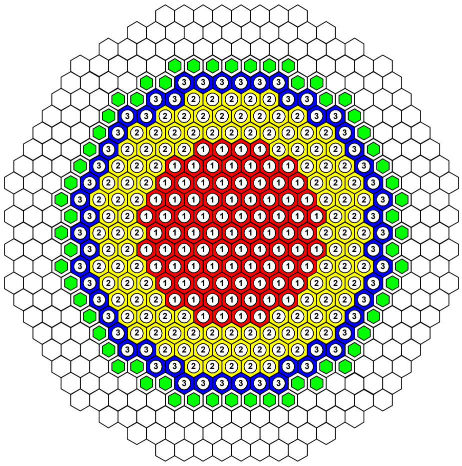


FIG. XXIII-7. RBEC-M core layout.

The fourth type of fuel assembly is used in the lateral blanket. Mixed uranium-plutonium nitride fuel with a density of 13.3 g/cm^3 and plutonium content of 13.7 % is used in all core zones. Axial and lateral blankets contain pellets of depleted uranium nitride with a density of 13.3 g/cm^3 (Table XXIII-7).

The core, including subassemblies of the lateral blanket and lateral reflector, has an effective diameter of 4.24 m and full height of fuel assemblies equal to 1.8 m.

TABLE XXIII-7. MAIN PARAMETERS OF RBEC-M CORE*

	Core-1	Core-2	Core-3	Lateral blanket
Number of assemblies	85	114	54	60
Fuel assembly pitch, mm	178			
Number of pins in a fuel assembly	252			120
Fuel rod pitch in fuel assembly, mm	10.8			15.3
Fuel rod pitch-to-diameter ratio	1.54	1.44	1.26	1.39
Fuel pellet outer diameter, mm	5.7	6.2	7.2	9.7
Fuel material	$(U_{0.863}+Pu_{0.137})N$			UN
Fuel pellet density, g/cm^3	13.3			
Radial fuel-clad gap width, mm	0.15			0.10
Fuel rod cladding thickness, mm	0.5			
Outer cladding diameter, mm	7.0	7.5	8.6	11.0
Active core height, mm	1000			

* Dimensions are given at 293 K

The mixed U-Pu nitride fuel is assumed to be manufactured to provide the level of admixtures of oxygen and carbon in the fuel below 0.1 weight % each, and fuel pellet density not lower

than 90% of the theoretical value. Nitrogen is used with an enrichment of 99.9% by ^{15}N, providing both good neutron-physical characteristics and acceptable amounts of ^{14}C generated in the reactor core for the total fuel irradiation period.

During the initial stages of development, it was assumed that the isotopic composition of plutonium in the fresh fuel would correspond to “reactor-grade” plutonium extracted from the cooled spent fuel of a typical light water reactor (e.g., from a 900 MW(e) PWR fuel irradiated up to a burn-up of 33 MW·d/kg U, reprocessed after 10 years of cooling) and loaded in the RBEC-M reactor in two years. The isotopic composition of uranium corresponds to depleted uranium with a content of 0.1 weight % ^{235}U.

The fuel rod free volume is filled with helium at 1 MPa pressure. A gas plenum of 500 mm height is located in the lower part of the fuel and fertile rods to mitigate the effect of fission gas release. Because of the high pin pitch-to-diameter ratio in the RBEC-M core, fuel and fertile rods are fixed with spacer grids (Fig. XXIII-8). Fuel assemblies of all three active core zones have no shrouds (ducts). A subassembly of the lateral blanket has a shroud and differs from the core fuel assemblies by a lower number of higher diameter pins.

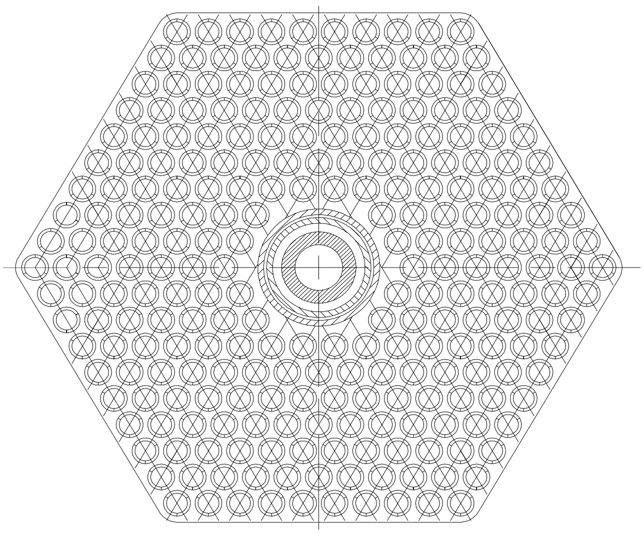


FIG. XXIII-8. RBEC-M fuel assembly design (in the center there are tubes for CPS absorbing elements).

Control and protection systems

According to the regulations adopted in Russia, the RBEC-M reactor has two independent and diverse control and protection systems (CPS), active and passive. Each system can shutdown the reactor and keep it subcritical from any nominal or accident state even in case of a failure of the most effective CPS element.

To obtain high efficiency of CPS elements, the absorber rods of active and passive systems are installed in 120 central tubes of fuel assemblies in the RBEC-M core (Fig. XXIII-9).

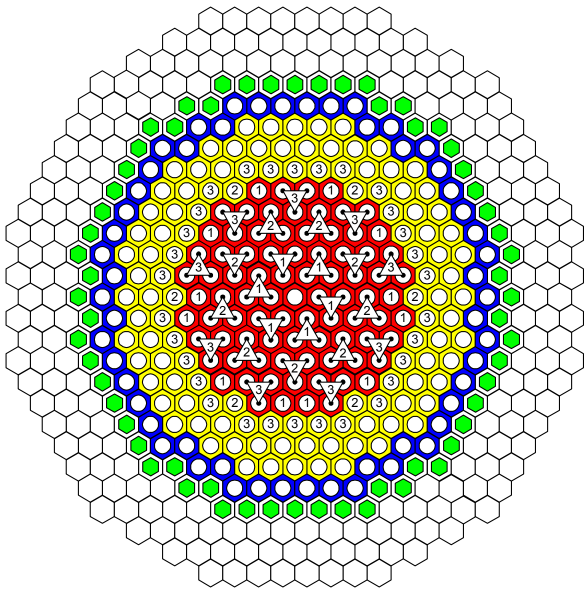
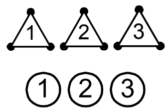


FIG. XXIII-9. Positions of CPS elements in the RBEC-M core.



Clusters of active CPS

Rods of passive CPS

The active CPS consists of absorber elements located in the central tubes of 72 fuel assemblies of the first core zone. The absorber rods of the active CPS are moved by the drives controlled by electric motors. The drives of the active CPS are combined in 24 clusters, which perform the following functions:

- Control and automatic maintenance of the power level; automatic or manual reactivity insertion into the core to reach a certain reactor power level; and planned reactor shutdown for refuelling.
- Compensation of temperature, power and neptunium reactivity effects; and compensation of burn-up reactivity swing between refuellings.
- Stabilization of the power distributions.
- Reactor scram.

The passive CPS consists of absorber elements in 12 fuel assemblies of the first core zone and 36 fuel assemblies of the second core zone. The passive absorber rods are kept in the same position below the core as the active absorber rods using special triggers that are bimetallic plates made of steel of ferritic-martensitic and austenitic grades with differing coefficients of thermal expansion. The trigger is installed on top of the fuel assembly central tube. When coolant temperatures at the fuel assembly outlet exceed 900 K, the thermal deformation of a trigger reaches its critical value leading to release of the shaft and flow up of the passive absorber rod.

Coolant and structural materials

The use of lead-bismuth as a primary coolant at relatively low core outlet temperature and coolant heatup allows the use of ferritic-martensitic steel EP-823 (12%Cr-Si) as a structural material for the core and steam generator. This steel was checked in practice for resistance against radiation swelling and radiation creep.

It is assumed that the corrosion resistance of the fuel cladding material is provided at chosen operational temperatures by a system for maintenance of the oxygen concentration in the coolant.

Basic systems of the primary circuit, reactor module

The RBEC-M reactor module has an integral layout (Fig. XXIII-10). All primary systems are located in the double-walled cylindrical vessel. The dimensions of the main and guard vessels are ∅8400×80 mm and ∅8000×80 mm, respectively. The vessel height is about 10 m. The gap between the main and guard vessel is filled with gas, with its pressure being controlled. The gas gap is designed to heat the reactor module before the start of operations by the forced convection of hot gas. In addition, the guard vessel localizes the radioactive coolant in case of a hypothetic failure of the main vessel.

Gas lift system

Application of the gas lift system and abandoning of main reactor pumps were adopted in the proposed RBEC-M concept. The elimination of pumps is to a great extent caused by the absence of qualified pump design for a reactor of medium power with heavy metal coolant. Besides, this measure decreases the primary circuit hydraulic resistance and increases the level of natural circulation.

The gas lift system is designed for argon supply in the lead-bismuth coolant under the core, for the creation of a required gas void distribution in the core and lateral blanket, as well as for the organization of the primary coolant circulation.

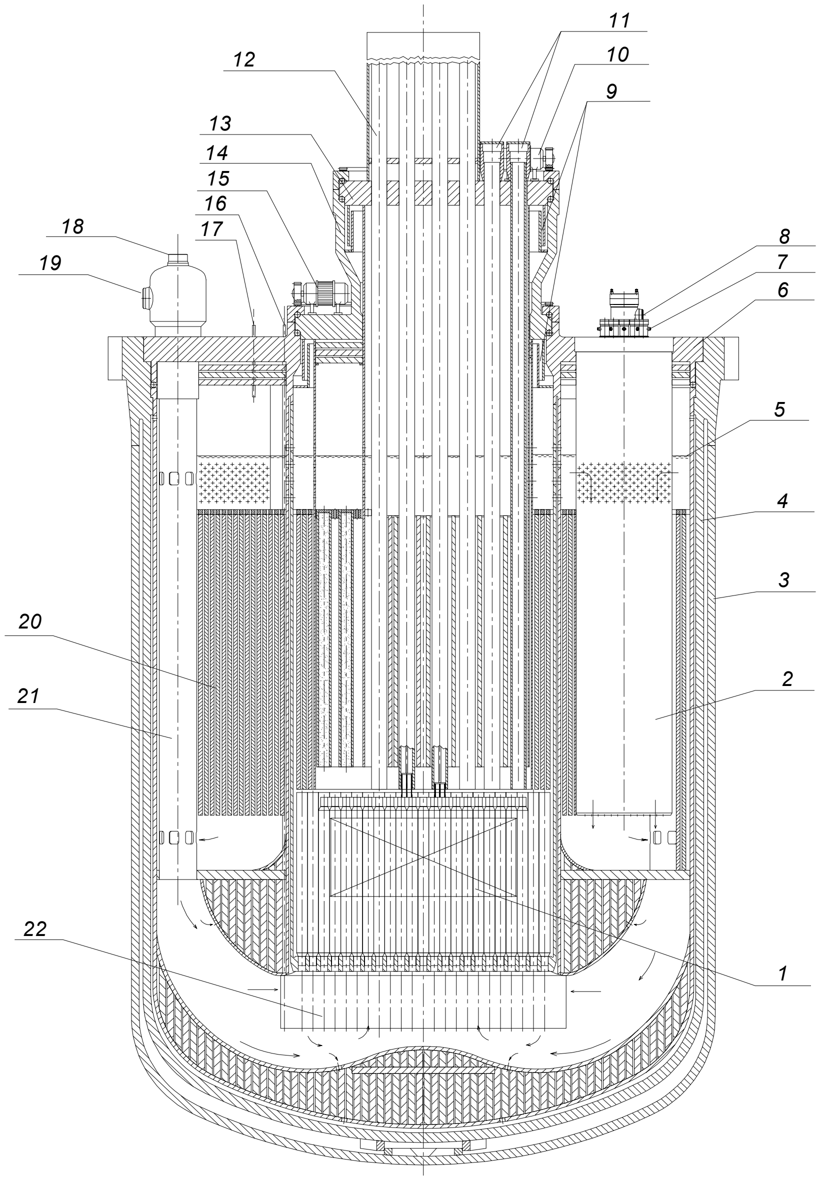
This variant of gas supply in a heavy-metal coolant was chosen because of design simplicity, the possibility of restarting the system after coolant entry into the gas flow channels, simplification of the system actuation and gas flow control, more modest hydraulic resistance, etc.

Application of the gas lift system does not introduce negative changes in the reactor neutron balance, solves the problem of a positive void reactivity effect, and promotes coolant circulation without main reactor pumps. Additionally, the gas lift system can be used as part of the coolant chemistry system to transport and introduce the necessary gas components (oxygen, hydrogen, etc.) into the coolant directly under the core.

Reverse flow is established in the core radial reflector because argon in the gas lift system is not bubbled under the radial reflector. Hot Pb-Bi coolant flows down in the core lateral reflector and mixes with cold coolant in the reactor lower plenum. The reverse flow through the core radial reflector flattens the radial temperature distributions of coolant and structures in the chimney above the core. Flattening of the coolant temperatures at the outlet from three active core zones is essentially secured by a rather flat power distribution in the active core.

Steam generators

Twelve steam generators of the tube-in-shell type are installed inside the RBEC-M reactor module; their parameters are given in Table XXIII-8.



1 – Core	10 and 15 – Electric drives
2 – Steam generator (SG)	11 – Refuelling channels
3 – Reactor guard vessel	12 – CPS drives (24)
4 – Reactor main vessel	13 and 14 – Small and large rotation plugs
5 – Coolant free level	16 and 17 – Gas inlet in and outlet from the gas lift system
6 – Reactor cover	18 and 19 – Air inlet in and outlet from PRACS
7 – Steam outlet from SG	20 – Displacers
8 – Feedwater inlet in SG	21 – Passive reactor auxiliary cooling system (PRACS)
9 – Seals	22 – Gas lift system

FIG. XXIII-10. General view of RBEC-M reactor module.

TABLE XXIII-8. STEAM GENERATOR PARAMETERS AT 293 K

PARAMETER	VALUES
Inner/outer diameter of SG vessel, mm	1300/1330
Inner/outer diameter of SG central tube, mm	340/370
Number of tubes	234
Inner/outer diameter of a tube, mm	13/16
Tube length, mm	42735
Mass of tubes, kg	5296
Number of coils	16
Axial / radial tube step, mm	19 / 30
Tube inclination angle, degrees	6
Tube bundle height, mm	4500
Heat transfer area, m^2	5466

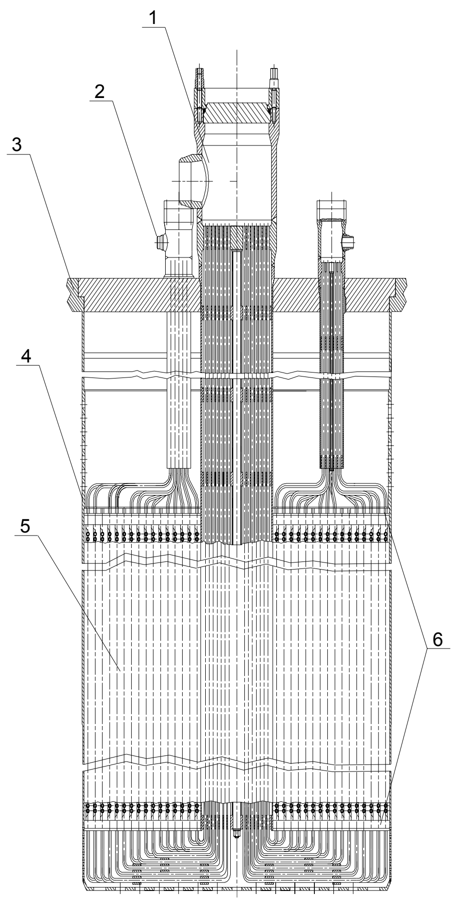
The steam generator is utilized to generate steam of 762 K and 15 MPa using the once-through scheme. The SG is of a counter-flow type: lead-bismuth flows in the inter-tube space downward; the steam-water mixture flows inside the tubes and upward (Fig. XXIII-11). The tube bundle is of a coil-type.

A system of SG emergency depressurization in case of tube failure requires future consideration.

Passive reactor auxiliary cooling system

The passive reactor auxiliary cooling system (PRACS) consists of 37 coaxial tubes of ∅50×1.5 mm, installed with a 65 mm pitch in a cylindrical vessel with a diameter of 147 mm and height of about 6 m, Fig. XXIII-12. The PRACS operation is based on the natural circulation of air, flowing downward in the inner tube of ∅39×3.0 mm and upward in the annulus of the coaxial tube. Primary coolant flows downward in the inter-tube space. After passing PRACS the coolant in the lower plenum is mixed with coolant that has passed the steam generator.

The nominal coolant flow rate through 12 PRACSs amounts to 15% of the total nominal reactor flow rate. The working fluid is the air of the environment. In this, about 1.4 % of the generated power in nominal conditions is transferred to the air of the environment. The flow rate of air in nominal conditions through one PRACS is 4.8 kg/s; air temperature at the outlet reaches 530 K.



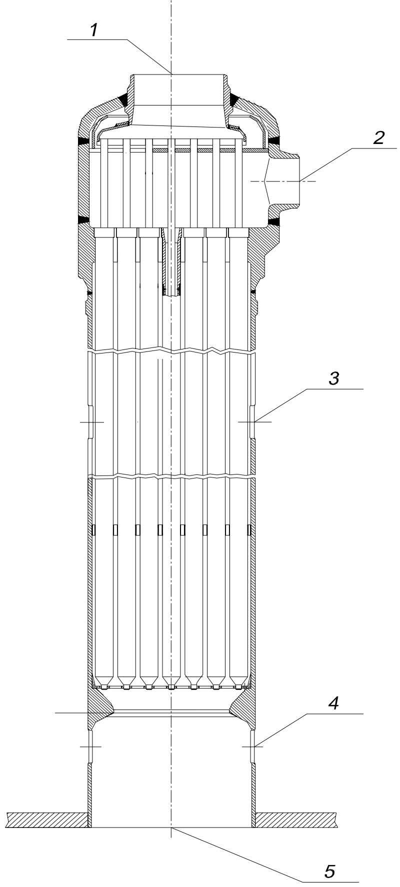
1: Feedwater header	4: Steam generator shell
2: Steam collector (13)	5: Tube bundle
3: Reactor vessel cover	6: Tube plate

FIG. XXIII-11. Steam generator of RBEC-M.

Main heat transport system

The basic function of the main heat transport system is to remove nuclear heat generated in the fuel elements of the active core and blankets by heavy metal coolant in natural circulation, driven by the gas lift system in all modes of operation, i.e. under normal reactor operation and accident conditions. Figure XXIII-13 shows the schematic diagram of heat removal paths under normal operation and in accidents.

The lead-bismuth coolant circulates according to the following scheme. The cold coolant from the downcomer and lower plenum enters the gas lift system. The coolant in the gas lift system is mixed with argon and directed into the assemblies of the active core and lateral blanket. After passing the core, the heated two-phase mixture of liquid metal and argon elevates in the channels formed by the displacers and shrouds of the CPS drives and reaches a free coolant level where gas is separated from lead-bismuth. A dominant fraction of gas bubbles are assumed to be separated on the coolant mirror in the cover gas volume because the vertical component of the velocity of coolant flowing down in the steam generators (SG) and the passive reactor auxiliary cooling system (PRACS) is not high (about 0.5 m/s).



1: Air inlet 2: Air outlet 3: Hot coolant inlet
4: Inlet of coolant from steam generator 5: Coolant outlet

FIG. XXIII-12. Passive reactor auxiliary cooling system (PRACS).

The reverse Pb-Bi coolant flow is established in the core radial reflector because argon in the gas lift system is bubbled only under assemblies of the active core and lateral blanket. After the gas is separated on the coolant-free level, hot single-phase coolant flows down in the gaps between displacers around the core barrel circumference, comes downward through the core lateral reflector and specially designed tubes in the gas lift system and enters the lower plenum. Thus, the coolant in the lateral blanket is additionally heated and the coolant temperature at the core inlet is raised. The flow rate through the lateral reflector is about 8% of the core flow rate.

After the gas is separated, the coolant flows through perforations in the CPS column shroud and in the core barrel and enters the chamber at the SG and PRACS inlets. The coolant flow is separated here. About 85% of the flow enters SGs and goes downward in the space between the tubes, transferring heat to the secondary steam-water mixture. About 15% of flow through the spilling windows enters 12 PRACSs where it also goes downward, transferring heat to the environmental air. In this regime about 1.4 % of the generated power is lost in 12 PRACSs. Optimizing the location of the spilling windows with respect to the coolant-free level can reduce the power losses under nominal conditions. In this way, a passive feedback could be designed between coolant temperature and PRACS power.

After passing the SG, cold coolant enters the chamber, which is common for all 12 SGs, and through lower windows of PRACS goes into the PRACS plenum to mix with hotter coolant, which passed the PRACS. After passing the downcomer, about 10% of the flow mixes in the reactor lower plenum with the flow from the core lateral reflector and goes upward inside the tubes of the gas lift system being saturated with gas. The larger fraction of the cold coolant enters the gas lift system through the lateral windows and mixes with the two-phase flow in the gas lift system plenum.

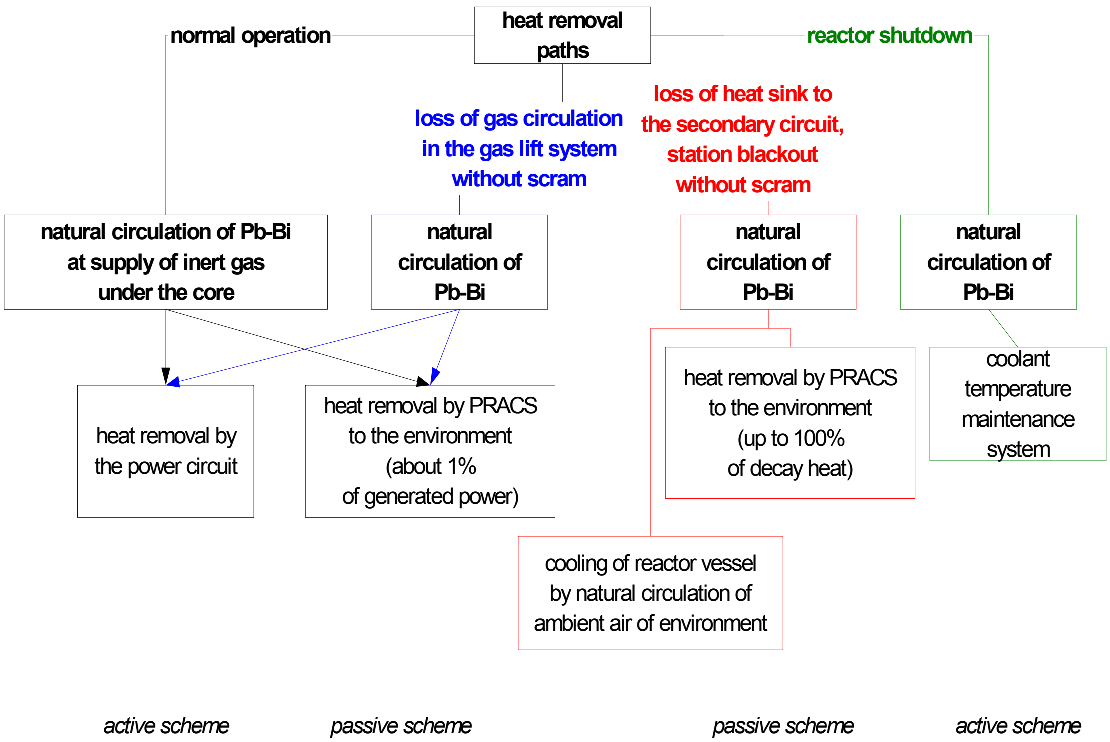


FIG. XXIII-13. Scheme of main heat transport system.

XXIII-2.2. Description of the turbine generator plant and systems

The parameters of the turbine circuit and turbine generator were chosen to be as close as possible to the parameters in the basic three-circuit RBEC design [XXIII-2]. The schematic of the RBEC-M power cycle and secondary circuit systems is shown in Fig. XXIII-1. Table XXIII-9 presents basic characteristics of the secondary circuit of the RBEC-M plant.

TABLE XXIII-9. BASIC CHARACTERISTICS OF THE SECONDARY CIRCUIT

Working fluid	Water/steam
Number of steam generators (SG)	12
Feedwater temperature, K	561
Feedwater flow rate per SG, kg/s	450
Generated steam pressure, MPa	15
Generated steam temperature, K	762

XXIII-2.3. Systems for non-electric applications

No detailed consideration was performed.

XXIII-2.4. General plant layout

No detailed consideration was performed.

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LIQUID METAL COOLED FAST REACTOR FOR WASTE TRANSMUTATION AND POWER PRODUCTION (PEACER)

Seoul National University, The Republic of Korea

XXIV-1. General information, technical features and operating characteristics

XXIV-1.1. Introduction

PEACER (Proliferation-resistant Environment-friendly Accident-tolerant Continuable-energy Economical Reactor) is a liquid metal cooled fast reactor for power production and waste transmutation.

There is growing concern that serious difficulties may result from the energy crisis within this century due to shortages of fossil energy and limitations in renewable energy. Even LWR technology will have problems ranging from the depletion of uranium resources to the radiological hazards of spent nuclear fuels. Hence, a new approach in nuclear technology is essential to find a pathway to sustainable nuclear power. In resource-stricken Korea, it is of high importance to establish national energy security by developing nuclear options. Technology development for nuclear waste has recently been initiated to transmute long-lived wastes from spent nuclear fuel into short-lived, low to intermediate level wastes using lead bismuth cooled fast reactor technology.

The core technology for partitioning and transmutation, including pyro-processing and Pb-Bi coolant technology, has either been successfully developed or is being explored in leading nuclear nations. It is difficult to safely dispose of spent nuclear fuel containing plutonium in geological formations and to assure international trust on proliferation-control due to the scarcity of suitable sites and dense population in Korea. However, long-lived transuranic waste elements can be utilized in fast neutron spectra and produce useful energy along with valuable resources.

Nuclear power technology for the 21st century is radically different from today’s practices. The advent of new technology may also be accelerated from earlier projections of 2010 to 2020, taking recent hikes in oil prices into consideration.

The concept of PEACER [XXIV-1], a fast lead-bismuth cooled reactor for electricity generation and waste transmutation, was proposed in 1998. The present research includes innovative reactor design development for the transmutation of spent nuclear fuel, the development of three-dimensional (3D) virtual reality technology and the demonstration of Pb-Bi coolant technology.

Design and technology development for the PEACER is performed under the auspices of the Korean Ministry of Commerce, Industry and Energy at the Nuclear Transmutation Reactor Engineering Center Korea (NuTRECK), Seoul National University.

XXIV-1.2. Applications

The PEACER is designed to produce electricity as well as to burn transuranic elements and fission products.

XXIV-1.3. Special features

The PEACER is a land-based nuclear power station.

XXIV-1.4. Summary of major design and operating characteristics

The key features of the PEACER design are as follows:

- Eutectic Pb-Bi coolant and a pancake-type core with a square lattice.
- Forced circulation by a main coolant pump (MCP) and the Rankine cycle for power generation.
- As with other Pb-Bi cooled fast reactor concepts, the operating coolant temperature is low, around 300–400°C, to achieve corrosion-resistance conditions and a longer reactor lifetime.
- Metallic fuel with a high thermal conductivity. In the initial stage of design, swelling posed a threat to the metallic fuel. However, recently a ternary alloy of U-Pu-Zr has shown superior behaviour with a strong potential for its use in the PEACER.

Installed capacity

The PEACER concept provides for two reactor designs of different capacity. The PEACER-550 has a 1560 MW(th) core, following the basic integral fast reactor design. The PEACER-300 is designed to produce 850 MW(th) or 300 MW(e).

Mode of operation (basic, load follow)

The PEACER could operate in both base load and load-follow modes.

Some major design characteristics of the PEACER are given in Table XXIV-1.

TABLE XXIV-1. MAJOR DESIGN CHARACTERISTICS OF PEACER

ITEM	PEACER-550	PEACER-300
Fuel type	U-TRU-Zr (57.2%-31.9%-10.9%) metal	
Smeared density (%)	67	67
Cladding material	HT-9	HT-9
Average power density (MW/m^3)	204.5	204.5
Coolant	Pb-Bi	Pb-Bi
Fuel enrichment	Inner core (15 weight % ^{235}U) Outer core (17 weight % ^{235}U)	Inner core (15 weight % ^{235}U) Middle core (17 weight % ^{235}U) Outer core (19 weight % ^{235}U)
Types of structural materials	HT-9	HT-9
Core type	Pancake type core with square lattice array	Pancake type core with square lattice array
Number of circuits	2	2
Turbine circuit	Superheated water	Superheated water

Load factor/ Availability:

Load factor (target):	80%
Availability (target):	90%

Simplified schematic diagram

A schematic diagram of the PEACER plant is given in Fig. XXIV-1. There is no intermediate heat transport system. Though the primary circulation is forced, Pb-Bi has a reasonably high natural circulation capability, providing an important inherent safety feature of the system.

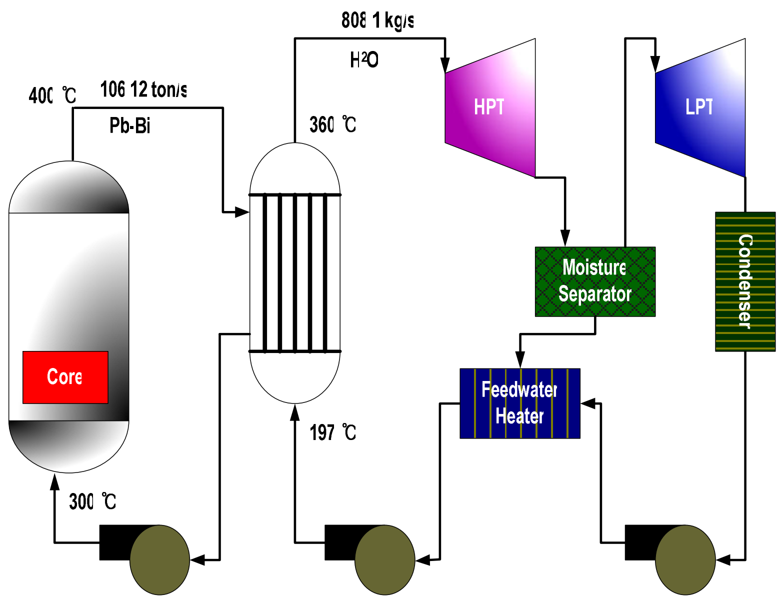

FIG. XXIV-1. Schematic diagram of PEACER plant.

In the turbine circuit, when there was no feedwater heater, the temperature difference was sizable between the inlet and outlet of the steam generator (SG), leading to severe thermal expansion of the structural material. The steam at the turbine inlet is superheated to 633.15 K and 8 MPa; steam is extracted from the high pressure turbine (HPT) at 0.345 MPa with a quality of 0.855. A moisture separator is installed between the HPT and the low pressure turbine (LPT) to minimize erosion of the LPT blades.

Neutron-physical characteristics

To assess core neutronic performance, the equilibrium cycle performance was calculated with a cycle length of 330 days and the K_{eff} of 1.002 at EOC. In burn-up calculations, actinides heavier than ^{245}Cm and lighter than ^{232}Th were ignored. For external fuel cycle calculations, all actinides in the chain were assumed to be extracted by pyro-processing and repeatedly recycled in the reactor.

The excess reactivity behaviour for an equilibrium cycle is shown in Fig. XXIV-1. The excess reactivity at BOC turned to be high, about 5% Δk/k. To cope with it, some adjustments of the design are planned within the next phase of the project.

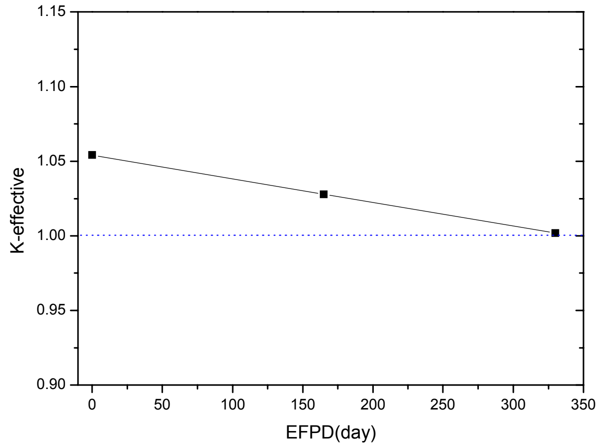

FIG. XXIV-2. Excess reactivity curve.

As shown in Fig. XXIV-3, higher relative power peaks are observed in areas between the inner core and the outer core. However, it was found that the maximum power peaking was less than 1.3 without control rods. It is anticipated that locating the control assemblies next to the hottest assembly could effectively control power peaking.

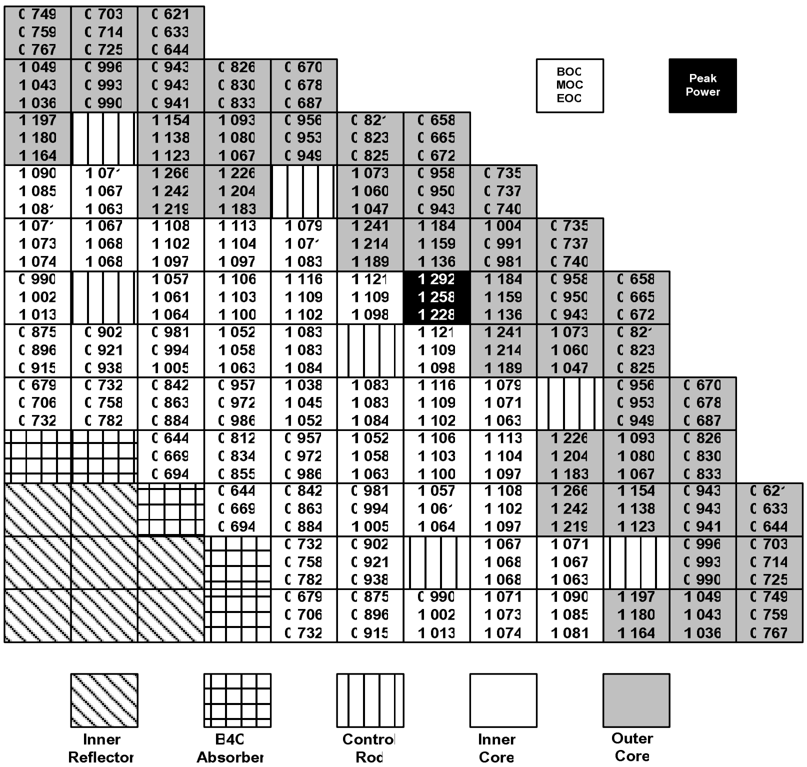

FIG. XXIV-3. Core map with assembly power distributions in equilibrium cycle.

Since Pb-Bi has a much higher boiling point (1670°C) than sodium, the positive void coefficient is expected to pose no serious problem for the PEACER.

Reactivity control mechanism

The PEACER incorporates an active reactivity control and shutdown system (motor driven) and the passive reactor shutdown system (driven by gravity).

The active reactivity control and shutdown system consists of 28 control assemblies that are used for power control, burn-up compensation and reactor shutdown according to signals from the plant control and protection systems. Each control assembly unit consists of an array of tubes containing B_4C.

Cycle type

Indirect, with Rankine cycle in the turbine circuit; the thermal efficiency at 8 MPa is 35.3%.

Thermal-hydraulic characteristics

The PEACER is designed to operate with forced circulation enhanced by an electromagnetic pump. A residual heat of about 10% of the nominal power can be removed by natural circulation of the primary coolant.

Figure XXIV-4 illustrates the thermal-hydraulic characteristics of the PEACER core. The temperature distribution shows that the maximum coolant temperature of 460°C exceeds the specified design temperature range (300–400°C). Figure XXIV-5 shows the relative fuel rod power distribution in the hottest assembly.

Figure XXIV-6 schematically shows the PEACER primary system with a nodal scheme for pressure drop calculation. Flow velocity data for the pressure drop calculation have been determined under steady state normal operation. The primary pressure drop distribution is listed in Table XXIV-2.

Table XXIV-3 summarizes thermal-hydraulic characteristics of the PEACER cores [XXIV-2]. The fuel temperature limit is set as 1000°C for both steady state and transient conditions.

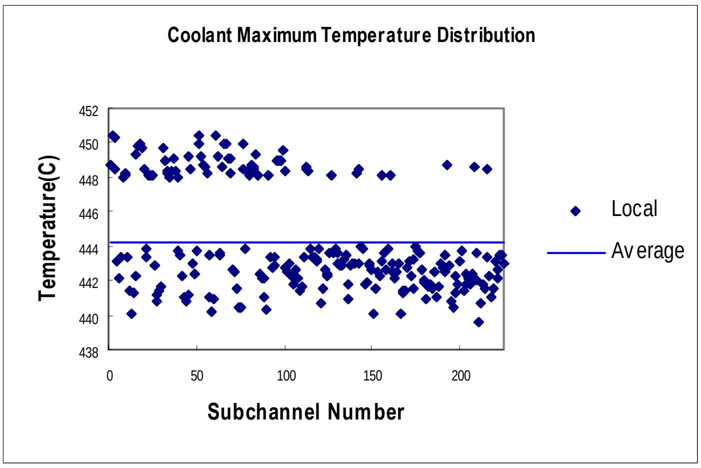

FIG. XXIV-4. Coolant temperature distribution in PEACER-550 hottest assembly.

TABLE XXIV-2. PRESSURE DROP IN PRIMARY CIRCUIT OF PEACER-550 UNDER FORCED CIRCULATION

Node		Flow velocity (m/s)	Pressure loss (Pa)
1		1.66	8229
2		1.66	445
3		1.66	11 547
4		0.93	1184
SG	5 (Friction)	0.93	23 279
	5 (Grid spacers)	1.32	19 579
6		0.93	1754
7		1.66	7454
8		1.66	2804
9		1.66	0
10		1.66	445
11		1.66	2804
12		1.66	2384
13		1.47	9651
14		1.10	1989
Core	15 (Friction)	1.10	9359
	15 (Grid spacers)	1.45	7447
16		0.55	885
17		0.26	11
Total pressure drop			111 250

TABLE XXIV-3. THERMAL-HYDRAULIC CHARACTERISTICS

Characteristic	PEACER-550	PEACER-300
Thermal power, MW(th)	1560	850
Core pressure, MPa	1	1
Inlet/outlet temperature, ^{o}C	300/400	300/400
Number of fuel pins	70 560	70 560
Core flow rate, kg/s	106 120	58 060

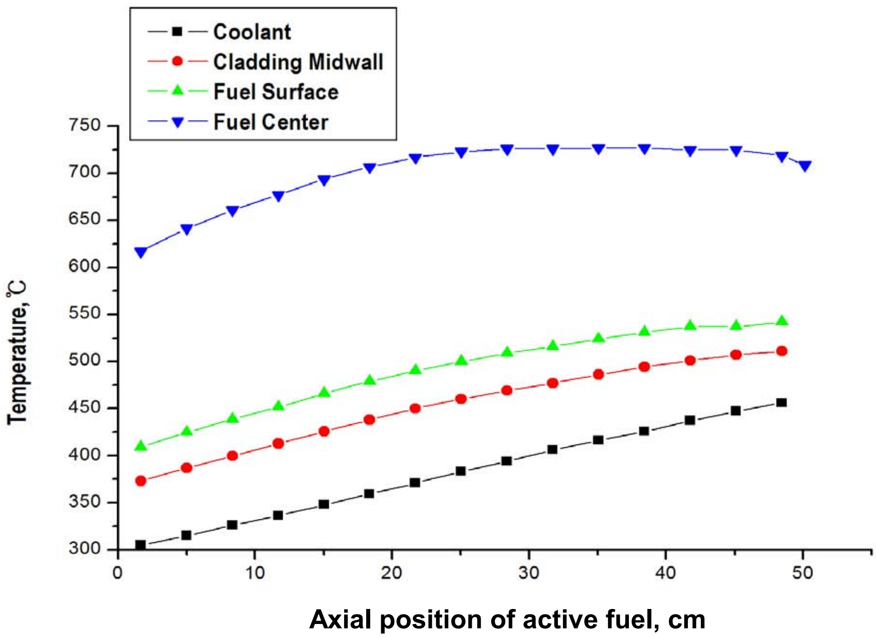

FIG. XXIV-5. Fuel pin temperature distribution at low burn-up (~0.5 % of fissile materials).

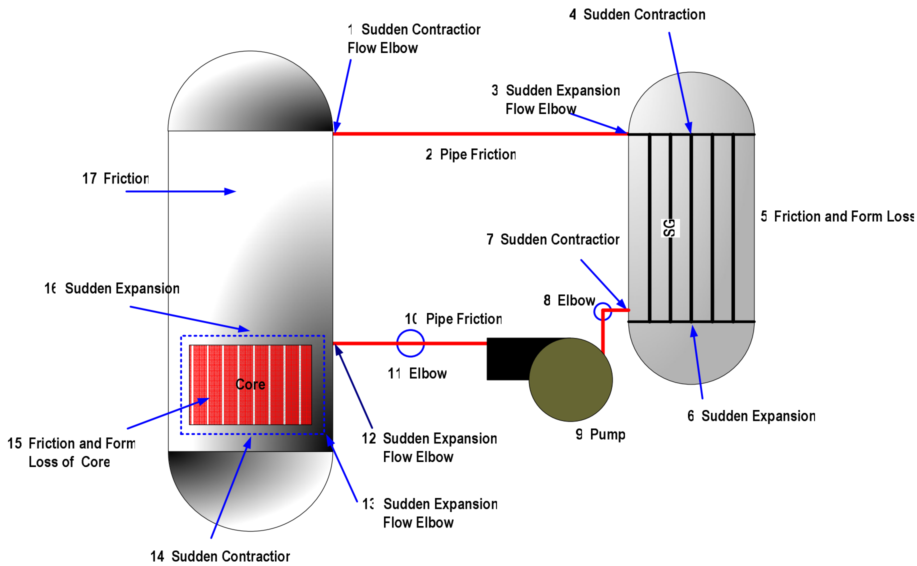

FIG. XXIV-6. Primary system of PEACER.

Fuel lifetime/period between refuellings

On average, 75 fuel assemblies will be replaced per year, assuming 330 effective full power days (EFPD) of operation per year and the average discharge burn-up of 75 MW·d/ kg.

Design basis lifetime for reactor core, vessel and structures

The design service life of all non-replaceable structures (including vessel, major piping and concrete structures) is 60 years. Components and equipment with a lower design life will be made easily replaceable during routine shutdowns.

Economics

Preliminary estimates indicate that the overnight capital cost of the PEACER plant will be ~US $2750 per kW(e), on a 2001 basis. The estimated construction period from first pour of concrete to criticality is six years. The unit energy cost for the electricity generated with the PEACER is expected to be competitive with that for alternative energy options available within the time frame and within the geographic regions of deployment of this reactor.

XXIV-1.5. Outline of fuel cycle options

Standard fuel cycle

The PEACER is designed to operate in a closed fuel cycle starting with U fuel and going on with the recycle of all actinides, Fig. XXIV-7. The PEACER could also transmute components of spent fuel produced by other reactors, e.g., light water reactors (LWRs).

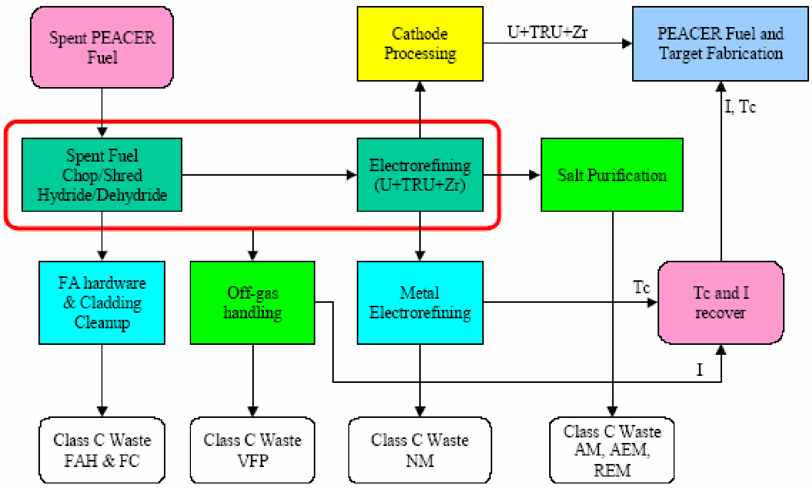

FIG. XXIV-7. Flow chart of PEACER fuel cycle.

Metal fuel pyro-processing

The PEACER will discharge the spent metallic fuel. After the fuel assembly structures are disassembled, the fuel pin is chopped and the fuel is separated from the cladding by the hydride/dehydride process. The fine powder produced is transferred to the electrolysis cell. U and TRU are recovered through the electro-refining process, along with Zr. After the U and TRU electro-refining process, the noble metal fission products are electro-refined to extract Tc and to decontaminate U and TRU. The molten salt enriched with fission products is purified with the electro-refining process as used for pyro-processing of the oxide fuel. The electro-refiner cell shown in Fig. XXIV-8 consists of a liquid metal anode, liquid metal or solid cathode, and molten salt electrolyte. The regions of molten salt and liquid metal shown in Fig. XXIV-8 are stirred to enhance the material transport. Each region is maintained at turbulent flow conditions and constant temperature and pressure. Well-defined diffusion layers are assumed to be developed with uniform thickness at the interfaces between an electrode and the molten salt. Using the assumption of well-defined diffusion layers, the region of interface between the liquid metal electrode and molten salt contains two distinct diffusion layers, Fig. XXIV-8. With a solid electrode, only one diffusion layer is present on the molten salt side. In the bulk of the molten salt and liquid metal, the chemical species are mixed by turbulent

convection to develop uniform concentration profiles. In each diffusion layer, the transport of chemical species takes place by both diffusion and migration. For the electrochemical cell where the geometry effect is neglected by the above assumption, only the diffusion layer could be considered. The cell like the one depicted in Fig. XXIV-8 transforms into an arrangement of continuously connected diffusion layers assuming that the bulk of the molten salt is maintained at constant properties.

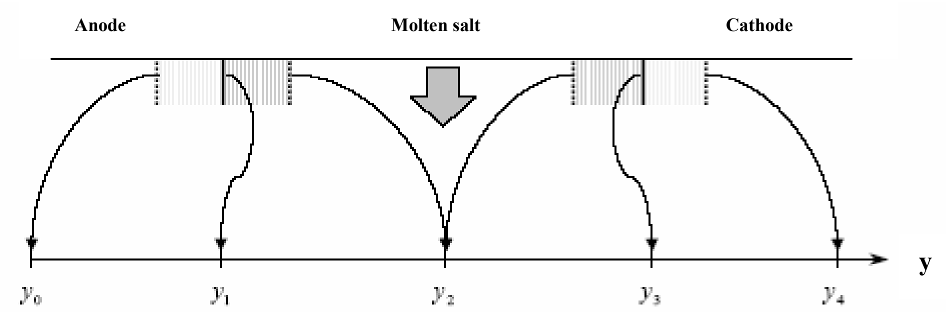

FIG. XXIV-8. Schematics of electro-refining process (as used in modelling).

Salt purification

After several runs of the electrolysis process, the active metal fission products such as alkali, alkaline earth and rare earth metals are accumulated in the molten salt. The accumulated fission products must be removed from the molten salt because they will affect the recovery efficiency of U and TRU. Periodically, the molten salt is removed from the electrolysis cell, purified using the salt purification process and recycled to the electrolysis cell. However, the molten salt always contains U and TRU with the fission products because the electrolysis is used to recover pure U and TRU without fission products. Therefore, fission products removed from the molten salt are always accompanied by some amount of U and TRU. It is necessary to optimize between the loss of U and TRU and the quantity of fission products removed because an increased removal of the fission products results in an increased contamination by the TRU in the waste stream.

The salt purification process is illustrated in Fig. XXIV-9. A fraction of the molten salt is removed from the electrolysis cell and is placed in contact with lithium-rich liquid cadmium. By the exchange reaction between Li and salt-borne TRU and the fission products, the less stable species in the molten salt are transferred to the liquid Cd. Generally, U and TRU are less stable than the rare earth metals and are first transferred to the liquid Cd. The Li concentration in the liquid Cd must be increased to decrease the contamination of the molten salt by TRU. Then, concentration of the fission products is also increased in the liquid Cd. After a forward reductive extraction process, the decontaminated salt with the salt-borne fission products passes through zeolite beds that replace nearly all of the alkali, alkaline earth, and rare earth metals with K and Li by ion exchange. The residual actinides in the molten salt are also adsorbed in the zeolite. The molten salt leaving the zeolite is free of actinides and fission product ions. The purified salt is mixed with an oxidizer such as $CdCl_2$ and is contacted with liquid Cd that contains U and TRU by the forward reductive extraction process. $CdCl_2$ will contain U and TRU to be oxidized. U and TRU are transferred to the molten salt from the liquid Cd. The molten salt with U and TRU is recycled to the electrolysis cell. The liquid metal is also recycled to the forward reductive extraction process.

Waste stabilization

Figure XXIV-10 depicts a process flowchart for decontamination of the PEACER fuel assembly hardware and fuel claddings with electro-polishing.

Surfaces of the fuel assembly hardware and fuel claddings are contaminated with radioactive nuclides, so electro-polishing is applied to decontaminate those surfaces. The surface is electro-dissolved in the molten salt to remove U and TRU, as shown in Fig. XXIV-10. The thickness of dissolution should be sufficient to permit disposal of the fuel assembly hardware and fuel cladding as the U.S. NRC Class C waste. The decontaminated metal is washed with pure LiCl - KCl, melted and cast into metallic waste forms. The salt purification process purifies the contaminated salt. Metallic fission products collected as particles from the pyro-process can be oxidized and mixed with zeolite from the salt purification process. Then, the zeolite with fission products is mixed with glass frit. Consolidation of the mixture into a monolithic body at a temperature near the melting point of the glass leads to a chemically stable sodalite. Then, the zeolite with fission products is mixed with glass frit. Consolidation of the mixtures into a monolithic body at a temperature near the melting point of the glass leads to chemically stable sodalite.

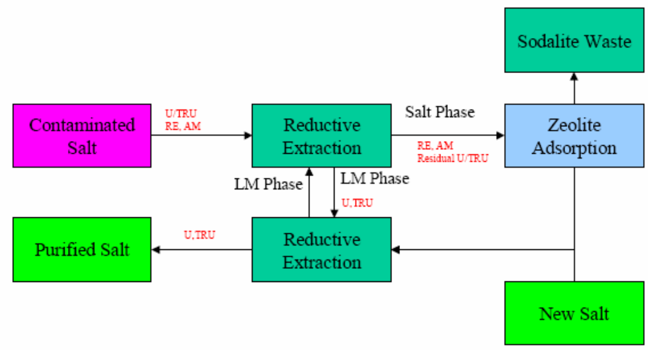

FIG. XXIV-9. Salt purification process.

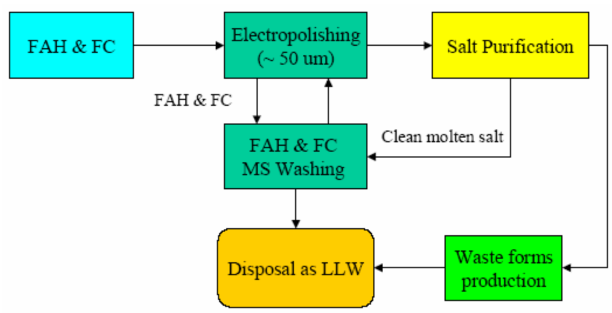

FIG. XXIV-10. Decontamination process.

Energy park concept

The transmutation process requires partitioning that inherently raises concern about the possibility of diversion or theft of sensitive nuclear materials. Since the PEACER is designed to have a high transmutation rate, it is assumed that the reactor and fuel cycle facilities would be centralized within a dedicated park that could be operated under international control. Figure XXIV-11 presents a possible operation chart of the fuel cycle facilities (fabrication and reprocessing) located within such park.

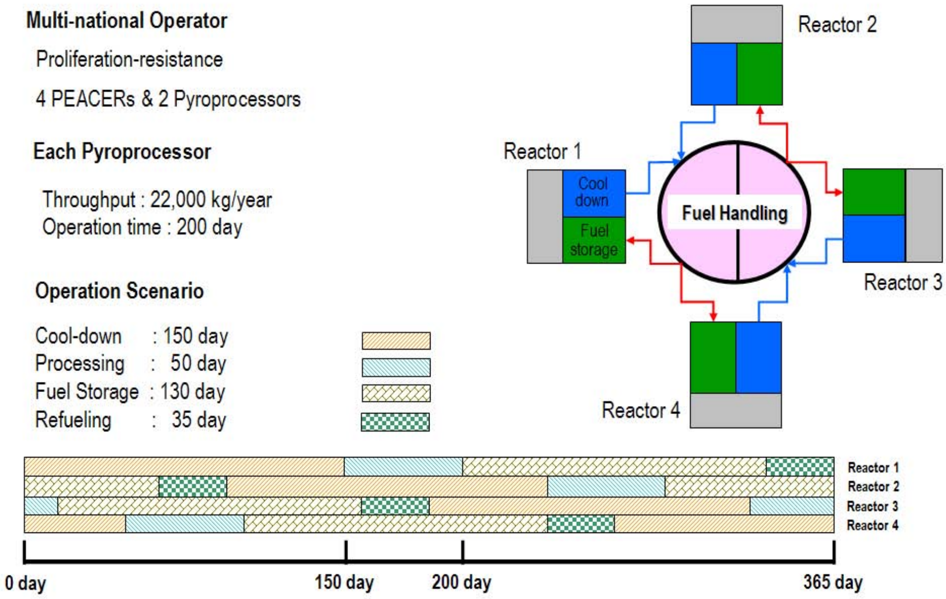

FIG. XXIV-11. International PEACER Park.

XXIV-1.6. Technical features and technological approaches that are definitive for PEACER performance in particular areas

XXIV-1.6.1. Economics and maintainability

The high capital cost of NPPs with liquid metal cooled reactors has become one of the hindrances to their wide commercial use. The projected future situation in the Republic of Korea requires the introduction of commercial liquid metal cooled reactors in the 2030s to maintain a sustainable nuclear electricity generation programme. To satisfy the nation's needs in 2030s, the PEACER should be able to demonstrate economic viability as a commercial liquid metal cooled reactor.

The PEACER design is considerably simplified by taking advantage of the chemical inertness of the Pb-Bi coolant. The main design features of the PEACER leading to reduced capital cost per MW(e) are:

- Minimization of MCPs and associated equipment.
- Elimination of an intermediate loop and no double wall piping.
- Substitution of steam generators with steam drums of simple construction.
- The 60-year service life of all non-replaceable structures.

XXIV-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

The conversion capability of the PEACER was evaluated by applying simple indices of the overall fissile inventory ratio (FIR) and fissile gain (FG):

FIR = (Fissile fuel amount in discharged fuel)/(Fissile fuel amount fresh fuel)

FG = (Fissile fuel amount in discharged fuel – Fissile fuel amount fresh fuel) / (Fissile fuel amount fresh fuel)

In evaluating the breeding capability of the PEACER core, comparisons were made with other reactors: PWR, CANDU and sodium cooled fast reactors (Na-LMR). Table XXIV-4 data show that the PEACER core has a goal for high conversion, not breeding.

TABLE XXIV-4. COMPARISON OF CONVERSION CAPABILITIES FOR DIFFERENT REACTORS

	PWR	CANDU	NA-LMR	PEACER
FIR	0.35	0.72	1.03	0.92
FG (%)	-64.61	-28.18	3.00	-8.00

In the pyro-chemical process, the decontamination factor (DF) for TRU is introduced as an indicator of process performance. The overall DF in the pyro-chemical process is defined as the ratio of the mass of the TRU loaded in the process to the TRU lost into the waste stream, expressed as:

DF = (Amount of TRU loaded in pyro-chemical process)/ (TRU lost into waste stream)

In an earlier study [XXIV-1], a PEACER pyro-processing system with a DF of $\sim 10^5$ was conceptually designed; a DF of 2.3×10^5 was suggested to meet several requirements of the U.S. NRC Class C limit, for a facility disposal volume of 1.6×10^5 m^3.

To evaluate the total waste produced from pyro-processing, a system was considered comprising 12 PEACER reactors and 20 LWRs, each of 1 GW(e) capacity, a 40-year service life, and a 33 000 MW·d/t U discharge burn-up with 30 years spent fuel cooling time. The masses of actinides in the PEACER were analyzed in an equilibrium state by considering the time interval between each stage in the pyro-processing.

Regarding fission products; ^{90}Sr, ^{135}Cs, ^{137}Cs and ^{151}Sm were assumed to be recovered with a 95% removal efficiency, to satisfy regulations on heat load and volume of the disposal facility. Table XXIV-5 and Table XXIV-6 list the evaluated total TRU and fission product waste production from pyro-processing for a DF = 2.3×10^5 and a 95% fission product removal efficiency, respectively.

TABLE XXIV-5. TRU WASTE PRODUCTION FROM PYRO-PROCESSING WITH DF=2.3 10^5 FOR A SYSTEM WITH 20 LWRS AND 12 PEACER REACTORS

NUCLIDE	INITIAL INVENTORY [g]			DF	MASS IN WASTE [g]		
	PEACER	LWR	Total		PEACER	LWR	Total
Cm-244	8.10E+01	2.09E+05	2.93E+07	2.3E+05	1.27E+02	9.09E-01	1.27E+02
Pu-240	2.20E-01	5.29E+07	5.11E+08	2.3E+05	1.99E+03	2.30E+02	2.22E+03
U-236	6.47E-05	1.06E+06	1.11E+07	2.3E+05	4.35E+01	4.61E+00	4.81E+01
Pu-238	1.71E+01	3.61E+06	4.71E+07	2.3E+05	1.89E+02	1.57E+01	2.05E+02
Pu-242	3.90E-03	1.34E+07	1.40E+08	2.3E+05	5.52E+02	5.83E+01	6.10E+02
U-234	6.25E-03	1.37E+04	9.23E+06	2.3E+05	4.01E+01	5.96E-02	4.01E+01
U-238	3.36E-07	2.02E+08	2.34E+09	2.3E+05	9.30E+03	8.78E+02	1.02E+04
Pu-241	1.01E+02	4.85E+06	1.04E+08	2.3E+05	4.33E+02	2.11E+01	4.54E+02
Am-241	3.43E+00	2.85E+07	3.03E+07	2.3E+05	7.70E+00	1.24E+02	1.32E+02
Np-237	7.06E-04	1.39E+07	5.28E+07	2.3E+05	1.69E+02	6.04E+01	2.30E+02
U-233	9.69E-03	1.81E+00	4.63E+02	2.3E+05	2.00E-03	7.87E-06	2.01E-03
Am-243	1.92E-01	2.85E+06	3.75E+07	2.3E+05	1.50E+02	1.24E+01	1.63E+02
Pu-239	6.13E-02	1.22E+08	5.51E+08	2.3E+05	1.87E+03	5.30E+02	2.40E+03
U-235	2.16E-06	1.84E+06	5.90E+06	2.3E+05	1.77E+01	8.00E+00	2.57E+01

TABLE XXIV-6. FISSION PRODUCT WASTE GENERATION FROM PYRO-PROCESSING WITH 95% REMOVAL EFFICIENCY FOR SELECTED FISSION PRODUCTS IN A SYSTEM WITH 20 LWRS AND 12 PEACER REACTORS

NUCLIDE	INITIAL INVENTORY (g)			1 – Removal efficiency	MASS IN WASTE (g)		
	PEACER	LWR	Total		PEACER	LWR	Total
Se-79	2.93E+04	1.55E+05	1.84E+05	1.00E+00	2.93E+04	1.55E+05	1.84E+05
Sr-90	1.50E+06	5.34E+06	6.84E+06	5.00E-02	7.50E+04	2.67E+05	3.42E+05
Zr-93	2.78E+06	1.89E+07	2.17E+07	1.00E+00	2.78E+06	1.89E+07	2.17E+07
Tc-99	3.97E+06	1.99E+07	2.39E+07	5.00E-02	1.99E+05	9.95E+05	1.19E+06
Pd-107	2.52E+06	5.91E+06	8.43E+06	1.00E+00	2.52E+06	5.91E+06	8.43E+06
Sn-126	2.84E+05	7.23E+05	1.01E+06	1.00E+00	2.84E+05	7.23E+05	1.01E+06
I-129	1.25E+06	4.72E+05	1.72E+06	5.00E-02	6.25E+04	2.36E+04	8.61E+04
Cs-135	8.01E+06	9.22E+06	1.72E+07	5.00E-02	4.01E+05	4.61E+05	8.62E+05
Cs-137	6.77E+06	1.24E+07	1.92E+07	5.00E-02	3.39E+05	6.20E+05	9.59E+05
Sm-151	5.52E+05	2.64E+05	8.16E+05	5.00E-02	2.76E+04	1.32E+04	4.08E+04

The fuel cycle described above results in an order-of-magnitude reduction in the quantities of actinides that are introduced into the waste stream per unit of energy generated. The net output could be a low-waste generating nuclear energy system that would also mitigate the LWR nuclear waste problem, as illustrated by Fig. XXIV-12.

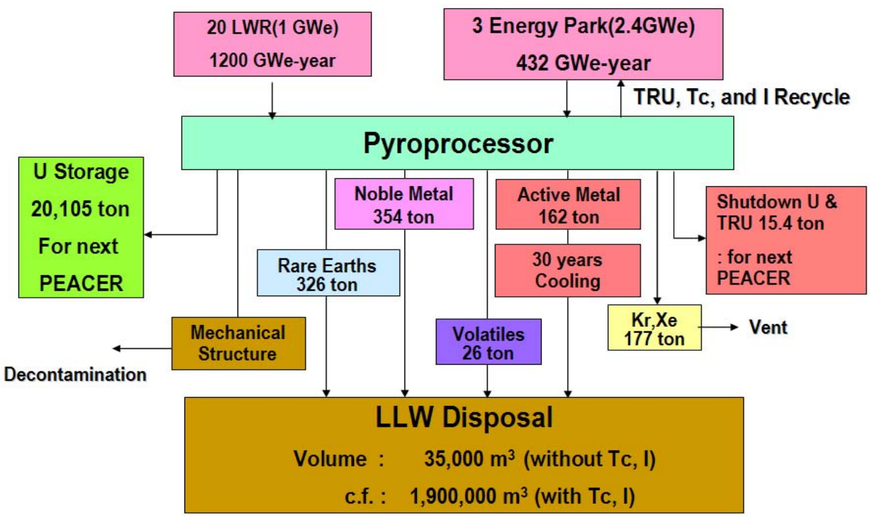

FIG. XXIV-12. Waste management strategy for a system with 20 LWRs and 12 PEACER reactors.

The economics of a closed fuel cycle within the already mentioned system with 20 LWRs and 3 PEACER parks with 12 PEACER reactors (PEACER system) was evaluated in comparison with the present-day fuel cycle of an advanced LWR (ALWR). A preliminary economic analysis has been conducted using cost figures suggested in the accelerator-driven transmutation of waste (ATW) roadmap; the cost of pyro-processing within the PEACER system was conservatively assumed to be twice the value suggested in this roadmap. The overall fuel cycle cost of the PEACER system was preliminarily evaluated to be about 24% lower than that of the ALWR fuel cycle, as presented in Table XXIV-7.

TABLE XXIV-7. COMPARISON OF FUEL CYCLE COSTS FOR THE PEACER SYSTEM AND AN ALWR

ITEM	UNIT COST, US $/kg	MATERIAL FLOW, kg/yr	TOTAL COST, US $/yr
PEACER SYSTEM			
LWR spent fuel pyro-processing	402	30 000	1 260 000
PEACER spent fuel pyro-processing	11 640 (2×ATW)	4700	54 708 000
PEACER fuel fabrication	2700	5300	14 310 000
PEACER waste disposal	50	30 000	1 500 000
LWR spent fuel disposal credit	-1500	30 000	-45 000 000
		Total cost	37 578 000
		Levelized by 1000/550	68 323 636
ALWR			
ALWR fuel fabrication	1500	30 000	45 000 000
ALWR waste disposal	1500	30 000	45 000 000
		Total cost	90 000 000
Fuel cycle cost ratio (PEACER system)/ALWR		0.76	

XXIV-1.6.3. Safety and reliability

Safety concept and design philosophy

The PEACER safety concept draws upon the experience with lead-bismuth coolants and metallic fuel for fast reactors. Specifically, a Pb-Bi coolant is not only chemically inert but has a high natural circulation capability that could lead to certain inherent safety features. For example, core melting can be prevented by natural circulation of the Pb-Bi during a loss-of-flow accident. Since the Pb-Bi coolant has a much higher boiling point (1670°C) than sodium, the role of a positive void reactivity coefficient becomes essentially diminished. The safe nature of the metallic fuel in a fast reactor was demonstrated in the EBR-II programme in the 1980s, and these results are also applicable to the PEACER reactor concept.

The PEACER is designed to be self-controlled so that the response to anticipated transients without scram (ATWSs) always leads to a safe shutdown state without exceeding the limits assuring the core integrity. The design goals are as follows:

- Establish the limits securing the core integrity and ensure that these limits will not be exceeded during the unprotected loss-of-flow (ULOF), unprotected loss-of-heat-sink (ULOHS) and unprotected transient overpower (UTOP) scenarios.
- Rely on inherent and passive safety features including the use of enhanced thermal expansion materials, a fast expansion module, a very low reactivity swing, and other reactivity feedback mechanisms to achieve a high degree of reactor self-control in the ATWS scenarios.
- Ensure reliable removal of the decay heat.

The capacity for natural circulation, especially for Pb-alloy cooled fast reactors with a loose-pitch core, offers significant potential for simplifying the heat transport system. Conventional active safety systems can also be used, but the design goal is to achieve a safe shutdown state without reliance on the active safety systems. Since the inherent safety features provide a high degree of confidence in function, independent of human or machine actions, the primary thrust of the design effort will be directed to maximize the reliance on such features.

In the extreme and hypothetical cases of a severe accident, when the core integrity is not maintained, the reactor system design targets to:

- Prevent a reactivity excursion due to core compaction and,
- Reach a stable state without radioactivity releases to the environment, relying on natural phenomena and barriers.

Active and passive systems and inherent safety features

Table XXIV-8 outlines the safety design features and targets of the PEACER concept. The approach behind Table XXIV-8 data was to minimize the investment risk.

TABLE XXIV-8. DESIGN FEATURES AND TARGETS FOR PEACER

SPECIFIC OBJECTIVE	DESIGN FEATURE OR TARGET	RATIONALE
Accident prevention (Inherent safety features)	Design simplification	Simplicity is emphasized in all aspects of the design, construction, operation, and maintenance; complexity of plant designs has been one of the main sources of high capital costs.
	Large thermal capacity of the primary system	Large thermal capacity makes a system transient slower and provides more time to cope with abnormal events.
	Negative power reactivity coefficient	Maintaining the core power reactivity coefficients negative during all modes of plant operation is crucial for the overall plant safety.
Accident mitigation (Passive heat removal mechanisms)	Target: large radioactivity release frequency of less than 10^{-7}/reactor-year to be achieved by reliable containment design and passive heat removal mechanisms	Passive mechanism is, in general, preferable for mitigating an accident and could be more reliable.
Core damage prevention (Passive heat removal systems)	Target: core damage frequency (CDF) less than 10^{-6}/reactor-year	PEACER should have a CDF lower than that of advanced LWRs
	Highly reliable and diversified decay heat removal	Emphasis should be given not only to safety grade decay heat removal systems but also to non-safety grade decay heat removal systems.

Specifically, a reactor vessel auxiliary cooling system (RVACS) is adopted that can remove 100% of the decay heat passively, even in total loss of the normal heat sink. In the RVACS, heat generated in the core is conveyed to the reactor vessel by natural circulation of the primary coolant, is conducted across the vessel and the guard vessel and is finally transferred to the atmospheric air naturally flowing on the outer surface of the guard vessel.

The second reactor shutdown system is passive (driven by gravity). The primary reactivity control and reactor shutdown system is active, based on control rods.

Structure of the defence-in-depth

Four physical barriers exist within the PEACER to prevent the release of fission products to the environment: these include (1) the fuel matrix itself, (2) the fuel pin cladding, (3) the primary coolant system, including a guard vessel, and (4) the outer containment.

Design basis accidents and beyond design basis accidents

The preliminary list of design basis and beyond design basis accidents considered for the PEACER is given in TABLE XXIV-9.

TABLE XXIV-9. LIST OF DESIGN BASIS ACCIDENTS

INITIATING EVENTS	DESCRIPTION
Design basis accidents	
1. Single seal failures	Minor liquid metal leaks.
2. Radioactive waste system failures	RAPS (Radioactive argon processing system*) /CAPS (Cell atmosphere processing system) valve leaks; RAPS surge tank failure; Cover gas diversion to CAPS; Liquid metal tank leaks.
3. Transients outside the expected range of variables	Fuel failures during normal operation.
4. Events resulting in the release of radioactivity from the primary system	Class 4 events and heat exchanger leaks.
5. Refuelling accidents inside the containment	Drop of a fuel element; Crane impact on fuel assembly head; Inadvertent floor valve opening; Leak in fuel transfer cell/chamber.
6. Accidents with spent fuel outside the containment	Shipping cask drop; EVST (Ex-vessel storage tank) / FHC (Fuel handling cell) system leaks; Loss of forced cooling to EVST
8. Accident initiation events considered in design-basis evaluation in the safety analysis report	SG leaks; Fuel failure propagation; Rupture of primary piping; Pump failure or reactivity transients
Beyond design basis accidents	
1. Hypothetical sequences of failures more severe than class 8 events	Successive failures of multiple barriers normally provided and maintained

* RAPS is a system that purifies contaminated core gas (argon)

Provisions for safety under seismic conditions

The PEACER structures, systems and equipment are designed for high level and low probability seismic events such as an operating basis earthquake (OBE) and safe shutdown earthquake (SSE). These are also called S1 and S2 level earthquakes, respectively, and seismic instrumentation is planned to meet national and international standards [XXIV-3, XXIV-4].

Probability of unacceptable radioactivity release beyond the plant boundaries

The targeted probability of unacceptable radioactivity release beyond the plant boundary is less than 1×10^{-7}.

XXIV-1.6.4. Proliferation resistance

The transmutation process requires partitioning, which inherently raises concern about the possibility of diversion or theft of sensitive nuclear materials. Both technical and institutional solutions to proliferation-resistance are sought for the on-site waste partitioning process.

Utilization of the pyro-processing technique could enhance proliferation resistance, since it has inherent limits in the degree of partitioning and produces an intense radiation field. The most effective technical barrier is the use of fully remote control systems allowing waste processing without long-term pre-cooling. Also, denaturing of the sensitive nuclear materials could essentially complicate the process of nuclear weapon production, e.g., because of a high rate of spontaneous fission resulting in a high level of residual heat. In addition to this, the fuel separation facility using pyro-chemical processing could be made compact and housed within the physical boundary of the transmutation complex.

For institutional barriers against proliferation, it is proposed that a waste transmutation complex consisting of multiple units of the PEACER be run by an international organization in compliance with the IAEA safeguard procedures. Another approach is related to the denaturing of the fissile materials with isotopes that would complicate the use of the reactor fuel for weapon programmes. As a measure of the extent of denaturing, the odd ratio is defined as follows:

Odd ratio (OR) = (Total mass of isotopes with odd mass numbers)/(Total mass of a selected actinide element)

The technical barriers represented by the OR were examined as a function of core design parameters. As shown in Fig. XXIV-13, the OR of plutonium decreases as both core height-to-diameter (L/D) and the fuel volume fraction (FVF) decrease. This indicates that a low power density and high leakage core design could also enhance proliferation resistance. Such behaviour could be explained by the slower breeding of Pu-239 from U-238 resulting from high leakage and the faster destruction of odd number isotopes, due to the lower capture-to-fission ratio. In the PEACER, the OR of Pu decreases to about 0.5 at the end of the equilibrium cycle (EOEC), as shown in Fig. XXIV-13. This is much lower than that in the LWR spent fuel (0.7). The difference between the two OR values at the beginning of the equilibrium cycle (BOEC) and at EOEC is fairly constant at about 0.04 within the range of this study. The maximum fuel discharge burn-up is found to be 11.5% and is within the proven performance range of the EBR-II core.

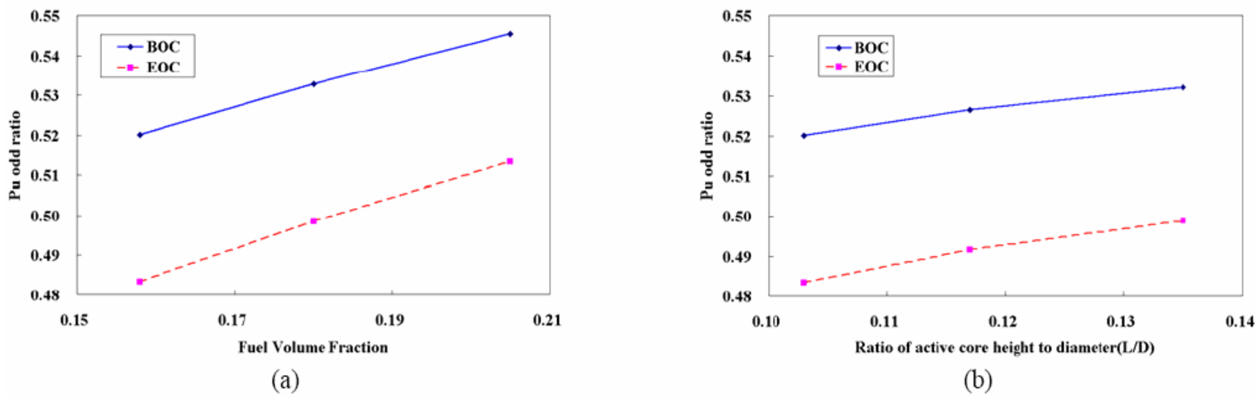

FIG. XXIV-13. Pu odd ratio of U-TRU-Zr metallic fuel: (a) L/D = 0.103, (b) FVF = 0.158.

The overall goal of denaturing is to obtain a dirtier Pu composition at discharge compared to the spent LWR fuel. Therefore, diluting of other fissile isotopes present in the discharged fuel, such as ^{235}U or ^{233}U, could also be foreseen.

XXIV-1.6.5. Technical features and technological approaches used to facilitate physical protection of PEACER.

The double containment based reactor building of the PEACER provides a significant barrier to reactor damage arising from external impacts, both natural and human-induced.

XXIV-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of PEACER

No information was provided.

XXIV-1.8. List of enabling technologies relevant to PEACER status of their development

A list of the enabling technologies relevant to the PEACER concept is given in Table XXIV-10.

TABLE XXIV-10. LIST OF ENABLING TECHNOLOGIES AND THEIR DEVELOPMENT STATUS

DESIGN AREA	ENABLING TECHNOLOGY	STATUS OF DEVELOPMENT
Design approaches and calculation technologies for neutronic core design	Data for evaluation of TRU extinction characteristics of a transmutation reactor	Nuclear data for fast reactors collected
	Neutronic and depletion analysis codes	REBUS code system is under validation and testing
	Tools for 3D visualization of neutronic performance	Interface development completed
	Fuel management scheme	Preliminary study in progress
	Optimized core design	R&D in progress
Design approaches and calculation technologies for thermal-hydraulic and safety design	Thermal-hydraulic design of the core	PEACER-300 and 550 core analyses completed.
	System analysis of the PEACER energy park	System analysis using Dynamic Simulator for Nuclear Power Plants (DSNP) completed
	Tools for transient thermal-hydraulic and fuel performance analysis.	Code development in progress: DSNP, MATRA: Multi-channel Analyser for steady state and Transients in Rod Arrays
	Safety design	Natural circulation capacity and RVACS, LOHS, LOFA analyses completed
Pyro-processing technology	Conceptual design of pyro-processing plant	R&D in progress
	Optimization of pyro-process design	R&D in progress
Primary circulation	Design and technology of submersible electromagnetic pumps for the primary circuit	Further R&D needed
Fuel, coolant and structural materials technology	Database of materials properties	Analysis process established
	Metal fuel design	R&D in progress
	Pb-Bi coolant technology	Technology loop: experimental facility under construction
	Technologies to ensure reliable core operation	Corrosion test loop will be built to define and validate a technology for the protection of structural materials from corrosion / erosion in a flow of Pb-Bi coolant; experimental facility is under construction. Demonstration loop: scaling analysis completed, ongoing experimental programme.

XXIV-1.9. Status of R&D and planned schedule

R&D for the PEACER is fully supported by the Korean Ministry of Commerce, Industry and Energy. The R&D is mainly performed at NuTRECK (Nuclear Transmutation Reactor Engineering Center Korea), Seoul National University.

The conceptual design of the reactor and the basic design of major nuclear systems have been completed. It is projected that, under favourable conditions, the design could be sufficiently complete to enable initiation of prototype-construction related actions by the end of 2010, subject to the availability of funds and regulatory and other statutory clearances.

Based on the conceptual design of a liquid metal cooled reactor developed in 1998, the present research encompasses innovative transmutation reactor design development for spent nuclear fuel disposal, the development of 3D virtual reality technology, and Pb-Bi coolant technology demonstration.

XXIV-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

Since the PEACER will feature a host of unique design and engineering features, construction of a demonstration prototype would be needed.

XXIV-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

In the Russian Federation, a variety of Pb- and Pb-Bi cooled power reactors are under development, e.g., BREST-300 and 1200, SVBR-75/100, etc. Some of them, e.g., SVBR-75/100, can operate for a long time without reloading and reshuffling of fuel. Namely these Russian advances have sparked an interest at the INEEL and MIT (both, USA) in the fall of 1998 to investigate this type of reactor for future energy production.

Elsewhere, Pb-Bi cooled fast reactor cores have continued to be investigated in Japan, Korea, and the United States at the Argonne National Laboratory, Lawrence Livermore National Laboratory as well as the University of California, Berkeley, CA.

XXIV-2. Design description and data for PEACER

XXIV-2.1. Description of the nuclear systems

Reactor core and fuel design

The fuel pin geometry with an 11 mm diameter and 1 mm cladding thickness is similar to that used in PWRs.

Figure XXIV-14 depicts the cross-section of a single fuel assembly of the PEACER-550; each fuel assembly consists of 196 fuel rods creating a 14×14 rectangular array. Figure XXIV-15 gives the cross-section of a single fuel assembly of the PEACER-300; each fuel assembly is made up of 180 fuel rods and 9 skeleton rods in a 17×17 rectangular array.

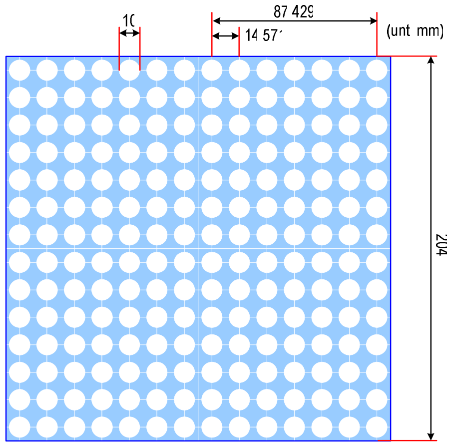

FIG. XXIV-14. Cross-section of PEACER-550 fuel assembly.

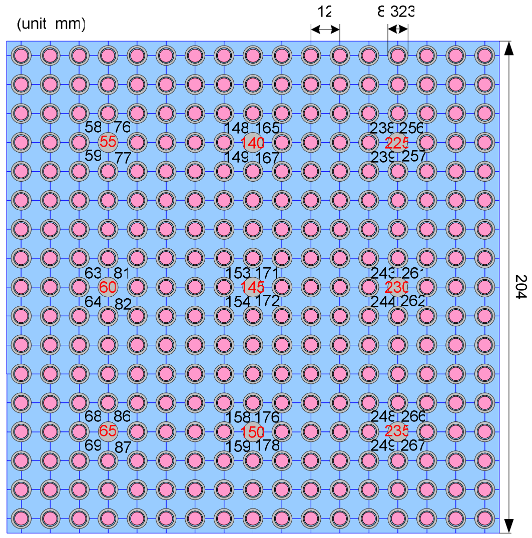

FIG. XXIV-15. Cross-section of PEACER-300 fuel assembly with sub-channel numbers around skeleton rods.

The major core design characteristics of the PEACER-300 and PEACER-550 are presented in Table XXIV-11.

The PEACER-300 core consists of 252 fuel assemblies, 20 control assemblies, 156 reflectors, and 4 secondary control assemblies. The core height is 130 cm, of which the effective heated height is 45 cm with an upper gas plenum and a lower axial reflector; the active diameter is 385 cm including the reflectors and shields. To flatten power distribution, the PEACER-300 has three fuel types: inner, middle and outer fuels.

The PEACER-550 has two fuel types: inner (15 weight % of ^{235}U) and outer (17 weight % of ^{235}U) fuels, see Fig. XXIV-16. The core regions consist of an inner reflector, inner core, outer core, outer reflector and shield. A blanket was not included because breeding is not desirable in the reactor intended for transmutation. The core dimensions were determined as 0.5 m in height and 2.5 m in radius, like a pancake. A short and wide core shape reduces the total fuel volume fraction and increases the neutron leakage to prohibit the production of TRU due to neutron capture. From the standpoint of safety, the large neutron streaming effect from the pancake type core leads to a negative void reactivity coefficient.

TABLE XXIV-11. MAJOR CORE DESIGN CHARACTERISTICS OF PEACER REACTORS

ITEM		PEACER-550	PEACER-300
Active fuel height (m)		0.501	0.45
Total fuel height (m)		1.353	1.3
Number of assemblies	Fuel	360	252
	Inner core	184	84
	Middle core	No	84
	Outer core	176	84
	Inner radial reflector	32	-
	Outer radial reflector	100	156
	Radial shield	192	184
	Control/Shutdown	20	20
Fuel pins per assembly	Inner core	14×14	17×17
	Outer core	14×14	17×17

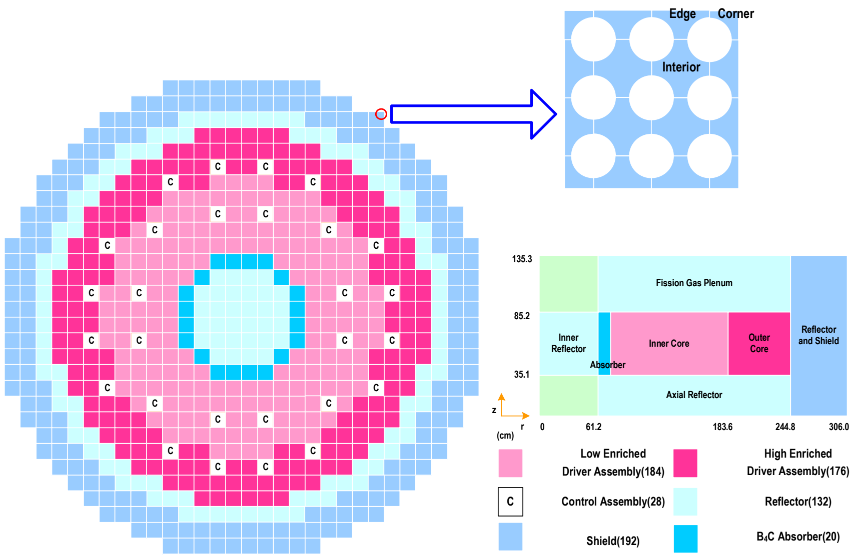

FIG. XXIV-16. Schematic diagram of PEACER-550 core.

Main coolant pumps

Four submersible electromagnetic pumps (EMPs) will circulate the primary coolant through the reactor.

Steam generators

In the PEACER, the steam generators appear as steam drums of simple design, see Fig. XXIV-17. The Pb-Bi velocity in the steam generators cannot exceed the limit of 2 m/s. The steam generator has three distinct zones; the first one is the sub-cooled liquid zone (single phase), the second is from the pre-dryout to post-dryout zone (two phases) and the third is the superheated steam zone (single phase).

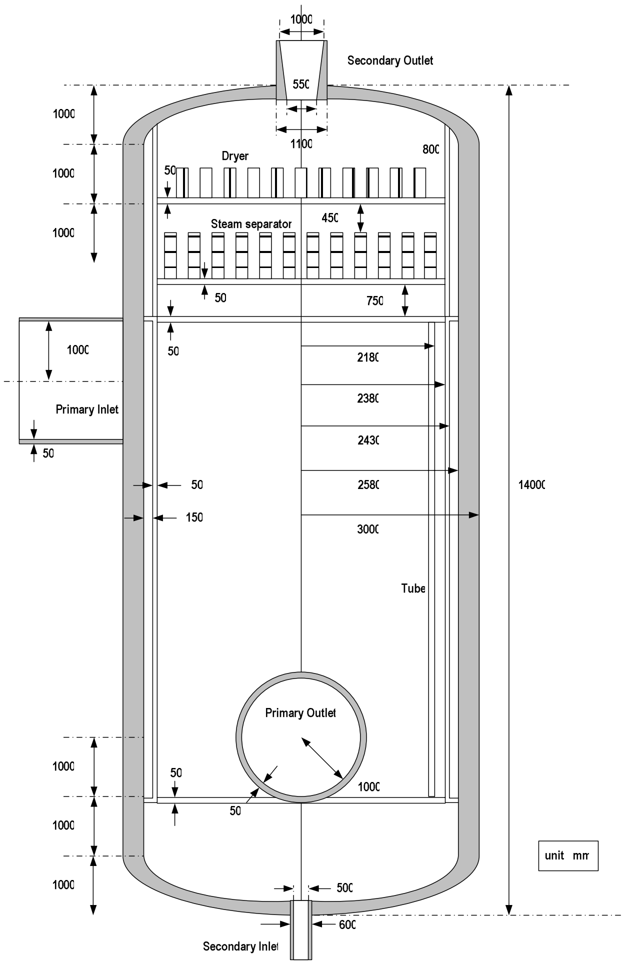

FIG. XXIV-17. Steam generator.

The tubes in the steam generators are of a once-through type. The primary Pb-Bi flows on the shell side, while the secondary (water) coolant flows on the tube side. Table XXIV-12 lists the inlet and outlet temperatures calculated using the energy balance equation and the thermal efficiency for the steam generator spanning from 5 MPa to 15 MPa.

TABLE XXIV-12. TEMPERATURE AND THERMAL EFFICIENCY OF A STEAM GENERATOR

PRESSURE (MPA)	Superheated zone temperature (°C)	Saturated zone temperature (°C)	Sub-cooled zone temperature (°C)	Efficiency (%)
5	360.15	264.06	214.33	32.69
6	360.15	275.7	208.96	34.31
7	360.15	285.94	203.43	35.69
8	360.15	295.12	197.44	36.78
9	360.15	303.46	191.02	37.73
10	360.15	311.11	184.13	38.52
11	360.15	318.2	176.69	39.19
12	360.15	324.8	168.58	39.77
13	360.15	330.98	159.68	40.19
14	360.15	336.79	149.72	40.6
15	360.15	342.28	138.48	40.66

Reactor vessel auxiliary cooling system

A characteristic feature of the PEACER-300 is the reactor vessel auxiliary cooling system (RVACS). Figure XXIV-18 presents the schematic diagram of this system. For a reactor with an RVACS-type decay heat removal system with liquid Pb-Bi in the gap between the reactor vessel and the guard vessel, the static load is accommodated by the guard vessel, because the reactor vessel 'floats' in the liquid Pb-Bi.

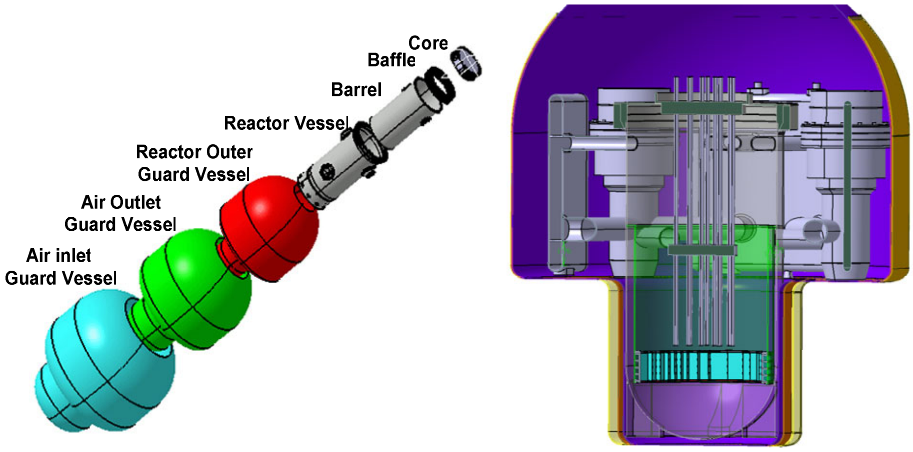

FIG. XXIV-18. Schematic diagram of PEACER-300 RVACS.

XXIV-2.2. Description of the turbine generator plant and systems

A superheated steam turbine is to be used, see Fig. XXIV-1.

XXIV-2.3. Systems for non-electric applications

The PEACER design includes no systems for non-electrical applications.

XXIV-2.4. Plant layout

No information was provided.

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[XXIV-3] PARK, C.K. et al., KALIMER conceptual design report, KAERI/TR-2204/2002, Korea Atomic Energy Research Institute, Daejon, Korea (2002).

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ANNEX XXV

MEDIUM SCALE LEAD-BISMUTH COOLED REACTOR

JNC, Japan

SHORT DESCRIPTION

XXV-1. Basic summary

Medium Scale Lead-bismuth Cooled Reactor is being developed by the Japan Nuclear Cycle Development Institute (JNC).

Core design

This reactor has a two-region homogeneous core with nitride fuel; the average burn-up target is 150 GW d/t to reduce fuel cycle cost.

Plant design

This is a medium sized tank-type reactor without an intermediate heat transport system; the modular-type plant consists of 4 reactor units.

Safety features

Three reactor auxiliary cooling systems are located at the upper plenum; these systems are driven by natural convection. The inherent safety features of the core are enhanced to avoid a core-disruption accident even in anticipated transients without scram (ATWSs).

XXV-2. Major design and operating characteristics

Main characteristics of the reactor core are summarized in Table XXV-1. Major characteristics of an NPP with the Medium Scale Lead-Bismuth Cooled Reactor are given in Table XXV-2. A general view of the reactor is shown in Fig. XXV-1.

TABLE XXV-1. CORE CHARACTERISTICS

ITEM	SPECIFICATION
Fuel type	Nitride type (100% ^{15}N enriched)
Core type	2- region homogeneous
Fuel assembly type	Duct type
Number of fuel assemblies (inner/outer core)	288 / 246
Number of fuel pins	331 per fuel assembly
Enrichment by Pu (inner/outer core)	14.7 / 17.8 weight %
Fuel burn-up	151 000 MW·d/t
Operation cycle length	20.2 months
Cladding outer diameter/ lattice pitch	7.16 / 8.85 mm
Pitch of fuel assemblies	176.5 mm

ITEM	SPECIFICATION
Core circumscribed radius	5.15 m
Core effective height	0.70 m
Average core power density	167.0 W/cm^3
Maximum linear heat rate	198.3 W/cm
Burn-up reactivity swing	0.79 %Δk/k

TABLE XXV.2. PLANT CHARACTERISTICS

ITEM	SPECIFICATION
Reactor type	Tank type
Electric output	710 MW(e)×4 units
Thermal output	1875 MW(th)×4 units
Primary coolant temperature	445 / 285°C
Main steam temperature / pressure	403.5^{o}C / 6.6MPa
Feedwater temperature	210°C
Plant efficiency	38 %
Primary coolant circulation	Forced convection
Steam generator	Helical coil type; located in reactor vessel
Decay heat removal systems	3 decay heat removal auxiliary cooling systems (DRACSs)
Containment system	Containment structure and liner

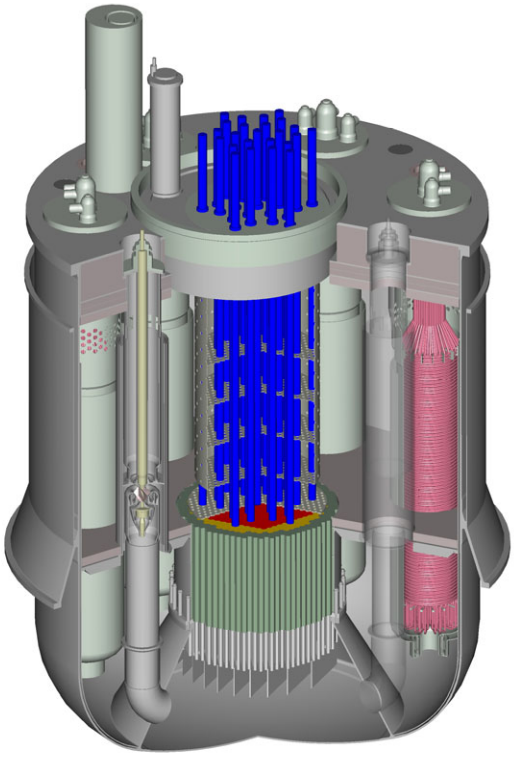

FIG. XXV-1. General view of a medium scale lead-bismuth cooled reactor (one unit).

XXV-3. List of enabling technologies and their development status

A list of the enabling technologies for the Medium Scale Lead-bismuth Cooled Reactor is presented in Table XXV-3.

TABLE XXV-3. LIST OF ENABLING TECHNOLOGIES FOR MEDIUM SCALE LEAD-BISMUTH COOLED REACTOR

ENABLING TECHNOLOGY	DEVELOPMENT STATUS
Nitride fuel technology	Conceptual design
Corrosion resistant operation of claddings in lead-bismuth coolant.	A corrosion test has been conducted for 10 000 hours in stagnant lead-bismuth.
An oxygen control system to protect structural materials operating in Pb-Bi from corrosion.	Conceptual design.
Control rod with a heavy metal sinker.	Conceptual design.
Three-dimensional seismically isolated reactor building.	Experiments on the laboratory scale have been performed.

DESIGN DESCRIPTIONS OF NON-CONVENTIONAL SMRs

ANNEX XXVI

THE ADVANCED HIGH TEMPERATURE REACTOR (AHTR)

Oak Ridge National Laboratory,
Sandia National Laboratories and
University of California, Berkeley, United States of America

XXVI-1. General information, technical features and operating characteristics

XXVI-1.1. Introduction

The Advanced High Temperature Reactor (AHTR) is a new reactor concept that combines four existing technologies in a new way:

- Coated particle graphite matrix nuclear fuels (traditionally used for helium cooled reactors).
- Brayton power cycles.
- Passive safety systems and plant designs from liquid metal cooled fast reactors.
- Low pressure liquid salt coolants with boiling points far above the maximum coolant temperature.

The coolant is a liquid (molten) salt; however, the AHTR is not a molten salt reactor (MSR). The term "molten salt reactor" refers to reactors that have the fuel dissolved in the coolant (i.e. MSRs are liquid fuel reactors). The liquid salt in the AHTR is a clean coolant with a coolant cleanup system to remove impurities. The coolant cleanup system ensures low rates of equipment corrosion and minimizes radioactive contamination outside the reactor core. The AHTR uses a traditional solid fuel. The Appendix to Annex XXVI provides more detailed information on the characteristics of liquid salts as coolants and the relevant industrial experience.

This new combination of technologies enables the design of a 300 to 1200 MW(e) [600 to 2400 MW(th)] high-temperature reactor, with reactor-coolant exit temperatures between 700 and 1000°C and passive safety systems for economic production of electricity or hydrogen. A lower-temperature AHTR (700°C) can use existing qualified metals of construction, whereas very high temperatures will require development of new materials. The high temperature liquid cooled system minimizes the size of the safety systems and power conversion equipment per kilowatt (electric) output. Because of the similar fuel, core design, and power cycles, about 70% of the required research is in common with that for high temperature gas cooled reactors. The reactor concept is being developed in the USA co-jointly by Oak Ridge National Laboratory, Sandia National Laboratories, and the University of California at Berkeley. Several commercial reactor vendors are currently evaluating the concept.

XXVI-1.2. Applications

The AHTR is designed to produce electricity and/or high temperature heat. The heat may be used for hydrogen production or other applications.

XXVI-1.3. Special features

The AHTR design concept and physical characteristics may provide three desirable features for the utility:

Economics

The AHTR may potentially enable the development of a low cost medium sized reactor using factory assembled modular units with ease of transport. Several technical factors provide the basis for this conclusion:

- As a low pressure reactor, AHTR has no heavy pressure vessels with associated transport difficulties.
- As in a liquid cooled reactor, the equipment sizes are small relative to those for gas cooled reactor concepts.
- The medium sized AHTR offers passive safety — a feature usually associated with smaller reactors.
- The Brayton power cycles are traditionally modular units — a characteristic that is assisted by their high power density per unit power output.

Dry cooling

The AHTR is a high temperature reactor with high efficiency in the conversion of heat to electricity (~50%). This minimizes the cost of dry cooling and may allow siting at locations with minimum water resources.

Passive safety

The AHTR is a passively safe reactor in sizes larger than the traditional designs of passively safe high temperature gas cooled reactors.

XXVI-1.4. Summary of major design and operating characteristics

The AHTR (Fig. XXVI-1, Table XXVI-1) uses coated particle graphite matrix fuels and a liquid fluoride salt coolant [XXVI-1, XXVI-2 and XXVI-3]. The fuel is the same type that is used in modular high temperature gas cooled reactors (MHTGRs), with accident fuel failure temperatures in excess of 1600°C. The transparent liquid salt coolant is a mixture of fluoride salts. Depending upon the specific salt mixture, the freezing point is between 350 and 500°C and the atmospheric boiling points exceed 1200°C. Several different closely related fluoride salts are being evaluated (see the Appendix) as the primary coolant, including lithium-beryllium and sodium-zirconium fluoride salts. The reactor operates at near atmospheric pressure. At operating conditions, the heat transfer properties of the salt are similar to those of water. Heat is transferred from the reactor core by the primary salt coolant to an intermediate heat transport loop. The intermediate heat transport loop uses a secondary liquid salt coolant to move the heat to a thermochemical hydrogen production facility to produce hydrogen or to a turbine hall to produce electricity. If electricity is produced, a multi-reheat nitrogen or helium Brayton power cycle is used.

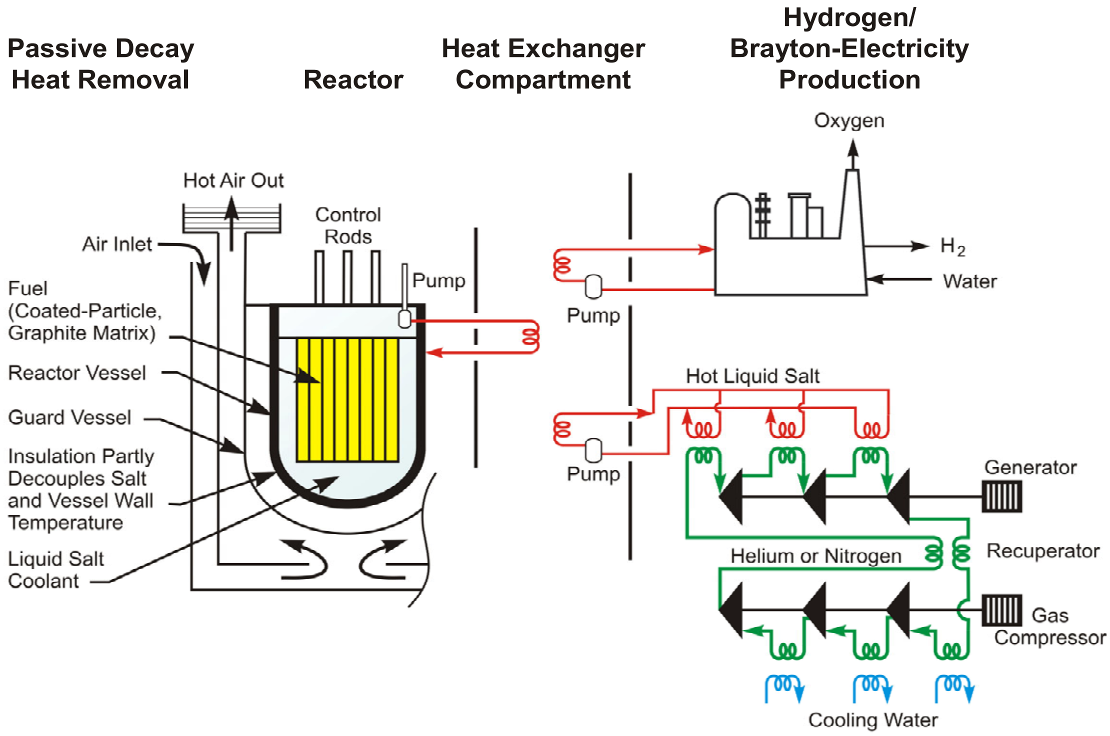

FIG. XXVI-1. Schematic of the AHTR for electricity production.

The baseline AHTR facility layout (Fig. XXVI-2) that was developed is similar to that of the S-PRISM sodium cooled fast reactor designed by General Electric. Both reactors operate at low coolant pressures and high temperatures; thus, they have similar design constraints. The major difference in facility design is that the AHTR operates at much higher temperatures and thus uses a Brayton (helium or nitrogen) power cycle rather than a Rankine (steam) cycle. The vessel size determines the core size, which, in turn, determines the power output. In the initial baseline studies, it was assumed that the fuel and power density (8.3 W/cm^3) were essentially identical to those of the MHTGR. This is a conservative assumption because higher power densities are possible with liquid coolants. The General Electric design studies indicated that reactor vessels up to 9.2 m in diameter could be practically constructed — the size of the proposed S-PRISM reactor vessel. Using the same vessel size and a facility design similar to that of the S-PRISM, the AHTR power level could be as large as 2400 MW(th). This power output was used in most of the initial studies because the detailed design and cost information on S-PRISM was used for the AHTR pre-conceptual studies. The actual reactor size will depend upon the application. As a new reactor concept, only limited studies have been conducted. These studies examined plant sizes from 600 to 2400 MW(th). As discussed later, the characteristics of a high temperature low pressure liquid cooled reactor may enable the design of cost effective medium sized reactors.

Three peak coolant temperatures were evaluated: 705, 800, and 1000ºC, for the AHTR — Low Temperature (AHTR-LT), the AHTR — Intermediate Temperature (AHTR-IT), and the AHTR — High Temperature (AHTR-HT), respectively. The respective thermal-to-electric efficiencies are 48.0, 51.5, and 56.5%. The AHTR-LT uses existing code qualified materials, the AHTR-IT uses existing materials that have not been fully tested, and the AHTR-HT uses advanced materials. The AHTR-LT has a metallic blanket system that separates and insulates the reactor vessel from the reactor core so that the fuel and coolant can operate at higher

temperatures than the vessel. The AHTR-HT and AHTR-IT have similar systems using a graphite blanket system. The insulation ensures long vessel life (minimizing long term creep) and minimizes heat losses during normal operations.

TABLE XXVI-1. SUMMARY OF AHTR PRE-CONCEPTUAL DESIGN PARAMETERS

Power level	600 to 2400 MW(th)	Power cycle	3-stage multi-reheat Brayton
Core inlet/outlet temperature (options)	900°C/1000°C 700°C/800°C 670°C/705°C	Electrical efficiency (heat to electricity vs. temperature)	56.5% at 1000ºC 51.5% at 800ºC 48.0% at 705ºC
Coolant (several options)	2^7LiF-BeF_2 (NaF-ZrF_4)	Power cycle working fluid	Nitrogen (helium is longer term option)
Fuel		Vessel	
Kernel	Uranium carbide/oxide	Diameter[a]	9.2 m
Enrichment	10. wt % ^{235}U	Height[a]	19.5 m
Form	Prismatic	Reactor core Shape	 Annular
Block diameter	0.36 m (across flats)	Diameter[a]	7.8 m
Block height	0.79 m	Height[a]	7.9 m
Columns	324[a]	Fuel annulus[a]	2.3 m
Decay heat system	Air cooled	Power density	8.3 W/cm^3
Volumetric flow rate	5.54 m^3/s	Reflector (outer)[a]	138 fuel columns
Coolant velocity	2.32 m/s	Reflector (inner)[a]	55 fuel columns
Power cycle	3-stage multi-reheat Brayton	Range of estimated overnight capital costs	US \$816–US \$930/kW(e)[a]
Annual consumption of natural uranium[b]	144 metric tons per GW(e) year		

[a] Size for 2400 MW(th) output
[b] Assumes 0.3% ^{235}U tails assay in the enrichment process, 119 500 MW d/MT U and 50% conversion efficiency of heat to electricity.

In the current pre-conceptual design, the AHTR has an annular core through which the coolant flows downward. The liquid salt coolant flows upward through the non-fuel graphite section in the middle of the reactor. The coolant pumps and their intakes are located above the reactor core with appropriate siphon breakers; thus, the reactor cannot lose its coolant except by failure of the primary vessel. The guard vessel is sized so that even if the primary vessel fails, the core remains covered with salt.

The unique characteristic of the AHTR is that it is a liquid cooled high temperature reactor whereas traditional high temperature reactors use gas cooling. This technical characteristic gives the AHTR unique characteristics in terms of efficiency of converting heat to electricity and capabilities to produce high temperature heat for applications such as hydrogen production.

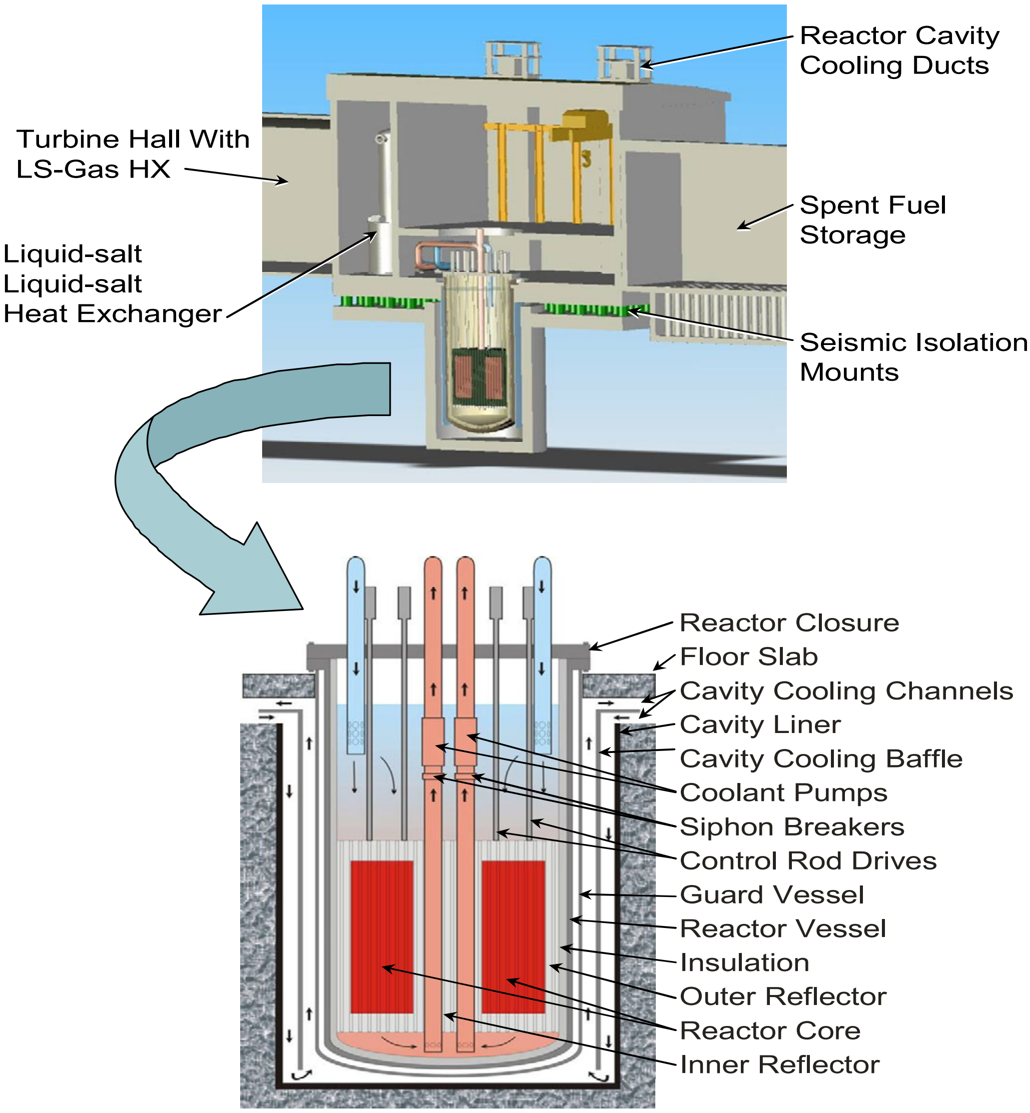

FIG. XXVI-2. Schematic of the AHTR nuclear island and vessel.

Electricity Production

For electricity production, recuperated gas (nitrogen or helium) Brayton cycles (Fig. XXVI-1) are proposed with two to three stages of reheating and three to eight stages of inter cooling [XXVI-4, XXVI-5]. Brayton power cycles are the only efficient power cycles that have been developed that match the temperatures of high temperature reactors. The multi- reheat Brayton cycle is somewhat similar to the multi- reheat Rankine steam cycle used in many coal fired power stations. The gas pressure is reduced through multiple turbines in series, with reheating of the gas to its maximum temperature with hot liquid salt before it reaches each turbine. The gas then flows through a recuperator and is compressed in multiple stages with inter stage cooling. Both nitrogen and helium Brayton cycles are being considered.

The AHTR is a liquid cooled, high-temperature reactor, a fact that significantly increases the electrical efficiency (4 to 8%) relative to that of gas cooled reactors with the same exit coolant temperatures. Gas cooled reactor systems have high pumping costs relative to those of liquid cooled systems. Because gas cooling has high pressure losses, practical designs of gas cooled

reactors [such as the General Atomics helium cooled MHTGR with a direct gas turbine (GT) cycle and the British carbon-dioxide cooled Advanced Gas Reactor (AGR)] have large temperature increases across the reactor core and deliver their heat to the power cycle over a large temperature range (Fig. XXVI-3). Typical temperature increases across the core are 350°C. For example, while the proposed General Atomics MHTGR has an exit temperature of 850°C, the average temperature of delivered heat is only 670°C and the lowest temperature of delivered heat is 491°C. In contrast, liquid cooled reactors such as the French sodium cooled Super Phoenix liquid metal fast breeder reactor (LMFBR) and pressurized water reactors (PWRs) have low pumping costs and are designed to deliver their heat from the reactor core to the power cycle over a small temperature range, 20 to 150°C. The same is true for the AHTR. Efficiency is roughly dependent upon the average temperature delivered to the power cycle, not the peak temperature; thus, an AHTR with a peak coolant exit temperature of 750°C delivers heat at a higher average temperature to the power cycle than an MHTGR with a gas coolant exit temperature of 850°C. Liquid cooling results in higher power plant efficiencies at lower peak reactor coolant temperatures.

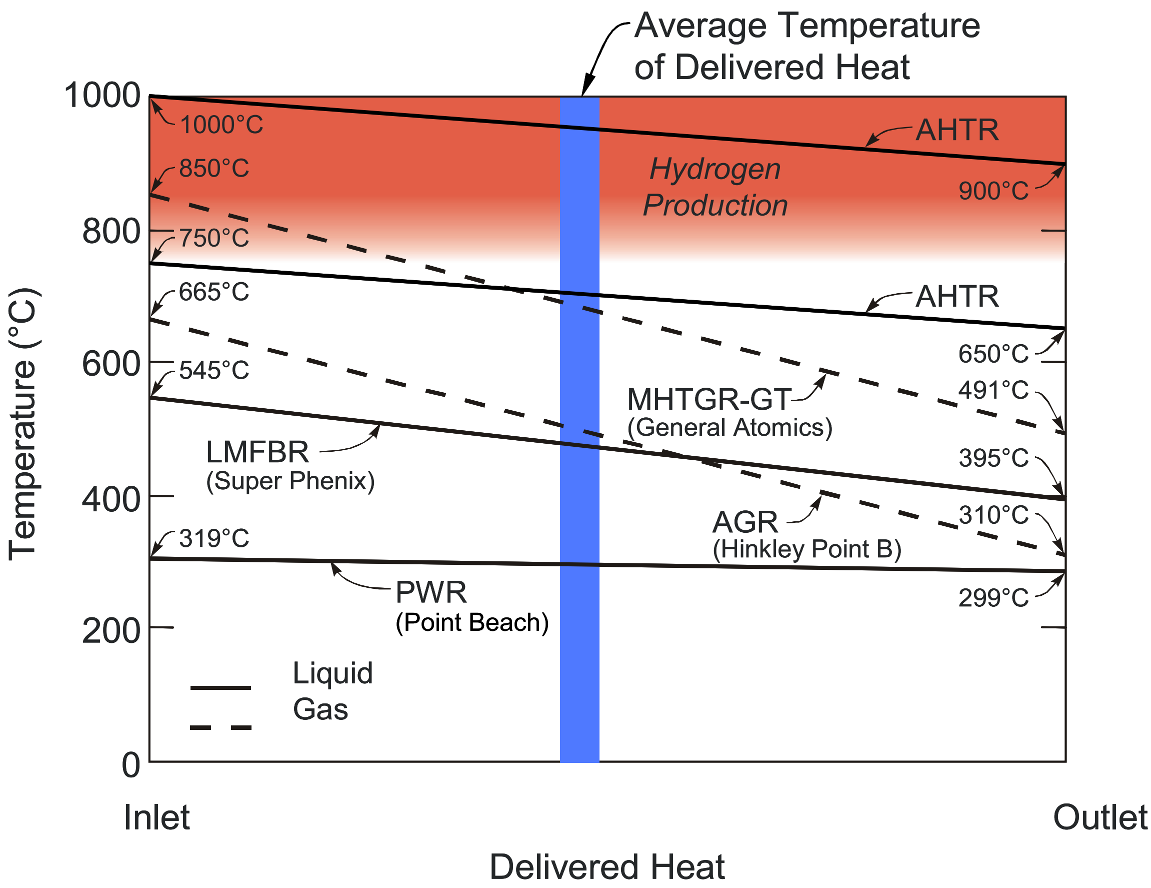

FIG. XXVI-3. Temperature of heat delivered to power conversion cycle for different reactors.

Hydrogen Production

The leading technologies for the low cost production of hydrogen using nuclear energy are high temperature thermochemical cycles [XXVI-6], in which, through a series of chemical reactions, high temperature heat and water yield hydrogen and oxygen. The primary technical challenge is that heat must be provided, depending upon the process, at temperatures between 700 and 850°C. The reactor temperatures must be significantly higher to transfer heat from the reactor fuel, through an intermediate heat transfer system and then to the thermochemical hydrogen plant. The required reactor temperatures are at the limits of conventional

engineering materials. As a liquid cooled reactor, the AHTR [XXVI-7] has peak reactor coolant temperatures 100 to 200°C lower than those of gas cooled reactors for heat delivered at the same temperatures to a thermochemical hydrogen plant. These is because liquids are better coolants than gases and have smaller temperature drops (1) from fuel to coolant and (2) across the two heat exchangers in the intermediate heat transport loop. As a consequence, the high temperature materials requirements to deliver high temperature heat may be significantly less for the AHTR than for gas cooled reactors.

XXVI-1.5. Outline of fuel cycle options

The AHTR uses the typical gas cooled reactor prismatic fuel. The neutron physical characteristics (flux levels, neutron energies, etc.) are nearly identical to those of traditional high temperature gas cooled reactors. As a consequence, the fuel cycle options and characteristics are essentially identical to those of traditional gas cooled reactors as well.

XXVI-1.6. Technical features and technological approaches that are definitive for AHTR performance in particular areas

XXVI-1.6.1. Economics and maintainability

Preliminary overnight capital costs (Table XXVI-2) of a 2400 MW(th) AHTR for several exit temperatures were determined relative to other higher temperature reactor concepts [i.e., the S-PRISM and the gas turbine - modular helium reactor (GT-MHR)] based on the relative size of systems and quantities of materials. The economic analysis used the larger size AHTR because the initial studies used the basic S-PRISM facility design where relatively detailed system design and cost information was available. This approach provides relative, but not absolute, costs. Only the construction of multiple reactors can provide reliable absolute costs. The lower capital costs are a consequence of several factors: economics of scale [a 2400 MW(th) reactor vs. four 600 or 1000 MW(th) reactors], passive safety in a large reactor system, and higher thermal efficiency.

TABLE XXVI-2. COMPARISON OF ESTIMATED OVERNIGHT CAPITAL COST (2002 US$/KW(e)) OF THE AHTR-IT AND AHTR-HT, AS A PERCENTAGE OF THE COSTS OF THE S-PRISM AND GT-MHR (WITH MULTI-MODULE OUTPUT OF 1145 MW(e))[a]

		S-PRISM $ 1681/kW(e)	GT-MHR $ 1528/kW(e)
AHTR-IT	$ 930/kW(e)	55%	61%
AHTR-HT	$ 816/kW(e)	49%	53%

[a] The General Electric S-PRISM consists of four reactor modules, each producing 1000 MW(th) and 380 MW(e). The peak sodium temperature is 510ºC. The General Atomics GT-MHR consists of four reactor modules, each producing 600 MW(th) and 285 MW(e). The peak helium temperature is 850ºC.

The size of most passively safe high temperature reactors is limited by the design characteristics of the passive decay heat removal systems to about 600 MW(th) with power outputs of 200 to 300 MW(e). The AHTR does not have this technological size limitation (see section XXVI-1.6.3). As a consequence, it can be built as a medium (600 MW(e)) or as a large reactor. This offers economics of scale.

The AHTR has two coupled physical characteristics that may potentially enable the development of a low cost medium sized AHTR using factory assembled modular units with ease of transport. The low pressure liquid cooling for a medium sized reactor implies small reactor vessel and heat exchanger sizes relative to those for gas cooled reactors. The high temperature allows the use of Brayton power cycles. Brayton turbines have much higher power densities than steam turbines and are consequently much smaller in size per unit output. Brayton cycle turbines are typically manufactured and shipped as modular units and have lower costs than traditional steam cycles per unit output. These options have not yet been investigated.

XXVI-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

Fuel sustainability and waste management

The AHTR uses the same fuel and has the same fuel cycle options as gas cooled reactors with coated particle fuels. As a consequence, the provisions for sustainability and waste management are essentially identical to those for high temperature gas cooled reactors.

Enabling technology for future fuel sustainability

While the AHTR has fuel cycle characteristics that are essentially identical to those of traditional gas cooled reactors, it is potentially the bridge or gateway to three different reactors with long term energy sustainability. The central characteristic of the AHTR is that it uses a low pressure, high temperature liquid coolant. That coolant technology is directly applicable to other high temperature reactors. There are strong incentives to develop high temperature nuclear energy systems with coolant temperatures between 700 and 1000°C. Power plant efficiency increases with temperature. Consequently, the size and cost per kilowatt (electric) of energy conversion systems (heat to electricity), heat rejection systems (cooling towers, etc.), and nuclear safety systems (decay heat removal) become smaller as the reactor temperature increases. The higher thermal efficiency implies lower nuclear fuel costs and less generation of waste per unit of energy.

While the above benefits of high temperature reactors have been understood for decades, it is the new goals and enabling technologies that are creating a renewed interest in such reactors and the potential for practical economic high temperature reactors:

- Hydrogen production. There is a growing interest in hydrogen production using thermochemical hydrogen cycles that require high temperature heat to convert water to hydrogen and oxygen. Preliminary cost estimates indicate that hydrogen production costs using these cycles may be ~60% of the cost of hydrogen from electrolysis. Depending upon the system [XXVI-6], heat must be delivered between 700 and 850°C.

- Brayton power cycles. Brayton power cycles using nitrogen or helium allow efficient conversion of high temperature heat to electricity. Only in the last decade have the efficient Brayton power cycles, derived from aircraft engines, been developed for utility applications. For over 80 years, the traditional utility power conversion cycle has been the Rankine steam cycle. Steam cycles have served the utility industry well; however, the upper limit of practical steam cycles is between 500 and 600°C. Earlier high temperature reactors were coupled to steam cycles and thus were unable to efficiently utilize the high temperature heat that was available from the reactor core for electricity generation.

Nuclear reactor types can be classified by power output and by the peak temperatures of their coolants (Fig. XXVI-4). Light water reactors (LWRs), such as the General Electric Economic Simplified Boiling Water Reactor (ESBWR), are low temperature, high pressure reactors.

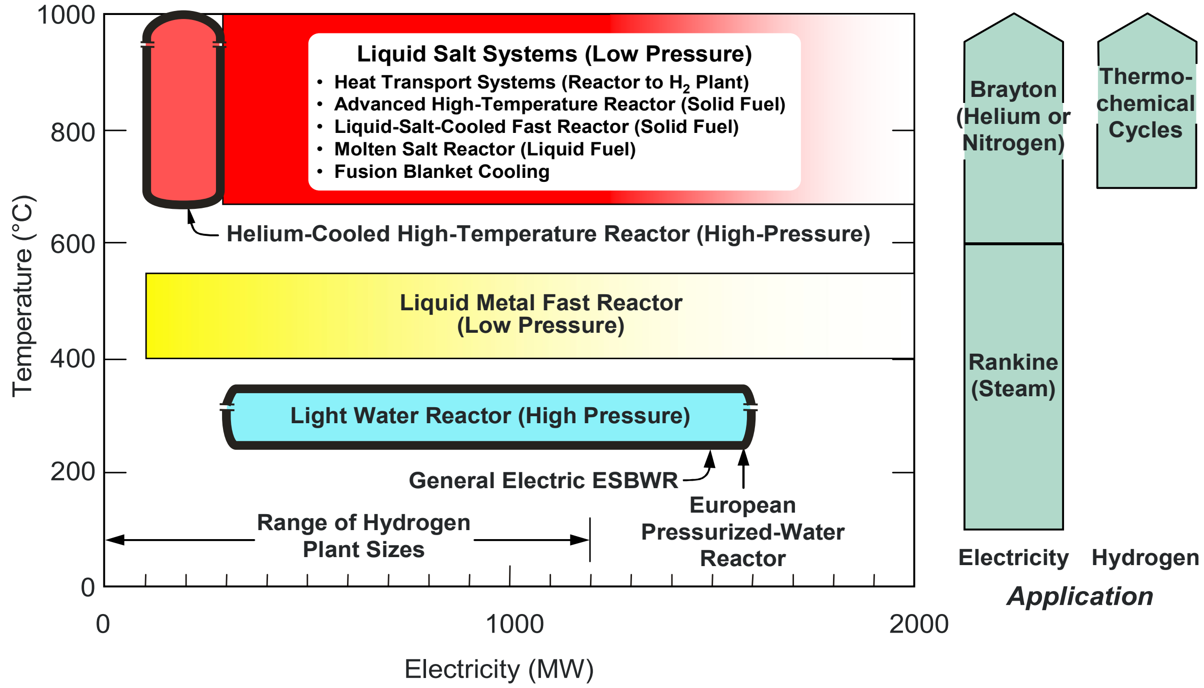

FIG. XXVI-4. Reactor type vs. temperature and power output.

Traditional fast reactors cooled with liquid sodium operate at medium temperatures and low pressures. Two options exist for high temperature reactor coolants:

- High pressure gases, and
- Low pressure liquids with boiling points above the peak coolant temperatures.

Helium is the traditional high temperature, high pressure gas coolant. Liquid fluoride salts are a traditional high temperature, low pressure liquid coolant. The only other potential candidates are liquid metals, particularly molten lead or lead alloys for fast spectrum reactors. Because of their relatively low boiling points, traditional liquid metals such as sodium are not candidates for high temperature operations.

Liquid fluoride salts can be used for multiple nuclear applications. The AHTR is the simplest reactor application (excellent compatibility between salt and graphite based fuel, once through fuel cycle, etc.) and thus the likely first commercial use for liquid salts in reactor applications. The base technology, however, leads to three long term sustainable advanced reactor options:

- Liquid salt cooled fast reactor. Liquid salt cooled fast reactors are similar in design (fuel, plant, etc.) to sodium cooled fast reactors, except they are designed to operate at higher temperatures and use Brayton power cycles. These reactors use fast reactor fuel cycles and may be breeder reactors. Only limited exploratory work [XXVI-8] has been conducted on such reactors; thus, there are many uncertainties.

- MSR. The MSR is a thermal neutron breeder reactor that uses the ^{233}U-Th fuel cycle with very low production of actinides. The MSR is a liquid fuel reactor in which uranium, fission products, and actinides are dissolved in a liquid fluoride salt. The fuel

salt flows through a graphite reactor core, which acts as a moderator. Fuel salt carries the heat to an intermediate heat exchanger. The intermediate heat transport loop transfers the heat to a Brayton power cycle or a hydrogen production facility. This reactor concept was partly developed in the 1970s, and two test reactors were successfully developed [XXVI-9, XXVI-10].

- Fusion reactors. Liquid salts are major candidates for cooling inertial and magnetic fusion energy systems [XXVI-11].

Water and land resources

Water consumption and land use are major sustainability issues. Power production often conflicts with these other critical sustainability challenges:

- Water. Existing power plants use very large quantities of water as cooling water for rejection of heat from the power cycle. In the United States, the largest uses for water are irrigation and cooling water for power plants, with each application using a similar amount of water. Worldwide, power production is in direct conflict with other uses of water.
- Land use. Cooling water for power production has traditionally been the first requirement for siting a nuclear power plant. However, almost all of the world's population lives close to oceans, lakes, or rivers because of the requirements for water. The most critical natural areas are typically associated with water. The cooling water requirements of nuclear power plants conflict with other primary land uses.

The AHTR capabilities minimize the consumption of water via the efficient use of dry cooling systems for heat rejection [XXVI-12]. The challenge of heat rejection using dry cooling is economics. While fossil Rankine steam power plants (totalling 30 000 MW(e)) have been built with dry cooling where water was not available, the costs have been high. These penalties can be drastically reduced with higher temperature multi-reheat Brayton cycle nuclear power plants:

- Less heat rejection. Current LWRs have operating temperatures of ~270°C, with an efficiency of ~33%. The AHTR is significantly more efficient because of its higher temperature multi-reheat power cycle. For peak coolant temperatures of 705, 800, and 1000ºC, the respective plant efficiencies are 48, 51.5, and 56.6%. While the LWR rejects 2 kW(t) of heat per kilowatt (electric), the three AHTR designs reject, respectively, 1.08, 0.94, and 0.77 kW(t) per kilowatt (electric). The higher efficiency reduces the heat rejection system capacity requirements by about a factor of 2 relative to LWRs.
- Reduced penalty for higher heat rejection temperatures. The capital costs of dry cooling systems can be reduced by rejecting heat at a higher temperature but with the penalty of lower plant efficiency. That penalty becomes smaller as the peak temperature of the power cycle increases. For the AHTR Brayton cycle with a minimum helium temperature of 35°C, the losses in efficiency for a 10ºC rise in the compressor inlet temperature were calculated to be 1.5, 1.3, and 1.1%, respectively, for AHTR peak coolant temperatures of 705, 800, and 1000°C.
- Heat rejection over a temperature range. Dry cooling involves heating air (i.e., raising the temperature). If the heat from the power cycle can be rejected over a temperature range rather than at a single temperature, the appropriate design of counter current dry cooling tower heat exchangers results in a constant temperature drop across the heat

exchangers, which reduces their required size. With dry cooling, Brayton cycles have major advantages over Rankine (steam) cycles that deliver rejected heat at a constant temperature. This Rankine cycle characteristic is consistent with evaporative cooling, in which water is vaporized at a nearly constant temperature. In contrast, a Brayton cycle delivers rejected heat over a temperature range that matches dry cooling. In the Brayton cycles described herein, the heat is rejected over a 50°C range, with the helium being cooled from ~85 to 35°C.

The technical basis for the match between the AHTR and dry cooling is shown in Fig. XXVI-5, the temperature entropy diagram for a three stage multi-reheat Brayton power cycle.

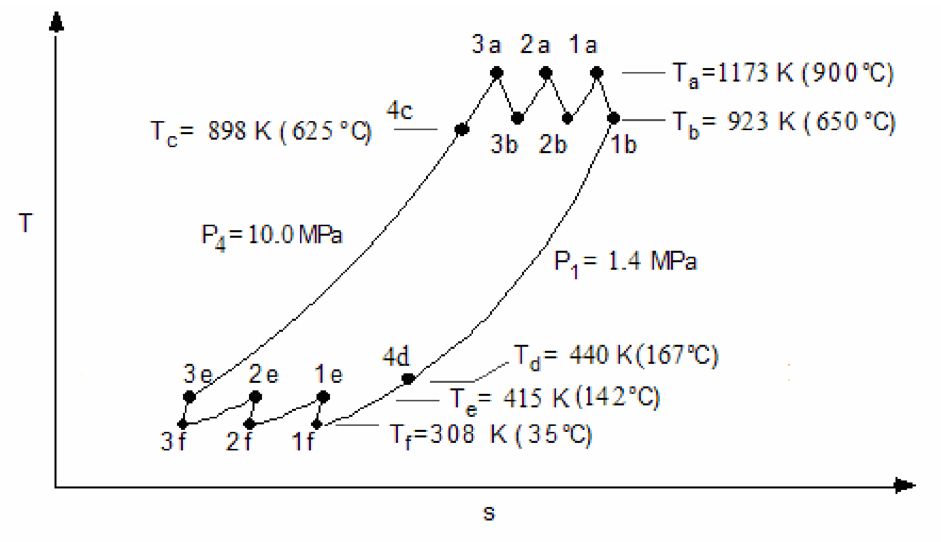

FIG. XXVI- 5. Temperature (T) entropy (S) diagram for the very high temperature, three stage multi-reheat Brayton power cycle.

The high efficiency of the power cycle with low rates of heat rejection is a consequence of the power cycle approximating an ideal Carnot cycle, where heat is inputted at one high temperature and rejected at one low temperature. Figure XXVI-5 also shows that the heat is rejected over a temperature range and thus matches the needs of dry cooling systems.

XXVI-1.6.3. Safety and reliability

Safety concept and design philosophy

The safety strategy for the AHTR is to use (1) passive safety systems for normal and accident conditions and (2) inherent safety features for beyond design basis accidents. This is to ultimately allow the AHTR to be licensed without off site emergency planning. Figure XXVI-6 shows the heat removal pathways for the AHTR.

Normal operations

Under normal operating conditions, the heat is transferred to the Brayton power conversion system that produces electricity. Reject heat from the power conversion unit is dumped to the environment via cooling towers.

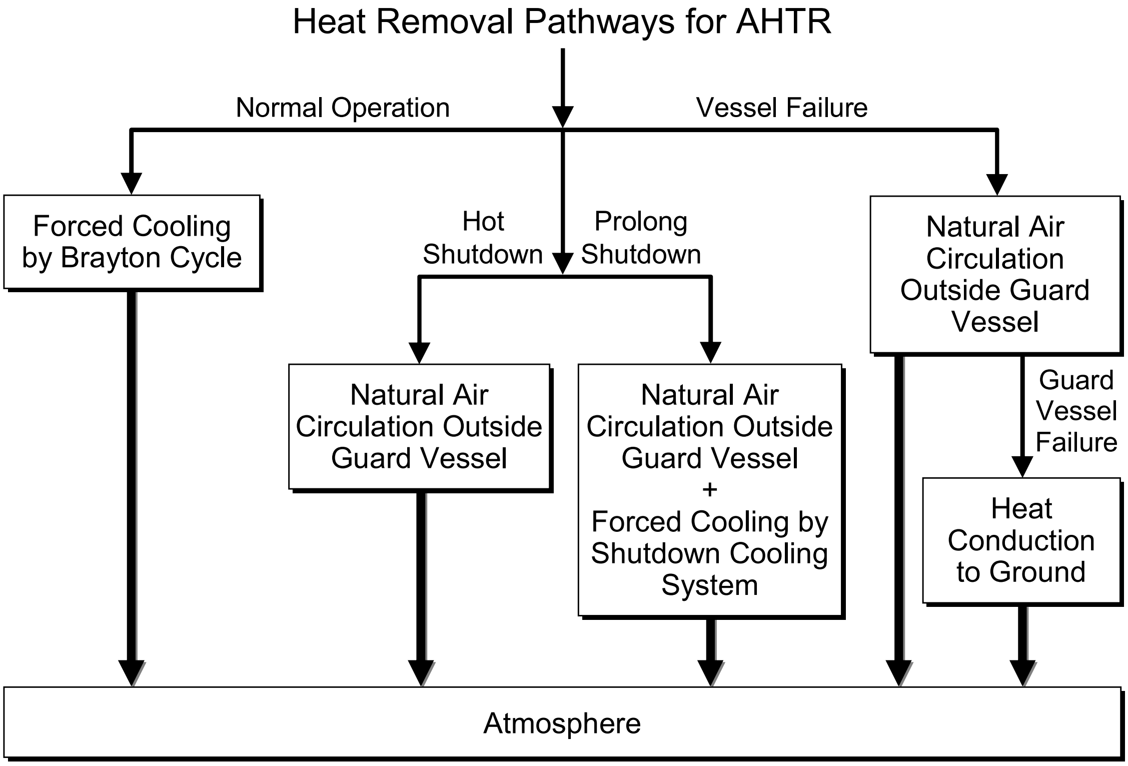

FIG. XXVI- 6. Normal, accident, and beyond design basis accident decay heat removal.

Shutdown operations

There are two decay heat removal systems when the reactor is shut down. The first is a passive decay heat removal system used for all shutdowns (planned, unplanned, accident) and the second is an active non-safety grade system used for refuelling and other shutdown operations that require close control of the reactor core temperature.

The reference AHTR design (Fig. XXVI-1) uses passive reactor vessel auxiliary cooling (RVAC) systems similar to that developed for decay heat removal in the General Electric sodium cooled S-PRISM. The reactor and decay heat removal system are located in a below grade silo. In the AHTR, RVAC system decay heat is:

- Transferred from the reactor core to the reactor vessel graphite reflector by natural circulation of the liquid salts.
- Conducted through the graphite reflector and reactor vessel wall.
- Transferred across an argon gap by radiation to a guard vessel.
- Conducted through the guard vessel, and then
- Removed from outside of the guard vessel by natural circulation of ambient air.

Primarily the radiative heat transfer through the argon gas from the reactor vessel to the guard vessel controls the rate of heat removal. Radiative heat transfer increases by the temperature to the fourth power (T^4); thus, a small rise in the reactor vessel temperature (as would occur upon the loss of normal decay heat removal systems) greatly increases heat transfer out of the system. Under accident conditions such as a loss of forced cooling accident, natural circulation flow of liquid salt up the hot fuel channels in the core and down the edge of the core rapidly results in a nearly isothermal core with about a 50°C temperature difference

between the top and bottom plenums. For a typical simulation of the reactor with an average coolant exit temperature of 1000°C, the calculated peak fuel temperature in such an accident is ~1160°C, which will occur at ~30 hours after loss of pumped coolant flow, with a peak reactor vessel temperature of ~750°C at ~45 hours. The average core temperature in this accident rises to approximately the same temperature as the hottest fuel during normal operations.

The AHTR uses a guard vessel for two functions. The first function is part of a thermal switch mechanism that ensures low losses of heat during normal operations but efficient removal of decay heat during shutdown or off normal conditions. The second function of the guard vessel is to provide a backup in the event of vessel failure and catch the core and the liquid salt within the core.

In terms of passive decay heat removal systems, a major difference is noted between the liquid cooled AHTR and gas cooled reactors. The AHTR can be built in very large sizes (>2400 MW(th)), while the maximum size of a gas cooled reactor with passive decay heat removal systems is limited to ~600 MW(th). The controlling factor in decay heat removal is the ability to transport this heat from the center of the reactor core to the vessel wall or to a heat exchanger in the reactor vessel. The AHTR uses a liquid coolant, where natural circulation can move very large quantities of decay heat from the hottest fuel to the vessel wall with a small coolant temperature difference (~50°C). Unfortunately, under accident conditions when a gas cooled reactor is depressurized, the natural circulation of gases is not efficient in transporting heat from the fuel in the center of the reactor to the reactor vessel. The heat must be conducted through the reactor fuel to the vessel wall. This inefficient heat transport process limits the size of the reactor to ~600 MW(th) to ensure that the fuel in the hottest location in the reactor core does not overheat and fail under accident conditions.

Mitigation of beyond design basis accidents

Beyond design basis accidents involve catastrophic failures that make active and passive safety systems non-functional. The AHTR has a combination of accident mitigation capabilities that may ultimately eliminate the need for off site emergency planning. This capability is based three inherent characteristics of the system:

- Fuel. The AHTR uses the same fuel as the GT-MHR. This high temperature fuel has the same excellent high temperature fission product retention capabilities.
- Coolant. Most fission products (excluding primary krypton and xenon) and all actinides escaping the fuel are soluble in the liquid salt and will remain in the liquid salt at very high temperatures. Caesium and iodine remain in the salt. The liquid salt provides an effective method to transport heat from the fuel to the silo and then to the environment. The coolant prevents air ingress and access to the graphite matrix fuel, thus avoiding the potential of oxidation of the fuel matrix.
- Low energetics. The chemical inertness and low pressure of the liquid salt coolant eliminate the potential for damage to the confinement structure by rapid chemical energy releases (e.g. sodium) or coolant vaporization (e.g. water).

In a beyond design basis accident, it is assumed that the air cooled passive decay heat removal system has failed and that significant structural failures (vessel failure, etc.) have occurred. Decay heat continues to heat the reactor core but decreases with time. To avoid the potential for catastrophic accidents (accidents with significant release of radionuclides), the temperature of the fuel must be kept below that of fuel failure by (1) absorption of decay heat in the reactor and silo structure and (2) transfer of decay heat through the silo walls to the environment. For

the MHTGR, the maximum size of reactor that can withstand this accident without major fuel failure is ~600 MW(th).

Work has begun to define the maximum size AHTR that can withstand this type of accident based on the earlier work on MHTGRs. The choice of (1) a high temperature fuel and (2) a low pressure (relatively chemically inert), high temperature coolant enables construction of larger reactors with this capability. The beyond design basis strategy can be understood by following the sequence of expected events and defining the mechanisms to prevent massive fuel failure (Fig. XXVI-7).

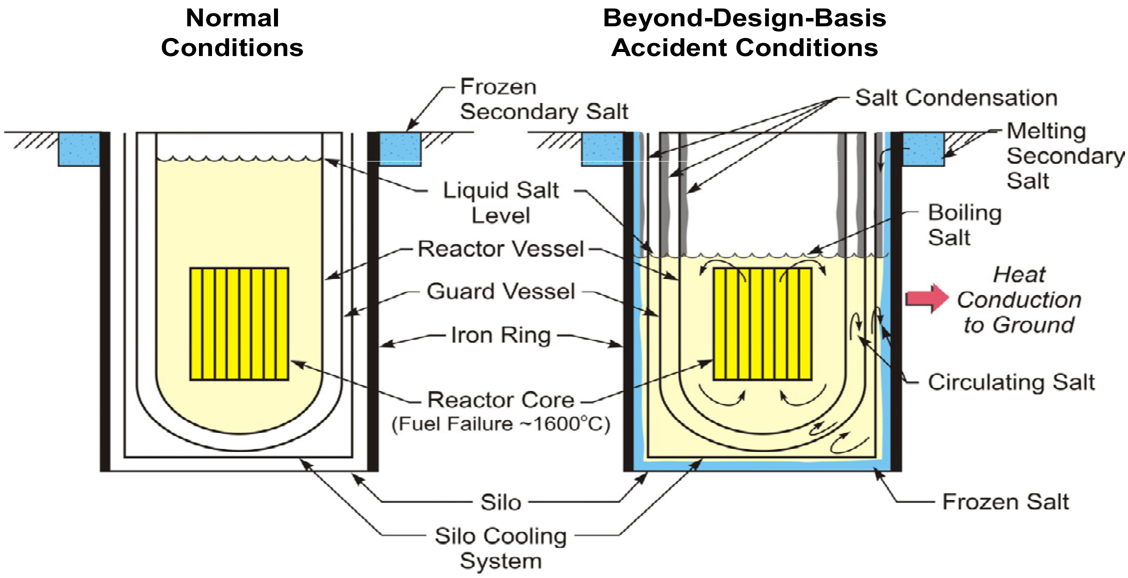

FIG. XXVI-7. Normal and beyond design basis accident states for the AHTR.

Reactor vessel heat up

After loss of decay heat cooling, the initial event is heat up of the reactor vessel. The AHTR thermal inertia per megawatt thermal in the reactor vessel exceeds that of the MHTGR; that is, the peak fuel temperatures increase at a slower rate after loss of all cooling. This slower increase occurs despite the fact that the AHTR vessel volume (2400 MW(th), 9.2 m diameter, 1260 m^3) is almost identical to that of the MHTGR (600 MW(th), 8.4 m diameter, 1210 m^3) and reflects the more efficient use of the thermal inertia of materials within the reactor vessel.

Under design basis depressurization loss of cooling conditions in the MHTGR, large radial and vertical temperature gradients exist within the reactor vessel. Under depressurized conditions, the MHTGR peak fuel temperature reaches 1560°C after 60 hours, while the peak temperatures of the reactor vessel are less than 600°C. Large temperature gradients are needed to remove the decay heat by conduction. If the reactor remains pressurized with better heat transfer in the reactor vessel, the core temperature peaks at only 1240°C at 50 hours, because of the more uniform core temperature caused by natural convection of the high pressure helium coolant. Most of the mass in the reactor is far below allowable peak temperatures and not efficiently used to maximize the effective thermal inertia.

The larger total thermal inertia of the AHTR is a consequence of (1) the liquid salt circulation, which ensures almost isothermal conditions within the reactor core, and (2) the higher heat

capacity of reactor core. The conceptual design of the 2400 MW(th) AHTR has a 9.2 m diameter, 5 cm thick vessel with a 0.65 m thick graphite liner and reflector and an effective annular core diameter of 7.8 m. Conversely, the effective core diameter of the MHTGR is only 4.9 m, because of the 0.22 m thick vessel wall, the inner core barrel and shell for helium inlet down flow and vessel thermal conditioning, and the graphite reflector. In both reactors, the centres of the annular cores are filled with graphite, which is included in the calculations of core heat capacity. In the vertical direction, the MHTGR heats the 1.6 m thick graphite reflector, located above the 7.9 m high core. Conversely, the AHTR provides a 6.8 m deep molten salt pool above the core. Thus, for the AHTR, the ratio of the active volume to absorb heat relative to that of the MHTGR is 4.1. Furthermore, in the MHTGR a significant fraction of the thermally active volume is occupied by helium, which has negligible heat capacity. Conversely, in the AHTR, all of the active volume is occupied by graphite or by molten salt, which has a larger specific heat capacity than graphite.

Vessel failure

High temperatures ultimately cause the vessel to fail. Liquid salt coolant from the reactor vessel fills the bottom of the silo. The reactor vessel contains sufficient salt to keep the reactor core flooded. The circulating liquid salt between the reactor vessel and the silo efficiently transfers heat from the reactor vessel to the silo wall. Several different liquid salts are being considered as reactor coolants. The freezing points are typically 350°C, or somewhat higher. When the salt contacts the cold silo wall, it freezes. Unlike water, the salt will not leak out. Furthermore, no major chemical reactions that generate heat or gases will occur, which is not the case with sodium.

Silo wall heat conduction

The silo wall contains low cost thick steel rings that are similar to those used in the mining industry to line deep mine shafts and prevent their collapse. In the mining industry, these rings are referred to as tubing or “ausbau.” The diameter of the AHTR silo is similar to that of large mine shafts, but the depth is only 20 m. Under operating conditions, the rings are cooled by exposure to outside air that is drawn down in the silo and then flows up on the other side of a partition to remove heat from the guard vessel. Following vessel failure, the rings conduct heat up the silo wall and distribute it above the coolant salt layer.

Secondary salt melting

Near the top of the silo is an annular ring of a secondary solidified salt. As the temperature of the secondary salt increases, the salt melts, flows into the silo, and floods the silo to a higher level. The melting, heating, and boiling of the secondary salt can provide a significant source of thermal inertia:

- Heat absorption. Typical fluoride salts have a volumetric heat capacity of 4000 kJ/(m^3°C). If the secondary salt were heated to 1000°C, it would absorb 0.046 MW·d/m^3. The heat of vaporization for typical fluorides is about 0.16 MW·d/m^3. Depending upon design, the heat up and selected boil off of secondary salt components can absorb several days of decay heat.

- Salt selection. Unlike the reactor coolant salt, the secondary salt has no requirement for low nuclear cross sections to minimize neutron absorption. A variety of chloride and fluoride salts are potential candidates. Studies have not yet been conducted to define the preferred salt based on cost and performance requirements (compatibility with coolant salt and melting point). If appropriate low cost salts are found, the option exists for the secondary salt inventory to absorb days to weeks of decay heat.

Heat conduction to earth

Heat is conducted to the earth surrounding the silo and ultimately to the environment. The 600 MW(th) MHTGR uses the same approach for ultimate heat rejection in a beyond design basis accident. However, significant differences are noted between gas cooled and liquid salt cooled reactors in their ability to reject heat to the ground:

- Heat transfer area. The flooding of the silo with liquid salt increases the effective surface area of heat transfer from the reactor vessel to the silo wall. If the silo is full of liquid salt, the entire silo wall, not a small section of the wall, rejects heat to the environment. The placement of the reactor core at the very bottom of the reactor vessel allows full utilization of the complete silo area. Because liquid salt heat fluid is used for heat transfer, heat rejection rates can be further increased by (1) increasing silo depth or (2) designing the top of the silo with its shorter pathway for heat rejection to the environment. The effective heat transfer area is thus doubled.
- Uniform temperatures. Natural circulation of the liquid salt results in a relatively uniform temperature throughout the silo. The vertical temperature gradient will be only a few tens of degrees.
- Temperature drops. The peak temperature of the fuel is fixed by the need to avoid fuel failure. Temperature drops occur from the fuel to reactor vessel wall, from the vessel wall to the silo wall, and from the silo wall into the earth. Liquid cooling (reactor coolant and secondary salt) minimizes the first two temperature drops. This allows for higher silo temperatures, which, in turn, allow greater heat rejection to the ground.

Extrapolations from the MHTGR (considering heat capacity, effective silo surface area, and available temperature to drive heat from the silo wall to the environment) indicate that a 2400 MW(th)-AHTR with beyond design basis accident capabilities could be built. However, major uncertainties remain because such systems imply high temperatures near the silo and reactor facilities. There are many design choices and trade offs, including options that may not require a secondary salt.

Provisions for safety under seismic conditions

The AHTR facility design is similar to the General Electric S-PRISM; thus, the AHTR uses the same seismic safety strategy. The nuclear island (Fig. XXVI-2) is a seismically isolated platform with seismic isolation pads to separate it from large ground accelerations during an earthquake. The reactor vessel is hung from the nuclear island platform.

XXVI-1.6.4. Proliferation resistance

The proliferation resistance of the AHTR is essentially identical to that of gas cooled reactors using graphite matrix coated particle fuel because the same fuel and fuel cycle are used. The intrinsic proliferation resistance characteristics of this fuel include high burn up, a difficult to process fuel matrix, and a low ratio of fissile to fuel block mass.

XXVI-1.6.5. Technical features and technological approaches used to facilitate physical protection of AHTR

The AHTR has the potential for extraordinary capabilities to ensure protection of the public against severe accidents or sabotage. The combination of a high temperature fuel, the liquid salt coolant and an underground silo facility design implies that failure of the reactor vessel and other containment structures will not result in significant release of radionuclides to the

environment. While extreme events can result in a loss of investment, the design goal is to ensure no significant off-site consequences.

XXVI-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of AHTR

No information was provided.

XXVI-1.8. List of enabling technologies relevant to AHTR and status of their development

About 70% of the R&D required for the AHTR is shared with that for helium cooled high temperature reactors. This includes fuel development, materials development, and Brayton power cycles:

- Fuel development. The fuel development requirements for the AHTR are somewhat less than for equivalent gas cooled reactors. Liquids have better heat transport capabilities than gases; consequently, AHTR fuel temperatures will be 100 to 200°C lower than those of gas cooled reactors. This reduces the high temperature performance requirements for the fuel. However, as is true for gas cooled reactors, significant R&D is required to develop large scale manufacturing of high quality fuel.
- Graphite. Liquid salts (see the Appendix) have been shown to be chemically compatible with graphite in reactor radiation fields; thus, the graphite R&D requirements of liquid salt and gas cooled reactors are essentially identical. The R&D goals are to develop a more radiation resistant graphite to reduce graphite swelling with time and thus reduce the costs associated with the periodic replacement of graphite in the reactor core.
- Materials for vessels, pipes, and heat exchangers. The AHTR material development requirements are similar, but not identical, to those for gas cooled reactors. There are four challenges in the development of high temperature structural materials for reactors: air corrosion; coolant corrosion; long term creep at high temperatures; and long term changes in material properties because of the combined effects of radiation, time, and high temperatures. Three of these four challenges are identical for gas cooled and salt cooled systems. The major difference is minimizing corrosion caused by impurities in helium and liquid salts. Up to about 750°C, existing code qualified materials of construction exist for liquid salt systems. A number of potentially viable commercial metals are available that may allow higher temperature service but require material qualification for this service. At 1000°C, major material development programs are required for the AHTR. The Appendix discusses these material challenges in further detail.
- Brayton power cycles. The fundamental Brayton cycle development programmes for gas cooled reactors and the AHTR are almost identical.

The AHTR shares its facility design with liquid metal reactors. This includes passive decay heat removal systems, general plant layout, and refuelling systems. The major requirement is further development of these systems to allow higher temperature operations. The close coupling of fast reactor and AHTR facility designs implies that advances in either reactor facility concept assist in the development of the other.
Other areas that require significant R&D include:

- The reactor vessel insulation system.
- Optimisation of core design.

- Refuelling and maintenance operations in the reactor vessel at 350 to 500°C.
- Selection of the preferred liquid fluoride salt.

As a new reactor concept, the AHTR is early in its development, with significant technical uncertainties remaining.

XXVI-1.9. Status of R&D and planned schedule

The AHTR is part of the U.S. Department of Energy Generation IV reactor program and is being actively investigated. At the same time, commercial reactor vendors are conducting parallel studies. Detailed development plans are being prepared. As a new reactor concept, the AHTR would require a test reactor. If the AHTR is selected for large scale development, the goal would to be to have an operating test reactor by 2012. A medium sized pre-commercial demonstration reactor would follow this.

XXVI-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

The AHTR will require a test reactor as part of its development because no liquid salt reactors have ever been built. A test reactor provides the information required to design a pre-commercial reactor. The likely size of the test reactor would be ~50 MW(th) to provide sufficient energy output to test a multi-reheat Brayton cycle and potential thermochemical hydrogen production methods.

XXVI-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

The development of the AHTR is tightly coupled to MHTGRs because about 70% of the R&D is in common. This includes fuel development, materials development, and Brayton power cycles.

Appendix to ANNEX XXVI

Liquid salt coolants

There is a massive experience base with liquid salts [XXVI-3]; however, the nuclear reactor community does not know of most of this experience base. This Appendix provides background information on this key technology that is the basis of the AHTR.

History of liquid salt coolants

A significant experience base exists for only three high temperature liquids: molten iron, molten glass, and molten (liquid) fluoride salts. Since the 1890s, the Hall electrolytic process has produced essentially all aluminium. In the Hall process, aluminium oxide is dissolved in a mixture of sodium and aluminium molten fluoride salts (cryolite: $3NaF\text{-}AlF_3$) at ~1000°C in a graphite lined bath. Massive graphite electrodes provide the electricity that converts aluminium oxides to aluminium metal.

In the 1950s, at the beginning of the cold war, the USA launched a large program to develop a nuclear aircraft for delivery of nuclear weapons. Nuclear submarines were being developed at that time, and the U.S. Air Force wanted an equivalent aircraft with unlimited range. MSRs

were initially developed to provide a heat source with the heat transferred via an intermediate loop to a jet engine. MSRs were chosen for this application to minimize aircraft weight. The high temperature, low pressure fluid fuelled reactor avoided the need for heavy pressure vessels and the high temperatures maximized jet engine efficiency. Furthermore, because there is no potential to burn out the fuel, liquid fuelled reactors can have much higher power densities than solid fuel reactors, a factor that reduces reactor size and weight. The nuclear aircraft programme was ultimately cancelled because of the peacetime risks of aircraft crashes and the high shielding weight required protecting the crew.

In the 1960s and 1970s, the MSR was investigated as a thermal neutron breeder reactor [XXVI-9]. The liquid fuel offered a unique advantage: on-line processing of the fuel salt, which enabled the design of a thermal neutron breeder reactor using a thorium-^{233}U fuel cycle. The programme was ultimately cancelled when the USA decided to concentrate on the development of a single breeder reactor concept.

These billion dollar programmes developed the technology base for use of liquid salts in nuclear systems. Two experimental reactors were built and successfully operated. The aircraft reactor experiment (ARE) was the first MSR. It was a 2.5 MW(th) reactor that was operated in 1954 at a peak temperature of 860ºC and used a sodium-zirconium fluoride salt. This was followed in 1965 by the molten salt breeder reactor (MSBR) Experiment, an 8 MW(th) reactor that used a lithium-beryllium fluoride salt and demonstrated most of the key technologies for a power reactor. In addition, test loops with liquid salts were operated for hundreds of thousands of hours, materials of construction were code qualified to 750ºC, and a detailed conceptual design of a 1000 MW(e) MSBR was developed. Over 1000 technical reports were produced.

Properties of liquid salt coolants

Unlike water, sodium, or helium, liquid fluoride salts are a family of coolants with similar general properties. The choice of a specific molten salt for a specific application is determined by functional requirements and costs. Many salts have been examined. Table XXVI-3 shows the properties for several different liquid salts and traditional reactor coolants under typical conditions. Table XXVI-4 lists leading candidates for various nuclear liquid salt applications and their key physical properties. The remainder of this Appendix discusses the various salts and the constraints that limit the choice of salt.

Stability of liquid salts and materials of construction

Liquid salts and metals

A major challenge for all reactors is the development of materials of construction for components and heat exchangers that are compatible with the environment. For high temperature service, there are four issues:

- Strength over time (avoiding changes in metallic structure from various thermal mechanisms).
- Long term creep.
- Corrosion resistance to air.
- Corrosion resistance to molten fluoride salts.

TABLE XXVI-3. CHARACTERISTICS OF REACTOR COOLANTS [a]

Coolant	T_{melt} (°C)	T_{boil} (°C)	ρ (kg/m^3)	C_p (kJ/kg°C)	ρC_p (kJ/m^3°C)	K (W/m°C)	$v \cdot 10^6$ (m^2/s)
Li_2BeF_4 (Flibe)	459	1430	1940	2.34	4540	1.0	2.9
0.58NaF-0.42ZrF_4	500	1290	3140	1.17	3670	~1	0.53
Sodium	97.8	883	790	1.27	1000	62	0.25
Lead	328	1750	10 540	0.16	1700	16	0.13
Helium (7.5 MPa)			3.8	5.2	20	0.29	11.0
Water (7.5 MPa)	0	100	732	5.5	4040	0.56	0.13

[a] Parameters are as follows: ρ is density; C_p is specific heat; K is thermal conductivity; v is viscosity.

While only coolant metal compatibility will be discussed herein, this may not be the primary materials challenge for high temperature reactors. The experience with high temperature liquid salt corrosion test loops is that air corrosion has often been more of a limitation than salt corrosion, particularly above 800ºC. The other three materials issues are common challenges for all high temperature reactors. In liquid salt systems, Hastelloy-N has been code qualified for service to 750ºC. Because of limitations from long term, high temperature creep, this is near the upper long term temperature limit for Hastelloy-N. Although many other candidate alloys exist, these have not been qualified for nuclear service.

Fluoride salts are fluxing agents that rapidly dissolve protective layers of oxides and other materials. To avoid corrosion, liquid salt coolants must be chosen that are thermodynamically stable relative to the materials of construction of the reactor; that is, the materials of construction are chemically noble relative to the salts. This limits the choice to highly thermodynamically stable salts. Table XXVI-5 shows the primary candidate fluorides suitable for a liquid salt, along with their thermodynamic free energies of formation. The general rule to ensure that the materials of construction are compatible (noble) with respect to the salt is that the difference in the Gibbs free energy of formation between the salt and the container material should be >20 kCal/(mole ºC). The corrosion strategy is the same as that used in sodium cooled reactors where the materials of construction are noble relative to metallic sodium.

In high temperature systems, there are also temperature induced mechanisms for corrosion that are dependent upon large temperature differences in the heat transport system. The classic example is the transport of carbon in the form of various oxides in gas cooled systems, resulting in depletion of carbon in one part of the system and the deposition of carbon in another part. Parallel mechanisms have been seen in liquid salt test loops. In this system, the primary concern has been the selective chemical transport of chromium (an alloy constituent of many alloys) from hot to cold locations in the reactor system by uranium, where the equilibrium is temperature dependent:

$$2UF_4 + Cr = 2UF_3 + CrF_2$$

TABLE XXVI-4. PHYSICAL PROPERTIES OF SELECTED LIQUID SALTS [a, b]

Salt (mol%)	Form weight (g/mol)	Melting point[c] (°C)	Density (g/cm^3), T(°C)	Heat capacity at 700°C (cal/g°C)	Viscosity (centi-Poise), T(K)	Thermal conductivity (W/cm°C)
Alkali fluorides (IA): Non-toxic						
LiF-NaF-KF (46.5-11.5-42)	41.2	454	2.53–(7.3 10^{-4}) T	0.45	0.04exp(4170/T)	0.006-0.01
LiF-RbF (43-57)	70.7	475	3.30–(6.9 10^{-4}) T	0.284 (estimate)	0.021exp(4678/T)	~ 0.06
Alkali + Alkaline earth fluorides: (IA + IIA)						
$LiF-BeF_2$ (66-34)	33.1	458	2.28–(4.884 10^{-4}) T	0.57	0.116exp(3755/T)	0.011
$NaF-BeF_2$ (57-43)	44.1	360	2.27–(3.7 10^{-4}) T	0.52	0.034exp(5164/T)	~ 0.01
Alkali + ZrF_4: Non-toxic and low tritium yield						
$NaF-ZrF_4$ (50-50)	104.6	510	3.79 –(9.3 10^{-4}) T	0.28	0.071exp(4168/T)	~0.01 (estimate)
$NaF-KF-ZrF_4$ (10-48-42)	102.3	385	3.45–(8.9 10^{-4}) T (estimate)	0.26 (estimate)	0.061exp(3171/T) (estimate)	~0.01
$Li-NaF-ZrF_4$ (42-29-29)	71.56	460	3.37–(8.3 10^{-4}) T	0.35	0.0585exp(4647/T)	~0.01

TABLE XXVI-4. (continued)

Salt (mol%)	Form weight (g/mol)	Melting point[c] (°C)	Density (g/cm^3), T(°C)	Heat capacity at 700°C (Cal/g °C)	Viscosity (centi-Poise), T(K)	Thermal conductivity (W/cm °C)
Fluoroborates: Secondary salt candidates [d]						
$NaF-NaBF_4$ (8-92)	104.4	385	$2.252–(7.11\ 10^{-4})$ T	0.36	0.0877exp(2240/T)	~0.005
$KF-KBF_4$ (25-75)	120.48	460	$2.258–(8.02\ 10^{-4})$ T (estimate)	> 0.32	Similar to KBF_4	~0.005
$RbF–RbBF_4$ (31-69)	151.25	442	$2.494–(8.7\ 10^{-4})$ T (estimate)	-	-	-
$NaBF_4$	109.8	408	$2.263–(7.51\ 10^{-4})$ T	0.36	0.0832exp(2360/T) 0.0787exp(2406/T)	~0.005
KBF_4	125.9	570	$2.228–(8.15\ 10^{-4})$ T	0.32	0.0946exp(2280/T)	~0.005
RbF_4	172.27	582	$2.795–(10.4\ 10^{-4})$ T	-	-	~0.005

[a] Other mixtures of interest: $NaF-RbF-ZrF_4$ (8-50-42; melting point 400°C) and LiF-NaF-RbF (45-10-45; melting point 430°C).
[b] The early development of liquid-salt technology emphasized MSRs. As a consequence, much more work has been done on salts optimized for this application than for other applications.
[c] The use of four component salts may further depress the melting point.
[d] If low-cost methods for isotopic separation of boron are developed, these salts may be used as primary salts in the AHTR, a fast reactor, or a fusion machine. Boron-11 has a very low nuclear cross-section (0.05 barn).

Chromium is the least thermodynamically stable element among the materials of construction. This and other corrosion mechanisms resulted in the development of Hastelloy-N as a material of construction, which offers very good corrosion resistance in liquid salt systems.

TABLE XXVI-5 THERMODYNAMIC STABILITY OF COMPONENTS OF LIQUID SALTS AND STRUCTURAL MATERIALS

Constituent	Free energy of formation at 1000ºK (kCal/mol ºF)	Cation thermal capture cross-section (barns)
Majority constituents (>99.9 mol%)		
^7LiF	-125	0.033 (^7Li)
MgF_2	-113	0.063
NaF	-112	0.52
RbF	-112	0.70
KF	-109	2.1
BeF_2	-104	0.01
ZrF_4	-94	0.18
$^{11}BF_3$	~-95	0.05 (^{11}B)
$A1F_3$	-90	0.23
SiF_4	-	0.16
F	Not applicable	0.01
Structural metal constituents (Trace)		
CrF_2	-75.2	3.0
FeF_2	-66.5	2.5
NiF_2	-55.3	4.5
MoF_6	-50.9	2.5

In this context, avoiding corrosion in an MSR with metallic components is significantly more difficult than avoiding corrosion in clean salt coolant applications (heat transport loops, the AHTR, fast reactors, and fusion). In an MSR, the dissolved uranium in the fuel salt cannot be removed from the system; therefore, alloys such as Hastelloy-N that are resistant to such corrosion mechanisms must be developed. In clean salt applications, these types of corrosion mechanisms can be reduced or eliminated by (1) using purified salts that do not contain chemical species with multiple valance states that can assist in the transport of chromium and (2) operating under chemically reducing conditions. Under chemically reducing conditions, chromium fluoride has an extremely low solubility, which limits chromium transport. Higher temperature test loops indicate low corrosion rates with liquid fluoride salts when these conditions are met. The practical engineering conclusion is that the development and qualification of metallic materials for higher temperature clean liquid salt systems (>750°C) - such as heat transport loops, the AHTR, liquid salt cooled fast reactors, and fusion reactors - will be significantly easier than for a higher temperature MSR with its fuel salt. Liquid salt systems are similar to water and a helium coolant system in that corrosion control depends upon control of coolant chemistry.

Liquid fluoride salts and graphite

A large experience base has demonstrated the compatibility of liquid fluoride salts and graphite in radioactive and non-radioactive systems. The MSR programmes of the 1950s and 1960s investigated the compatibility of molten salts with graphite in chemical tests, loop tests, and reactors. In an MSR, the reactor core is made of bare graphite (the moderator) with the

liquid fuel salt flowing through channels in the graphite. In an MSR reactor, where the uranium and fission products are dissolved in the fuel salt, the fuel salt is dumped to storage tanks during shutdown. For safety and maintenance purposes, it is essential to know exactly where all the salt, fission products, and uranium are located. As a consequence, the interactions of salt and graphite were carefully investigated.

Post-irradiation examination from the MSRE showed no interactions (erosion or corrosion) between the salt and the graphite. The original machining marks were still clearly visible. Out of reactor tests were conducted to 1400ºC with no interactions between the salt and graphite. Experiments show the non-wetting behaviour of the fluoride salts of interest and demonstrate that liquid fluoride salts will not penetrate small cracks in the graphite. For the AHTR, this has the practical implication that the clean coolant will not contact the fuel micro spheres that are embedded in a carbon matrix.

Reactions of liquid salts with other fluids

Liquid fluoride salts do not react with helium or nitrogen but will react slowly with water. Experiments have been conducted to determine the impact of injecting liquid salt into water, such as might occur in some types of systems that have water. In the largest experiment, 230 kg of molten salt at 815°C was injected into a water bath over 45 to 50 seconds with the discharge pipe 1.5 m (5 ft) under the water level. No steam explosion occurred, only a negligible amount of steam reached the surface of the water, and no entrainment of water or steam into the air was observed.

Nuclear criteria for salt selection

Low nuclear absorption cross section

The neutron absorption cross-sections of any liquid salt for reactor applications must be low to avoid excessive parasitic capture of neutrons. For thermal and intermediate neutron spectrum reactors, this probably eliminates chloride salts with their higher nuclear cross sections, even if the high cross section ^{35}Cl is removed. Only fluoride salts are candidates. A wide variety of atoms have low cross sections; however, the realistic candidates are also restricted by the requirements of thermodynamic stability to ensure viable materials of construction for the container. Table XXVI-5 shows the primary salt options and their cross sections. If either lithium or boron is used as a salt component, isotopically separated lithium and boron are required to have a salt with a low absorption cross section.

Radiolysis of liquid fluoride salts

Unlike water, fluoride salts do not undergo radiolysis in radiation fields when liquid, and no fluorine will be generated. No radiolysis was detected in flowing loops of liquid salt operated in intense radiation fields of the Materials Testing Reactor. Liquid fluoroborate coolant salts (containing BF_3 + alkali fluorides) were also tested for their radiolysis response, and none was found. In fact, the reverse reaction—recombination—typically counteracts primary radiolysis events far below the melting point of the salt. For solid $2LiF-BeF_2$ salt in a radiation field, the temperature that inhibited a net radiolysis response was ~150°C. For other solid fluoride salts (ZrF_4), no radiolysis response was found above room temperature, and radiolysis at room temperature was not observed for $NaF-ZrF_4$ salts. This radiation stability is partly a consequence of the materials requirement that only very stable fluoride salts can be considered to ensure that the metals of construction are thermodynamically stable with respect to the salt.

Environmental and occupational considerations

Several environmental and occupational factors impact the choice of salt for specific applications:

- Waste management. The choice of salt can have major impacts on waste management. In addition to their relatively high neutron absorption cross sections, chloride salts create significant waste management challenges. In a reactor the high cross section ^{35}Cl is activated to ^{36}Cl, a long lived radio nuclide with a half life of 300 000 years. Most chlorides are soluble in water, which makes it more difficult to avoid long term release of ^{36}Cl from a repository to the environment. In contrast, fluorides have low neutron cross sections and do not activate to radionuclides that create major challenges in waste management systems and form many insoluble waste forms. If a chloride salt was used, isotopically separated ^{37}Cl would probably be required.
- Chemical toxicity. The toxicity of the molten fluoride coolant depends upon the specific salt and varies from the fluoride salts used in toothpaste for prevention of tooth decay to toxic materials.
- Neutron activation. Some salts produce tritium under radiation (lithium and beryllium). Other salts such as sodium produce gamma-emitting radionuclides. The choice of coolant impacts the need for tritium control systems and radiation shielding in the primary system.

Physical Properties

The chemical and nuclear criteria define the allowable elements for a liquid salt coolant. Physical property requirements are used to define the candidate salts for specific applications. The requirements include:

- A good coolant with coolant properties generally between those of water and sodium.
- A coolant freezing point as low as possible.
- Application specific requirements.

In all cases, binary or more complex fluoride salt mixtures are preferred because the melting points of fluoride salt mixtures are much lower than those for single component salts. For example, the molten salt Li_2BeF_4 has a melting point of 457ºC, whereas pure LiF has a melting point of 847ºC and pure BeF_2 has a melting point of 544ºC. Other traditional candidate salts include NaF-ZrF_4 (50 mol % NaF, 50 mol % ZrF_4), with a melting point of 510ºC, and NaF-RbF-ZrF_4 (8 mol % NaF, 50 mol % RbF, and 42 mol % ZrF_4), with a melting point of 400ºC. With some three component mixtures such as ^{7}LiF-BeF_2-NaF, and potentially with four component mixtures, it is possible to reduce melting points to ~350ºC. At operating conditions, the thermo physical properties of liquid salts are similar to those of water except for the very low vapour pressure.

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