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Current status and future development of modular high temperature gas cooled reactor technology

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FOREWORD

During the 1980s, the modular high temperature gas cooled reactor (HTGR) concept was developed, primarily in Germany and the United States of America. This concept utilized characteristics of HTGR technology to arrive at a design wherein safety issues were addressed through the inherent response characteristics of the system. These initial modular HTGR designs were primarily directed toward electricity generation using a steam turbine (Rankine cycle).

Substantial developments have occurred over the past decade in the modular HTGR programmes of Member States of the IAEA's International Working Group on Gas Cooled Reactors (IWG-GCR). The 1990s were witness to the initiation of plant designs that incorporate this advanced nuclear reactor coupled to a gas turbine power conversion system for the production of electricity (Brayton cycle). This design replaces the steam cycle components with fewer gas turbine cycle components, and with an attendant benefit of increasing net plant electrical efficiency from approximately 40% into the range of 45 to 50%. The resulting plant simplification and increased thermal efficiency provides the promise of competitive capital and O&M costs at relatively low unit ratings (100–300 MW(e)).

Significant programmatic changes are also taking place in the investigation of the modular HTGR as the high temperature heat source for industrial co-generation and non-electric applications to realize products including hydrogen and synthesis fuels as well as the production of electricity.

This report was developed by IAEA for the purpose of providing Member States with a detailed reference on the current status and future plans for utilization of the modular HTGR as an energy source for industrial applications and the generation of electricity. The international HTGR programmes described herein involve substantial international collaborative efforts of IWG-GCR Member States including technical personnel and research facilities.

This report was developed by H.L. Brey from materials obtained from Member States participating in the IWG-GCR. It has received an international review by experts in HTGR development from China, Japan, South Africa and the USA, as indicated in the Contributors to Drafting and Review section. The IAEA officers responsible for this publication were J. Kupitz and J.M. Kendall of the Division of Nuclear Power.

CONTENTS

Chapter 1

INTRODUCTION

1.1. NUCLEAR ENERGY FOR TOMORROW

The constructive utilization of energy is of paramount importance in the enhancement of society's standard of living. Worldwide, the consumption of primary energy is expected to increase by 2.1%/a[1] due to population increase and rapid economic growth in some regions of the world. Currently, 90% of the energy consumed originates from the burning of fossil fuels, with 30% of the total used as primary energy for electricity production [1-1]. Most of the remaining 70% is used either for transportation or converted into hot water, steam and heat. Nuclear energy is now being used to produce about 17% of the world's electricity.

The burning of fossil fuels to satisfy our ever increasing needs represent significant hazards to society and to the environment. Most of the world's fuel supply is of hydrogen combined with carbon (hydrocarbons). Burning of these hydrocarbons results in the liberation of carbon oxide gases with undesirable side effects: CO is toxic to life, CO_2 is currently labeled as one of the primary causes of the greenhouse effect. Although subsequent global warming has been recognized, there still exists a discrepancy between the worldwide trend of CO_2 emission and the global target of CO_2 reduction. By agreement of 154 countries at the 1992 Rio Earth Summit, sustainable development in the reduction of greenhouse gases was to be pursued with attainment of the 1990 level by the year 2000 [1-1]. Yet, this reduction has not occurred. The trend has actually been in the opposite direction of ever increasing CO_2 emissions.

World consumption of energy is not going to decrease in the foreseeable future. The ability to supply and utilize energy in a more environmentally friendly manner is a necessity in resolving the global warming issue. This can be addressed through a number of avenues including cleaner, more efficient production and burning of fossil fuels, significantly increasing the use of energy sources such as hydro, wind, solar and biomass and greater utilization of nuclear power in the world's fuel mix. In the majority of scenarios, nuclear power is an essential contributor to successfully addressing environmental issues

The focus of this report is on utilization of the modular high temperature gas cooled reactor (HTGR) to support the goal of meeting the energy demands of the future in an efficient, safe and more economic and environmentally acceptable manner than the present methods of energy production and utilization. The international status and planning associated with development of the HTGR for the production of electricity and utilization in achieving a wide range of process heat applications is examined herein as an advanced source of energy for the twenty-first century.

1.2. THE MODULAR HTGR

The modular HTGR is expected to achieve the goals of being a safe, efficient, environmentally acceptable and economic high temperature energy source for the generation of electricity and for industrial process heat applications such as the production of hydrogen. All HTGRs incorporate graphite moderated, helium cooled cores with ceramic coated fuel

[1] EIA Reference Projection for 1997–2000 = 2%/a.

particles capable of handling temperatures of 1600°C. The most current HTGR designs are capable of continuous operation at average core helium outlet temperatures between 900° and 950°C.

Among the HTGR advances currently under investigation by Member States of the IAEA's International Working Group on Gas Cooled Reactors (IWG-GCR) is the closed cycle gas turbine concept which (through the Brayton Cycle configuration) exhibits the capability of achieving net plant efficiencies in the range of 47% at 850°C core outlet/gas turbine inlet temperature. In the area of high temperature process heat, the HTGR is being considered as the energy source for applications including steam and CO_2 reforming of methane for the production of hydrogen as a fuel and/or its subsequent synthesis to other fuels such as methanol. Utilizing the HTGR as the energy source eliminates the need to burn fossil fuels in order to achieve the heat required for these industrial processes to occur.

1.3. THIS REPORT

This report includes an examination of the international activities with regard to the development of the modular HTGR coupled to a gas turbine. The most significant of these gas turbine programmes include the pebble bed modular reactor (PBMR) being designed by ESKOM of South Africa and British Nuclear Fuels plc. (BNFL) of the United Kingdom, and the gas turbine-modular helium reactor (GT-MHR) by a consortium of General Atomics of the United States of America, MINATOM of the Russian Federation, Framatome of France and Fuji Electric of Japan. Details of the design, economics and plans for these plants are provided in Chapters 3 and 4, respectively.

Test reactors to evaluate the safety and general performance of the HTGR and to support research and development activities including electricity generation via the gas turbine and validation of high temperature process heat applications are being commissioned in Japan and China. Construction of the high temperature engineering test reactor (HTTR) by the Japan Atomic Energy Research Institute (JAERI) at its Oarai Research Establishment has been completed with the plant currently in the low power physics testing phase of commissioning. Construction of the high temperature reactor (HTR-10) by the Institute of Nuclear Energy Technology (INET) in Beijing, China, is nearly complete with initial criticality expected in 2000. Chapter 5 provides a discussion of purpose, status and testing programmes for these two plants.

In addition to the activities related to the above mentioned plants, Member States of the IWGGCR continue to support research associated with HTGR safety and performance as well as development of alternative designs for commercial applications. These activities are being addressed by national energy institutes and, in some projects, private industry, within China, France, Germany, Indonesia, Japan, the Netherlands, the Russian Federation, South Africa, United Kingdom and the USA. Chapter 6 includes details associated with these R&D programmes. Also, support of specific HTGR related research projects is included in the European Union's Fifth Framework Program beginning in 2000. Further opportunities and capabilities of the HTGR in the development of co-generation and non-electric applications are presented in Chapter 7.

Spent fuel disposal and decommissioning are key issues that are significantly influencing the future of nuclear power. Chapter 8 addresses the anticipated manner of handling these areas within the PBMR and GT-MHR. Also addressed are the activities associated with spent fuel disposal and decommissioning of HTGRs previously shut down.

The development and commissioning of any new nuclear plant concept is subject to risks and challenges to its commercialization. This is also evident in the closed cycle gas turbine, particularly with regard to the design and development of the power conversion system (PCS). The GT-MHR and the PBMR (as well as many other designs under consideration) incorporate state-of-the-art components in their PCS that must operate safely and efficiently for this concept to succeed. These components include magnetic bearings on the rotating machines, large compact plate-fin recuperator modules and seals between PCS components that have size, orientation or environmental operating characteristics yet to be fully demonstrated and proven. These challenges to the commercialization of the GT-MHR and PBMR are discussed in Chapter 9.

The IAEA is advised on its activities in development and application of gas cooled reactors by the IWGGCR which is a committee of leaders in national programmes in this technology. The IWGGCR meets periodically to serve as a global forum for information exchange and progress reports on the national programmes, to identify areas of collaboration and to advise the IAEA on its programme [1-2]. Countries with representation in the IWGGCR include Austria, China, France, Germany, Indonesia, Italy, Japan, the Netherlands, Poland, the Russian Federation, South Africa, Switzerland, the United Kingdom and the USA. Representation from international organizations includes the European Commission and the OECD-NEA. Activities of the IAEA in support of HTGR technology development are presented in Chapter 10.

REFERENCES TO CHAPTER 1

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Chapter 2

HTGR DEVELOPMENT AND GENERAL FEATURES

2.1. BACKGROUND

Commercial experience with gas-cooled power reactors (GCRs) began in 1956, with the generation of electricity from the Calder Hall plant in the United Kingdom. The commitment by the UK to GCRs for electricity production subsequently extended to 26 Magnox and 14 advanced gas-cooled reactors. Early commercial GCR experience also included plants in France, Japan and Spain.

Development of the HTGR began in the 1950s to improve upon GCR performance. HTGRs utilize ceramic fuel particles surrounded by coatings and dispersed in a graphite matrix, along with a graphite moderator. Either prismatic type graphite moderator blocks (block type reactor) or spherical fuel elements (pebble bed type reactor) are employed. Helium is used as the coolant to permit an increase in the operating temperature, and flows through coolant holes in the block type elements, or through the interstices present in the pebble bed core. HTGRs can operate at very high core outlet coolant temperatures because of the use of an all ceramic core [2-1].

2.1.1. Early HTGR plants

2.1.1.1. Early steel vessel HTGR prototype plants

The early HTGR plant designs (Dragon, AVR and Peach Bottom) utilized steel primary system vessels. The general performance of these plants was very good, and they provided an excellent basis for further development of the HTGR.

The Dragon Reactor Experiment in the UK first operated at power in July 1965, and reached its full capability of 20 MW(t) by April 1966. Initiated in April 1959 as an international project of the European Organisation for Economic Co-operation and Development, the primary objective of this plant was to demonstrate the feasibility of the HTGR and to launch the development of a technology which had already begun at a low level in various national laboratories. The reactor incorporated graphite fuel elements with high enriched uranium-carbide coated fuel particles. The core exit and inlet helium temperatures were 750°C and 350°C, respectively. The plant operated for long periods at full power and demonstrated successful performance of many components [2-4]. Although it did not produce electric power, it served as a productive research tool for the development of helium gas cooled reactors and advanced fuel particle coatings.

In Germany, the 15 MW(e), Arbeitsgemeinshaft Versuchsreaktor (AVR), a pebble bed type HTGR began operation in 1967. The AVR had a steel containment vessel and used particle fueled, graphite spheres 6 cm in diameter that traveled downward through the core. Although the AVR initially included a core outlet temperature of 850°C, this was subsequently raised to 950°C without decrease in plant performance. The AVR was the main fuel development tool for the pebble bed concept, and it and supplementary laboratory fuel testing became the major support of Germany’s position that an LWR-type containment barrier was not needed for future HTGRs [2-2]. The operation of the AVR continued until December 1988. In completing 21 years of service, the plant had accumulated more than

122,000 hours of operation with a 66.4% overall availability and had generated 1.67 billion kW•h of electricity [2-1].

Although several GCRs were built in the USA during the 1950s and 1960s, mainly for propulsion purposes, the helium cooled, 40 MW(e) Peach Bottom (No.1) HTGR provided the major background for the continued commercialization of HTGR technology in the USA. Operated by the Philadelphia Electric Company from 1967 to 1974, the plant achieved a gross capacity factor of 74% and an overall availability of 88% (exclusive of planned shutdown for R&D programmes). This HTGR introduced coated particle fuel and initiated the development of licensing criteria for this type of GCR [2-3]. Two reactor cores were utilized in the operation of Peach Bottom 1. The fuel particles of core 1 were coated with a single layer of anisotropic carbon to prevent hydrolysis of the carbide fuel kernels. Fast neutron induced dimensional changes resulted in fracture and distortion of the coatings, which eventually resulted in 90 cracked fuel element sleeves out of 804. Plant operation, however, was not impaired, and primary circuit activity reached a maximum of 270 Ci, well below the design activity level of 4225 Ci. core 1 accumulated 452 equivalent full power days before it was replaced with core 2. core 2, which provided buffer isotropic pyrolytic carbon (BISO) coatings on the fuel particles, operated with no fuel failures for its full design life of 897 equivalent full power days. The primary circuit activity averaged only .5 Ci during this entire period. Throughout the operation of Peach bottom 1, excellent agreement was found between the predicted and actual plant design characteristics, verifying the methods used and providing a reference data base for application to larger HTGR plants [2-4].

2.1.1.2. PCRV demonstration power plants

The follow-on HTGR plants, Fort St. Vrain (FSV) and the thorium high temperature reactor (THTR-300), both featured primary systems enclosed in prestressed concrete reactor vessels (PCRVs). These plants came on line at a time when the commercial industry was in decline, limiting the resources available to resolve issues associated with the commissioning and operation of new power plant designs.

The FSV HTGR was operated by Public Service Company of Colorado as part of the United States Atomic Energy Commission’s Advanced Reactor Demonstration Program. This plant was designed with several advanced features including: a) a prestressed concrete reactor vessel (PCRV) containing the entire primary coolant system, b) a core of hexagonal, graphite block fuel elements and reflectors with fuel in the form of ceramic coated (TRISO) particles, c) once-through steam generator modules producing 538°C superheated main and reheat steam, and d) steam-turbine-driven axial helium circulators [2-4]. Although ~5.5 billion kW•h of electricity was generated at FSV, the plant operated at low availability primarily because of excessive downtime due to problems with the water-lubricated bearings of the helium circulators. In spite of this low availability, the plant was a valuable technology test-bed, successfully demonstrating the performance of several major systems, including the reactor core with TRISO coated fuel particles in hexagonal graphite blocks, reactor internals, steam generators, fuel handling and helium purification [2-3].

The THTR-300 power plant was sponsored by Germany (FRG) and Nordrhein Westfalen (NRW). Construction of this 300 MW(e) plant began in 1971, but primarily due to increasing licensing requirements, the plant was not completed until 1984. This pebble bed reactor plant was connected to the electrical grid of the utility Hochtemperatur-Kernkraftwerk GmbH (HKG) in November 1985. In August 1989, the decision was made for the permanent shutdown of the THTR-300. This action was not due to technical difficulties associated with the plant, but was the result of an application by HKG for early decommissioning based on a projected short fall in funding and contractual changes in the allocation of decommissioning costs between the FRG, NRW and HKG that would take effect upon the termination of the demonstration phase in 1991. Operation of the THTR-300 was successful in validating the safety characteristics and control response of the pebble bed reactor, primary system thermodynamics and the good fission product retention of the fuel elements.

2.1.1.3. Other HTGR plant designs

Continued interest in development of larger steam cycle HTGR plants included the German HTR-500, the Russian VG-400 and the US HTGR-steam cycle (SC) plant. The HTR-500 made considerable use of the technology development for the HTGR-300, with simplifications and optimizations based on practical experiences with the THTR-300 [2-1]. This plant featured a simple design with the primary system components located within a single cavity PCRV, and included a pebble bed reactor power level of 1390 MW(t) to produce 550 MW(e) of electricity.

The Russian VG-400 design included a pebble bed reactor with a final power level of 1060 MW(t) for the co-generation applications of electricity production and had an intended final helium core outlet temperature design of 950°C.

In the USA, the focus in the early 1970s was on HTGR-SC designs of 770 to 1160 MW(e). Contracts to General Atomics from US utilities included 10 plants that did not materialize due to the 1973 oil crisis and corresponding economic setback and collapse of the US nuclear power market of 1975. These plants incorporated cores of hexagonal graphite blocks with TRISO coated fuel particles similar to FSV.

2.1.2. Modular HTGR plant development

The 80 MW(e) HTR module (HTR-MODUL) concept, developed by Siemens/Interatom, was the first small, modular type HTGR concept to be proposed. Although initially developed in the early 1980s for industrial process heat applications, the passive safety features of the side-by-side concept, coupled with the other attractive characteristics of the modular concept, soon led to the proposed electricity generation application. Work on a generic, site independent safety assessment of the HTR module was initiated with the filing, in 1987, of the safety analysis report in the State of Lower Saxony. The HTR module safety concept is characterized by comprehensive protection of the environment by passive system characteristics even under extreme, hypothetical accident conditions. The safety features of the HTR module were based on the design condition that, even in the case of failure of all active cooling systems and complete loss of coolant, the fuel element temperatures would remain within limits such that there is virtually no release of radioactive fission products from the fuel elements. Such a condition guaranteed that the modular HTR power plant would not cause any hazard to the environment either during normal operation or in the case of an accident.

According to these principles, the most important design features of the HTR module plant were as follows:

— The use of spherical fuel elements with TRISO coating, which are capable of retaining all radiologically relevant fission products up to fuel element temperatures of approximately 1600°C.

- The reactor core was designed such that a maximum fuel element temperature of 1600°C is not exceeded during any accident.
- Active core cooling was not necessary for decay heat removal during accidents. It is sufficient to discharge the decay heat by means of passive heat transport mechanisms (heat conduction, radiation, natural convection) to simple surface coolers. The surface coolers consist of a water cooled system installed outside the reactor pressure vessel in the primary cell.
- Reactor shutdown was effected solely by absorber elements, which can drop freely into boreholes in the reflector, limiting the cylindrical core diameter to ~3 m.
- The uranium content of the fuel elements amounting to 7 g of uranium was selected such that water ingress into the primary circuit as the result of an accident will cause a lower reactivity increase than the accidental withdrawal of all reflector rods.
- The faulty withdrawal of all reflector rods as a covering accident was controlled by simply switching off the primary circuit blower, whereby the permissible fuel element temperature of 1600°C is not exceeded.
- Graphite, which has been successfully tried and tested in gas reactors, was used in core areas with high temperatures (fuel elements, core internals). Temperature incurred failure of this material is impossible at the maximum occurring temperature of 1600°C.
- The single phase noble gas, helium, which is neutral from a chemical and neutron physical viewpoint, was used as coolant.
- Because of the high activity retention in the fuel elements and the accident response characteristics of the design, a pressure tight reactor building was not appropriate. The reactor building was accessible to repair work at any time after accidents as a result of the low activity release.
- The reactor core and the steam generator were installed in separate steel pressure vessels in such a way that there was no danger of component overheating in the case of failure of the primary circuit cooling. This arrangement also increases the accessibility of the components to maintenance and repair.

The small core diameter stems from the requirement for reactor shutdown from all operating conditions using only free falling control rods in reflector borings. The requirement to keep the maximum fuel element temperature for all possible accidents inherently below 1600°C, a temperature at which all radioactive fission products are contained within the fuel elements, leads directly to a mean power density of 3 MW/m^3. In order to gain as much power as possible from the core, the core height was chosen as large as possible [2-1].

With the objective that nuclear power plants utilizing small HTGRs can provide economic, environmentally favorable and reliable electricity and heat for community and industrial purposes, Brown, Boveri und Cie and Hochtemperatur-Reacktorbau initiated the design of the HTR-100 pebble bed plant. This design featured a 285 MW(t) HTGR with a net electrical output of 100 MW on the basis of the AVR concept and utilized the advanced technologies of THTR-300 components and systems. The primary system included the reactor, steam generator and helium circulator in a single, vertical steel pressure vessel. The reactor core included 317 500 spherical elements of TRISO type particles and a power density of 4.2 MW/m^3. In the equilibrium cycle, 55% of the spherical elements were fuel, with the remainder being graphite. The basic design for this plant incorporated two HTR-100 units with overall capability of producing 170 to 500 t/h of industrial steam at 270°C and 16 bar, with 100 to 175 MW gross electric output [2-8].

In the Russian Federation, the conceptual design of the modular plant (VGM) that has evolved was very similar to the side-by-side HTR module. A notable difference is the incorporation of both an intermediate heat exchanger (for process heat experiments) and a steam generator in the main heat transport system. The stated intent was to operate initially with an outlet temperature of 750°C by mixing an adjustable core bypass helium flow with the higher temperature core outlet helium flow and generate steam only. Later, the mixed outlet temperature would be adjusted to 950°C, by changing the core bypass flow, to generate high temperature helium for process heat used in the intermediate heat exchanger followed by steam generation [2-1].

In 1983, the US organization representing utility/user interests in the HTGR programme, the Gas Cooled Reactor Associates, conducted a survey to determine the utility nuclear generation preference for the future. This survey resulted in a strong interest for smaller generation increments. This was an important input leading to the evaluation and subsequent selection of the modular HTGR in 1985 [2-3]. Following a detailed evaluation in the spring and summer of 1985, a side-by-side concept similar in configuration to a German module design was chosen as the reference concept for further design and development by the US program. The basic module was designed to deliver superheated steam at 17.3 MPa and 538°C. An initial module power level of 250 MW(t) was selected, but subsequent detailed safety analyses showed that this power level could be increased with the hexagonal graphite block core design without compromising margins. Adopting an annular core allowed the power level to be increased initially to 350 MW(t). Other reactor design changes and analysis refinements subsequently allowed the power to be increased to 450 MW(t), while maintaining adequate margins to component and safety limits [2-5].

Central to the Modular HTGR (MHTGR) passive safety approach was the annular reactor core of prismatic fuel elements within a steel reactor vessel. A low-enriched uranium, once-through fuel cycle was used. For a standard steam cycle MHTGR plant, the steam output from each of the four modules was connected to an individual turbine generator. The four module plant consists of two separate areas, the nuclear island and the energy conversion area.

A preliminary safety information document based on the 350 MW(t) steam cycle MHTGR design was submitted to the US Nuclear Regulatory Commission (NRC) in autumn 1986. Review meetings between the NRC staff and the program participants were initiated in autumn 1986 and continued through 1987. NRC conclusions from the review were documented in a safety evaluation report, a draft of which was completed in June 1988, and reviewed by the Advisory Committee on Reactor Safeguards (ACRS). An ACRS letter on the safety evaluation report, issued in October 1988, generally agreed with the NRC staff position. A draft version of the safety evaluation report was issued in 1989. This draft was generally favorable toward the MHTGR concept on major items including containment approach and emergency planning requirements. However, formal issue of the safety evaluation report was delayed indefinitely due to NRC questions regarding design philosophy differences between the MHTGR design reviewed and the new production reactor variant of the MHTGR (for the production of tritium), and then due to the interruption of the US Department of Energy HTGR programme in 1994.

It was the development of the above mentioned modular HTGR steam plants that provided the key emphasis to initiation of the HTGR coupled to a gas turbine power conversion system. With few exceptions, it is the German HTR-MODUL and the HTR-100 reactor designs that are being utilized by ESKOM as the base for the PBMR. The MHTGR, with its annular core arrangement within a steel vessel, forms the basis for the GT-MHR reactor design.

2.2. GENERAL FEATURES

2.2.1. Safety

The overall good safety characteristics of all HTGRs are due to: the high heat capacity of the graphite core; the high temperature capability of the core components; the chemical stability and inertness of the fuel, coolant, and moderator; the high retention of fission products by the fuel coatings, the single phase characteristics of helium coolant; and the inherent negative temperature coefficient of reactivity of the core.

The modular HTGR adds 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 characteristic has been achieved by deliberately decreasing the core power level and configuring the reactor so that natural heat removal processes can limit fuel temperatures to levels at which the release of fission products from the reactor system to the environment is insignificant for postulated accidents. Even for extreme accidents having very low probabilities of occurrence, the cumulative fission product release at the site boundary is estimated to be below those acceptable under defined protective action guidelines.

The most fundamental characteristic of the MHTGR that separated it from previous reactor designs was the unique safety philosophy embodied in the design [2-6]. First, the philosophy requires that control of radionuclides be accomplished with minimal reliance on active systems or operator actions; the approach to safety is to rely primarily on the natural processes of thermal radiation, conduction, and convection and on the integrity of the passive design features. Arguments need not center on an assessment of the reliability of pumps, valves, and their associated services or on the probability of an operator taking various actions, given the associated uncertainties involved in such assessments.

Second, the philosophy requires control of releases by the retention of radionuclides primarily within the coated fuel particle rather than reliance on secondary barriers (such as the primary coolant boundary or the reactor building). Thus, ensuring that the safety criteria are met is the same as ensuring that the retention capability of the coated fuel particles is not compromised.

The assessment of the capability of the MHTGR to control accidental radioactivity releases shows that the doses are a small fraction of the US10CFR100 requirements even for the bounding analyses which consider only the systems, structures and components that require neither operator action nor other than battery power. In fact, the exposures are so low that the protective action guidelines would require no evacuation or sheltering plans for the public as specified in the utility/user requirements. The evaluation confirms that accident dose criteria can be met with a containment system that places primary emphasis on fission product retention within the fuel barriers.

Safety associated with the PBMR and the GT-MHR closed cycle gas turbine plants is addressed in Chapter 3 and Chapter 4, respectively.

2.2.2. Coated fuel particles

The fuel designs for the HTGR, even though they are being developed in various countries, use the same generic coated fuel particles for the retention of fission products. The coated fuel particle is a microsphere of about 0.8 mm diameter consisting of a fuel kernel (either fuel oxide, carbide or a mixture of oxide and carbide) surrounded by a low density carbonaceous layer (the buffer), followed successively by layers of pyrolytic carbon (PyC), silicon carbide (SiC), and PyC. The resulting microsphere is termed a TRISO coated fuel particle. In Germany, the TRISO particles are surrounded by an 'overcoat' of carbonaceous material, and the overcoated particles are pressed together into a spherical shape resulting in a 'pebble'. The USA employs similar TRISO particles and places them into a fuel rod die and injects carbonaceous material into the interstices to form a fuel 'rod'. Japan employs overcoated fuel particles similar to those of Germany, but presses them into an annular rod which is placed inside a cylindrical graphite sleeve. Figure 2.1 (a-c) illustrates the fuel element designs of Germany, the USA/the Russian Federation and Japan.

Information gathered from extensive irradiation and heating tests has provided a fundamental understanding of the performance capabilities of present day TRISO fuel and has allowed advances in the modeling of fuel performance. Several performance limiting mechanisms for particle failure (i.e. the significant release of fission products from TRISO particles) have been identified; these as well as means for controlling them are given below.

(a) *Pressure induced failure.* The buildup of pressure inside the particle coat- ings due to the generation of fission gases results in a tensile stress in the SiC load bearing layer. If this stress exceeds the strength of the layer, the result is a simultaneous failure of all the coating layers. Pressure induced failure is controlled by product specifications on the distribution of fuel kernel diameters, coating thickness and on the 'void' volume provided in the buffer layer.

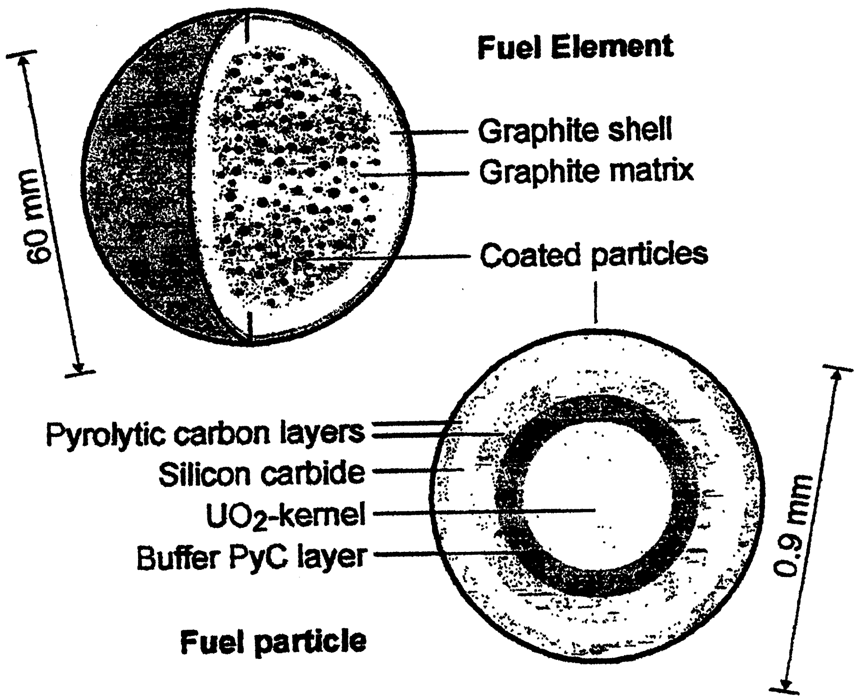


FIG. 2.1 (a). German (pebble) HTGR fuel [2-10].

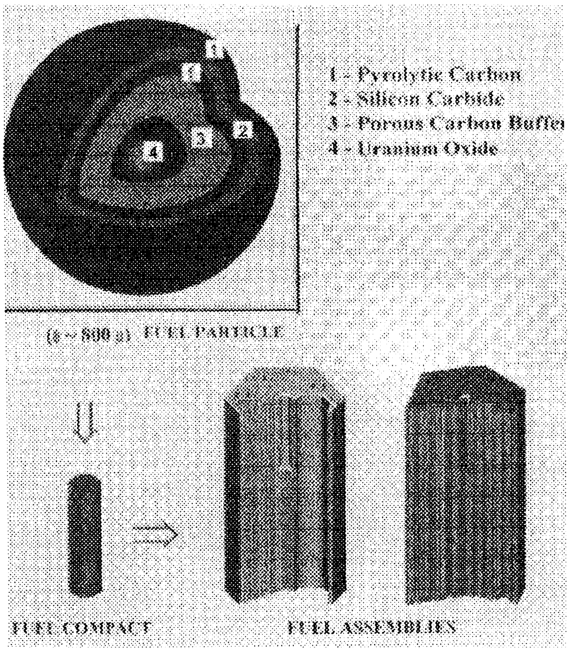
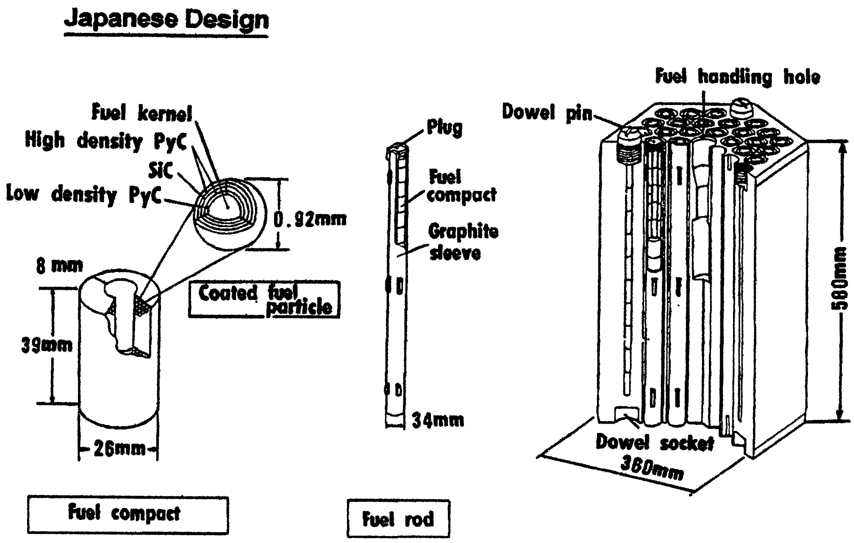


Fig. 2.1 (b). US/Russian (block) HTGR fuel.



c.) Japanese HTGR fuel [10]

Fig. 2.1 (c). Japanese (pin-in-block) HTGR fuel [2-10].

(b) *Kernel-coating interactions.* Carbon transport in the presence of a thermal gradient may result in kernel-SiC layer contact and coating degradation. Design selections of core operating temperatures and associated temperature gradients limit the kemel-coating interactions to acceptable values.

(c) *SiC dissociation and increase in SiC porosity at high temperatures.* Thermal damage and/or increased porosity of SiC may occur at very high temperatures, resulting in increased fission product release after extended times. Thermal dissociation can lead to complete SiC degradation at 2200°C. To mitigate these effects, reactor design conditions can be selected to keep fuel temperatures below 1600–1700°C under severe accident conditions, at which temperatures the above mechanisms have a negligible effect on fuel coating failures.

(d) *SiC-fission product interactions.* Fission products that are released from the kernel and reach the SiC layer may interact chemically with the SiC and result in its degradation and, ultimately, in coating failure. Testing under severe conditions with high thermal gradients specifically intended to concentrate fission products at the SiC surface has shown that palladium (a fission product) interacts with and degrades the SiC. To mitigate these effects, reactor design parameter values are selected to limit core thermal gradients and maximum temperature to values such that SiC-fission product interactions are negligible.

(e) *Irradiation effects on coating integrity.* If binding occurs between the fuel matrix material and the outer coating of the fuel particle, irradiation stresses can lead to coating failures due to the relatively high shrinkage of matrix material and the 'tearing away' of portions of coating layers during irradiation. Such types of coating failures can be avoided by designing the fuel element so that only a weak physical coupling exists between the outer coating of the fuel particle and the surrounding matrix material, under which conditions the above mechanism is insignificant. Also, dimensional changes at the PyC layers under irradiation has led to fracturing of layers, delamination and damage to the SiC layer. This is addressed by controlling the coating conditions.

Quantitative performance models considering the above mechanisms for fuel failure (except for item (e), which is avoided through proper choice of the fuel fabrication process), have been developed on the basis of data from irradiation tests, heating tests, and material properties studies; these models are utilized in the design of reactor cores so as to result in a high degree of reactor safety. The validity of these models can best be determined by comparing the calculated results with results from pertinent experimental measurements. Fuel performance data from normal and off- normal testing conditions are given below.

For normal reactor conditions, irradiation testing of high quality fuel has been performed in material testing (or research) reactors, as well as operating HTGRs. Parameters such as heavy metal burn-up, operating temperature, and fast neutron fluence are varied to assess fuel performance. Continuous monitoring of released fission gases during irradiation tests gives a direct indication of the integrity of fuel coatings. The deconsolidation of fuel elements and subsequent individual particle gamma spectroscopy allow the determination of the fission product retention (fuel performance) variabilities within a large population of particles [2-1].

In the early 1980s, both the USA and German fuel programmes embarked on improving development based on the TRISO coating design. The German programme focused on the UO_2 kernel, whereas the US programme concentrated on the UCO kernel.

Investigation into the UCO kernel by the USA was initiated in an attempt to achieve high burnup of low enriched fuel. Revisions in the US regulatory requirements mandated a change in the enrichment level to a maximum of 20%. Studies indicated that fuel economics improved with an increase in enrichment. However, systematic and technical developmental problems within the US programme prevented the achievement of the UCO kernel to reach the intended enrichment/burnup goals.

Within Germany, a systematic incremental improvement programme on UO_2 kernel development continued in the 1980s with the top enrichment level at 10%. Nine irradiation tests comprising over 200 000 coated fuel particles were performed; no coating failures were observed as a result of in-reactor gas release. Statistically, the above corresponds to a median probability (50% confidence level) that the fuel coating failure fraction in a much larger batch of coated particles is less than 3×10^{-6}, and to a 95% confidence level that the coating failure fraction is less than 1×10^{-5}. Calculated failure fractions based on fuel performance models have been obtained for MHTGR normal operating conditions, and are in the range of 2 to 5×10^{-5} [2-1]. German fuel utilizing the UO_2 kernel now forms the basis for fuel development of all HTGR programmes.

Currently, only Japan has the fuel fabrication facility to provide limited quantities of coated particles on a commercial basis. The HTTR fuel from this facility consists of TRISO coated UO_2 kernels, 600 μm in diameter. The quality of the first loading of HTTR fuel exhibited an average bare uranium fraction and SiC defective fraction of the fuel compacts at 2×10^{-6} and 8×10^{-5}, respectively [2-9].

Fuel testing under off-normal conditions has provided fuel performance information as a function of fuel temperature, up to 2500°C. Radiologically, the most significant nuclide released as a gas from the fuel is iodine. Fractional krypton release is easier to measure than fractional iodine release and gives a conservative measure of fractional iodine release. The German programme measured krypton fission gas releases from high quality fuel during temperature ramp tests; Fig. 2.2 shows the results obtained. Each test involved heating an AVR fuel pebble to as high as 2500°C, with each pebble containing 10 000 to 17 000 particles. As shown, fuel temperatures of about 2200°C were required before fission gas release fractions approached the equivalent of one particle coating failure.

Not all fission products are gases, and so Fig. 2.2 cannot be applied to fission products in general. From irradiation and annealing experiments, the sequence of fission product release from HTGR fuel as a function of temperature is generally observed to be, in order: silver, caesium, strontium, gases (e.g. krypton), cerium and ruthenium/zirconium. Specific US results from a temperature ramp test similar to Fig. 2.2 tests, but based on testing fuel particles by themselves, are shown in Fig. 2.3. The fission gas release results (i.e. krypton) are in good agreement with Fig. 2.2 results for AVR fuel, which illustrates that the fission gases are being retained within the coated fuel particles to very high temperatures (~2200°C). Figure 2.3 also shows that silver isotope, Ag110m, is not retained well at high temperatures and diffuses through SiC and pyrocarbon [2-1]. The amount and level of influence this isotope has on the metallurgical and maintenance considerations for the power conversion system components on the closed cycle gas turbine HTGR plants over the long period of operation is not yet fully determined. Also, if operation is allowed to exceed a fuel temperature of 1250°C. over an extended period of time, SiC coating thickness deterioration will occur due to palladium attack [2-7].

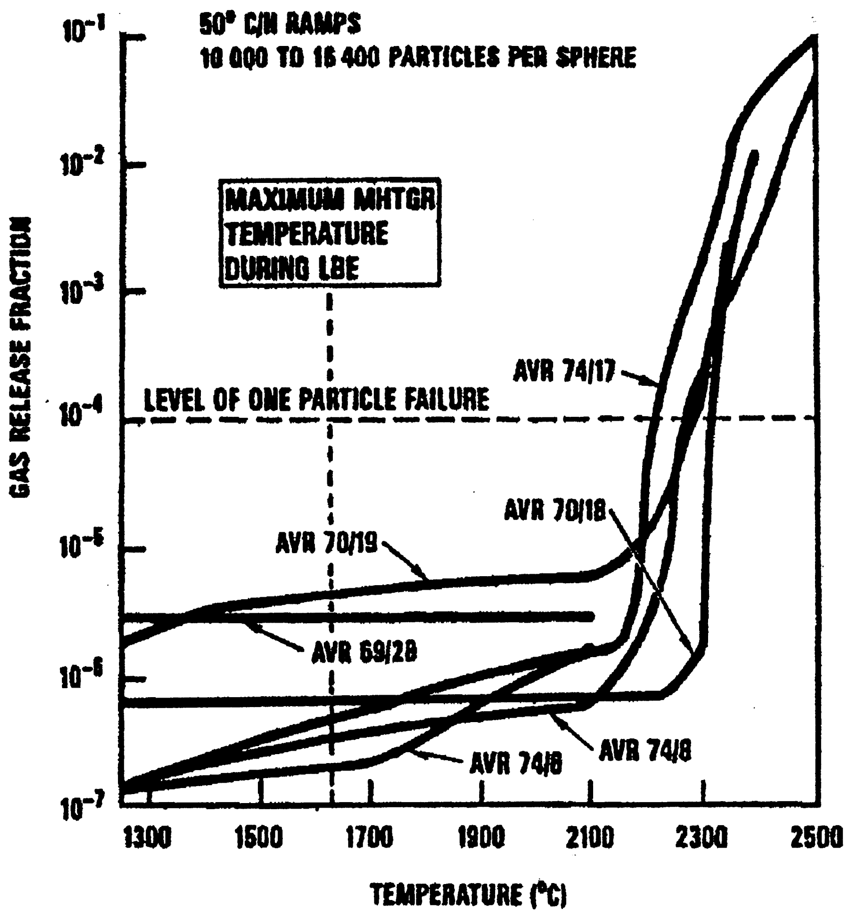


FIG. 2.2. Fission gas release results accompanying heat-up of irradiated AVR fuel pebbles in temperature ramp tests of 50°C/h, up to 2500°C.

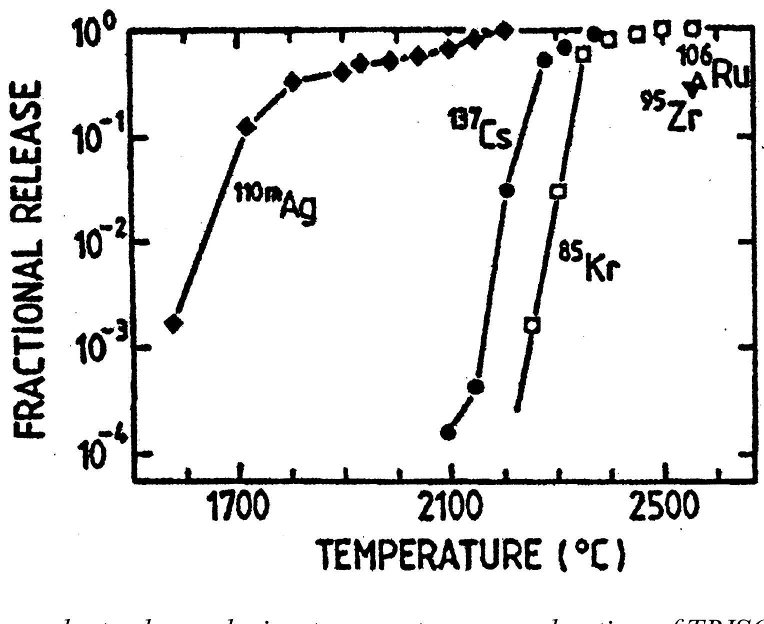


FIG. 2.3. Fission product release during temperature ramp heating of TRISO particles.

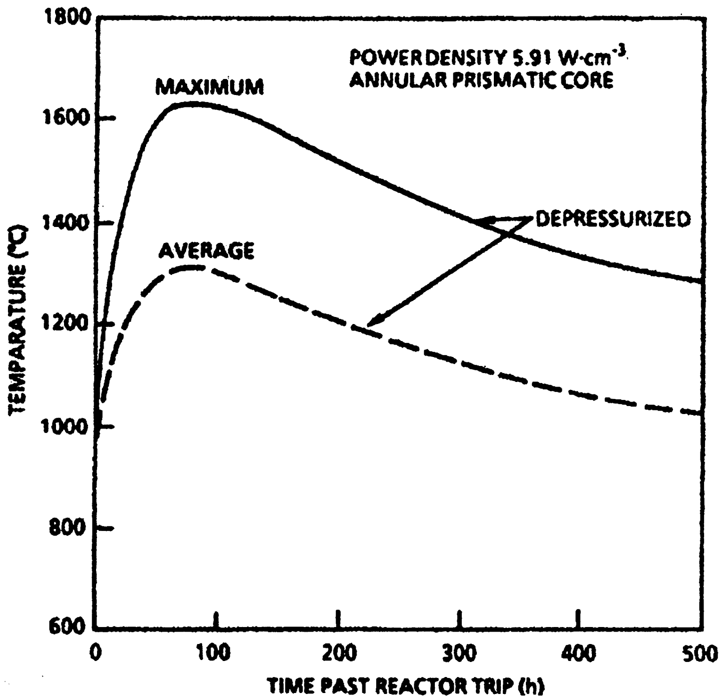


FIG. 2.4. Typical temperature transient after a loss of coolant in a high performance MHTGR, with the passive heat removal system operational.

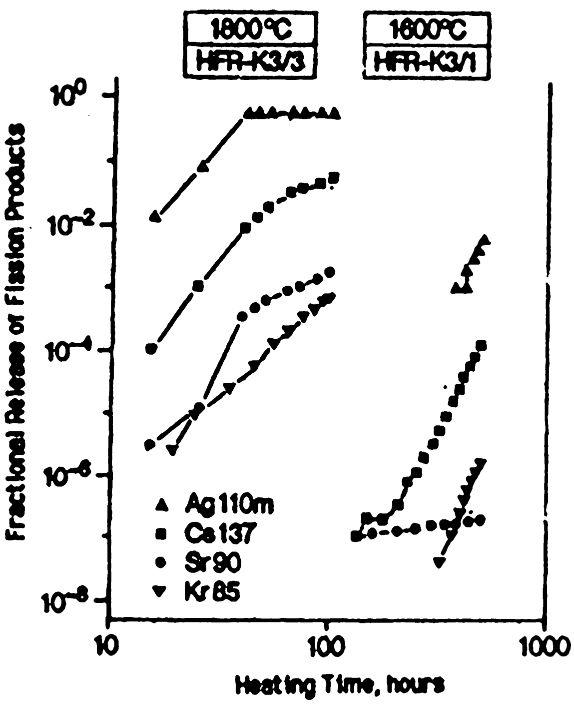


FIG. 2.5. Fission product release from typical irradiated HTGR fuel when exposed to constant temperatures for various times at 1600°C and 1800°C.

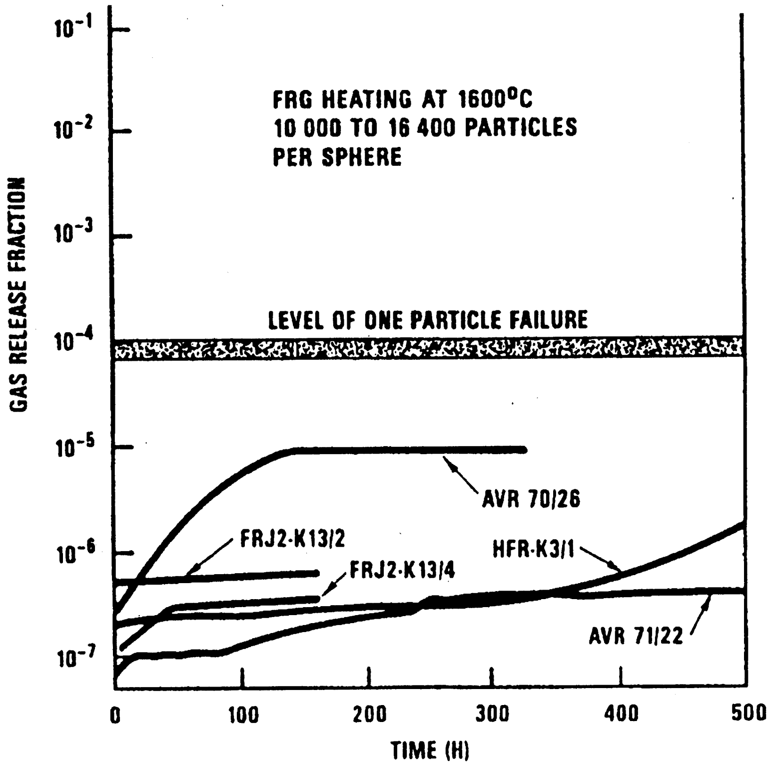


FIG. 2.6. Fission gas release as a function of time at 1600°C for pebble bed reactor fuel.

The maximum fuel temperatures (including uncertainty estimates) in a typical MHTGR under a loss of coolant accident are illustrated in Fig. 2.4. As shown, the peak temperature of about 1600°C occurs at about 50 h after the start of the event and temperatures higher than 1500°C exist for about 200 h. Consistent with the above, it is reasonable that fuel performance tests expose fuel to high temperatures (about 1600°C) for 200 h to determine fission product retention in MHTGRs under reactor accident conditions. In Germany, such measurements were made on pebble fuels; Fig. 2.5 provides fuel performance information for temperature exposures of up to 500 h at 1600°C, and up to 100 h at 1800°C.

The following conclusions can be drawn:

(a) Fission product releases are very low after 299 h at 1600°C.

(b) Increasing the exposure temperature to 1800°C has a very substantial effect on reducing the fuel performance.

(c) Increasing the exposure time at fixed temperature has a very significant effect on increasing the fission product release.

(d) Fission gas retention is much more effective than that of metallic fission products such as silver and caesium for times of hundreds of hours at 1600°C, and up to one hundred hours at 1800°C.

Corollaries to the above conclusions are: (1) results from experimental tests performed on MHTGR fuel have demonstrated satisfactory attainment of performance goals at 1600°C; and (2) a somewhat higher temperature is acceptable, but it is closer to 1600°C than to

1800°C. Substantiating the very high retention of fission gases, Fig. 2.6 gives pebble bed reactor fuel test results for fission gas release for times up to 500 h at 1600°C; the fission gas release is extremely low (fractions of 10^{-5} to 10^{-6}), which implies extremely high iodine retention in the fuel.

The above results show that HTGR fuel elements essentially retain all fission products (from a 'public exposure' viewpoint) at temperatures of up to ~1600°C. The results shown in Figs 2.2 and 2.3 imply that nearly all of the retention is within the coated particle fuel [2-1].

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Chapter 3

REVIEW OF THE PEBBLE BED MODULAR REACTOR (PBMR) PLANT

The successful introduction of an advanced nuclear power plant programme depends on many key elements. It must be economically competitive with alternative sources of energy, its technical development must assure operational dependability, the support of society requires that it be safe and environmentally acceptable, and it must meet the regulatory standards developed for its use and application. These factors interrelate with each other, and the ability to satisfy the established goals and criteria of all of these requirements is mandatory if a country or a specific industry is to proceed with a new, advanced nuclear power system [3-1].

ESKOM has determined that the PBMR meets these requirements for the introduction of the HTGR in South Africa. The utility, in partnership with BNFL and the Industrial Development Corporation of South Africa (IDC), is currently in the process of performing the final evaluation for the introduction of this advanced nuclear power reactor coupled to a closed cycle gas turbine as additional generation capacity on their electric grid and for commercialization in the international market place.

3.1. PBMR BACKGROUND AND INITIAL EVALUATION

South Africa has recently experienced significant political and economic changes. In the era of apartheid, South African industry became very inwardly focused. As the majority of the economically active population, and therefore, customers, were close to Johannesburg, industry also tended to be based close to this area and away from the coast (on average ~1000 km away). This was a convenient situation for ESKOM and its customers as the heavy industrial electrical load and the country's large coal deposits are located in the same general area.

As South Africa opened itself up to the world market after the fall of apartheid, the need to base export orientated industrial plant near the coast has become apparent. Recent examples include the Hillside aluminium smelter in Richards Bay (850 MW(e) load) and the Saldanha steel mill near Cape Town (200 MW(e)), with plans for several more major plants. This concentration of load on the coast will place extensive demands on the ESKOM transmission system as well as increasing the problems of quality of supply [3-3]. This re-concentration of the load growth to the coastal areas of South Africa has changed the focus of additional generation capability from large stations concentrated at the coal fields to smaller units with capacity addition commensurate with the increase in electrical demand, hence the PBMR.

Another factor in considering the PBMR for national electrical capacity additions as well as for export in the international market place is the quest for development of new technology in support of future economic growth in South Africa. This situation can best be summarized by the following excerpt from P. Maduna, former Minister of Minerals and Energy Affairs of the Government of South Africa, written as part of the opening address to the IAEA Technical Committee Meeting on High Temperature Gas Cooled Reactor Technology Development, held in Johannesburg in November 1996. "The government's macro-economic plan is aimed at uplifting the economy, and to ensure that the social and economic inequalities that exist are overcome. If this is to be done, it must be based, among other things on the development and application of high technology, as this is the ultimate long term wealth creator in any country."

By any standards, the nuclear power industry requires one to tap some of the best technological ideas known to humanity. In the past, South Africa has devoted enormous resources to the development of nuclear technology at the Atomic Energy Corporation. While the returns are still difficult to quantify and perception is that they have not been as good as could have been hoped for, my department is currently looking at ways and means of improving performance in certain areas. The area of high temperature reactors is one where past investments in nuclear technology might yet bear fruit.

It is therefore in the light of South Africa's need for a technology-based future, and a technology base developed in the nuclear and related fields, that one must consider South Africa hosting this meeting.

In the modern world, the increasingly competitive global economy forces every country to carefully assess its competitive advantages. In the case of South Africa, one of these is the lowest cost of electricity in the world. This low cost, of below 2 U.S. cents per kilowatt hour, is one of the driving forces behind a number of major projects, which are helping expand the economy. It is an advantage that needs to be maintained into the future to allow industrialists the confidence to make large, long-term fixed investments" [3-4].

3.1.1. Background

ESKOM, as *the* South African state electricity utility, with an installed capacity of 38 397 MW(e) at the end of 1996, accounts for ~98% of all national electricity generating assets. It is almost entirely coal based with ~5% being nuclear (two PWR units of 965 MW(e) at the Koeberg station located near Cape Town).

The option of adding further LWR capacity was investigated extensively in the late 1980s and rejected due to the cost penalty when compared to coal fired options. The average price of coal delivered to ESKOM's stations is in the order of $7.50/ton, with the overall cost of utility operations as 2.1 cents(US)/kW•h in 1996. This includes all generation, transmission and distribution costs [3-2].

3.1.1.1. Utility requirements

As part of the ongoing process to review generation options for future expansion, ESKOM, under their Integrated Electric Planning Process, considered the following supply side options:

- fluidized bed coal combustion,
- combined cycle gas turbines,
- coal bed methane,
- compressed air energy storage,
- high head pumped storage,
- pebble bed modular reactor [3-5].

The study involved looking ahead for up to fifteen years in considering load growth, demand profile projections and generation options. The key requirements for supply side additions were identified as follows:

(1) Cost: The capital and operating cost must match (or improve upon) that achieved by large (4000 MW+) coal stations. The target capital cost, including interest and owners costs, is under $1000/kW(US), which is in line with the costs of a coal fired station on ESKOM's system. The O&M costs are to also be in line with coal fired stations and target at 2 cents (US)/kW•h.
(2) Lead Time: The lead time must be as short as possible to avoid the type of over capacity ESKOM experienced at the end of the 1980s.
(3) Load Following: The station must be able to load follow to compensate for the limitations on ESKOM's current capacity.
(4) Availability: This must be as high as possible. ESKOM's current target for existing stations is 90% (7% planned outage and 3% forced).
(5) Location: The plant should be able to be located where the load is without impacting overall costs.
(6) Environment: Although not mentioned above, it is vital that any new plant must be environmentally (and publicly) acceptable.
(7) Fuel Diversity: Although not vital, it would be valuable to increase ESKOM's diversity of fuel supply (currently 92% coal).
(8) Unit Size: It would be valuable, in light of the South African geographical size, to limit the unit size in relation to the local demand to allow multi-unit sites, and therefore reduce the transmission requirements for redundancy [3-6].

A series of screening processes were then undertaken to narrow down the supply side options for further consideration. Beginning in 1993, ESKOM reconsidered the nuclear option (specifically for load centers away from the coal fields) and again discounted LWRs. Further investigation was performed and two key nuclear issues where identified. These were cost and public acceptance. It was considered that these were driven by safety issues related to potential accidents. It was determined that the only way to obtain competitive costs with nuclear power was to remove the potential (however remote) for accidents with significant offsite consequences [3-8].

3.1.2. Evaluation of the PBMR

3.1.2.1. Initial assessment of the PBMR

Initial evaluation of the HTGR began in 1993 with results that indicated further study and assessment was warranted based on ESKOM's needs and requirements. The only reactor type that was seen to have the capability to overcome the above mentioned nuclear related issues was the small HTGR, using coated particle fuel [3-8]. As the result of the preliminary investigation and evaluation process, the modular HTGR coupled with a closed cycle gas turbine was selected as the concept that had the potential to meet ESKOM's requirements for supply side additions as well as the nuclear related issues of cost and public acceptance. This then became the PBMR project.

The overall objectives of the investigations up to 1998 were to establish whether such a plant could reasonably be included in ESKOM's expansion planning, what specific advantages it would bring over current options and what technical risk (and associated cost) would be involved in its implementation [3-6]. A key aspect of this investigation was the commissioning of the South African engineering firm, Integrators of System Technology Ltd (IST), to perform a detailed technical/economic evaluation of the PBMR plant. This evaluation was subjected to a rigorous technical review programme and performed in close cooperation with the Institute for Nuclear Safety and Reactor Technology of the Research Center, Juelich, Germany [3-7]. The evaluation by IST was completed in 1997 and basically supported the PBMR as the nuclear energy source to meet ESKOM's needs.

Studies throughout the period 1993–1998 showed that the technology being adopted for the PBMR base line design had been adequately demonstrated to avoid fundamental technical risk. This was supported by technology contracts with the original commercial developers in Germany (Siemens and ABB) through their subsidiary (HTR GmbH) as well as the related research institute (KFA in Juelich). To support other key technology areas, there has been detailed involvement of overseas experts, including GEC-Alsthom on the helium cooled generator and Mitsubishi Heavy Industry on the main reactor vessel, both from a technical and costing point of view.

The safety analysis of the plant was undertaken by KFA (now FZJ) Juelich in Germany, who performed the original safety work for the German program. The standards being asked for in the PBMR design were more demanding and this team provided the corresponding analysis for their justification. The increased level of safety (over current designs) is a fundamental aspect to achieving cost reductions over other nuclear designs. By demonstrating a 'catastrophe free' design, the requirements for both safety grade backup systems and an offsite emergency plan are removed.

The operational requirements were established at the start of the project by the issuance of user and owner requirements. These objectives were developed in light of the potential of a reactor linked to a closed cycle gas turbine, the current cost requirements by ESKOM and the increasing need for load following plants. Once these requirements had been established they were maintained throughout the evaluation phase as non-negotiable. This led to a major effort being put into consideration of load following ability and maintenance requirements. In no case was it found that meeting the requirements significantly impacted the cost estimate for the plant [3-6].

3.1.2.2. Influence of the PBMR on ESKOM's system

The PBMR studies were initiated to meet a future need for distributed generation at a competitive cost to ESKOM's current coal generation. The following combination of advantages of the PBMR over any other identified options were concluded to be:

- distributed generation,
- short construction period,
- small unit size,
- excellent load following capability,
- low environmental impact,
- competitive economics,
- potential for high level content.

The major impacts of the PBMR option on ESKOM's expansion planning would therefore be:

— To allow for the construction of multi-unit power stations near to coastal load centers and therefore limit the need for extensive transmission system strengthening,
— To reduce the uncertainty, risks and the costs associated with the current long term planning requirements,
— To reduce ESKOM's exposure to negative environmental claims, such as carbon dioxide emissions and use of highveld water resources,
— Improvement of quality of supply at remote locations (e.g. Eastern Cape) without the need for new line compensation equipment,
— To provide an economic mitigation strategy for greenhouse gas reductions [3-8].

The reason for the level of detail in the concept design was to ensure that the safety arguments were valid and that the costing could be effectively performed. The key requirements for this programme were that:

— An adequate technology level exists within local industry,
— A large enough utility to provide backing for the project,
— A non-prescriptive licensing regime,
— A cost structure for power generation that excludes current technology.

ESKOM has determined that these requirements appear to be met within the South African situation, and that ESKOM is well placed both in terms of its size and its legal position to advance this nuclear programme [3-2].

3.1.2.3. Initial PBMR programme reviews

There have been numerous independent reviews of the technical and commercial aspects of the plant (Table 3.1), including reviews funded by the project and those requested by potential joint venture partners. The only specific concerns raised were over the back end fuel cycle costs (by an IAEA expert on nuclear costing) and the performance of the recuperator. In the case of the back end fuel cycle costs, the figures for the PBMR are based on an agreed rationale for Koeberg. The current cost of spent fuel disposal is less than 1% of the fuel cost and therefore even a large (factor of 10) increase would not significantly increase the levelized cost of power. In the case of the recuperator, the concern is principally over the compactness of the design and this has been addressed by doubling the available volume for the recuperator. This also has not had a significant impact on the overall costs. During the period of these reviews, the available core power was increased by changes to the fuelling regime from 226 to 265 MW(t), while increasing the available margins (i.e. lowering the peak fuel temperature during loss of cooling events). At the same time the reflector structure was made substantially simpler.

The reviews have included a market survey for this class of plant covering 19 countries. This market research indicated a substantially larger overseas market than that used for the economic evaluations. This is due to the strong cost advantage of the PBMR design over other options where very cheap coal is not available.

The IAEA has been formally requested by the South African government to investigate and advise on the technical and economic feasibility, safety and proliferation aspects of the PBMR. This study is currently in progress with the final report to the South African government expected in 2000 [3-8].

TABLE 3.1. REVIEWS OF PBMR PROJECT (THROUGH JUNE 1998) [3-5]

REVIEW ORGANIZATION	AREA REVIEWED
AEA Technology	Fuel & Plant Technical & Cost
Nuclear Fuel Industries of Japan	Fuel Technical & Cost
IAEA	Overall Project Costs
Mitsubishi Heavy Industry	Plant Technical & Cost
ESKOM's Generation Projects	Project Analysis
Hunter Simonsen & Associates	Technical & Economics of the PBMR
EPE (pty) Ltd.	Macro Economic Impact Analysis
Northern Research & Engineering Co.	Turbo-machinery
Japan Atomic power Co.	Fuel & Plant Technical & Costs
Weir Group	Plant Technical
Prof. J. Gittus (UK)	Technical & Commercial Audit + Market
ESKOM Corporate Technical Audit Mgr.	PBMR Project Review
GEA	Power Conversion Unit Technical & Costs
Fuji Electric	Plant Technical & Costs
Sumitomo Corp.	Fuel & Plant Technical & Costs

3.2. PLANT ECONOMICS AND COMMERCIAL STRUCTURE

3.2.1. PBMR cost assessment

A major aspect of the technical/economic investigation performed by IST for ESKOM included a detailed assessment of the development, capital and operating costs for the PBMR and associated support facilities such as the fuel manufacturing plant. This investigation was finalized in early 1997 and encompassed the conceptual design that was in existence at that time. Further cost refinement is taking place as an integral aspect of PBMR development as this process progresses through the more detailed phases of design.

It is expected that the specific details of system and component design will continue to become more refined as the final design approaches. To attempt to address a cost breakdown by system and major component at this time in the plant design is not realistic. However, an overview of the current (March 2000) cost analysis for the PBMR can be provided for the following general categories:

— The basic development cost is estimated at R 432 000 000 (in 1999 Rands). Of this value, the design cost is ~2/3 of the total with the remaining ~1/3 applied to other aspects of development such as development of the Environmental Impact Assessment and a portion of the initial cost of licensing.
— The cost of the first plant is estimated at R 800 000 000 (in 1999 Rands), which includes plant specific design, fabrication, installation, commissioning and the construction of a 400 000 sphere/a fuel plant.
— The n^{th} plant cost is estimated at less than US$1000/kWe (installed) and can generally be broken down based on the following major plant areas:
    - Machinery ~57%
    - Electrical, C&I ~11%
    - Building ~12%
    - Services ~20%.

— The fuel cost is estimated at ~ US$4/MWhr, with the capital cost/reactor for the fuel plant of the n^{th} plant estimated at between $1.5 and 2.0 million.
— The total cost of generation is estimated at 1.67 US cents/kW•h based on 6% discount, 40 year life and 93% load factor. This includes all current estimated costs for capital, fuel O&M and decommissioning.

3.2.2. PBMR commercial structure

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/a. for local construction and 20 units/a. for export. The South African content of the local plant would be 81%, with 50% for the export plant. These content values were based upon a breakdown of plant equipment and an assessment by ESKOM of current local manufacturing capability. The project has been subjected to an input/output analysis for only the construction of the units (without the development and fuel programmes) [3-8].

3.2.2.1. National aspects of PBMR project

The analysis showed that when the project had matured (in ~10 years), the effect on the South African economy from the local and export market was:

	LOCAL	EXPORT	TOTAL
Permanent Jobs Created	92,710	111,836	204,546
GDP/a.*	R 7,734 million	R 1 0597 million	R 18,331 million
BOP/a,*	-R 791 million	R 6,488 million	R 5,697 million

* 1998 Rand

The choice of 20 units/a. for export was based upon an assessment of the world market performed by J. Gittus[1] which would equate to a 2% share of the overall world power market or (over the 20 years considered), approximately 14% of the replacement market for current nuclear plants.

The 10 units/a. for the local market was based on the long-term growth trend at ESKOM of 3.53% (1980–1993), which equates to 1,500 MW/a. on a base of 41 000 MW. This equates to the long-term medium to high growth assumptions of ESKOM.

In both cases (local and export) the impact is on a linear basis, i.e. the impact of 1 unit/a. for local market is 10% of the impact of 10 units/a. The above table can be scaled to match the assumption basis. To retain the local content, however, there must be an adequate production level (5 to 10 units/a.) to maintain the economies of scale.

3.2.2.2. Utility aspects of PBMR project

The PBMR studies were 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 economically and offering low impact to the environment [3-6].

The impacts of the PBMR on the utility's expansion planning were previously addressed (Section 3.1.2.2), and basically support the key requirements set by ESKOM for any future supply side additions.

[1] Professor of Risk Management at the University of Plymouth, senior partner of the French nuclear consultancy NUSYS, Consultant to AEA-Technology (of which he was previously a director) and Principal Advisor to Cox Insurance Holdings (Lloyds of London)

In terms of economics, the development cost of the PBMR represents ~2.5% of the capital cost of a 4000 MW coal fired station on ESKOM's system, or the interest on a three month total project delay [3-8].

3.2.2.3. Investor aspects of the PBMR project

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 as indicated in Fig. 3.1. Therefore, it was determined by ESKOM that this represents an excellent export possibility.

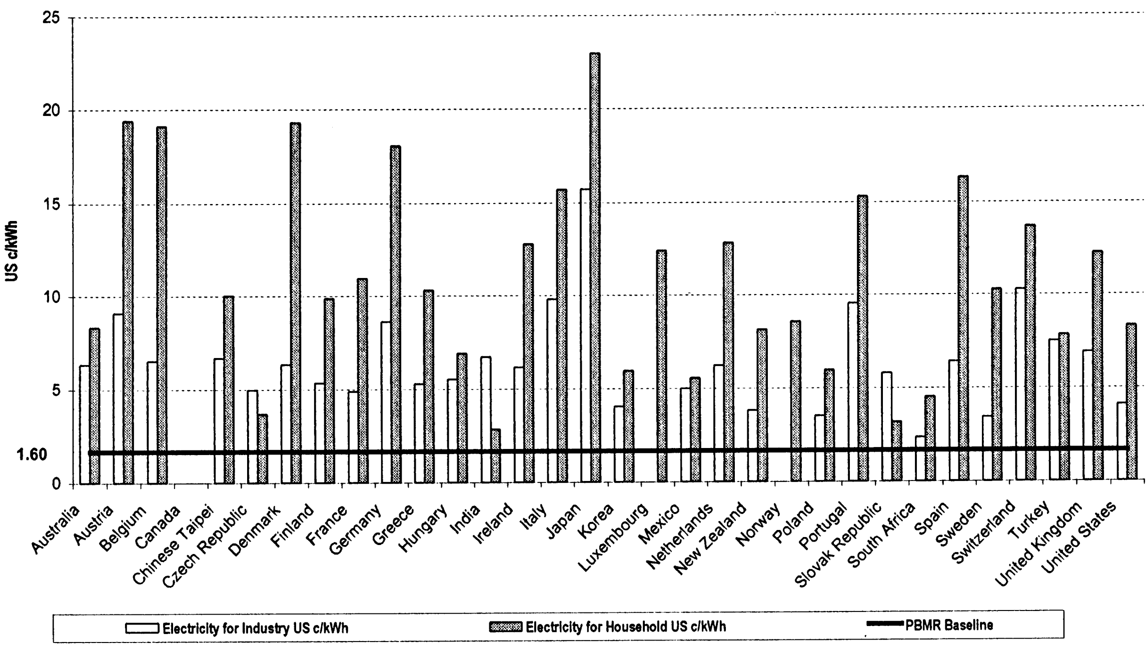

FIG. 3.1. Cost of electricity by country compared with the PBMR baseline [3-10].

In 1997, an analysis of the project investment returns was undertaken which assumed the PBMR captured ±2% of the world market for new power plants. This analysis also assumed that an owners generation cost of 1.6 cents(US)/kW•h would be attractive and resulted in a base case where ESKOM's Internal Rate of Return on invested equity and loan capital over a twenty five year period is 26% real after tax [3-6]. Beyond the first module no sales to ESKOM were included in this analysis and no external gearing was assumed. Analysis showed that the result was not highly sensitive to the startup cost, but was sensitive to the construction period gearing after the first unit.

On this basis, the project is seen by ESKOM to be a viable and attractive investment opportunity. It should be noted that there are several reasons why ESKOM has exploited specific competitive advantages that South Africa offers including having an adequate technology level within the local industry to support the PBMR project, a large utility to provide backing for the project, a non-prescriptive nuclear licensing regime, a cost structure for power generation that imposes a strong cost cap and a host utility having good public image and credibility [3-8].

3.2.2.4. Commercial structure and funding

Due to the good possibility for commercial sales of the PBMR, ESKOM has opened the project to perspective Joint Venture partners, both within South Africa and internationally. Joint Venture partners of the "Holding Company" now include ESKOM and IDC of South Africa and BNFL of the UK.

A financial model has been produced to extract potential investor returns over a twenty five year analysis period under the structure presented in Fig. 3.2.

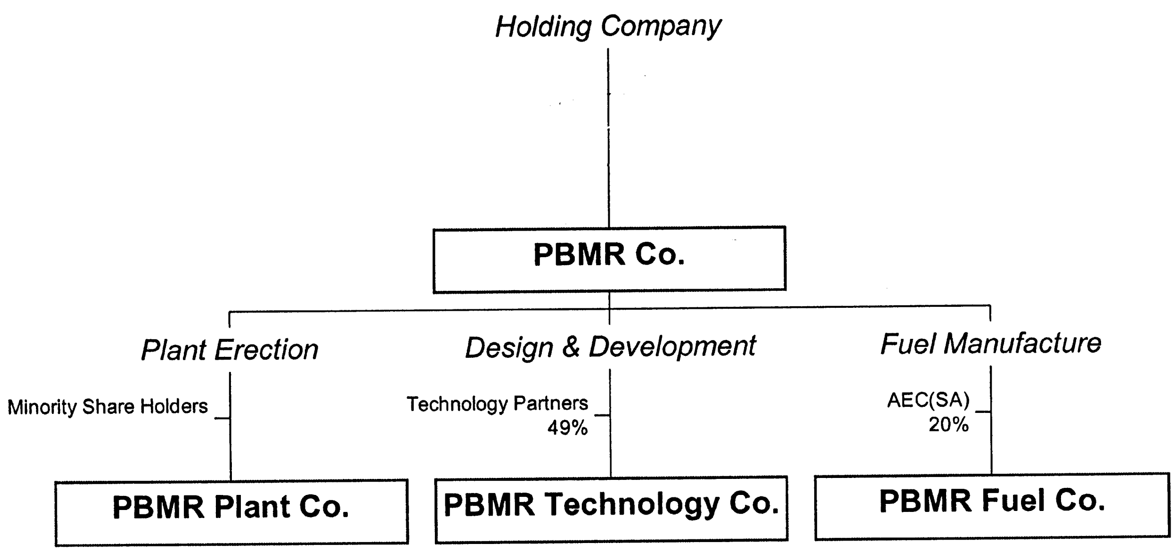

FIG. 3.2. Commercial structure for PBMR Project [3-10].

The business plan for the PBMR project is divided into three main activity phases, with Phase One consisting of the development of the technology towards establishing its commercial viability, and the completion of the engineering design for the reference plant. During this phase, the licensing process would be initiated with the NNR, a generating license will be applied for, and an Environmental Impact Assessment would be completed. Reaction of the public to the development and potential construction of the first plant will also be elicited and evaluated.

In the event of successful completion of the first phase, phase two would be entered into and include the construction of the reference plant, performance of design commissioning tests, and receipt of the design certification and operating license. Phase two would, in addition, include the construction of a fuel manufacturing plant.

While these two phases are distinct, some of the activities are to run in parallel in order to achieve the time schedule, which is to complete the development and construction of the first reference plant within five years. Parallel activities would include the early pre-qualification of the equipment suppliers for the reference plant and the early construction of some critical path items.

Phase three is the commercial exploitation of the technology through the establishment of an export infrastructure in South Africa, and the manufacture and sale of plant and fuel to utilities both locally and worldwide. (The Project does not sell electricity.)

As indicated in Fig. 3.2, ESKOM has proposed that the Project be housed in three separate limited liability companies, i.e. “Technology”, “Plant” and “Fuel”.

A joint “Technology Company” would be established between ESKOM and the providers of the intellectual property required for phase one, which is to produce the engineering design. Thereafter, this Company will sell the technology under exclusive license to a plant manufacturing company (Plant Company) and a fuel manufacturing company (Fuel Company) in return for royalty income. The Technology Company will continue in technology development after the completion of phase one, and will, by prior agreement, re-invest a fixed portion of its revenues in on-going research and development into the pebble bed technology.

The “Plant Company” will manufacture both the reference plant and other plants as it may be able to market and sell both locally and world-wide. This company will pay an on-going royalty to the Technology Company for exclusive use of the technology.

The “Fuel Company” will manufacture and sell PBMR fuel at a profit to the end users of the PBMR. This company will also pay a royalty to the Technology Company.

The three entities will be separate stand-alone operations, although they will have cross shareholdings [3-6].

3.3. PBMR DESIGN

As of the end of 1999, the conceptual design of the PBMR is nearly completed and the basic design process for selected systems is underway. Most of the information described herein resulted from developmental work during the conceptual design and is subject to further refinement as the detailed design phase is completed. The PBMR design team is located in dedicated facilities external to ESKOM in Pretoria, South Africa. This team has expanded in accordance with maturing of the design and licensing, and currently numbers 150 personnel.

3.3.1. General design criteria

3.3.1.1. Technical design philosophy [3-6]

The fundamental concept of the design of the PBMR is aimed at achieving a plant that has no physical process, however unlikely, that could cause a radiation induced hazard outside the site boundary. This is principally achieved in the PBMR by demonstrating that the integrated heat loss from the reactor vessel exceeds the decay heat production in the post accident condition, and that the peak temperature reached in the core during the transient is below the demonstrated fuel degradation point and far below the temperature at which the physical structure is affected. This is intended to preclude any prospect of a core melt accident. Heat removal from the vessel is to be achieved by passive means.

The other significant concern is a fire in the graphite core. This is avoided by showing that there is no method of introducing sufficient oxygen into a high temperature (>1000°C) core to achieve sustained oxidation. This is achieved primarily by the structural design of the reactor structure and building.

The use of helium as a coolant, which is both chemically and radiologically inert, combined with the high temperature integrity of the fuel and structural graphite, allows the use of high primary coolant temperatures (800 to 900°C) which yield high thermal efficiencies. With these high temperatures, the use of a closed cycle gas turbine is justified. This increases the efficiency over a steam plant (from ~35% to ~45%), thus reducing the unit capital cost. It also removes external sources of contamination of the nuclear circuit, as there is no system with a higher pressure than the helium. Without the possibility of leakage into the helium circuit the need for on-line clean up systems is largely reduced.

TABLE 3.2. PBMR KEY PERFORMANCE DATA/UNIT [3-12]

Description	Rating
Reactor core thermal output (maximum nominal)	265 MW(t)
Net electrical power output (maximum nominal)	116.3 MW(t)
Thermal hydraulic cycle efficiency	45.3%
Net plant efficiency	42.7%
Core inlet temperature	536ºC
Core outlet temperature	900ºC
Helium mass flow rate through core, (nominal)	140 kg/s
Nominal pressure at core inlet	7 MPa
Pressure drop over core	0.175 MPa
Average core power density (fuel spheres only),	4.3 MW/m^3
Brayton cycle pressure ratio	2.7
HP & LP compressor efficiency	89%
HP & LP turbines efficiency	89%
Power turbine efficiency	90%
Generator efficiency	98%
Ramping capability “up + down” between 0 → 100% power load,%/min	10
Step function,% of current power between 0 → 100% power level	10

3.3.1.2. Plant performance and operating principle

The key performance data are listed in Table 3.2. The dynamic performance of the design has been validated through engineering simulation specifically developed for the PBMR project.

Control is achieved by adding or removing helium to the volume inside the main circuit. This increases the primary system pressure and mass flow rate without changing the temperatures and pressure ratios. The increased pressure and subsequent increased mass flow elevates the heat transfer rate, thus raising plant power. Power reduction is achieved by evacuating gas from the circuit. The power control system revolves around a series of helium storage tanks ranging from low pressure (LP) to high pressure (HP) to maintain the required pressures. Short term control is achieved by the adjustable stator blades on the turbo-machinery and bypass flow (Fig. 3.3).

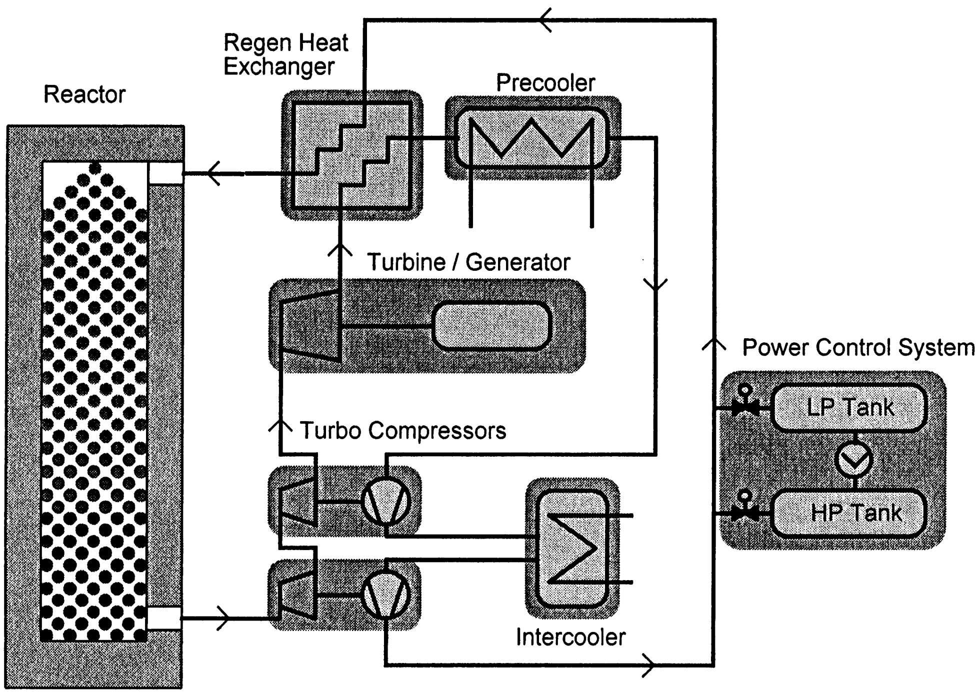

FIG. 3.3. Schematic layout of the PBMR.

The plant typically consists of a single building ~50 m × 26 m in plan and 42 m height of which 21 m would be typically below ground level. With the exception of the cooling water system (dry cooling towers or sea/river cooling) and the site control room, no other buildings are envisaged. The process cycle used is a standard Brayton cycle with a closed circuit water cooled intercooler and precooler. Separate turbo compressors and power turbine with adjustable stator blades are utilized in the power conversion unit (PCU). This separation simplifies the design and qualification process while the adjustable stator blades are used for short term control. All the bearings in the cycle are of the magnetic type which avoids any contaminants in the helium circuit and limits maintenance. The reactor systems are placed inside a reactor pressure vessel (RPV) with the PCU components located separately.

The PCU heat sink depends on the site details with the current design based on cooling water inlet at 22°C and outlet at 34.6°C. The present option includes seawater cooling with a closed-loop intermediate circuit and backup cooling [3-6].

The gross electrical output is planned at 117 MW(e) with a ramp rate of 10% per minute. Due to the safety analysis being based on the integral decay heat, it is possible to utilize the design margin of the turbo-machinery (for step changes in power) to allow short periods of operation above 100% nominal power.

3.3.1.3. PBMR general module description

The Brayton cycle and simplified system flow diagrams of the PBMR are provided herein as Fig. 3.4 and Fig. 3.5, respectively.

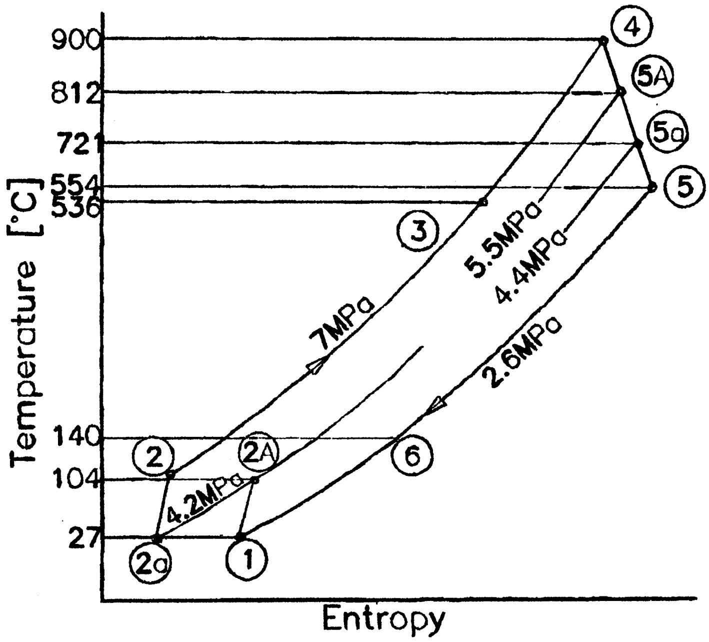

FIG. 3.4. PBMR Brayton cycle characteristics [3-12].

A thermal power level of 265 MW is calculated to provide a gross electrical generation of 117 MW(e). Helium, at a pressure of 7.0 MPa and a temperature of 900°C. leaves the pebble bed reactor and flows through the high and low pressure turbines providing energy to drive their associated compressors. The helium, at 721°C. and 4.4 MPa, then enters the power turbine providing energy to drive the electrical generator. The helium at 554°C then enters the recuperator which provides heat exchange to the helium returning to the core. After leaving the recuperator (at 140°C.), the helium passes through the precooler and exits this cooler at conditions of 27°C. and 2.6 MPa. It is then compressed by passing through the low pressure compressor, intercooler and then the high pressure compressor prior to entering the high pressure side of the recuperator. The recuperator heats the helium from 104°C. to 536°C. at a pressure of 7.0 MPa before entering the top of the reactor.

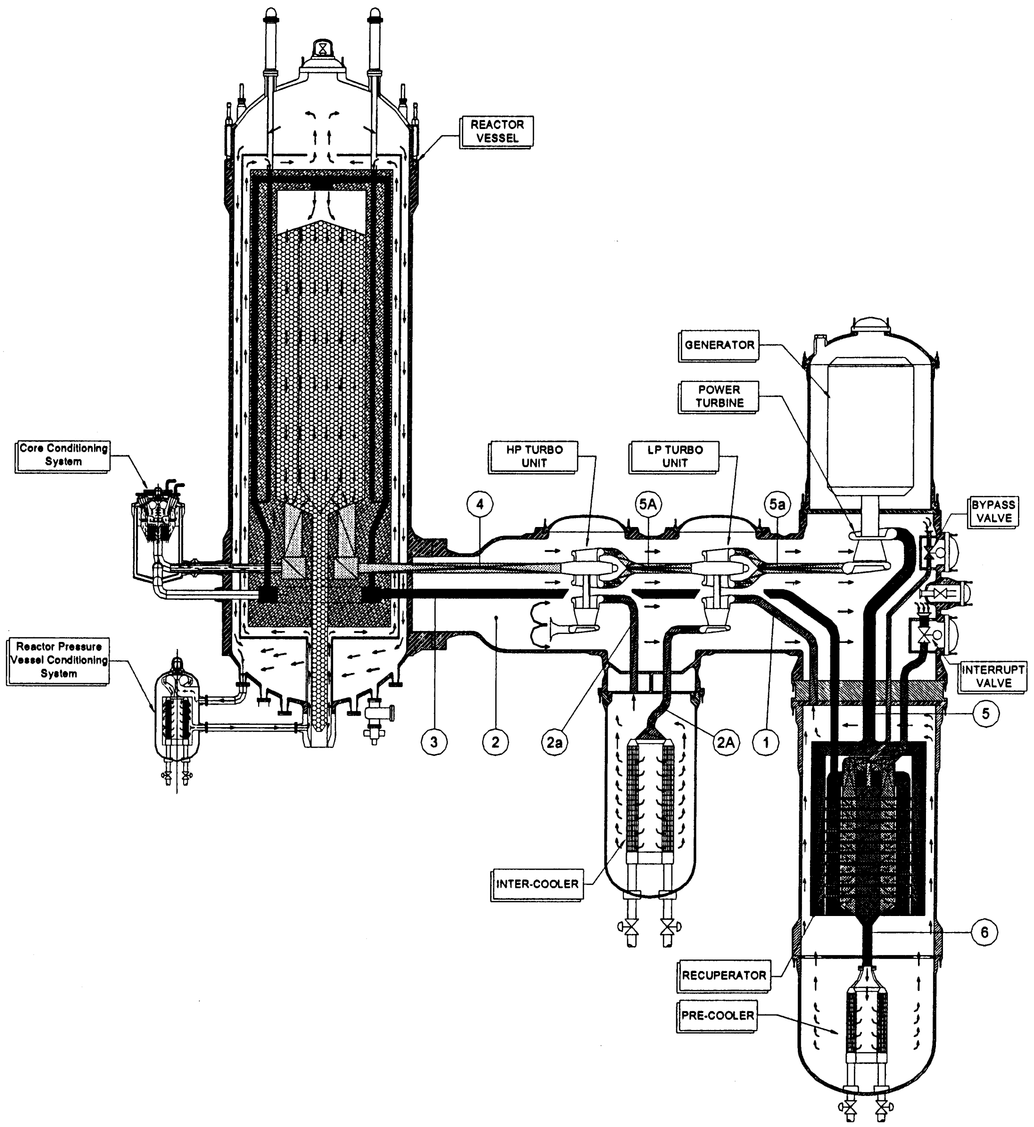

FIG. 3.5. PBMR system layout.

3.3.2. PBMR plant systems

The planned 116.3 MW(e) nominal power into the grid of the PBMR plant is achieved by the very efficient thermal hydraulic (direct) cycle design. This is possible due to the 900°C reactor gas outlet temperature combined with a highly effective recuperator. Of secondary importance is the utilization of efficient components from a performance and cost stand point, such as the turbo-units and the generator, as well as the choice of an optimal pressure ratio for the thermal hydraulic cycle.

The decision to incorporate a relatively high return temperature is based on the choice of an optimum pressure ratio. A higher pressure ratio would lead to a lower return temperature and therefore result in a lower cycle efficiency. An absolute pressure of 7.0 MPa was chosen to yield an optimum value in terms of cost versus performance of the components and system. The cycle efficiency is sensitive to the recuperator effectiveness because of the relatively low temperature of this cycle compared to open cycle gas temperatures. Also, choosing to control the pressure inside the generator compartment is deemed necessary to minimize energy losses but still guarantee di-electric isolation. The generator is basically a standard machine used in a new application but still provides an efficiency exceeding 98%.

3.3.2.1. Reactor system design and fuel development [3-12]

An annular pebble bed reactor core design with a graphite center column able to take control rods was initially selected for the PBMR. As the structural integrity of a center column capable of operation for 40 years without damage has not been proven, it was decided that a column of graphite spheres (without fuel) be incorporated to form an annular core configuration. The fuel spheres will be loaded from eight positions on the periphery, through the top reflector, to form the annulus. The advantage of this arrangement is that there is no power production in the centre of the reactor. The shifting of the peak power radially outward to the center of the annulus enables a significantly higher power output while still maintaining the required fuel safety margin. There is also a higher thermal neutron flux in the graphite reflector where the control systems are active, thereby automatically increasing the effectiveness of the control and shutdown systems.

The reactor core layout is shown in Fig. 3.6. The core diameter is 3.5 m and the equivalent core height is 8.44 m. The core is nominally subdivided into two distinct regions, the inner core primarily containing graphite spheres, and the annulus containing the fuel spheres. It was assumed that a mixing region will develop due to the sphere feeding system. However, experimental evidence indicated no more than one fuel sphere's width shifting during the downward movement. This mixing zone was statistically estimated at 25 cm but for the hot shutdown margin with the control system, 33 cm was used. The result is a fuel-poor zone of 77 cm radius, a mixing zone of 33 cm and a fuel zone of 65 cm. It is conservatively assumed that the mixing zone shows no fuel gradient from outside to inside.

The reactor core structures consist of ceramic and metallic parts. The metallic parts of the reactor structures are the:

- core barrel with its bottom support structure,
- support flange at the lower section and stiffening ring at the top,
- Helium flow channels,
- pipes and components of the fuelling system,
- fuelling tube, and
- lower discharge tube.

The ceramic parts are fabricated mainly from graphite, shaped in the form of brick elements. These ceramic parts are found in the bottom structures, on the sides forming a slender hollow cylindrical structure, and the top structure.

The bottom structures consist of the bottom reflector, cold and hot plenums and pipe shaped ceramic connections to the discharge and return tubes (pipes) to/from the PCU. Borated graphite insulation blocks are used at some locations to minimize neutron streaming to the manifold and prevent heat short-circuiting.

The cylindrical structure consists of a high purity graphite side reflector containing borings for helium flow, reactivity control (RCS) and cold shutdown systems (CSS). This is embedded in a heavily borated (4 vol.%) insulation ring that provides the functions of a thermal insulator and neutron shield for the metallic parts to protect them against activation.

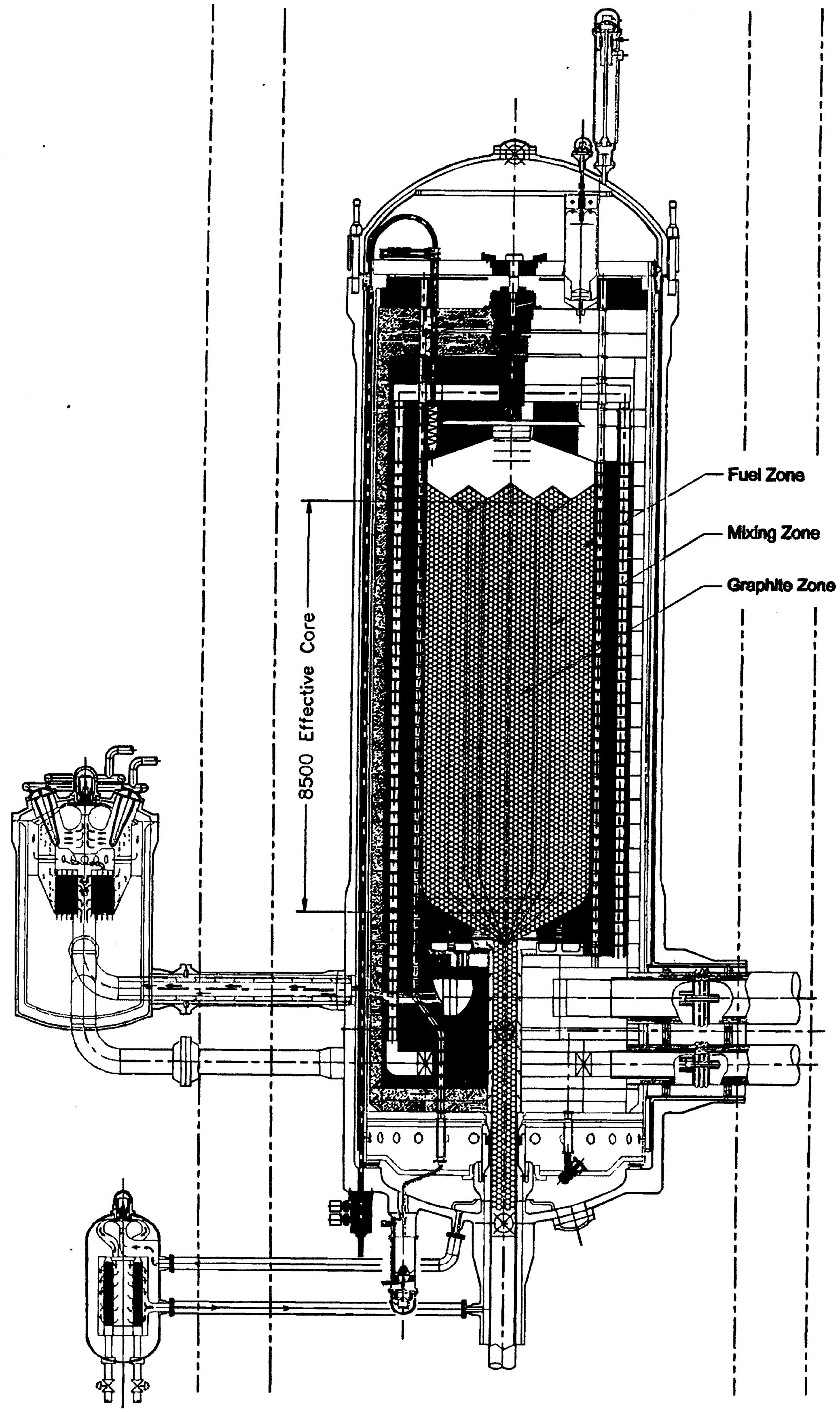

FIG. 3.6. Reactor layout.

The top structures consist of the top reflector with borings for the inlet plenum and flow holes/slots leading to the core cavity, thermal insulation, access plugs and fuel tubes.

The core cavity is a 36-sided polygon formed by the inner surfaces of the side reflector blocks. The average width across two opposing flat surfaces of the inside of the polygon is 3500 mm. The conical bottom surface of the cavity has an angle of 30° with respect to the horizontal plane.

In order to avoid the formation of stable lattices of spheres (clustering) during their movement along plane surfaces, all inner flow surfaces are provided with dish-like indentations (30 mm deep) to enhance a random distribution of the fuel spheres.

Neutrons are reflected back into the area of the pebble bed by the graphite reflectors. An effective thickness of 558 mm graphite material is expected to be sufficient for this task, however, the actual thickness of the reflector is on average 630 mm, which provides sufficient material to allow for holes and flow channels for reactivity control, coolant channels for helium flow. The coolant channel holes are located at a diameter of about 3,800 mm in the side reflector. This is an optimum position as the thermal neutron flux in the reflector reaches a maximum at a depth of approximately 150 mm.

The main helium flow path through the ceramic structures is as follows. The coolant gas enters the structures at the lower inlet plenum and flows upwards through gas channels/holes in the ceramic internals. It is redirected at the top via an upper plenum and flows downward through the reactor core. The helium leaves the core through the bottom structure and is collected in a hot plenum from where it is directed to the PCU.

The 36 borings in the side reflector for the coolant flow to the top plenum are lined with sleeves with an inner diameter of 190 mm. These sleeves provide for thermal insulation, good flexibility and sealing to reduce heat and gas leakage. Neutron shielding of metallic parts, e.g. RPV, is provided by the bottom, side and top reflectors and in particular by specific borated insulation layers in the bottom reflector, borated insulation outside the side reflector as well as the ring structures close to the return and discharge tubes. In addition, attention is given to gas flow routes in the vicinity of the hot gas duct connection in order to minimize neutron streaming.

Thermal insulation of the hot and cold helium streams through the core structures (having temperatures of 536 to 900ºC) is provided by layers and ring-type structures made of insulation material and a special insulation hood at the top. To minimize energy loss, a layer of insulation is fitted between the bottom hot and cold plenums.

The rate of nuclear energy production is controlled by the insertion or withdrawal of small absorber spheres in holes located inside the graphite reflector blocks of the core structures. Long-term shutdown of the nuclear reaction will also be achieved by the small absorber spheres that will be inserted into holes in the graphite structures.

All parts of the reactor core structures are designed for the service life of the reactor. However, provision is made for the replacement of the complete core structure. Visual inspections can be carried out on both the ceramic and metallic parts, if required. These operations are carried out with the main power system depressurized and core defueled, via openings in the top and bottom heads of the reactor vessel.

The following principle factors were evaluated in the selection of the values for fuel enrichment and heavy metal loading:

- Influence on control rod reactivity worth,
- Maximum power production per fuel sphere,
- Maximum power production per coated particle,
- Effect on Xe reactivity effects,
- Maximum fuel temperatures which can be reached.

The starting point for the fuel design was the choice made for the German MODUL reactor. This included an enrichment of 7.8%, and a heavy metal loading of 7 g per sphere. Increasing the enrichment increases the fuel costs, however, increasing the heavy metal loading will decrease costs since fewer fuel sphere will have to be manufactured and fewer spent fuel storage tanks needed. The 7 g per sphere for the MODUL was partly necessitated through the requirement that a minimal increase in reactivity should result in the event of the core being saturated with water vapor. The absence of a steam generator, and the annular core, largely mitigate the effect of the water (as far as reactivity is concerned) and places a lower restriction on the heavy metal loading. Up to now, a large number of calculations for different core geometries and fuel parameters have been performed in an attempt to optimize the core layout. As the core height also has a strong effect on the capital cost of the reactor, the best design as of April 1999 is the following:

Core height:	8.44 m
Core diameter:	3.5 m
Fuel enrichment:	7.8–8.3%
Heavy metal loading:	9 g/sphere
Number of control units:	12
Number of cold shutdown system (CSS) units:	24

The control units are needed for intermediate shutdown down to 400ºC and for reactivity control when load changes or Xe transients force adjustments. The Reactivity Control and shutdown system (RCSS) is used to:

- Control the reactor during normal operation,
- Maintain the reactor sub-critical down to a maximum temperature of 400ºC,
- Maintain the reactor sub-critical indefinitely in a cold (50ºC) condition.

The RCSS consists of two independent and diverse systems, namely the RCS and CSS. The RCS is used for control of the reactor during normal operational conditions and, when required, to place the reactor in the “hot shutdown” condition. Both systems (combined) are used to maintain the reactor sub-critical indefinitely in a cold condition. Although the CSS is also designed to hot shutdown the reactor, it will only be used for this function if the RCS is unavailable. The reason for this is that the CSS does not have an on-line system for the removal of small absorber spheres (SAS). Special operations are required to remove the SAS after they have been inserted into the holes in the side reflector.

The RCSS is seismically designed and will operate under a single failure criteria mode. The system is important to safety. However, the safety function is to ensure that the shutdown temperature of 400ºC is met, not that the system is available for maintenance.

Shielding plugs with staggered steps in the thermal shield prevents activation of the SAS containers on top of the reactor vessel by neutrons.

During operation, the RCS must compensate for reactivity changes in the core. This is accomplished by 12 subsystems referred to as control units. The control units are grouped together in six banks of two units, each forming an independent control bank. Control of the reactor is performed (with the control units, diagrammatically represented in Fig. 3.7) by adding/removing small absorber spheres (SAS) to/from the holes in the side reflector. To provide adequate excess reactivity, one bank is normally fully inserted. To ensure uniform long-term fluence, the inserted bank will be changed on a regular basis. This arrangement (of one complete inserted bank) reduces the peak neutron fluence on the reflector inner surface.

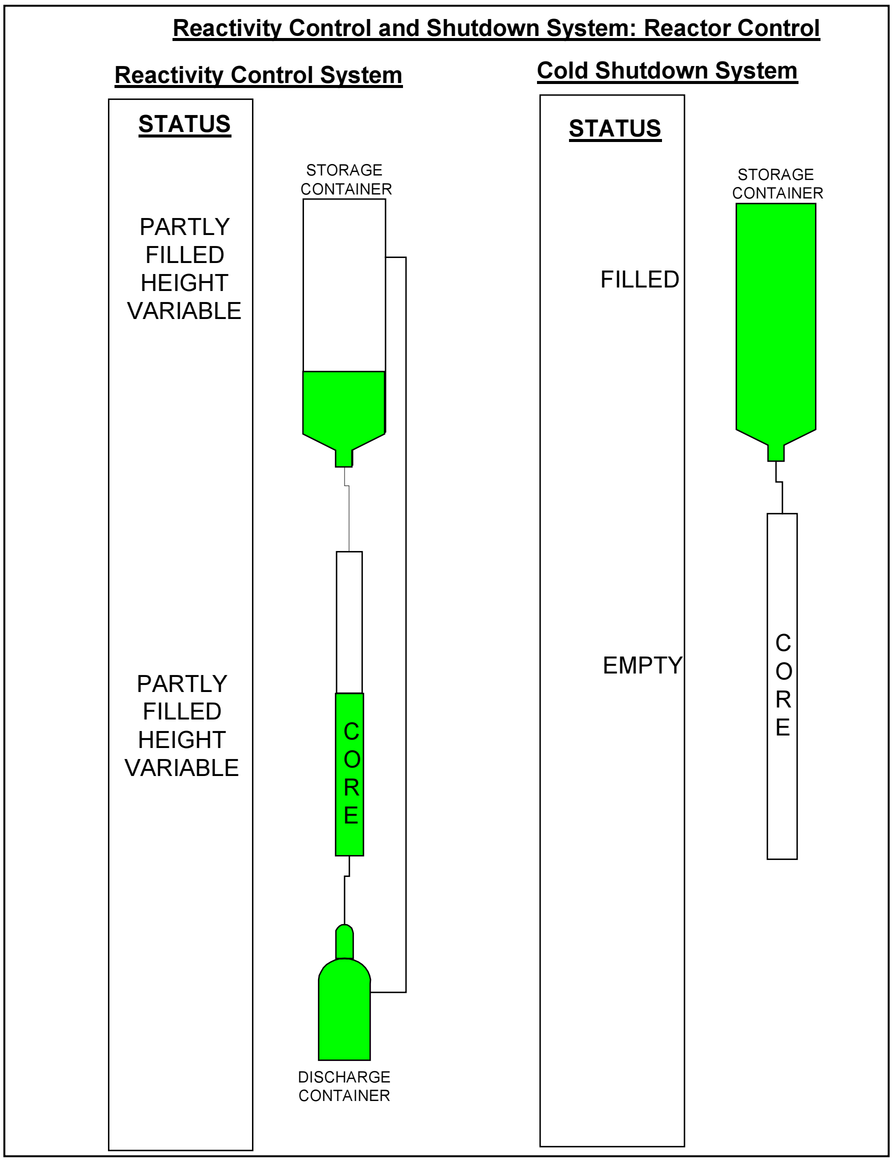

FIG. 3.7. RCS and CSS representation.

When hot shutdown of the reactor is required, all SAS in the storage containers of the RCS or CSS are released into the holes in the side reflector. The SAS fall by gravity from the storage container into the holes. Enough SAS are supplied to fill the holes in the side reflector to a position of one meter above the effective height of the core.

The control units are basically similar to the CSS units except that they are designed to deliver or remove small quantities of SAS into the control openings on the reflector. An immediate effect will be experienced upon insertion of the SAS, since the most effective area at the top will be covered less densely. There will be limits placed on the height of insertion of these units to ensure that a sufficient immediate shutdown margin is always available. Approximate control unit worth for the present design are given in Table 3.8.

The CSS enables the reactor to be placed in the maintenance shutdown mode for long periods. During operation, graphite spheres in boron carbide containers are stored in silos on top of the reactor lid. They are not automatically inserted on reactor scram signals, but are only used when the reactor needs to be shutdown for longer periods. On demand, all 24 bores for the CSS system will be filled.

The most important feedback reactivity effects that have to be taken into account in the design are the temperature coefficients and the build-up and decay of Xe following power changes. Reactivity control is also needed for the load following demands placed on the PBMR design. Pebble bed reactors exhibit a strong negative temperature coefficient that acts as an effective barrier to limit maximum temperatures of the fuel. The main components are the fuel, the moderator and the reflector coefficients, of which only the reflector coefficient is positive. The temperature coefficient remains negative over the entire temperature range of the reactor. It is not constant over the full temperature range, but attains a minimum value just below the operating temperature. This means that, when the temperature exceeds the operating limit, the coefficient that limits the excursion will become ever stronger. On the downside, the same is true when a severe under-cooling event take place. In that case, the reactivity will increase rapidly until the temperature starts increasing again. The reactivity data for the present design, including temperature effects, are given in Tables 3.3 and 3.4.

TABLE 3.3. HOT SHUTDOWN REACTIVITY REQUIREMENTS

(Core height 8.44 m, diameter 3.5 m, 9 g/sphere HM, 80 cm reflector with holes)

		ρ Hot Shutdown	ρ Cold Shutdown
Thermal Reactor Power	265 MW		
Load Following (without bypass)	100–40–100%	1.1%	
Xe Override	100–40–100%	1.2%	1.2%
Temperature Effect (100–0% power)	50ºC	0.2%	
Temperature (100% hot–0% cold)	600ºC		3.1%
Xe Decay to Cold			4.1%
Sub-criticality in Cold Condition			0.3%
Uncertainties	5%	0.1%	0.4%
Total Requirement		2.6%	9.1%

TABLE 3.4. REACTIVITY CAPABILITIES

Capabilities		ρ Hot Shutdown	ρ Cold Shutdown
12 control units (SAS)		8.5%	
Failure of Maximum Value unit		–0.75%	
24 CSS			11.3%
Failure of 1 CSS Line			–0.5%
Uncertainties	5%	–0.45%	–0.5%
Total Available		7.3%	10.3%
Reserve Balance		4.7%	1.2%

During reactor operation, the Xe will start building up approximately 15 minutes after power reduction is initiated. The result is that the SAS will have to be withdrawn to compensate for the loss of reactivity. After a certain time, equilibrium between Xe production from the Iodine and depletion through absorption of neutrons will be reached and no further compensation is needed. However, when the reactor power is reduced, the production of I-135 will decrease, while production of Xe will still continue at the old rate. At the same time burnup of Xe will also decrease, due to the lower neutron flux, and the result is that the Xe concentration will temporarily increase. This means that the control system must be able to add the required extra reactivity to keep the reactor operating. The larger the load following requirements, the larger this extra built-in reactivity will have to be. This reactivity must be kept as low as possible in order to limit the effects of a postulated event involving an uncontrolled withdrawal of SAS. These two opposing requirements have had the result that load following at continuous maximum efficiency has had to be limited to the range 100–40–100%. The reactivity requirements and control capability is provided in Table 3.3 and 3.4, respectively.

Central to the passive safety of the HTGR design is the coated particle fuel technology to be used in the PBMR. The approach to passive safety centers on the ability of the coated particles to retain all key radionuclides as long as a maximum fuel temperature of 1650ºC is not exceeded. This implies a reactor design that excludes reliance on any active safety systems inside the primary circuit for any postulated accident scenario.

The coated particle fuel used in the PBMR has a long history, starting with research in the UK for the Dragon project. The so-called TRISO fuel with a layer of SiC sandwiched between two layers of PyC was a development arising from this and fuel development has continued in different parts of the world (see Chapter 2 on TRISO coated fuel development). Fig. 3.8 depicts the fuel particle make-up for a PBMR sphere. Characteristics of the fuel to be used in the PBMR are given in Table 3.5.

The manufacture of fuel is a key element of the PBMR programme, and the quantities required exceed any previous HTGR project. Therefore, while there are facilities to construct HTGR fuel on a generally small scale currently in the world (in Japan and China, for example), the PBMR requires a new fuel manufacturing facility. The present intention is to construct it at the South African Atomic Energy Corporation (AEC) site, in the complex that built the fuel for Koeberg. The layout has been identified for a line to initially manufacture 1.4 million spheres/year, including the equipment specifications. The AEC, under an ESKOM

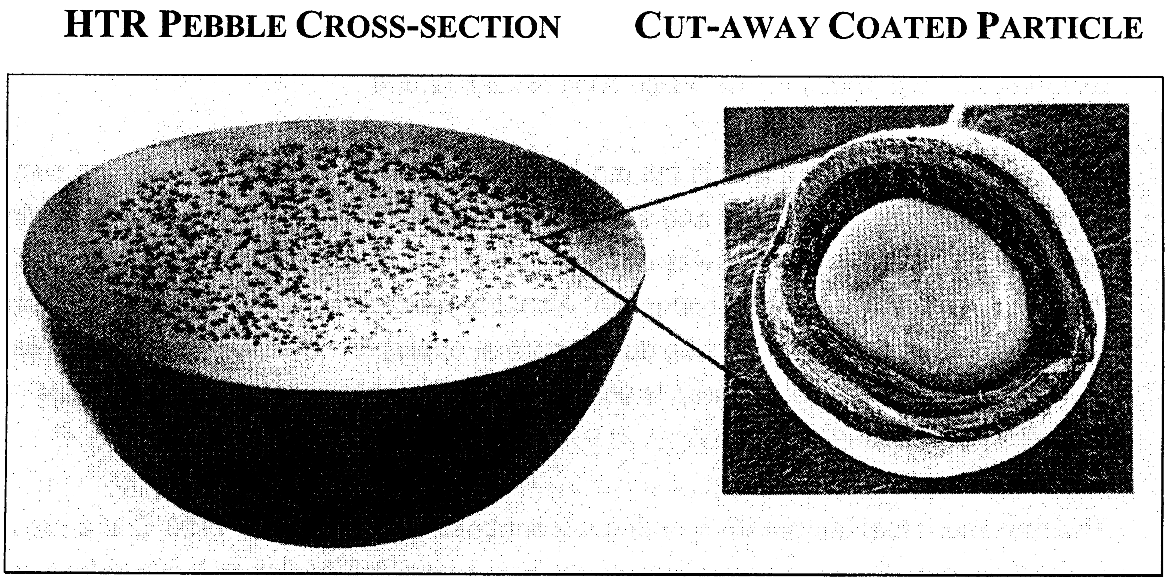

FIG. 3.8. PBMR fuel particle and sphere.

TABLE 3.5. PBMR FUEL CHARACTERISTICS

Kernels:	
Material of Kernels	UO_2
Enrichment	7.8–8.5% ($^{235}U/U$)
Density	10.4–10.75 g/cm^3
Diameter	500 μm
Coated Particles:	
Outer Diameter of Coated Particles	~920 μm
Material of Coatings	C/C/SiC/C
Density	1.05/1.90/3.18/1.90 g/cm^3
Thickness	95/40/35/40 μm
Spheres:	
Diameter	60 mm
Diameter of Fuel Zone	50 mm
Density	1.75 g/cm^3
Thermal Conductivity (Temperature and Irradiation Dependent)	0.17–0.39 W/cm· ºC
Graphite Material (Matrix Outer Shell, and Graphite Spheres)	A3 (proposed)
Loading:	
Heavy Metal Loading	9 g/sphere
No of Fuel Particles	15 000 n/sphere

contract, has started laboratory scale work including the manufacture of fuel kernels. This work is to support the external technology that is being obtained [3-2].

3.3.2.2. Power conversion unit [3-12]

The PCU (Fig. 3.9) includes the equipment necessary to convert the heat of the hot helium from the reactor into electricity.

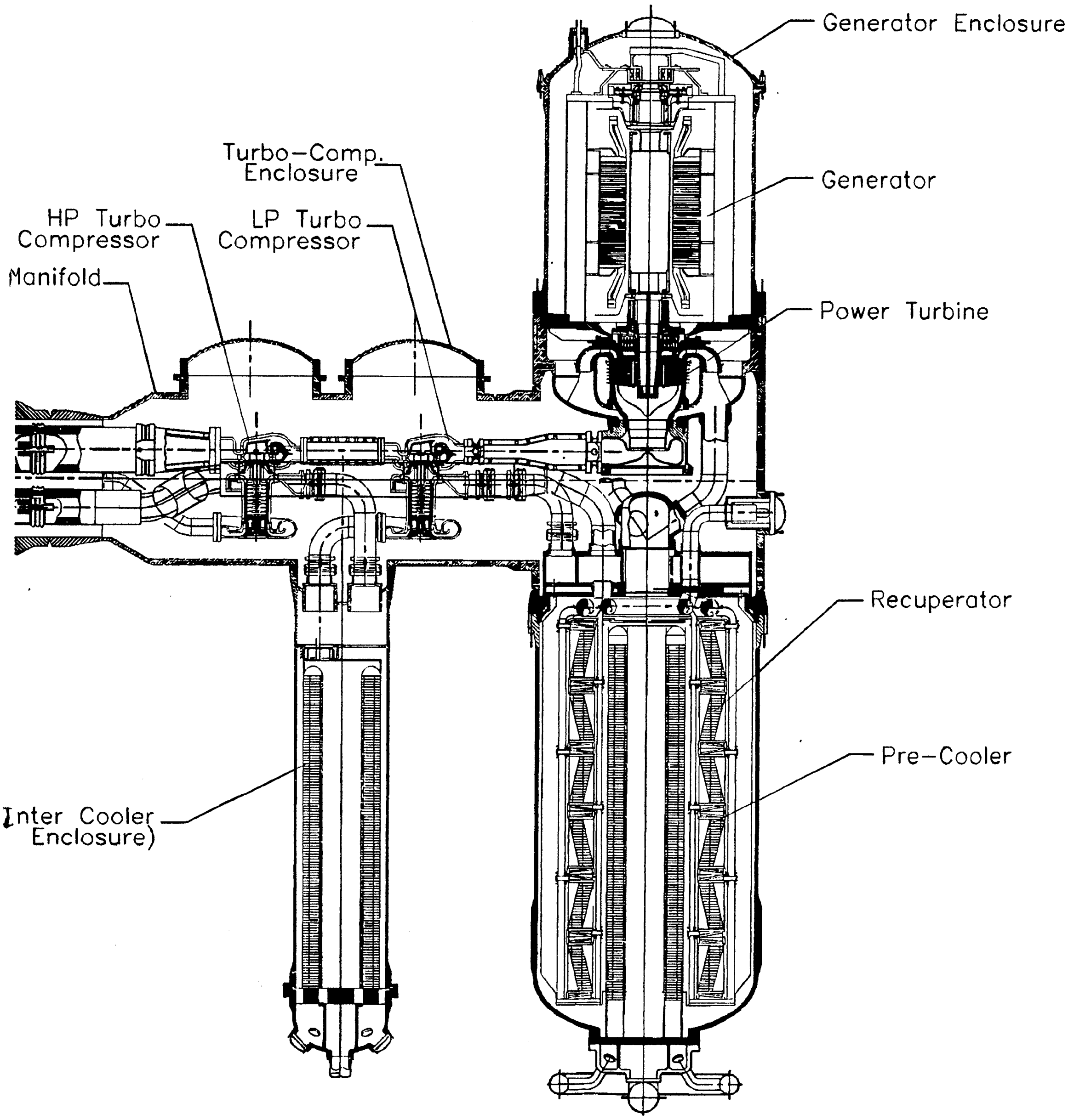

FIG. 3.9. PCU arrangement.

The PCU includes the following equipment:

- Low pressure turbo-unit (LPT)
- High pressure turbo-unit (HPT)
- Power turbine with generator (PTG)
- Recuperator (REC)
- Pipe and valve system
- Intercooler (IC)
- Precooler (PC).

The PCU converts heat received from the reactor to electrical energy. Waste heat of the Brayton cycle is rejected by internal heat exchange to the active heat removal system. The PCU is designed to deliver 114 MW nominal electrical energy to the grid. By using special control features, any generator output power setting from 2.5 MW (expected house load) to 116.3 MW is attainable. The PCU pressure boundary is seismically designed. The PC and IC, and other water-contained components are designed to retain fluid integrity during a seismic event, with respect to breaches internal to the PCU. All equipment is designed to survive a 0.5 g ground acceleration.

The PCU inventory control strategy is to control the helium inventory over a range from ~20 to 100%. The lowest helium inventory level under normal power mode is limited to guarantee a minimum pressure of 1 MPa in the generator compartment that follows the power turbine inlet pressure. This is the lowest allowable pressure to prevent flashover possibility. Lower power levels can only be obtained by use of the bypass valve.

Adjustable blade geometry on the turbo machines makes it possible to change the power level by 10% of current power within one second. For the purpose of the conceptual design, a value of ±20% mass flow change is used to mask uncertainties such as compressor surge line characteristics. The PCU control system will maintain system parameters such as turbo-speed and alternator enclosure pressure to the required levels, thereby matching the load, especially under transient conditions. The system performance requirements are met by running with a turbine inlet temperature of 900ºC and system pressure ratio of 2.7. Through the use of helium inventory control, both these parameters, as well as rotational speed of the turbo-machines, remain constant over the full power operating range. This implies minimal thermal stress as well as acceleration time independence under transient conditions. Table 3.6 provides full power system parameters.

TABLE 3.6. PCU SYSTEM PARAMETERS

Position	Parameter	Value
Cycle	Core mass-flow	140 kg/s
	Reactor thermal power	265 MW
	Reactor inlet temperature	536ºC
	Reactor outlet temperature	900ºC
	Generator electrical output	116.3 MW
	Overall pressure ratio	2.7 MW
	Cooling water temperature to heat exchangers	22 ºC
	Generator efficiency	98.5%
	Cycle efficiency (generator excluded)	45.3%
	System efficiency (generator included)	44.1%
	Plant net efficiency	42.7%

The function of the Turbo-units is to provide the pressure in the cycle to drive the system turbine. The two turbo-compressor units are of similar design and construction. For each unit, the 9-stage compressor and double entry turbine discs are constructed on the same shaft, suspended by electromagnetic bearings. The turbine discs overhang one bearing for easy removal while improving shaft resonant frequency to be well above the running frequency. The compressor discs are welded together, but the remainder of the rotor construction is bolted.

The turbo-units are inserted into fixed vertical barrels inside the manifold, accessed through the closures to allow for a hermetically controlled replacement. The compressor housing is a split-half construction and is bolted onto the inlet casing. The turbine housing is a 3-part casting, mounted on the compressor counterpart via an isolation ring, to minimize heat transfer. Removal of the outer turbine casing exposes the turbine double disc, which can then be unbolted and removed. Because of the electromagnetic bearing capability to tolerate unbalance, fitting of a new turbine disc assembly may not necessarily require re-balancing of the complete shaft unit. Turbo-compressor characteristics are provided in Table 3.7.

Material used for the individual components is as follows:

Turbo-turbine blades:	IN100
Turbo turbine discs:	IN100
Compressor blades:	PH13/8Mo stainless steel
Compressor discs:	17/4 PH stainless steel
Turbo-rotor:	17/4 PH stainless steel
Turbo-turbine housing:	INCOLOY 800
Turbo-compressor housing:	17/4 PH stainless steel

TABLE 3.7. TURBO-COMPRESSOR CHARACTERISTICS

	Unit	**HP comp**	**LP comp**	**HP turbo**	**LP turbo**
Shaft speed*	rev/min	15 200	14 200	15 200	14 200
Isentropic efficiency*	%	89	89	89	89
Inlet diameter*	mm	551	630	678	704
Blade length inlet*	mm	74.4	119.4	32.6	38
Load coefficient*	–	0.306	0.306	1.6	1.7
Flow coefficient*	–	0.50	0.50	0.75	0.75
No. stages*	–	9	9	1 + 1	1 + 1
Number rotor blades*	–	43	43	330*2	330*2
Number stator blades*	–	50	50	36	36
Power*	MW	58.4	58	58.4	58
Cooling flow*	kg/s			2.1	3.0

* To Be Finalized
HP comp = High pressure compressor
LP comp = Low pressure compressor
HP turbo = High pressure compressor turbine
LP turbo = Low pressure compressor turbine.

The LPT and HPT Turbo-units run at 14 200 and 15 200 rev/min respectively. Each unit shaft weighs ~600 kg. The turbo-machines will run at 30% under their first critical speed. The turbo-machines are housed in the primary pressure boundary and are subjected to a pressure differential across the housing. The turbo-machine housings are designed for a 100% increase in the pressure differential and are pressure tested to ensure functionality. Furthermore, the housings are designed such that a rotor disintegration due to an overspeed burst will be contained within its own housing.

The function of the power turbine (PT) is to absorb the energy in the high pressure and high temperature helium stream via the power turbine and supply the power to the electrical grid via the generator system. The PTG forms part of the active control of the recuperated Brayton Cycle, in conjunction with the HPT/LPT turbine units, bypass valve and helium inventory control system. The PTG consists of the following main subsystems:

- Power turbine,
- Generator system,
- Electromagnetic and Auxiliary Bearings,
- Force balancing system.

The PT is integrated on the same shaft as a 3000 rev/min generator. The unit consists of ten stages as compared to a high-speed turbine using a gearbox. The ten stages of rotor blades are fitted in annular grooves in the rotor. Stator blades and inlet guide vanes are adjustable. Each set of stator blades support a seal ring on the inside, which has a sealing surface towards the rotor center-line and upper disc. Cooling gas is passed through the hollow blade shank and seal ring to a cavity between the two seal faces. The quantity of gas flow is such that a temperature diluted stream will pass through the seals and adequate cooling of the disc will take place. The turbine housing is a split casing and allows assembly on the generator bearing housing after the turbine rotor is fitted to the shaft. The complete PTG unit fits inside the pressure housing and is supported on flanges machined on the vertical section of the manifold. Fig. 3.9 and Table 3.8 provides the arrangement and characteristics of the PTG, respectively.

TABLE 3.8. PTG CHARACTERISTICS

Unit	Power turbine
Shaft speed	3000 rev/min
Isentropic efficiency	98%
Inlet diameter	1560 mm
Blade length inlet	85 mm
Hub-tip ratio*	0.928
Blade tip speed*	245 m/s
Load coefficient*	1.87
Flow coefficient*	1.11
No. stages	10
No. rotor blades	347
No. stator blades	284
Power	116.3 MW
Cooling flow*	2.6 kg/s

*To be finalized.

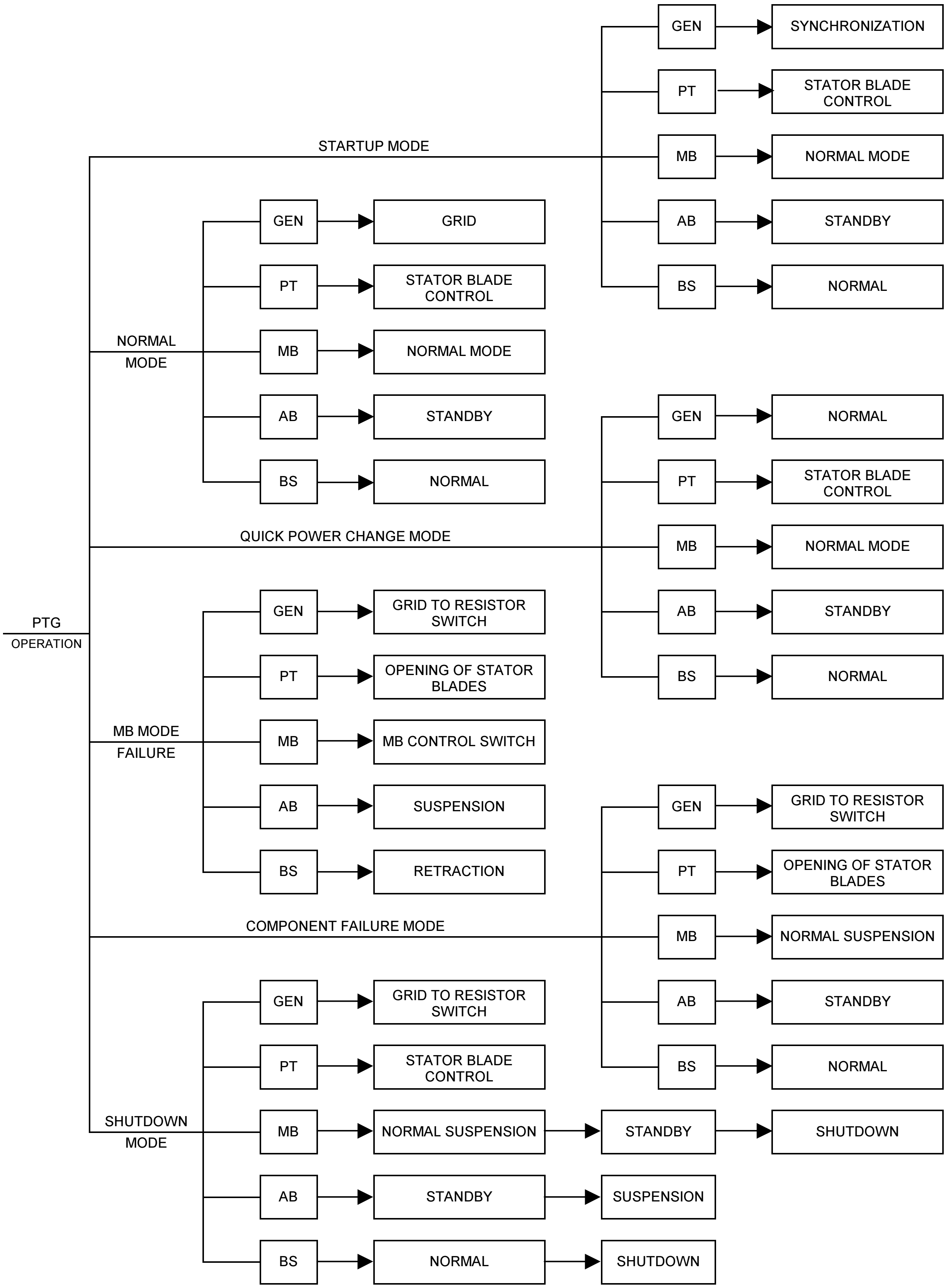

FIG. 3.10. Power turbine generator modes of operation.

Material used for the individual components is as follows:

—	Power turbine blades:	IN100
—	Power turbine rotor:	IN-713
—	Compressor blades:	PH13/8Mo stainless steel
—	Compressor discs:	17/4 PH stainless steel
—	Power turbine housing:	INCOLOY 800.

The PTG housing is designed for the maximum system pressure of 7.2 MPa. The PTG design is such that a rotor disintegration due to an over-speed burst will be contained within its own housing. The system is supplied with a UPS to ensure orderly rundown of the PTG takes place during power failure. A 10 MW resister bank has also been inserted in the generator electrical system to absorb energy upon load rejection after loss of a magnetic bearing in order to preserve the catcher bearing. The PTG modes of operation are depicted in the following diagram.

The PTG rotor is supported by eletromagnetic bearings which are anticipated to be relatively maintenance free and will not introduce contaminants into the cycle. Non-lubricated rolling element auxiliary (catcher) bearings are integrated to guarantee that the shaft is always supported in the case of an eletromagnetic bearing failure.

The PTG shaft weighs roughly 32 tonnes, is mounted vertically and runs at 3000 rev/min. The turbine is mounted on the generator shaft at the bottom end near the axial magnetic bearing to minimize differential expansion. The radial and axial capability of the bearings is 12 tonnes and 60 tonnes, respectively. The diameter at the radial bearings is 630 mm and the radial magnetic gap is 1.5 mm. The auxiliary (catcher) bearings are roller bearings, which will be designed to take 20 rundowns of the shaft before overhauling. The gap on the radial and axial auxiliary bearings is about 0.4 mm. The shaft will have to pass its first bending frequency and the first bending frequency of the stator during running up to full shaft speed.

The intent by ESKOM is to use a plate-fin recuperator composed of repetitive modules. The function of the recuperator is to return heat downstream from the power turbine back to the flow path ahead of the reactor. This raises the helium return temperature to the reactor. The high temperature return helium is kept apart from any pressure boundary component by forcing the helium through insulated pipes to the ceramic core structures. The recuperator is required to exchange maximum heat with minimum pressure loss. The plate-fin recuperator will contain 72 modules arranged in six layers. Each layer will have 12 modules arranged in an annulus inside a cylindrical container that is at the low pressure side of the recuperator. The high pressure side is connected to the module via a pipe network. Alternate layers of modules are to be arranged upside down to eliminate thermal differences where they fit together. The basic flow element is a pressed (corrugated) flat plate. Corrugation height and pitch and plate thickness is chosen to withstand the pressure difference.

The cycle has two heat exchangers, the PC located downstream of the recuperator and the IC between the two compressors. With a closed loop Brayton cycle, the PC anchors the temperature of the low pressure line. The IC reduces the volume flow to the second compressor, causing a reduction of compressive work. The heat load of the two exchangers is so close that similar layouts could be employed for both.

The PC is located in the pressure vessel and the IC is located in the center of the recuperator cavity. Both are mounted at the lowest positions of the PCU to decrease the likelihood of water ingress into the rest of the system. The vessel temperature is low at 27ºC, which represents the heat exchanger hot side outlet temperature. A two-pass layout of high finned tubes is incorporated in these coolers. Tube ends of U-shaped tubes are rolled and welded into bulkheads (tube sheets) which separate the water from the gas side.

Heat rejection from the PC and IC is via intermediate closed loops between the main gas stream and the heat sink. Water ingress into the main gas circuit is unlikely, as the helium pressure is always higher than the water pressure in the intermediate loop under reactor power operational conditions. Table 3.9 provides the PC and IC characteristics.

TABLE 3.9. PC AND IC DESIGN CHARACTERISTICS

Parameter	Pre-cooler	Intercooler	Unit
Mass-flow (gas side)	145.3	146.5	kg/s
Inlet temperature (gas)	140.5	104.4	ºC
Outlet temperature (gas)	27.9	27.6	ºC
Water inlet temperature	22.0	22.0	ºC
Water outlet temperature	54	44.4	ºC
Heat transfer capacity	85	58.1	MW
Tube inner diameter	15.75	15.75	mm
Fin diameter	34.0	34.0	mm
Triangular pitch	38.0	38.0	mm
Total heat transfer area	10.87	8.15	m^2*10^3
Water mass-flow	628	628	kg/s
Pressure drop (gas side)	152	76	Pa
Heat conduction	8.1	7.2	kW/m^2

3.3.2.3. Vessel system design [3-12]

The function of the pressure boundary system is to contain the helium coolant by maintaining boundary integrity. It also provides structural support and alignment for the components that are housed within the reactor unit pressure vessel (RPV) and the power conversion unit pressure vessel (PCUPV). The RPV contains the nuclear core and core support structures and the PCUPV contains the turbo-compressors, the turbo-generator, recuperator, precooler and intercooler. Support and restraint structures are considered part of the pressure boundary system (Fig. 3.11).

The pressure boundary system is being designed to an international pressure vessel code or standard capable of ensuring all of the functional, safety and reliability requirements can be met. The sizing calculations and material selection for the basic design were based on the ASME Boiler and pressure vessel Code, Section III, Division 1, Subsection NB. This code is presently the preferred design code.

The RPV and certain parts of the PCUPV will be cooled from the inside by the helium stream leaving the high pressure turbo-compressor. The RPV is kept at a uniform temperature of 250 to 350ºC by the RPV conditioning system. This ensures that the material properties remain stable under nuclear conditions. The PCUPV is maintained at a temperature of 120 to

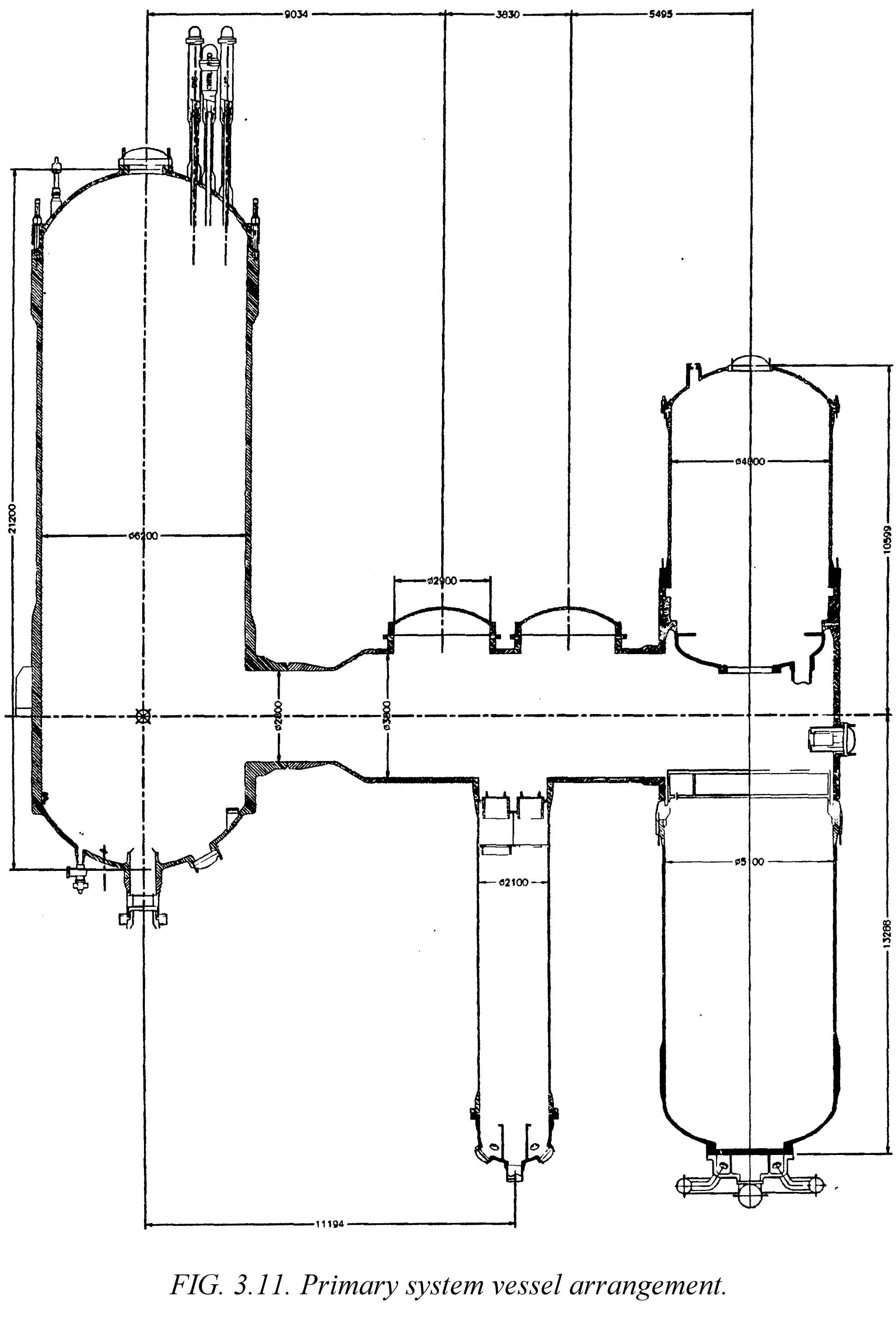

FIG. 3.11. Primary system vessel arrangement.

130ºC, depending on the internal helium flow. During a loss of forced cooling event, the wall temperature of the RPV will increase to approximately 350ºC. It should be noted that in case of a loss of forced cooling, the system pressure will equalize at approximately half the initial value.

The design of the pressure boundary system is based on:

— 7.0 MPa system pressure with a 300ºC wall temperature for normal operation,
— 4.5 MPa system pressure with a 350ºC wall temperature for a loss of forced cooling event.

The support principle used for the pressure boundary system is to restrain the centerline of the RPV tangentially, allowing unrestrained radial and axial thermal growth. The PCUPV is to be restrained perpendicular to its centerline but is allowed to move freely in line with it, to account for thermal expansion. Vertical support is provided at the same building level for both the RPV and PCUPV, to minimize the differential vertical thermal expansion at the different supporting locations. The vessels and supports are seismically designed. High stresses induced during an earthquake will be mitigated by seismic restraints or snubbers placed at the top of the RPV, at the top of the generator enclosure and at the bottom of the precooler enclosure. These seismic restraints or snubbers will accommodate slow thermal expansions during all design duty cycle events and will prevent any rapid motion of the vessel.

The nuclear safety function of the vessel system is to ensure that the core geometry is maintained within acceptable geometrical limits under all normal and postulated abnormal conditions. This safety function is derived from two safety requirements namely, the adequate removal of core heat and the control of heat generation. It also contains the radioactive cooling medium. The first requirement is satisfied by utilizing natural processes to transfer heat out of the core to the environment, in the event of the unavailability of active heat removal capability. The adequate transfer of heat by natural processes is in turn determined by the core geometry, i.e. the core slenderness ratio (diameter to length).

With regard to the second requirement, a change in the core geometry may have an impact on core criticality. A reduction in the core slenderness ratio, for instance (shorter and fatter), may reduce the built-in shutdown margin and, if decreased beyond a certain limit, it may not be possible to maintain the core sub-critical, resulting in an uncontrolled heat generation. This function is primarily controlled by the core barrel structure. The confinement of the helium is a secondary safety function of the pressure boundary system. The reason is because of the passive safety characteristics of the nuclear core (negative temperature coefficient and high temperature resistance), an intact primary coolant pressure boundary is not required to prevent a core melt.

The RPV consists of a main cylindrical section with torispherical upper and lower heads. The upper head is bolted to the cylindrical section and incorporates twelve stand-pipes, and twenty-four penetrations. These penetrations accommodate twelve fuelling actuators and twelve actuators for the cold shutdown system. An additional opening is also provided in the center of the upper head to allow access to the upper core structures. This opening is intended to be used during initial installation and, if necessary, for core barrel changes and major maintenance interventions. All control and instrumentation (C & I) penetrations are in the upper head. These will accommodate four nozzles for instrumentation channels. The lower head is welded to the main cylindrical section and will have ten openings for the refuelling system, twelve for RCS extraction and a large opening in the center for the fuel discharge system. An additional opening is provided for an access/RPV nozzle to the bottom core structures. This opening is only intended for use during initial installation.

The large nozzle forging for the PCUPV and the three support lugs are attached to the lower reinforced part of the cylindrical section. This portion is reinforced to withstand RPV support and manifold nozzle loads. The support lugs provide vertical as well as bottom horizontal support for the RPV. The shell flange at the top of the RPV accommodates the studs for bolting down the pressure vessel closure head. Welded lip seals are incorporated to assure leak tightness. Additional reinforcement is provided at the level of the upper attachment points for the upper seismic restraints. The inner vertical support for the core support structures is provided by a forged ring at the attachment level of the lower head and the cylindrical shell. The primary characteristics for the RPV are provided in Table 3.10.

TABLE 3.10. PRINCIPLE DATA FOR THE RPV

Design pressure	7.0 MPa
Design temperature	350ºC
Inside diameter	6200 mm
Flange outside diameter	6930 mm
Bolt circle diameter	6630 mm
Height of pressure vessel without lid	18 850 mm
Overall pressure vessel height	21 200 mm
Minimum thickness of cylindrical shell	140 mm
Reinforced thickness of bottom part	240 mm
Estimated pressure vessel weight	~600 tonnes
Pressure vessel material	SA 508

The PCUPV, together with the RPV, forms the pressure boundary system of the PBMR. Its main functions are the pressure containment of the helium working fluid and the provision of structural support and alignment for all the power conversion components.

The PCUPV consists of a large horizontal vessel with a short vertical section welded to one end. The other end is welded to the RPV. Removable enclosures are bolted to the top and bottom of the vertical section. A third enclosure is welded to the bottom of the horizontal vessel at a point approximately half way between the centerlines of the two turbo-compressors to form the complete PCUPV. The horizontal vessel with its short vertical end is called the manifold vessel.

The manifold vessel contains the two turbo-compressors in its horizontal section with the power turbine in the vertical portion. The helium transport pipes are also housed in the manifold vessel. The upper vertical enclosure contains the generator, and the lower vertical enclosure contains the recuperator/precooler combination. These enclosures can be removed to allow access for maintenance or replacement. The lower enclosure is provided with reinforced openings for the cooling water pipes and the top enclosure contains reinforced access openings and cable penetrations. The intercooler is located in the third enclosure welded to bottom the manifold vessel. This enclosure is also provided with reinforced openings for cooling water pipes. Two openings are provided at the top of the manifold vessel, above the turbo-compressors, to allow for their hermetically controlled removal. There are also internal pressure boundaries inside the PCUPV. These are the boundaries between the manifold vessel and the generator, the recuperator/precooler and the intercooler enclosures. These enclosures will have a design pressure of 5 MPa versus the 7.2 MPa of the manifold vessel. All components of the PCU will be mounted on support structures attached to the inside of the manifold vessel. Table 3.11 provides the principal enclosure data for the PCUPV.

3.3.2.4. Helium Services [3-12]

Under normal conditions, the helium level in the main power system (MPS) determines the power output level. The ratio between both levels is nearly 1, which means that, for example, 60% inventory circulating in the MPS allows about 60% power output at normal efficiency.

TABLE 3.11. PCUPV PRINCIPAL ENCLOSURE DATA

Manifold vessel	
Design pressure	7.2 MPa
Design temperature	200ºC
Maximum internal diameter (horizontal)	3,800 mm
Maximum internal diameter (vertical)	5000 mm
Maximum nominal wall thickness (horizontal)	165 mm
Maximum nominal wall thickness (vertical)	225 mm
Material	SA 508
Recuperator/Precooler enclosure	
Design pressure	5 MPa
Design temperature	150ºC
Internal diameter	4800 mm
Nominal wall thickness	65 mm
Material	SA 508
Generator enclosure	
Design pressure	5 MPa
Design temperature	150ºC
Internal diameter	4800 mm
Nominal wall thickness	65 mm
Material	SA 508
Intercooler enclosure	
Design pressure	5 MPa
Design temperature	150ºC
Internal diameter	2100 mm
Nominal wall thickness	30 mm
Material	SA 508

Therefore, varying the helium inventory level corresponds directly to load following with high efficiency. For instant load reduction or load rejection, the reactor, recuperator and turbines will be bypassed. For small (±10%) rapid load following, the vane settings on the turbo-machines will be adjusted. Both adjustments will reduce the efficiency of the Brayton cycle and will be used only as short term adjustments. For long term adjustments, the helium will be stored or injected from the storage system in order to restore efficiency.

Increasing the inventory from 20 to 100% will be possible at a maximum rate of 10%/min. This rate can also be achieved during inventory reduction from 100 to 40% and provides flexibility to the PBMR. In the range from 40 to 20%, a positive displacement compressor will be used at a rate of 0.33%/min inventory reduction.

It is possible to operate the PCU without extra compressors to store and retrieve helium. The turbo-units (LPT & HPT) will perform the work required to store helium in the pressure tanks. At the high pressure point in the MPS, ie, after the high pressure compressor, helium will flow out of the system into the storage tanks due to the pressure difference between system and tank. The helium will flow back into the system at the low pressure point in the MPS, ie, before the low pressure compressor, due to the pressure difference between tank and system. Ten tanks of similar design are proposed, providing a total storage capacity of ~56 m^3. Emptying or filling the tanks will cause the temperature in the tank to rise or drop because of the adiabatic compression or expansion of the helium in the tank.

The two positive displacement compressors have been dimensioned such that they reduce the helium inventory in the primary cycle from 40 to 20% in one hour, at 20 l/s per compressor. This requires 65 kW of work per compressor, to which the efficiency of the motor has to be added. It will take nine hours to reduce the inventory in the primary cycle further from 20 to 2%. The helium pumped out of the primary cycle will be stored in the last five tanks, which will then be at a pressure of 4.38 MPa.

If the compressors start filling a tank to a pressure which exceeds the design pressure of the tank, a relief valve will vent the excess of helium into the room where the tanks are located. The room is connected through a filtering system to the reactor hall, through which the overpressure can be reduced. The filter (active carbon) will sieve dust, iodine and other non-noble gases from the helium flow. The ventilation system of the reactor building subsequently vents the helium with its contaminants via the building filters to the atmosphere. An overpressure in the MPS (which can only occur when the tanks are all empty) will be relieved to a storage tank, preferably the highest pressure tank, in order not to disturb the order of filling and emptying. If this tank becomes over-pressurized, it will reduce its pressure as previously described.

The function of the helium purification system is to remove chemical and particulate contaminants from the primary coolant in order to provide the necessary degree of helium purity. The bypass flow to be purified is 50 kg/h. The flow will be extracted at the high pressure point in the system (at the recuperator/precooler volute) and injected into the MPS at the low pressure point of the system (before the low pressure turbo-unit). The bypass flow will first be filtered in order to retain particulate contaminants. It is followed by a heater that raises the temperature of the helium to approximately 250°C. This is the optimum operating temperature for the copper oxide reagent which oxidizes the contaminants HT, H_2, CO and CH_4. After cooling down the helium flow to about 40°C, a molecular sieve absorbs the oxidants HTO, H_2O and CO_2. At this point the bypass flow is considered clean and is led back to the MPS.

In contrast to previous HTR-designs, the MPS will not be cleaned for N_2 impurities (which stems primarily from the newly supplied helium) and also not for the inert fission gas isotopes of Xe and Kr. Compared to the German MODUL design, this results in a steady state coolant activity that is about six times higher after 30 years full power operation. In case of release to the atmosphere of the total PBMR inventory, this would amount to less than twice the activity at which the HTR-MODUL was designed (in the MODUL calculations, the absorbed fission gases in the filter were also expected to be released). In such an event, the importance of other nuclides (I-131) would be much larger, but still well under the standard dose limits for the population.

The purification system must be regenerated when the copper oxide has been reduced to copper, or when the molecular sieve is spent. An increase in certain contaminants downstream of the purification system will indicate when regeneration is necessary. Before regeneration, the purification system is disconnected from the primary system and the pressure is relieved. Regeneration of the molecular sieve is performed by back-washing with heated, new helium from the helium supply. In order to regenerate the catalyst, oxygen is injected from a small bottle intermittently and discontinuously into the purification flow.

After regeneration, the system will be evacuated in order to remove the last traces of the oxygen used for regeneration of the reagent. The gases used for regeneration are stored in a waste tank and released to the environment when the activity is sufficiently low. During the regeneration time (typically 20 h.) the MPS will not be purified. It is expected that a regeneration will be necessary once every half year.

3.3.2.5. Cooling systems [3-12]

There are several cooling systems designed for the PBMR plant as shown the following matrix.

COOLING SYSTEMS MATRIX

System	Sub-system	Active (Seawater) Heat Sink	Passive (Air)	Support Cooling Tower (Air)
Main Power System (MPS)	Pre-Intercooler	♦		
	Inter-Intercooler	♦		
	Generator	♦		
	Components	♦		
Reactor Pressure Vessel Conditioning System (RPVCS)		♦		♠
Core Conditioning System (CCS)		♦		♠
Fuel System	Fuel Handling (FH)	♦		
	Spent Fuel (SF)		♥	
Reactor Cavity			♥ ♥ ♥	
Helium inventory Control System (HICS)		♦		
Air Compressors		♦		♠
Heating, Ventilation & Air Conditioning (HVAC)		♦		♠
Initial Cleanup System		♦		♠

The function of the reactor cavity cooling system (RCCS) is to dissipate the heat from the reactor cavity during normal operation, including shutdown. The system also removes the decay heat during a loss of the heat transfer functions of the PCU (loss of forced cooling). The objective is to prevent the reactor vessel (including attachments), RPV supports, instrumentation and the concrete walls from exceeding their design temperature limits for all modes of operation. Natural processes including thermal radiation and convection, are relied upon to transport the heat from the non-insulated reactor vessel walls to the cooling panels of the RCCS.

No reliance is placed on the RCCS to protect the nuclear fuel from exceeding its maximum design temperature. The heat transport through the concrete structures of the reactor building is sufficient to ensure the maximum core temperature is not exceeded in the event of the RCCS is not available. The main purpose of the RCCS is to protect the investment into building and systems and it does not perform any nuclear safety functions. The RCCS is subject to a maximum heat flux during power operation of ~1.2 kW/m^2, and in the decay heat removal operation to ~3.6 kW/m^2.

The RCCS includes three independent systems each consisting of a low pressure, closed loop, natural-convection driven, self-acting water-based cooling system, with external water to an air heat exchanger. The systems do not rely on any active components such as pumps for operation. Each system provides 100% cooling capacity. In the case where all the cooling units are lost, the heat of the reactor is absorbed by heating up and then boiling off the water in the system. The systems are sized to provide this cooling function for up to three days.

A three-train configuration is achieved by joining every third raiser and down-comer panel to the same inlet and outlet headers, which are in turn connected to three separate heat exchangers. Each system consists of:

- heat transfer cooling raiser panels arranged around the RPV, which transports hot water to outside the reactor cavity,
- cooling tubes with ribs arranged in a cooling tower or other configuration, so that these can dissipate heat to the atmosphere with natural convection of air and at the same time cool down the hot water,
- down-comer panels, which transport cold water from outside the reactor building from the cooling panels to the reactor cavity,
- bottom collector inside the reactor cavity, which connects raiser panels and down-comer panels,
- upper collector inside the reactor cavity, which connects raiser panels and the collecting panels,
- water fill connections and maintenance drainage connections.

The RCCS is designed to remove dissipated heat from the reactor cavity to the atmosphere (Table 3.12). Under normal operations, the temperature of the RPV is less than 300ºC. Due to thermal radiation from the RPV surface and the natural convection of the air in the reactor cavity, heat is transferred to the raiser panels, where the water becomes heated and flows upward by thermal buoyancy in each closed water circuit. This hot water is cooled in the cooling tubes of each water-to-air heat exchanger (located on the external air duct outside the citadel), thereby transferring its heat to the atmosphere. Subsequently, the cooled water flows back down through the down-comer panels. Under normal operating or hot shutdown conditions, about 200 kW of heat will be transferred from the reactor cavity to the atmosphere.

In case of reactor shutdown or a loss of all forced cooling, or a pressurized loss of forced cooling, the temperature of the RPV is raised to the value of ~350ºC. When the plant is in a shutdown condition with the CCS and/or RPVCS in service, the heat load is less than at normal operation. In this case, decay heat from the reactor is removed with the self-acting RCCS system at a rate of 1000 kW/h. In case of failure of all three water-to-air heat exchangers, the water in the raiser panels will be converted over a period of three days to low pressure steam, which will be released into atmosphere from the roof of the module building.

TABLE 3.12. RCCS DESIGN VALUES

Heat Transfer Conditions between Reactor pressure vessel (RPV) and Cooling Panels	
Emissivity	0.762
Geometry factor between RPV and CPA φ_{12} [-]	0.99
Mean RPV-Temperature	350ºC
Mean CPA-Temperature	110ºC
Heat Transfer Coefficient, Natural Convection	5.0 W/m^2 K
Heat Flux Density	5.5 kW/m^2
Mean System Pressure p	5.5 bar
Water Mass flow	12.0 kg/s
Geometry of Cooling Panels	
Down-comer Panels:	
Total Number	3 × 6
Total Height	12.0 m
Tube Diameter	80.0 mm
Heat Transfer Panels:	
Total Number of Tubes	3 × 108
Tube Diameter	50.0 mm
Collecting Panels:	
Total Number	3 × 6
Total Height	2.5 m
Tube Diameter	80.0 mm
Data of Water/Air Coolers (per cooler)	
Total Number of Cooling Tubes	125
Total Length	15 m
Inner Diameter of Finned Tube	50 mm
Outer Diameter of Finned Tube	110 mm
Thickness of Fins	2 mm
Distance in-between Fins	20 mm
Heat Transfer Coefficient (natural convection in the atmosphere)	~8.5 W/m^2 K
Overall Cooling Data per System	
Cooling Capability under Normal Conditions	200 kW
Cooling Capability under Depressurization Accident	1000 kW

The core conditioning system (CCS) has two functions; the 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 (Fig. 3.5). The CCS consists of a blower, water cooler, gas to gas heat exchanger (recuperator) and valves and pipe connections to allow inlet/outlet gas mixing. The function of the gas mixing and recuperator is to decrease the temperature of the incoming gas to acceptable levels and the function of the water cooler is that of the final heat sink. The blower creates the necessary mass flow.

There are two CCS, each with a capacity of 50%. Each CCS is installed in a separate pressure vessel coupled to the primary pressure boundary via two 550 mm ID pipes. These pipes contain the gas transport pipes in a co-axial configuration. The hot and cold gas transport pipes are coupled directly to the hot and cold plenums in the core structures respectively. The operating pressures of the CCS pressure vessels and connecting pipes are the same as the primary pressure boundary as they are directly coupled. The pressure differential over the gas transport pipes and other systems of the CCS will be limited to the internal pressure drops due to flow. All the other CCS components are installed in the CCS pressure vessel.

The water coolers are fed by an intermediate loop to limit the amount of water that can leak into the system. However, water in-leakage is only a concern when the CCS is operated at very low primary system pressures. During most of its operating time, the gas side pressures will be significant higher than the water-side pressures.

The PBMR is equipped with a reactor pressure vessel conditioning system (RPVCS) with the function of maintaining the RPV at a homogeneous temperature, in order to limit thermal stresses due to localized high temperatures, and to maintain the RPV operating temperatures in range of PWR qualified materials (Fig. 3.5). The RPVCS consists of a blower, water cooler, valves and pipe connections. The water cooler will remove heat from the system in order to control the RPV wall temperature at approximately 280 to 300ºC. The blower creates the necessary mass flow.

There are two RPVCS, each with a capacity of 100%. Each RPVCS is installed in a separate pressure vessel coupled to the primary pressure boundary via two 150 mm ID pipes. All the other components are installed in a separate RPVCS pressure vessel. The flow channels and valves are installed in a barrel type structure inside this pressure vessel. The water coolers fit through openings in the vessel lid. The blower and its electrical motor are contained in an enclosure vessel and are fitted as a unit on top of the vessel lid.

3.3.2.6. Fuel handling and storage system (FHSS) [3-12]

The functions of the FHSS are:

— Loading of the core cavity with graphite spheres.
— Loading of new fuel spheres into the core.
— Removing erroneously discharged fuel spheres from the graphite sphere system.
— Preventing erroneously discharged graphite spheres initiating the loading of new fuel spheres, via radiation sensors fitted to the delivery line to the spent fuel storage tanks. A detected graphite sphere going the wrong way may not initiate the loading of a new fuel sphere.
— Removing fuel and graphite spheres from the discharge tube.
— Separating damaged spheres.
— Separating fuel, absorber and graphite spheres.
— Recirculating graphite spheres.
— Recirculating partially used fuel spheres through the core.
— Measuring burn-up of partially used fuel spheres, and discharging spent fuel spheres into the spent fuel storage system.
— Defueling and refueling of the core, by transfer of the core inventory from the reactor into separate graphite and fuel storage tanks located in an area adjacent to the reactor, during maintenance intervention requiring the venting of the main power system to atmosphere.
— Reloading the core from these tanks during refueling of the core.

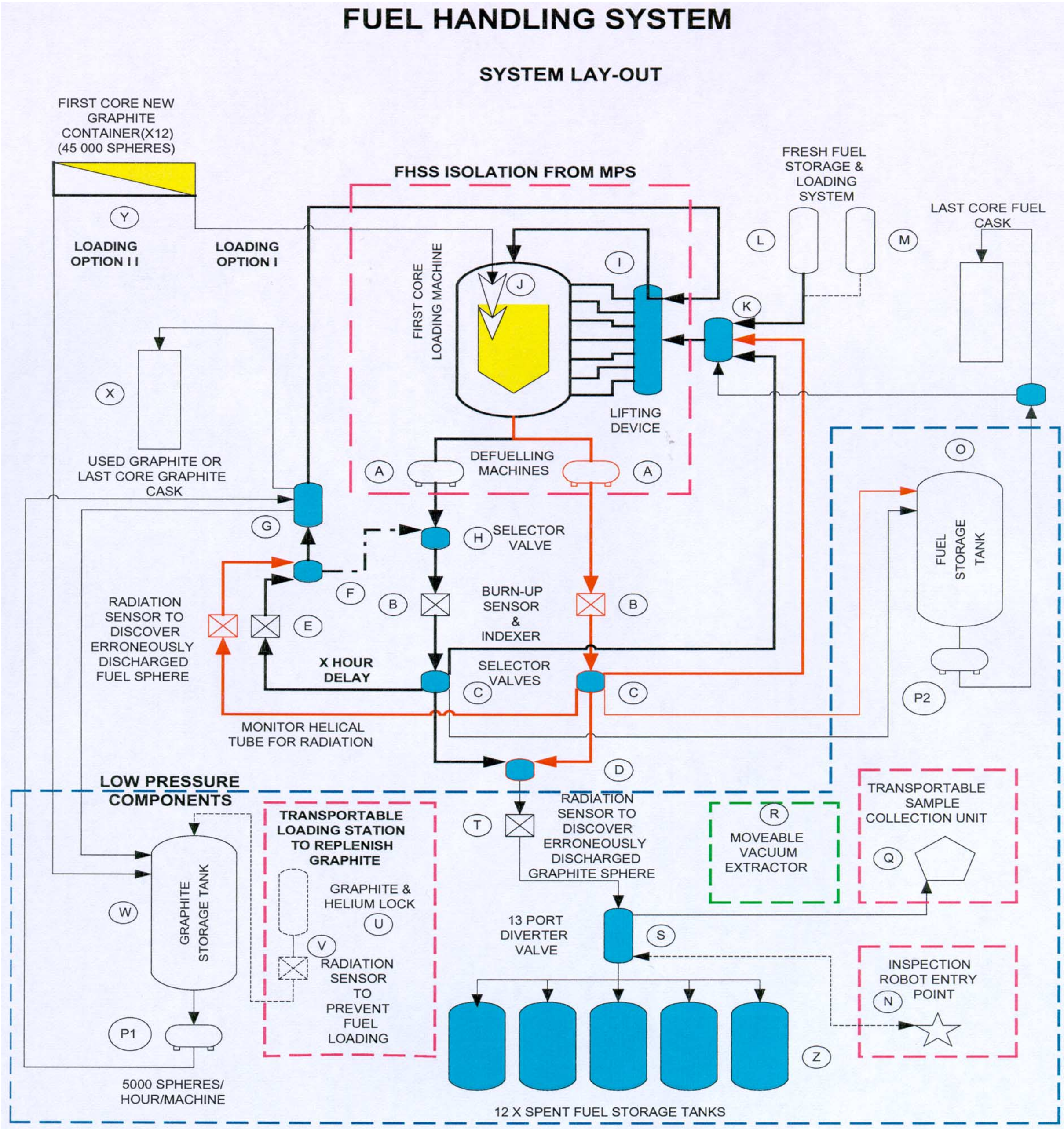

FIG. 3.12. FHSS system arrangement.

The PBMR core is to be operated according to the “multi-pass” fueling scheme; which means that fuel spheres are moved through the core more than once, e.g. 10 times, to reach the final burn-up level. The purpose of the multi-pass fueling scheme is to provide for a uniform distribution of the burn-up within the core, and thereby flattening the axial neutron flux profile and maximizing the thermal power output of the modular unit. A primary purpose of the FHSS is to keep the fuel and graphite spheres separate, after being separated by the selector valve that is activated by the radiation sensors (B). The fuel and graphite spheres are brought together above the pebble bed by the supply tubes arranged in a specific order to ensure the two-zone core loading with graphite spheres in the center and fuel spheres in the annulus surrounding the graphite.

The FHSS (Fig. 3.12), for the realization of the multi-pass fueling scheme, consists of the fresh fuel storage and feeding system, the fueling and defueling system, including the discharge equipment, the spent fuel system, and the fuel lifting system. The storage system consists of the new fuel storage, graphite storage, spent fuel storage and the damaged fuel storage. See Chapter 8 for details of the spent fuel storage system.

The main parts of the fuel handling system are located in shielded, individual compartments below the reactor. The spent fuel storage tank system is designed as a life-time spent fuel store and post operations intermediate store and is located in the lower part of the reactor building.

In the multi-pass fueling scheme, the recirculation of partially used fuel spheres is the primary function of the FHSS. The majority of the discharged spheres (approximately 4880 per day) is recirculated to the top of the core. The daily requirement for new fuel spheres is approximately 370 per full power load day, to replace the fully burned spheres that are removed. Fuel spheres are forwarded in horizontal and vertical tubes partly by gravity, but predominantly pneumatically, by mainly using primary coolant at primary system pressure. Monitoring of fuel sphere movement and of buffer and charge lock fill levels is performed with the aid of measurement and counting instruments, whose signals provide input to the control system which actuates the operating components and valves of the forwarding system.

The fueling and defueling system is mainly used to increase the effective use of the fuel, by evaluating the burn-up of the fuel spheres, and to recycle (multi-pass) them back to the core or to send the spent fuel spheres to the spent fuel system (Table 3.13). This system is also used to remove damaged or broken fuel spheres from the system. The system must also separate the fuel and the graphite spheres during defueling and place fuel and graphite spheres in their respective storage tanks. Fuel and graphite sphere carrier lines are always separate from each other to prevent fuel and graphite sphere mixing.

TABLE 3.13. DESIGN DATA FOR FUELING/DEFUELING SYSTEM

Fueling and Defueling	
Number of fuel elevator pipes to the core	9
Number of graphite elevator pipes to the core	1
Number of spheres (fuel and graphite) handled by system per day (EFPD)	4000–4880
Number of spheres (fuel and graphite) handled by system during defueling	10 000/h
Operating pressure	<7.0 MPa
Anticipated damage and failed fuel spheres at discharge tube/year	~180 (0.038 m^3)

3.3.2.7. Plant maintenance

Maintenance and service operations for most of the major equipment will take place at intervals of six years. A significant requirement of the PBMR maintenance and service strategy is to limit radiation exposure to personnel. This requirement is being incorporated into the design of the primary system building by requiring appropriate access to irradiated or contaminated equipment, while keeping radiation exposure ALARA.

The building design provides sufficient space (service hall) for the complete removal of components (e.g. turbo-compressors, generator, power turbine, recuperator, etc). Limited decontamination and hot workshop facilities are also provided, however, major components will be maintained offsite by the OEM. The following items will require major maintenance actions during the outage period each six years [3-12].

TABLE 3.14. EXPECTED MAINTENANCE REQUIREMENTS

ITEM	ACTION
Reactor vessel	Inspection of welds from the outside of the reactor vessel. (remote control inspection)
Turbo-compressors	Remove and replace with a service unit. The removed unit will be decontaminated and serviced (replacement of blades) for replacement during the next maintenance operation in a reactor module requiring them.
Power turbine	Inspection after 6 years and replacement of blades after 100 000 hours of service.
Generator	Attempt in-situ inspections, otherwise remove. Inspect and service in controlled area.
Fueling and defueling equipment	The building is designed in such a way that maintenance and repairs to most fueling and de-fueling equipment can be performed during normal operation of the plant. Access to high radiation areas is possible under special conditions.

3.4. PBMR SAFETY AND LICENSING

The safety approach established for the PBMR is taken from IAEA-TECDOC-801, Development of safety principles for the design of future nuclear power plants, and includes:

General objective:

> To protect individuals, society and the environment by establishing and maintaining in nuclear power stations an effective defence against radiological hazard.

Operation principle:

> To ensure, in normal operation, that radiation exposure within the plant and due to any release of radioactive material from the plant is kept as low as reasonably achievable (ALARA) and below prescribed limits, and to ensure mitigation of the extent of radiation exposures due to accidents.

3.4.1. Safety

3.4.1.1. Safety philosophy for PBMR

The safety philosophy of the PBMR differs from the traditional LWR philosophy in the approach taken to achieve effective defence against radiological hazard. For the PBMR, the fundamental principle in the strategy of defence in depth is to apply the principle of accident prevention (IAEA-TECDOC-801). A primary means for preventing accidents is to strive for such high quality in the design that deviations from the normal operational states are within prescribed design limits and do not weaken the capability of the nuclear fuel to retain radionuclides. This fundamental principle in the strategy of the defence in depth applied to the PBMR, allows the demonstration that for internal event accidents of Level 5, 6, and, 7 of the International Nuclear Events Scale, can be excluded. For external challenges, e.g. earthquakes, floods or airplane crashes etc, the principle is helpful but not sufficient, and therefore the traditional philosophy is applied [3-12].

ESKOM and the National Nuclear Regulator of South Africa (NNR, previously CNS) are currently in the process of developing a common safety design philosophy to apply to the PBMR. The key objectives of this statement would be:

> "To assure the nuclear safety of the PBMR so as to protect individuals, society and the environment by establishing and maintaining an effective defence against radiological hazards resulting from operation of the PBMR over its entire lifecycle.
>
> To ensure that in all operational states radiation exposures within the PBMR or due to any planned release from the PBMR is kept below prescribed limits and as low as reasonably achievable, and to ensure the mitigation of the radiological consequences of any accidents.
>
> To take all reasonable practicable measures to prevent accidents at the PBMR and to mitigate there consequences should they occur; to ensure, with high level confidence that, for all possible accidents taken into account in the design of the PBMR, including those of very low probability, any radiological consequences would be minor and below prescribed limits; and to ensure that the likelihood of accidents with serious radiological consequences is extremely low."

In this regard, the following "PBMR Fundamental Safety Design Philosophy Statement" is currently (7 March 2000) under consideration.

(1) "The Fundamental Safety Design Philosophy is based on the premise that the 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 Fundamental Safety Design Philosophy has been meet with adequate margins. The design has been systematically analysed 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.

(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.

(7) The PBMR design minimizes the generation of radioactive waste throughout its lifecycle (including decommissioning) and includes appropriate processing, conditioning handling and storage systems.

(8) Over its entire lifecycle the PBMR is supported by a quality management system."

The PBMR Safety Case suite of documents builds on the above philosophy.

3.4.1.2. Safety characteristics and plant barriers [3-12]

The plant's fundamental safety characteristics include:

— The utilization of a small normal operational excess reactivity, made possible by continuous fueling and defueling,
— 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.

The small excess reactivity at normal operation is a result of a core that is always in the equilibrium state due to continuous fueling and defueling. This means that no excess reactivity is needed in order to compensate for periodic excess fuel when new fuel is added in batches. Excess reactivity is therefore solely designed to allow for Xenon fluctuations and load following conditions.

The high temperature radionuclide retention capability is provided by the fuel kernel coatings consisting of multiple layers of PyC and SiC. These coated fuel particles have demonstrated excellent capability in containing radiologically significant gaseous and solid fission products under elevated temperature conditions (see Section 2.2.2 for details).

The large negative temperature coefficient of the fuel 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 U-238. This, together with the negative moderator temperature coefficient, add up to a strong total negative reactivity coefficient for temperature which means that the reactor will quickly counteract a rise in temperature with a reduction in power.

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. 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 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 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 Division 1. 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 is 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 containment atmosphere, reducing off site releases.

3.4.1.2. Accident prevention and mitigation

Simplicity of the reliance on passive safety features and inherent characteristics allow a simple overall PBMR plant design. The PBMR modules are operated as independent power units and interaction between them is minimized. The layout of the PBMR eliminates unnecessary components and systems, which simplifies normal and emergency operating procedures, inspection, testing, and maintenance. Reliance on control room and operating staff is minimized, since no operator actions are required to prevent fuel damage. Similarly, errors by the operating staff cannot upset the safety characteristics of the PBMR.

The continuous fueling of the reactor implies that no excess reactivity is necessary in order to compensate for burn-up effects. Nevertheless, a certain margin is required for the reactor control and to compensate for changes in the xenon concentration following changes in reactor power. A fast acting control system will serve to keep the reactor within normal operating limits.

Reactor cooling will be accomplished by the PCU or by the RCCS. The PCU is an active system that operates during power generation and provides the primary shutdown cooling when available. In the event that active heat removal systems are unavailable, the core design ensures a passive residual heat removal capability. The core geometry, limited core diameter, low thermal power rating, low power densities and the passive cavity cooling system limit the maximum core and fuel temperatures. Under these conditions, heat is transferred through the RPV wall by thermal radiation and natural convection to the cooling surfaces of the RCCS. The reactor vessel walls are non-insulated to facilitate this process.

3.4.2. National nuclear regulator (NNR) licensing approach [3-13]

The South African nuclear regulatory authority, the NNR, is currently at the first licensing stage of the safety review of the PBMR. ESKOM has officially submitted an application for a staged licensing process requesting that the first stage considers the concept design without any specific site. The NNR will evaluate the acceptability of the safety bases for the proposed PBMR.

The current PWR (Koeberg station) operating reactors were designed, manufactured, constructed and commissioned according to general design criteria and specific design rules and standards which were prevailing at the time e.g. US 10 CFR 50 and other French rules. These design standards have general international acceptance. The NNR licensing approach adopted at the time of licensing the Koeberg reactors required:

— That the design basis of the plant should respect prevailing international norms and practices (as indicated above) and
— That a quantitative risk assessment should demonstrate compliance with the NNR fundamental safety standards.

Based on the outcome of the safety assessment process to demonstrate compliance with the above criteria, conditions of the license were set which included requirements for maintaining a valid safety assessment, configuration and modification control and a series of general operating rules covering operation, maintenance, inspection, radiation protection, waste and emergency planning.

It has been proposed that a similar approach should be adopted in respect of the PBMR. However, unlike the LWR situation, the same level of international consensus has not yet been developed in respect to general design criteria and design rules which can be used as an "off the shelf' package for defining the design basis of the PBMR. Nevertheless rules and criteria have been developed during the licensing of some HTGRs e.g. in Germany.

Thus, as an integral part of the PBMR Licensing process it will be necessary to establish the general design criteria and associated design rules to assure that the PBMR design complies with the current CNS risk criteria which provide for, as a minimum, the same degree of protection to the operator, public and environment that is required for the current generation of nuclear reactors, and respects the observed societal trend to require higher levels of safety with time. As an initial (and key) aspect to provide regulatory guidance to the PBMR design , the NNR is currently considering the PBMR Fundamental Safety Design Philosophy Statement of Section 3.4.1.1.

3.4.3. Current licensing status

ESKOM has formally applied for a license for the PBMR design. This application initiated a programme aimed to achieve the initial "licensability' statement on the PBMR in early 2000. This will include the safety criteria that the plant must meet, the general and specific design criteria, the event list, the classification systems, and a review of the design basis. These activities are underway with the involvement of international consultants (both to support the NNR and ESKOM) [3-8].

Also, the South African Government formally requested the IAEA to investigate and advise them on the technical, economic, safety and proliferation aspects of the PBMR. In regard to this request, the IAEA investigative process was initiated in 1999 with the licensing phase of this review initiated in early 2000. Figure 3.13 provides the sequence for the PBMR licensing related programme.

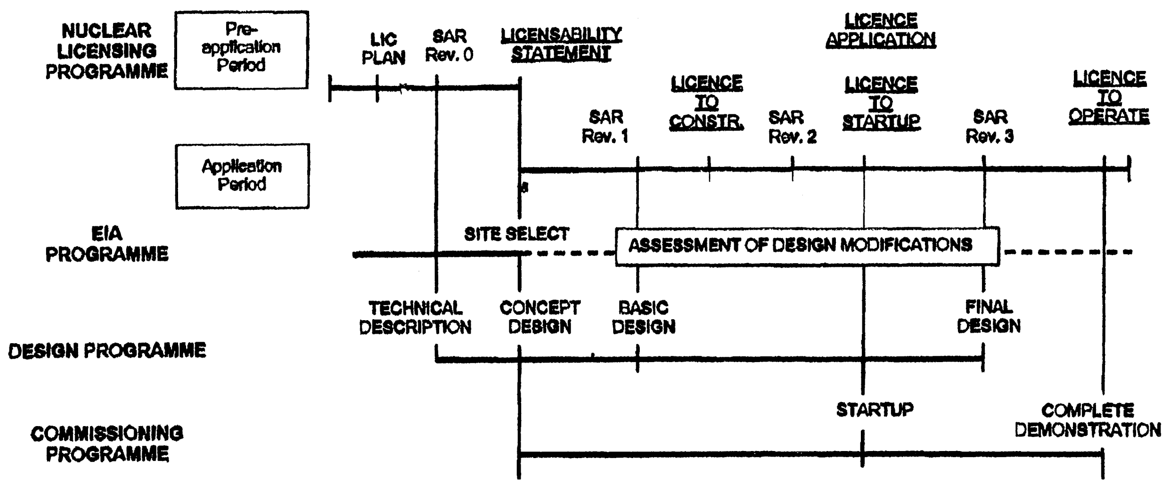

FIG. 3.13. PBMR licensing related programme sequence.

3.5. PLAN FOR PBMR PLANT DEPLOYMENT

3.5.1. Current status

The PBMR programme has achieved the following milestones:

- Application for nuclear license of the design with the NNR.
- Initiation of the environmental impact assessment (EIA) process to allow approval of the first site.
- Establishment of a single programme team (currently ~150 full time staff) outside the utility head office.
- Finalization of concept design.
- Prequalifiaction of key suppliers.
- Negotiations with potential joint venture partners.
- Initiation of tender process for detail design of long lead items.

This is intended, when combined with public and stakeholder consultations, to enable the decision on the potential construction of the first reactor to be taken in due course. This would include full consideration of the development, construction and operating costs, design parameters and the site location.

Under South African legislation there is a requirement for a full EIA for any new power plant. In light of this, ESKOM has started the process and has called for potential suppliers to submit their capabilities. A formal inquiry to the qualified suppliers is to be issued in early 2000. As part of the EIA process there will be a number of coastal sites considered, as well as the overall societal impact as to the value of the PBMR project. The decision has now been taken to only consider the existing Koeberg site for the first unit.

TABLE 3.15. EXAMPLES OF AREAS CURRENTLY UNDERGOING DESIGN RE-EVALUATION

ITEM	CONSIDERATION
Use of control rods rather then SAS for the Reactor control system	• Control rod material has been previously qualified • Reduce lower core structure complexity
Inlet/outlet manifold arrangement simplification	Repositioning of core inlet/outlet piping in order to reduce the core bypass flow has led to the need to move away from a single RPV nozzle.
Larger diameter turbo-machine volutes	To minimize differential pressure losses within the PCU
RCCS design re-consideration	Improve system dependability by optimizing active “non-safety” components versus passive “safety” components
Re-evaluation of core height	To optimize safety considerations such as shutdown margin, limiting the maximum fuel temperature and central column flow behaviour

The PBMR status addressed herein (as Section 3.2.2.) depicts the technical plant description as of April 1999, and includes the conceptual design phase of development. The PBMR programme is now progressing into the basic design phase. It is anticipated that this phase of development will result in certain changes to the plant not recognized during the 1999conceptual design.

Some areas currently under consideration for re-design in order to optimize plant safety, cost and/or operational efficiency are included in Table 3.15.

Those areas of the plant that are not covered to significant detail are those which employ standard "off the shelf" equipment and do not contribute significantly to the cost (e.g. the air compressor system). However, even in these cases, a performance specification has been generated. The load following capability has been a very specific emphasis of the work to date. There is an engineering simulator (based on a G2 platform) which has been developed to allow non-real time dynamic analysis. There are two other system analysis tools being used in the cycle development (FLOWNET and a MATHCAD based model). While these tools are sufficient to handle the concept and basic design phases, they are not seen to be appropriate for the detail design phase and a new engineering simulator has now been developed.

Extensive work has been accomplished on the maintainability of the design, and all components are classified by their life expectancy and difficulty of repair. This leads to some components having very easy access (e.g. the bypass valves) which can be maintained without breaching the helium circuit, and some which can be changed, but only in the same manner as the changing of the steam generators on a PWR (e.g. the recuperator). This analysis has allowed the maintenance cycles and removal routes to be established along with their impact on building design and crane requirements.

Test rigs are under construction in key areas. These are required to demonstrate very specific features that would be valuable to include in the design. Table 3.16 provides major technical areas under design and evaluation and associated suppliers.

TABLE 3.16. TECHNICAL AREAS/SUPPLIER

Technical Area	Supplier
Fuel Handling Design	PBMR Project/OKBM
Core Structure Design	AEA Technology
Core Design & Safety Studies	PBMR Project/FZJ, Juelich
Control & Shutdown Systems	PBMR Project/OKBM
Simulation	PBMR Project/Potch/AEA Technology
Operating Modes	PBMR Project
Event/System Classification	TUV, Hannover
Shielding & Pressurization	NRG Holland
Power Turbine Generator	PBMR Project/OKBM
Critical core Assembly	RRC Kurchatov, Moscow

The manufacture of fuel is a key element of the PBMR programme, and the quantities required exceed any previous HTGR project. The PBMR requires a new fuel manufacturing facility and the present intention is to construct it at the South African AEC site, in the complex that built the fuel for the Koeberg plant. This project has identified the layout for such a line (to initially manufacture 1.4 million spheres/year) and the associated equipment specifications. Discussions are now being held with vendors for the equipment.

In terms of the actual fuel technology, ESKOM has involved a number of suppliers and potential suppliers in various ways (such as partnership agreement, commercial contract, or, in some cases, negotiations are still underway). The AEC, under ESKOM contract, has initiated laboratory work with fuel kernels. This work is in support of the external technology that is being obtained on fuel particle development.

3.5.2. Schedule

Table 3.17 provides the schedule and target dates that have been established for the development and deployment of the first PBMR unit.

TABLE 3.17. PBMR PROJECT SCHEDULE

Date	Objective
November 1999	Safety Analysis Report (rev. OA) Complete
June 2000	Safety Analysis Report, (rev. 0) Complete
April 2000	“Feasibility of Licensing” statement from NNR
October 2000	Detailed Design Baseline Complete
July 2001	Site Available for First Unit
July 2001	Start of Construction on First Unit
January 2004	Start of hot non-nuclear testing
October 2004	First Criticality
December 2005	Commercial Operation of First Unit

The schedule in Table 3.17 requires close co-ordination of approvals from the South African Government as delineated in Table 3.18.

TABLE 3.18. SCHEDULE OF SOUTH AFRICAN GOVERNMENT APPROVALS

DATE	APPROVAL/RELEASE*
March 2000	Government “Approval in Principle”
July 2001	Release* for manufacturing and construction
October 2004	Release* for nuclear hot commissioning
2006	Release* for commercialization

Release* = Approvals from shareholders, government and NNR.

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Chapter 4

REVIEW OF THE GAS TURBINE-MODULAR HELIUM REACTOR (GT-MHR) PLANT

It has long been recognized that substantial gains in the generation of electricity from nuclear fission can be obtained through the direct coupling of a gas turbine to a HTGR. This advanced nuclear power plant is unique in its use of the Brayton cycle to achieve a net electrical efficiency in the range of 47% combined with the attendant features of low initial capital costs due to plant simplification, public acceptance resulting from the safety attributes of the HTGR, and a reduction in radioactive wastes [4-1]. Although an evaluation of this concept was initiated in the USA over twenty years ago, it was terminated due to the limited technical capabilities existing at that time primarily in the areas of magnetic bearing, compact plate-fin heat exchanger and turbo-machine development [4-2]. Recent technological advances in these components along with the international capability for their fabrication and testing has fostered a renewal in this plant concept as a future source for the production of electricity.

General Atomics (GA) and the Russian Federation Ministry for Atomic Energy (MINATOM) entered into a memorandum of understanding (MOU) in 1993 to co-operate on the development of the GT-MHR with the goal, following design and development, to construct, test, and operate a prototype in the Russian Federation. The primary objective at that time was to develop the GT-MHR as an export commodity for both the USA and the Russian Federation, as well as future deployment in these countries. Starting from this nucleus, it took another two years to establish a funded, well structured and organized program, during which the near term goal was changed to utilize the first deployment for destruction of weapons grade plutonium in the Russian Federation. Concurrent with the development of the program structure, negotiations were initiated to broaden the base of participants in the program. In January 1996, Framatome of France joined the GA/MINATOM cooperative program. In January 1997, Fuji Electric of Japan became the fourth participant and sponsor. Through 1998, funding support for work in the Russian Federation has come from GA, Framatome, Fuji Electric and matching funds from MINATOM, i.e. 50% of the work is funded by MINATOM [4-3]. Starting with fiscal year 1999, the programme is supported by the US Government.

4.1. INITIAL DEVELOPMENT ACTIVITIES

The gas turbine design was originally conceived in the 1970s, and had the potential for high plant efficiencies (40%) under dry cooling conditions and the possibility of achieving even higher efficiency (~50%) when bottoming cycles were incorporated. The focal point for the assessment of this concept was a potential HTGR Gas Turbine plant of 2000 MW(t)/800 MW(e) with a core outlet temperature of 850°C [4-4]. This large direct cycle unit was enclosed within a prestressed concrete reactor vessel (PCRV). The principal assessment finding of this plant was that it was feasible, but with greater development risk than that of the steam cycle HTGR with no incremental economic incentive to offset its development in favor of the steam cycle plant.

4.1.1. The MHTGR

In the 1980s, evaluation of the reasons for the dearth of new nuclear plant orders in the USA led to the conclusion that smaller, simpler nuclear power plants with inherent safety characteristics were needed for public acceptance (see Section 6.11.1). The modular high temperature gas reactor (MHTGR) was conceived to meet this need. The MHTGR employed a steam cycle power conversion system in common with the prior large HTGR plants. The plant design had unparalleled safety (meltdown proof), was considerably simpler because of the absence of the need for complex safety systems (completely passive decay heat removal system), and at 350 MW(t)/135 MW(e) per module was much smaller than the 1000 + MW(e) nuclear plant size common in that era. Furthermore, the design maintained fuel temperatures under accident conditions below 1600°C using completely passive means to retain fission products within the coated particle fuel. This provided substantial additional simplifications; the need for secondary containment was eliminated, as were the needs for emergency plans for sheltering and evacuation of the public. There was, however, increased importance for the coated particle fuel to meet high quality standards [4-5].

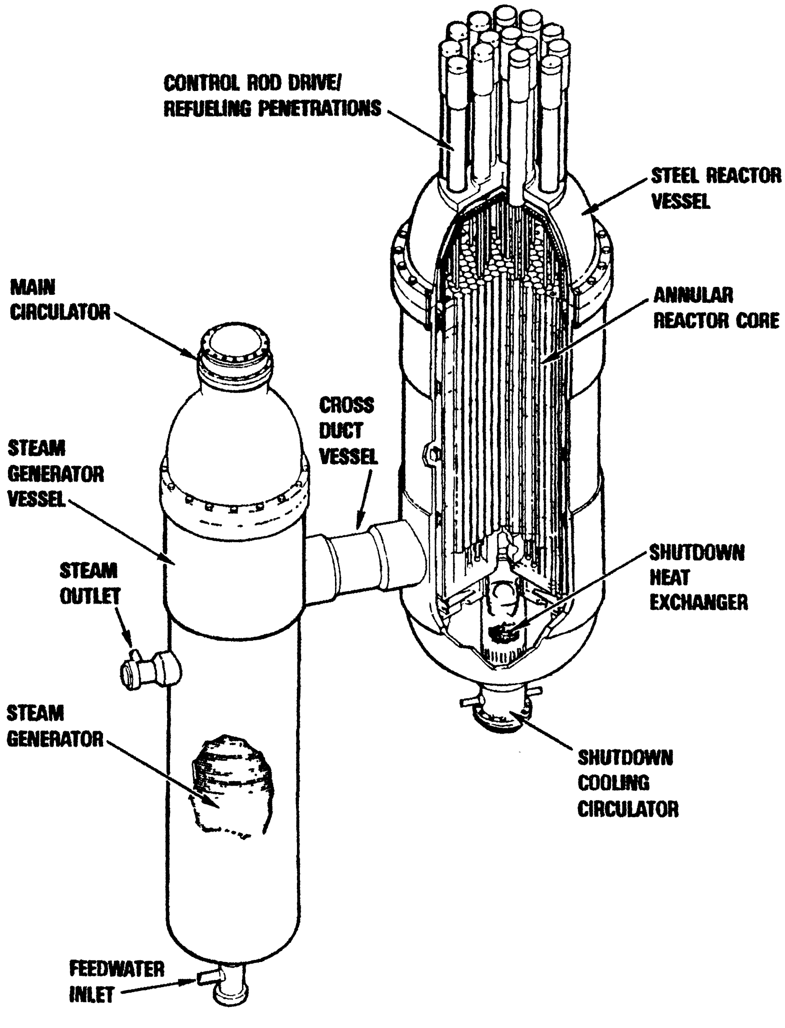


FIG. 4.1. MHTGR primary system arrangement [6].

Although the initial modular HTGR was of the small pebble bed design (Siemens/Interatom's HTR-MODUL), GA's MHTGR employed the prismatic block type core in an annular core configuration allowing for a higher power rating. The move toward the small modular plant was also very evident in the 1980s within the Russian HTGR programme with their initiation of the VGM (see Section 6.7.1). This small (200 MW(t)) plant featured a pebble bed reactor steam cycle system within steel pressure vessels and featured many characteristics similar to the German MODUL and US MHTGR units such as maintaining the fuel temperature limit below 1600°C.

Economic evaluation of the MHTGR indicated that a reference plant of four modules each of 350 MW(t)/135 MW(e) had power generation cost projections that were non-competitive with equivalent sized coal and LWR plants. A larger MHTGR module design was then developed rated at 450 MW(t)/175 MW(e) per module retaining the same safety and simplification features. The reference four module plant using this module size had projected generation costs essentially equivalent to comparably sized coal plants and large LWR plants. However, the technology for highly efficient combined cycle natural gas fired plants became the low cost new generation alternative. Being equivalent in cost to coal and LWR plants was judged to be an insufficient basis for committing the financial resources necessary to design and construct a first MHTGR plant and mature plants would not be competitive with new, more efficient combined cycle plants [4-5].

In the evolution of the design from the HTGR steam cycle plant of the 1960s to the MHTGR of the 1980s, a substantial change in basic primary system configuration took place. The PCRV featured for the primary system of Fort St Vrain and GA's initial design for the gas turbine plant was replaced by the simplified layout of steel vessels for the reactor, steam generator and cross duct. The general primary system arrangement of the MHTGR is shown below in Fig. 4.1.

4.1.2. Gas turbine cycle development coupled to the MHTGR

In order to be competitive, the thermal efficiency of nuclear power had to be markedly improved to compete with modern, high efficiency fossil plants. HTGR technology has always held the promise for electricity generation at high thermal efficiency by means of a direct Brayton cycle and fortuitously, technological developments during the past decade provided the key elements to realize this promise. These key elements are as follows:

- The HTGR reactor size had been reduced in developing the passively safe module design. At the same time, the size of industrial gas turbines had increased. The technology was now available for a single turbo-machine to accommodate the heat energy from a single HTGR module.
- Highly effective compact recuperators had been developed. Recuperator size and capital equipment cost are key economic considerations. Highly effective plate-fin recuperators are much smaller than equivalent tube and shell heat exchangers, provide for substantially less complexity and capital cost, and are a key requirement for achieving high plant efficiency.
- The technology for large magnetic bearings had been developed. The use of oil lubricated bearings for the turbo-machine with the reactor coolant directly driving the turbine was problematic with regard to the potential coolant contamination by the oil. The availability of magnetic bearings eliminates this potential problem [4-5].

A major requirement was for the plant to become substantially simplified in order to provide a significant reduction in the capital expenditure for new capacity additions. Fig. 4.2 provides a comparison of the simplification that can be achieved in moving from the MHTGR steam cycle to the basic direct gas turbine cycle.

The product of integrating these technologies into a closed Brayton cycle power conversion system coupled to a MHTGR is the GT-MHR. The rated power of the GT-MHR was increased to 600 MW(t), from 450 MW(t) used in the prior MHTGR. This was based on further evaluations that showed the higher power level was acceptable and compatible with the manufacturing capability of the increasing module vessel diameter. With a net efficiency of 47.7%, a single GT-MHR module produces 286 MW(e).

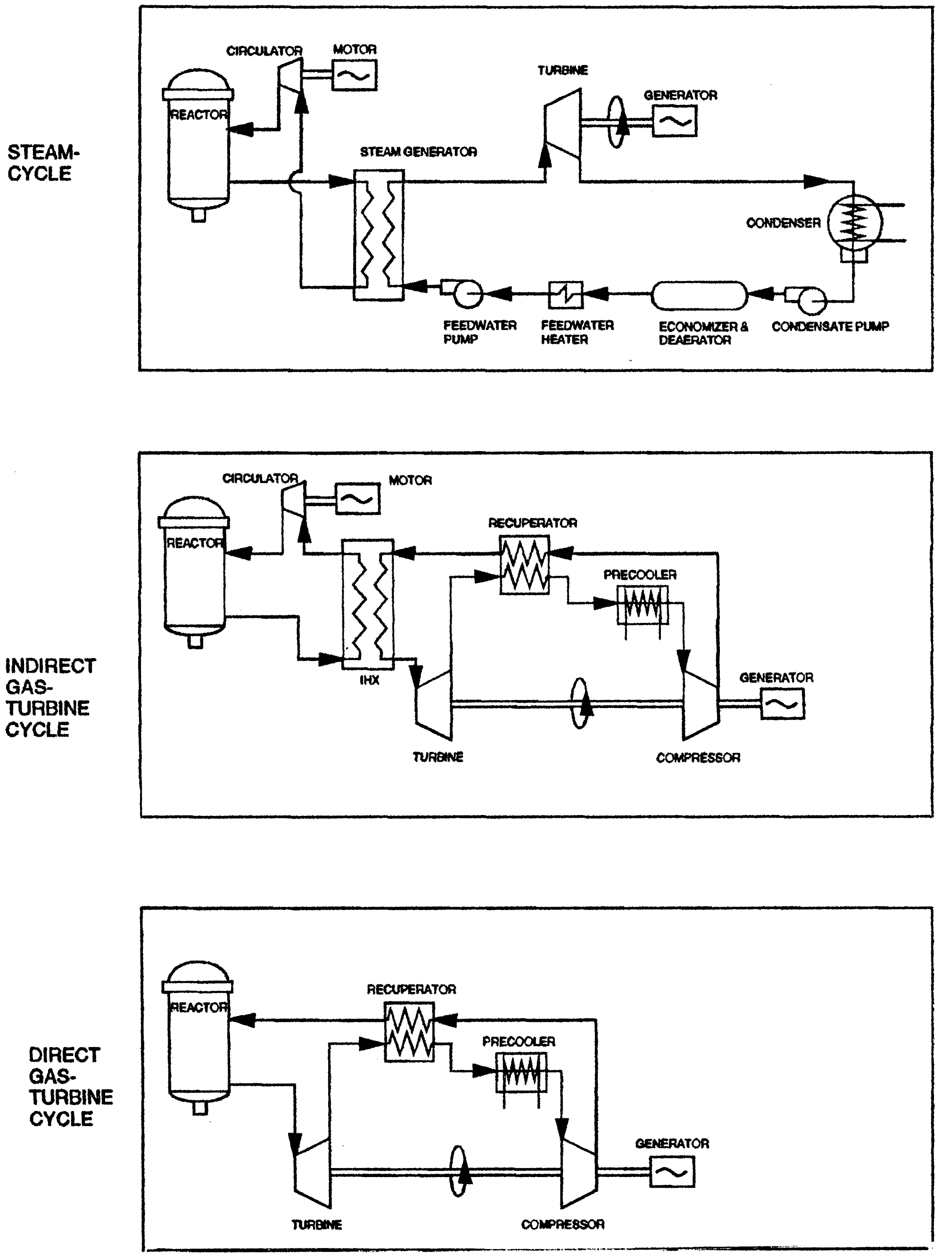


FIG. 4.2.System simplification, steam cycle to the closed cycle gas turbine plant [6].

The GT-MHR was formally selected in 1993 as the reference concept for development by the US gas reactor program for commercial deployment. However, driven by budget constraints, the US government decided to discontinue financial support of the GT-MHR starting in the fall of 1995. At that time, significant work was underway with participation of several industrial companies and national labouratories. Fortunately, the US government provided for documenting the design and development status through an orderly close out program. Government restrictions on providing other countries access to the documentation were also eliminated which opened the door for broad international cooperation.

Concurrent with the 1993 selection of the GT-MHR as the reference concept for development in the USA, GA and MINATOM signed a MOU for cooperating on the development of the GT-MHR for commercial deployment. Subsequently, in early 1994, MINATOM proposed that the cooperative program focus on development of the GT-MHR for disposition of Russian weapons plutonium. In the summer of 1994, GA and MINATOM agreed to initiate development of the GT-MHR for weapons plutonium disposition, and to jointly fund the preparation of the conceptual design, with the long term goal of utilizing the same design fueled with uranium for worldwide commercial deployment. Framatome and Fuji Electric subsequently joined with GA and MINATOM as participants in the cooperative program and in providing additional funding to support the GT-MHR conceptual design work in the Russian Federation.

A recent significant development in the advancement of the GT-MHR was the authorization of financial support by the US government on a matching resource basis with MINATOM to develop the GT-MHR for destruction of weapons plutonium in the Russian Federation. Although the GT-MHR is initially to utilize a plutonium fuel cycle which has the capability of achieving a burn-up approaching 95%, the versatility and flexibility of this core will allow for the application of a wide range of diverse fuel cycles [4-2] including burnup of plutonium discharge from commercial LWRs. Uranium is the fissile fuel presently anticipated for the commercial GT-MHR.

4.2. PROGRAMME MANAGEMENT

The GT-MHR is being developed on an international basis to optimize the HTGR capabilities and resources of many countries. Sponsoring organizations in the conceptual design of this plant include the Ministry of Atomic Energy of the Russian Federation, GA from the USA, Framatome of France and Fuji Electric of Japan. In 1999, the US Department of Energy (DOE) initiated support for the GT-MHR to dispose of weapons plutonium with the Russian contribution of matching funds.

4.2.1. GT-MHR programme management

During the conceptual design phase, the overall project management was derived from the GT-MHR Steering Committee. The membership of this committee included one member each from MINATOM, OKBM and the Kurchatov Institute (RRC KI), three members from GA and one member each from Framatome and Fuji Electric. The steering committee met twice a year at a location between the Russian Federation and the USA. The committee provided oversignt of project activities and established guidelines for the conduct of the work. It established the overall direction and strategy of the programme. The specific functions of the committee included:

— To control the execution of work under the Project implementation
— To establish procedures for review and approval of completed products
— To approve new work scope, revised work plans, and working group meetings
— To recommend, if appropriate, cost of living adjustments
— To develop strategy for promotion of the programme.

Within the Russian Federation, the participating organizations included:

— Experimental Machine Building Design Bureau (OKBM), as the leading design organization
— Kurchatov Institute (RRC-KI) to coordinate fuel and graphite related tasks
— All-Russian Scientific Institute of Inorganic Materials (VNIINM) for the design and development of the plutonium kernel

— Scientific Industrial Organization (SIA-Lutch) for the design and development of fuel coating and fuel compacts
— Siberian Chemical Complex (SCC) provides the site for the GT-MHR, fuel fabrication and handling facility, and plans for spent fuel storage and disposal.

During the conceptual design phase, interface and review of the detailed technical work was provided by two working groups; one on fuel development and one on overall plant design. Working group meetings were held approximately four times per year. The Fuel Working Group included representatives from MINATOM, GA, Framatome, Fuji Electric, RRC-KI, VNIINM, OKBM, SSC and LUTCH. The Russian institutes were responsible for design, development and testing of plutonium coated fuel particles, fuel fabrication processes, and the fuel fabrication facility design. The sponsoring organizations transferred the existing coated particle fuel technology to the Russian Federation and performed detail review of work products completed in the program.

The Plant Design Working Group included staff from MINATOM, GA, Framatome, Fuji and OKBM, and was responsible for the overall plant design activities, including reactor system, the power conversion system (PCS), and system integration and safety analysis. The Plant Design Working Group also reviewed plant design & construction schedule and design, development and plant costs.

Early in the conceptual design programme, the approach used was to develop the work scope for the design and development of the GT-MHR which included a work breakdown structure for tasks encompassing 480 individual products. The conceptual design work scope was established and seven major work areas were defined including the plant level design, system design, fuel development, reactor design, PCS design, controls and equipment handling and programmatic. For each major task, several deliverable products were established. These products included design specifications, system descriptions, design calculations, design drawings, computer codes, and some selected prototype components. All the products were produced both in the Russian and English languages so that program participants both in the Russian Federation and abroad can review and provide comments as the design proceeds.

As the conceptual design products were being completed, they were made available to the program participants for comments. These comments were incorporated in the design documents, as well as in the design components.

Indirect government support has been obtained through the International Science and Technology Center (ISTC) in Moscow. Several proposals submitted by the Russian laboratories have been accepted and funded, or are being reviewed by the ISTC. The accepted proposals typically focus on specific design or testing tasks that are directly pertinent to GT-MHR development, as well as to other reactor or non-nuclear applications. An example is the modular recuperator design developed by OKBM. The ISTC is currently co-funding manufacturing technology development and prototype testing of recuperator modules. Efforts are also underway in Europe, Japan, the USA and in the Russian Federation by the participants to attract further private and government support [4-3].

4.3. GT-MHR DESIGN AND ECONOMICS

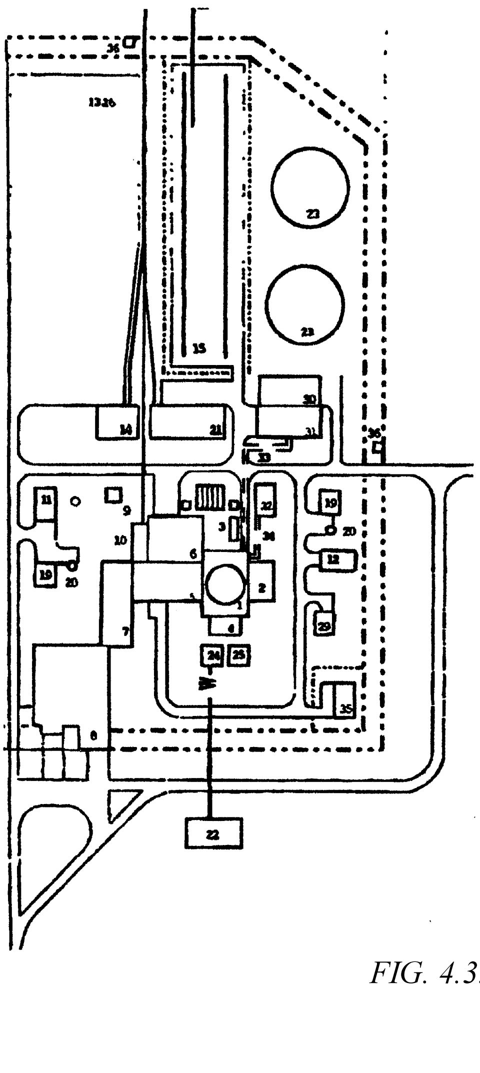
Conceptual design of the GT-MHR was completed in December 1997 and forms the base for the design information provided within this section. Present design activities currently include the development of the preliminary design of the GT-MHR, being led by the Experimental Machine Building Design Bureau (OKBM) in the Russian Federation.

4.3.1. General design description

4.3.1.1. Site arrangement

The prototypic GT-MHR NPP is planned for construction at the industrial site near the town of Seversk, the Russian Federation. The plant arrangement for the GT-MHR and associated auxiliary buildings is provided in Fig. 4.3, and provides for the following:

- Maximum possible capability for the prototype GT-MHR plant that will confirm the design of the four module commercial station
- Integration of the station's buildings and structures that reduces their quantity, area for GT-MHR disposition and length of related engineering trains
- Close proximity between technical facilities and support personnel
- Disposition of a depository for long-term storage of spent fuel at the station site
- Physical and functional separation of all the station systems and structures into two parts: "nuclear", where systems containing radionuclides and safety-related systems are located, and the conventional "non-nuclear" portion including all the remaining systems and structures.



1. NPP main building (reactor containment building, maintenance building)
2. Interim spent fuel storage (SFS)
3. Power unit auxiliary cooling system equipment(ventilated cooling towers of SFS, SCS and RCCS)
4. Electrical facilities building
5 Reactor equipment maintenance and repair building
6. Radioactive wastes treatment building
7. Sanitary - social building
8. Administrative building (Operation Center) with training center
9. Reserve control room building
10. Containers decontamination building
11. Fire pumphouse
12. "Clean" helium storage with compressor station
13. Open area for "clean" metallic items
14. Warehouse for "clean" reactor equipment
15. Area for spent nuclear fuel prolonged storage
16. Open area for heavy equipment
19. Reserve Diesel-electric station
20. Diesel fuel storage
21. Turbomachine maintenance building
22. Switchyard
23. Evaporating cooling towers of PCS heat removal systems
24. Unit transformers
25. Auxiliary systems heat removal structures
29. Refrigerating station
30. Return water supply pumping station
31. Intermediate circuit pumping and heat exchange facilites
32. "Dirty" helium storage
33. Intermediate circuit tunnel
34. "Dirty" helium tunnel
35. Guardroom
36. Check point

FIG. 4.3. Plant layout arrangement.

The plant layout solution influencing the arrangement of the reactor complex of buildings and the entire plot plan is based on underground location of buildings where the reactor module and equipment of the main safety-related systems are located (shutdown cooling system, helium service system, control and protection system boards, etc.). The following buildings and facilities are located underground:

(1) Reactor building
(2) Auxiliary reactor-related building
(3) Reactor equipment maintenance and repair building
(4) Temporary spent fuel storage.

Other buildings and structures of the GT-MHR are located either semi-underground or at ground level.

4.3.1.2. GT-MHR module description

The GT-MHR includes the nuclear heat source (i.e. the reactor system) and power conversion system consisting of equipment needed for electric power generation (turbo-compressor, recuperator, generator, precooler, intercooler and connecting pipelines). Components of the reactor and PCS are located in separate vertical steel vessels interconnected by the horizontal cross vessel as shown in Fig. 4.4. The bottom part of the reactor system vessel also houses the shutdown cooling system's "helium-water" heat exchanger and gas blower.

A simplified principal fluid diagram of the reactor module is given in Fig. 4.5. Helium is used as a working medium in the primary circuit and is circulated in the following manner.

The high pressure helium from the reactor upper collector plenum enters the reactor and is heated up as it passes through the core. The hot helium accumulates in the core lower collector plenum and flows through the inner concentric hot duct in the cross vessel to supply the PCS turbine. Helium enters the turbine at 850°C and 7.1 MPa. After expansion in the turbine, the helium at 510°C and 2.64 MPa is directed to the recuperator, where it flows through the hot side of twenty parallel heat-exchange modules and heats the helium flowing back to the reactor along the cold side of the recuperator. The helium is cooled by heat exchange in the recuperator to the temperature of 125°C and then enters the precooler where it is cooled to 26°C.

Downstream of the precooler, the cold helium is compressed from 2.57 MPa to 7.24 MPa in two successive stages (low and high pressure compressors). An intercooler between the compressors cools the helium to 27°C prior to entering the high pressure compressor unit. Downstream of the high pressure compressor the helium goes through the recuperator (along its cold side) where it is heated to 490°C and then collects in the PCS annular outlet chamber where it subsequently flows back to the reactor vessel through the annulus between the cross vessel and hot duct. Within the reactor vessel the helium moves upward to the upper collector plenum through flow channels outside the core barrel.

4.3.1.3. Plant design parameters and performance

The GT-MHR basic design parameters in the nominal operating mode at 100% power are given in Table 4.1. These parameters were selected to minimize the GT-MHR power cost and take into account the contemporary and expected near term basic design technologies of the plant’s primary system components. The combination of these factors provides for a plant efficiency of 48%.

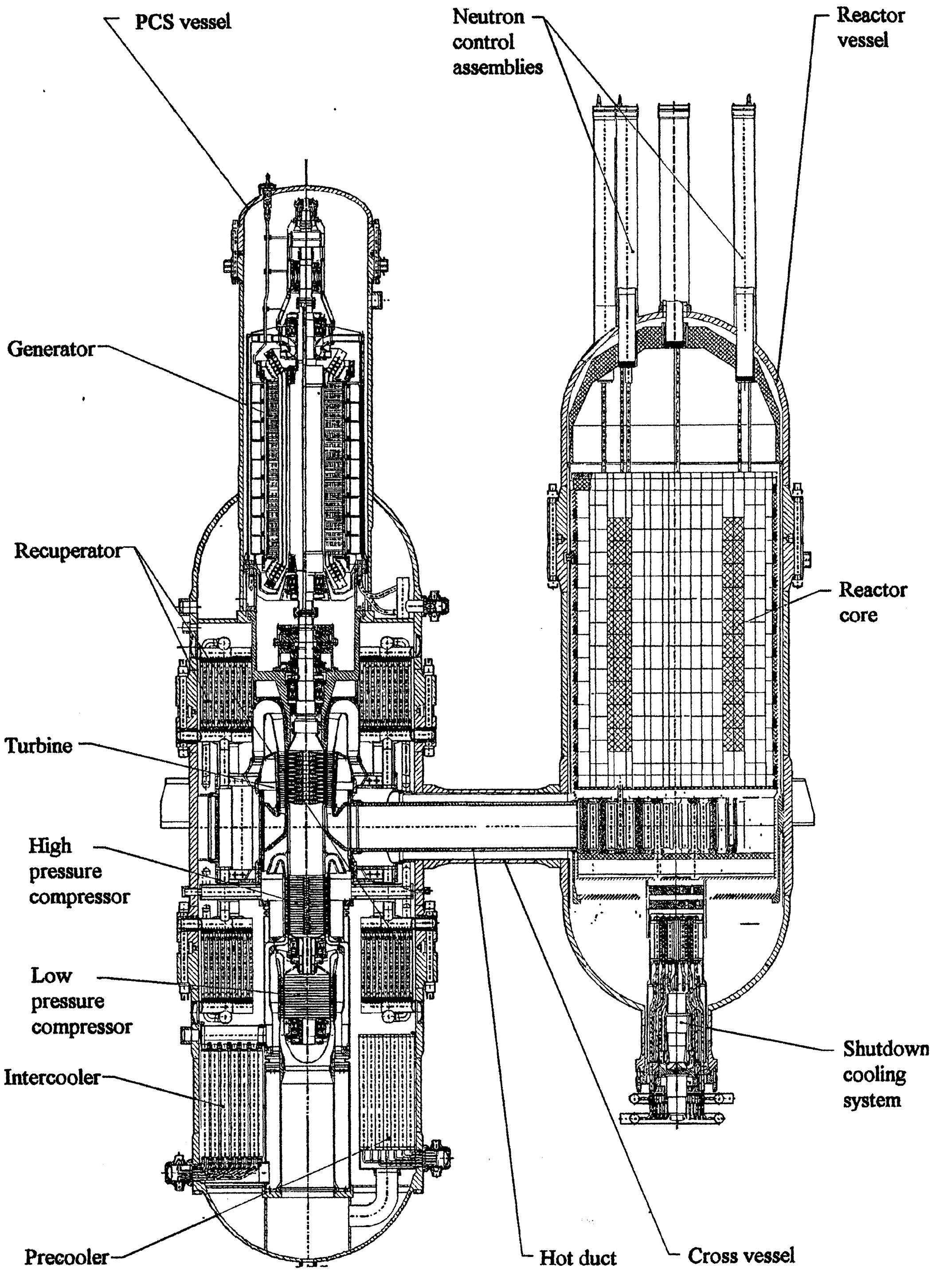


FIG. 4.4. GT-MHR module arrangement.

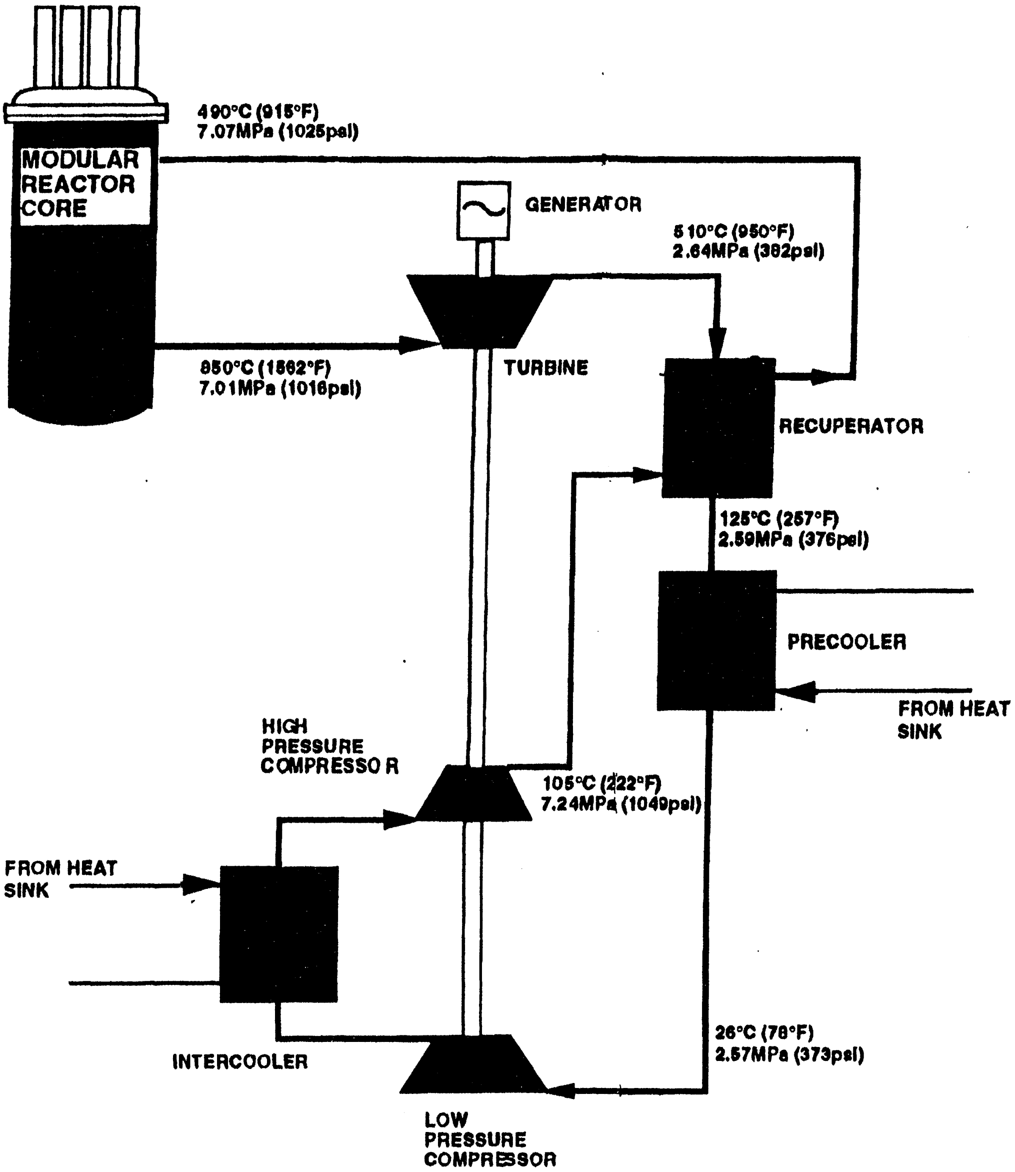


FIG. 4.5. GT-MHR flow diagram.

The most important parameters among those influencing efficiency which allow for some degree of freedom in selecting the design values are as follows:

— Helium core outlet temperature
— Compressor pressure ratio
— Thermal effectiveness of the recuperator.

Design values of the these parameters were selected with consideration of the following factors:

The upper level of the helium temperature at the core outlet was limited by the allowable service temperature level of the metal used for the turbine blades, hot duct and core support structure. Furthermore, the increase in helium temperature at the core outlet leads to an increase in fuel temperature. Taking into account the service characteristics of the contemporary high temperature metallic materials and operational experience of experimental and demonstration HTGRs (USA, Germany) with micro-fuel elements, the helium temperature at the core outlet was limited to the value of 850°C.

TABLE 4.1. GT-MHR DESIGN PARAMETERS AT 100% LOAD

PARAMETER	VALUE
Reactor	
Thermal power	600 MW
Helium temp. at core inlet/outlet	490/850°C
Helium flowrate through the core	316 kg/s
Helium pressure at core inlet	7.07 MPa
Turbomachine	
Rotor speed	3,000 rpm
Helium temp. at turbine inlet/outlet	850/510°C.
Gas expansion factor	2.7
Gas compression factor	2.8
Recuperator	
Thermal power	625 MW
Helium temp. on hot side inlet/outlet	510/125°C
Helium temp. on cold side inlet/outlet	105/490°C
Precooler	
Thermal power	173 MW
Inlet/outlet helium temp.	125/26°C
Helium flowrate	318 kg/s
Cooling water flowrate	996 kg/s
Intercooler	
Thermal power	133 MW
Inlet/outlet helium temp.	107/26°C
Cooling water flowrate	983 kg/s
NPP power unit	
Gross electrical output	285 MW
Net electrical output	278 MW
Efficiency	47%

The gas expansion factor in the turbine (and the compressor pressure ratio) was selected to limit the helium temperature at the core inlet to <490°C. This corresponds to the maximum allowable level of the reactor vessel metal service temperature.

The (net) plant efficiency at a relatively low temperature of helium at the turbine inlet compared to that of open cycle turbines depends significantly on the thermal effectiveness of the recuperator. The assessment of available experience in creation of effective heat exchangers allowed the conclusion that the recuperator can be developed with an effectiveness of 95% and is able to exhibit acceptable pressure losses within the desired dimensions.

4.3.1.4. Modes of plant operation

The power mode includes the GT-MHR automatic power control range of 30% to 100% of reactor thermal. This mode also includes plant operation at house load power level (about 3%). Both base load and load following modes can be provided while the GT-MHR operates in the automatic power control range. The plant ensures automatic following of a grid load change with a rate of up to ±5%/min.

Due to the fact that the turbo-compressor operates at synchronous speed (50 Hz) and because it is desirable to maintain the core outlet temperature constant to minimize thermal stresses and low cycle fatigue, load changes are achieved by adjusting the pressure ratio across the turbine. This is accomplished by bypass of helium from the compressor discharge to the turbine outlet. This has a negative effect on plant efficiency. Therefore, for long-term operation at part load, the helium inventory is adjusted in order to restore the pressure ratio to the design value.

The start-up/shutdown mode represent transients between the state of reactor outage and the power mode. These modes cover the reactor power range from the subcritical state to a power level which corresponds to the house load. In these modes, all of the auxiliary and service systems of the GT-MHR are operational. The start-up mode includes the following main stages: reactor start-up and power increase, helium temperature rise at the core outlet, turbo-machine start-up through the frequency converter and generator exciter, synchronization of the generator to the grid and matching the power of the reactor and turbo-machine. The shutdown mode is the reverse process to the start-up mode.

In a reactor outage mode, the reactor is in a subcritical state. The residual heat can be removed either by the PCS with the generator operating as a motor, or by the shutdown cooling system (SCS), or by the RCCS.

The refueling mode represents transferring the shutdown reactor to the state needed for core fuel reloading. This mode is characterized by low temperature of the core and low pressure of the helium (close to atmospheric) in the primary circuit. In this mode residual heat is removed by the PCS or SCS.

The design ensures a plant capacity factor of 0.8 if it is operated in the base load mode at full power with uranium fuel.

Monitoring and control of the technological processes in all modes of GT-MHR operation, including emergencies, are provided by the plant automated process control system (APCS). With respect to safety functions, the APCS is divided into an instrumentation and control system for both normal operation and protection safety (PS), including the reactor control and protection system (CPS).

The instrumentation and control system is divided into two subsystems: reactor-related and PCS-related. The PSs provided in the APCS structure ensure identification of deviations from normal operating conditions, automatic and remote control to the protective (including reactor trip and turbo-machine protection against over-speed), localization (including isolation of leaking coolers) and support safety systems.

4.3.2. Economic assessment of the GT-MHR

4.3.2.1. GT-MHR plant engineering development cost

The economic assessments obtained for the design and engineering development (ED) work on the prototype single module GT-MHR and its fuel were provided as part of the conceptual design report. The estimation of the design work including the cost of the successive design stages were taken into account and involved the preliminary design and technical (basic) design. Cost of the detailed design is taken into account in expenditures for the station construction.

Expenses for the station licensing procedure which was recently initiated in the Russian Federation have not been taken into account. The scope of both preliminary and technical designs documentation was adopted in conformity with the Russian Codes and Standards.

Cost assessments of engineering development work for equipment, systems and components were conducted on the basis of drawings and description of the systems and components developed during the GT-MHR conceptual design stage and utilizing norms existing at OKBM on work for nuclear reactor plants of the BN and HTGR types (previous generation of nuclear designs). Data from Russian manufacturers were used in the estimation of costs for the main equipment.

In order to take account contingencies, a margin of 20% (of the total cost) was provided in the cost of the technology development work. The total cost for this work on the plant is ~170 million US dollars.

4.3.2.2. Fuel development cost

Fuel engineering development costs include expenses for development of fuel particles, fuel and burnable poison compacts fabrication technology, construction of pilot installations, investigation into fuel particle and compact performance and quality under normal and emergency conditions of the GT-MHR, in-pile post reactor studies, calculation simulation of fuel behavior under irradiation conditions and fission fragments transport processes.

The scope of main fuel research and engineering design on fuel includes the following milestones:

(1) Laboratory phase of the fuel pilot specimens development, technical requirements formulation, restoration and fabrication of equipment, development of fabrication technology, conduction of initial investigations.
(2) Construction of a laboratory scale installation for investigation and selection of most productive fabrication processes.
(3) Bench-scale phase, construction of production installation, process modes refinement, fabrication of pilot batches. quality control, and production of batches for in-pile tests.
(4) The pilot-scale phase, including refinement of mock-ups of key apparatus and control methods, construction of the pilot installation, refinement of pilot batches production processes, including those for in-pile tests.
(5) In-pile tests, including pre-reactor and post-reactor studies of control batches.

The total estimated cost of fuel design and engineering development is 75 million dollars US.

4.3.2.3. Plant capital cost

Plant capital costs include expenses associated with the fabrication. And construction of the first prototype module and associated development of detailed documentation. The capital costs are conditionally divided into the following categories:

(1) Direct-expenses for immediate construction work and erection of plant equipment, systems, buildings and structures;
(2) Indirect-expenses for the work to support construction of structures and systems, which are not accounted in the direct costs, as well as expenses for site surveying work.

The capital cost figure is estimated at ~ 300 million dollars US. The assessment of this cost was performed addressing the following basic conditions:

- — The cost assessment was performed on the basis of the conceptual design information (drawings, system and component descriptions, technical requirements) developed at the given GT-MHR design stage;
- — The statistical data on NPP construction in the Russian Federation were used for the station components which had not been developed on the conceptual design stage;
- — Assessments of capital investment in the construction of auxiliary and service facilities, electrical facilities, water and heating supply networks and structures, etc. were performed using the data of the pre-design study of a co-generation nuclear district plant;
- — Construction labour cost estimation was performed for the conditions of the Seversk site;
- — Assessment of labour cost during main equipment fabrication was performed on the basis of available experience in reactor equipment fabrication;
- — The margin value for contingency expenses was taken as 10% of the sum of the accounted cost articles.

4.3.2.4. Fuel production capital cost

The capital costs associated with GT-MHR fuel production includes expenses for development of the pilot-industrial installation, fabrication and erection of equipment, construction and commissioning of the installation at the Seversk site.

This cost evaluation is based on a fuel production output of 1 050 000 compacts annually. The fuel production installation requires 10 000 m^2, including ~7000 m^2 for the main equipment items. In the determination of expenses for construction of the fuel production facility, coarse specific data on labour cost of work, energy resource expenditures, costs of equipment, buildings and structures were used from similar designs.

Total capital costs for construction of the fuel production installation at Seversk is 47 million US dollars.

4.3.2.5. GT-MHR annual operation and generation cost with plutonium fuel

Calculation of annual operation costs has been performed according to the following cost structure (accounts): 1. fuel cost, 2. operation and maintenance cost, 3. capital cost; 4. costs incurred by the site; 5. decommissioning allowances. 6. different funds and fees proceeding from the cost of nuclear power system generation cost. The costs have been calculated for the following conditions:

- — ED costs for GT-MHR equipment and fuel were referred to the capital costs for the single module NPP;
- — Cost for plant design is included in the capital costs of the commercial NPP;
- — The cost of fuel is adopted equal to a prime cost value;
- — Cost of site services for the NPP (i.e.: water, repair, guards, transport, etc.) which have not been included in O&M costs were taken into account in an article "Costs incurred by Seversk";
- — Costs for long-term storage of spent fuel and inflation were not considered in the calculations.

The basic technical and economic input characteristics of the NPP as taken into account in the calculation of O&M costs are given in Table 4.2.

TABLE 4.2. BASIC TECHNICAL AND ECONOMIC CHARACTERISTICS

DESCRIPTION	PROTOTYPE MODULE	FOUR MODULE NOAK PLANT
Thermal/electric output	600/285 MW	2400/1141 MW
House power	7.5 MW	30 MW
Annual full power hours (CF = .8)	7000 h	7000 h
Design & development cost	~$320 million US*	
Plant capital cost	~$273 million US*	~$928 million US*
Fuel fabrication capital cost	~$47 million US*	~$126 million US*
Annual operation cost, including the following:	~$44.1 million US*	~$102 million US*
Fuel cost	~$9.89 million US*	~$28.4 million US*
O&M cost	~$9.82 million US*	~$25.64 million US*
Capital cost	~$17.6 million US*	~32.64 million US*
User's overhead cost	~$3.73 million US*	~$8.7 million US*
Decommissioning fund	~$.53 million US*	~$1.24 million US*
Other funds	~$2.5 million US*	~$5.8 million US*
Annual net electricity production	1944×10^6 kW•h	7777×10^6 kW•h
POWER GENERATION PRIME COST	~2.2 cents US/kW•h	~1.3 cents US/kW•h

* Based on year 2000

Requirements of the following regulatory documents and statements related to nuclear power have been addressed in the calculation of annual generation cost:

(1) "Regulations for cost accounts in production and realization of products (work, services) to be included in the primary cost of products (work, services) and on a procedure of forming financial results addressed at profits taxation";
(2) The Federal law "on financing of particularly dangerous and nuclear hazardous productions and installations";
(3) "NPS costs component features".

The total annual costs of electrical energy production for both the prototypic single-module and commercial four module NPPs are about 44.1 million dollars and about 102 million dollars US, respectively. Therefore, the primary generation costs are about 2.2 cent/kW•h for the prototypic single module NPP and 1.3 cents/kW•h for generation from the four module commercial plant

4.3.3. Plant systems

Basic GT-MHR systems are described herein. Information of spent fuel and waste management associated with the GT-MHR is provided in Section 8.

The GT-MHR also eliminates many systems that are typical to a nuclear power plant. The PCS is the replacement for respective steam/water systems (steam generators, steam turbine plant, condensate/feedwater system) which are characteristic of NPPs with the steam cycle.

4.3.3.1. The vessel system

The basic functions of the vessel system: 1) provide the arrangement of the reactor system, PCS, supports, ducts and other primary equipment of GT-MHR; 2) primary coolant confinement; 3) protection of reactor module equipment against external impacts; 4) acts as the reactor module restraint in the reactor building cavities; 5) provides for heat transfer from the core to the RCCS; 6) prevention of both air ingress into the primary circuit and corrosion of core graphite elements; and 7) maintaining of proper configuration for the core in respect to NCA positions. The vessel system layout in the reactor building cavities is shown in Fig. 4.6.

The vessel system consists of the reactor vessel (RPV), cross vessel, PCS vessel and supports. The RPV and PCS vessel are installed vertically and interconnected by the horizontal cross vessel. To install and fix the vessel system on the concrete cavity foundations within the reactor building, four supports are welded to both the PCS and reactor vessels in the plane of the cross vessel axis. The reactor and PCS vessels also have supplementary supports for additional restraint of the vessels against seismic impacts. The vessel system withstands all operating loads including pressure and temperature effects emerging from both normal and anticipated accidental modes during GT-MHR operation. The main characteristics of the vessel system are given in Table 4.3.

The vessel system design addresses the requirement for limiting helium leakage at normal operation to not more than 10% of helium inventory in the primary circuit per year. In order to monitor and control the state of the vessel system and implement the "leak before break" concept, a strain-thermometer monitoring system (to be used during pre-operational tests) and an acoustic-emission system are provided which, in addition to the primary parameters monitoring system, permit the identification and characterization of defects to be made on-line.

The vessel system and its components are designed in conformity with the requirements of Codes and Standards of Russia. The USA requirements presented by GA have been treated as well. According to Russian code, the vessel system is a normal operation system important to safety.

TABLE 4.3. BASIC CHARACTERISTICS OF THE VESSEL SYSTEM

Characteristic	Reactor Vessel	PCS Vessel	Cross Vessel
Design temp.	440°C	150°C	440°C
Design pressure	8 MPa	8 MPa	8 Mpa
Hydraulic test pres.	13.5 MPa	13.5 MPa	13.5 Mpa
Wall thickness	270 mm (max.)	285 mm (max.)	100 mm
Inner diameter	7300 mm	4200 & 7300 mm	2,300 mm
Number of blocks in vessel	2	4	1
Total vessel mass	1362 t	1840 t	17 t
Vessel dimensions			
Height (length)	31 170 mm	37 000 mm	2900 mm
Outer diameter	8640 mm	8740 mm	2500 mm
Lifetime	60 a	60 a	60 a

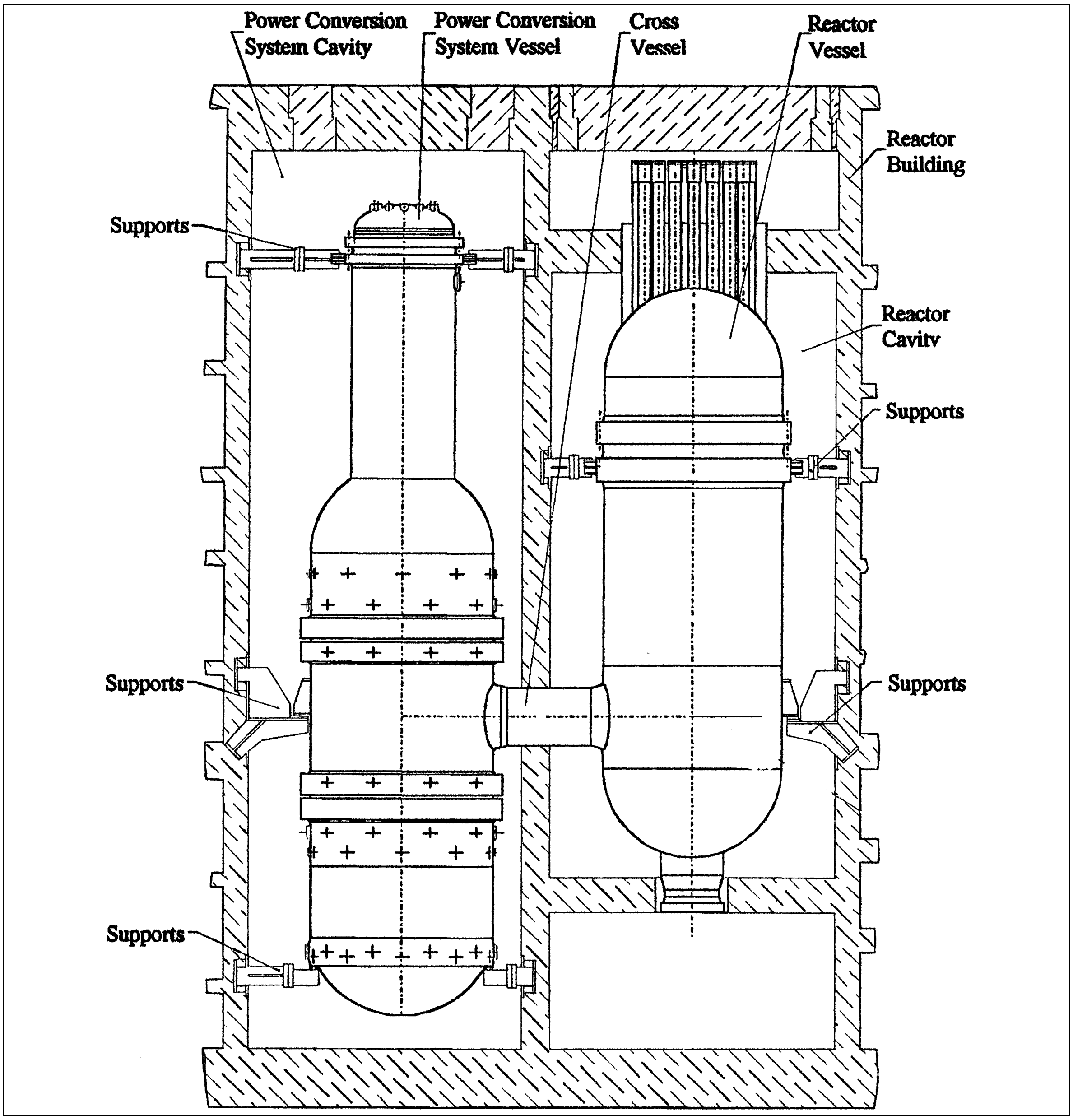


FIG. 4.6. GT-MHR vessel system layout.

In the GT-MHR, both the reactor and cross vessel have a normal operation temperature of ~440°C. Under emergency conditions this temperature can increase to ~540°C.

The Russian heat resistant steel, 10Cr9MoVNb, may be considered as a suitable material for these temperatures. In all phases of the metallurgical process, the 10Cr9MoVNb steel exhibits good technological properties and is presently included in the Russian Federation's "Rules for Design and Safe Operation of Steam and Hot Water Pipelines". Although this steel is not included for nuclear power plant use, samples have been irradiation tested at temperatures of 400 to 550°C. However, full testing of these samples is still required. The American steel, 9Cr-1Mo-V, is one of the foreign analogs of the 10Cr9MoVNb steel and has a similar level of experience in industry.

4.3.3.2. Reactor system and fuel

The basic functions of the reactor system (RS) are the generation of heat from the release of energy of nuclear fission, heat transfer to the primary coolant helium, monitoring and control of the neutron generation process in the core and confinement of radioactive products. Furthermore, the RS provides the necessary operating conditions for nuclear vessel material protection against adverse thermal and radiation effects of the core, as well as an opportunity for removing and replacement of the reactor components.

The RS (Fig. 4.7) consists of the core, reactor internals, neutron control assemblies and instrument transducers. The RS components are arranged primarily within the reactor vessel system. The nominal thermal output of 600 MW is transferred to the PCS. During cooldown of the RS when the PCS is not operable, the heat is transferred to the SCS. In this case helium in the RS is circulated by the SCS gas circulator. In accidents associated with failure of the above mentioned active heat removal systems, the reactor cooldown can be provided passively by the Reactor Core Cooling System (RCCS) through the wall of the RPV.

During reactor operation at nominal power, heat losses via both the SCS and RCCS systems do not exceed 1% $_{nom.}$. During normal plant operation coolant flows into the RS from the PCS through an annular gap between the cross vessel and hot duct. It then flows sequentially through the bottom metallic support, annular cooling passage between two shells, top collector plenum, upper core restraint, upper reflector, core assembly (where it is heated) and the bottom collector. Heated helium is returned to the PCS from the bottom collector through the hot duct. When the SCS is operating, the helium circulation train in the RS is similar with the exception that the heated helium flows through the SCS header, which is located beneath the bottom core collector, rather than through the hot duct.

The reactor components are made of the following structural materials:

- Graphite components; high strength nuclear graphite of GR-1 grade
- Metallic components; high-nickel alloy 55CrNiMoVWZr and steels including 03Cr2lNi32Mo3Nb and 10Cr9MoVNb.

Graphite materials are used for thermal and radiological shielding, alumina-silica fiber material (Al_2O_3 - 52%, SiO_2 - 45%, etc.) is used for thermal insulation of the gas duct.

The reactor core (Fig. 4.7) consists of an annular arrangement of hexahedral graphite fuel assemblies, neutron control assemblies (CPS control rods), absorber elements of the reserve shutdown system (RSS), replaceable reflector blocks and neutron source. The annular core arrangement is built of hexahedral prismatic fuel assemblies stacked up in 102 columns of ten fuel assemblies each. The top and bottom reflectors are arranged above and below the fuel assemblies arrangement, respectively. The selected core structural materials are corrosion and erosion resistant under operating conditions and not subjected to self-welding during service life. The core design and configuration provide safe decay heat removal from the core through the reactor vessel without any engineered heat removal means. In this event, fuel temperature does not exceed 1600°C.

The main core parameters are presented in Table 4.4. The control rods are subdivided into three groups according to their intended functions: emergency protection and reactivity compensation rods (12 rods) used during reactor startup; power control and emergency protection (3 rods), and reactivity margin compensation and emergency protection (33 rods). The rods are disposed in blocks of the core and replaceable reflector. All the control rods are

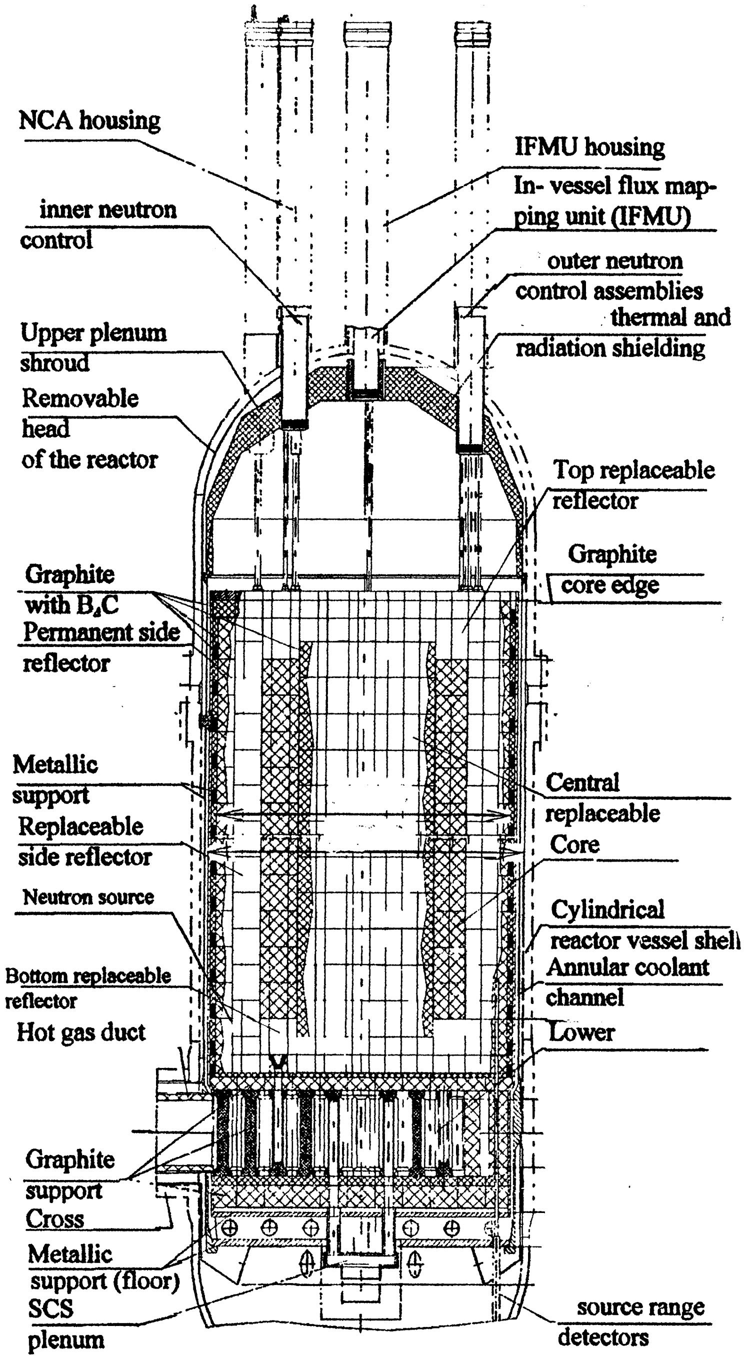


FIG. 4.7. GT-MHR reactor longitudinal view.

identical in their design. A CPS control rod consists of twenty ring-shaped graphite links hinged one to another. Nineteen links contain absorber material B_4C, the upper one has a sleeve for coupling with the drive grip and a ring to support the rod against the core graphite structure at the drive grip release. The control rod drive mechanism is electro-mechanical and consists of an electric motor, induction brake, a rack mechanism and position sensor.

The RSS absorber elements are used to shutdown the reactor and maintain it in a subcritical state if the rod system fails to shut the reactor down. The RSS spherical absorber elements are of an absorbent and protective cladding. The reactor is shutdown by gravity flow of the RSS absorber elements into the fuel assembly channels.

TABLE 4.4. GT-MHR CORE TECHNICAL PARAMETERS

Name of Characteristic	Value
Thermal Capacity, MW	600
Number of Fuel Assembly Columns	102
Average Power Density, MW(t)/m^3	6.5
Fuel Assembly Dimensions, mm	
- height	800
- width across flats	360
Fuel Assembly Arrangement height, mm	8000
Number of Fuel assemblies	1020
Number of Control Rods	48
Number of RSS Channels	18
Core Fuel Residence Time (core fuel life), days	~840
Refueling Interval, days	280
Number of Fuel Assemblies Replaced in a Cycle	340
Average Fuel Burnup, MWd/kgPu	642
Burnup of Initially Loaded Pu-239, %	88.6
Refueling Duration, days	20.7
Maximum power Peaking Factors:	
- axial (refueling interval mean)	<1.5
- over fuel compacts	<1.6/1.25*
Isothermal Temperature Reactivity Coef. In Working Point (beginning of cycle/end of cycle), pcm 1/°C	-7.5/-5.3
Worth of Reactivity Control Members (beginning of cycle/end of cycle), %Δk/k:	
-36 rods in side reflector	7.9/8.1
3-rod group in side reflector	0.9/0.93
12 in-core rods	6.8/7.3
-entire rod system	16.0/16.2
-reserve shutdown system	11.6/12.0
Plant Capacity Factor, %	80

The initial deployment of the GT-MHR will be utilized for the burning of plutonium. The commercial plant will utilize low-enriched uranium-235 as the fissile fuel. The basic arrangement of the core for both plants will be generally similar. A fuel assembly (FA) consists of a graphite block, fuel compacts, burnable poison compacts, guide dowels and plugs. Two types of fuel assemblies are used in the core distinctive in the presence or absence of channels for control rods or RSS absorber elements. Type 1 without holes for control rods or RSS absorbers is shown in Fig. 4.8. The main technical data for the fuel assemblies are given in Table 4.5.

The fuel compacts are placed in fuel holes of a fuel assembly (Fig. 4.8). A fuel compact consists of fuel particles mixed uniformly in a pressed graphite matrix. Artificial graphites on the basis of non-calcinated petroleum cokes with a low content of neutron absorber impurities are used as a filler for the matrix composition. 15 fuel compacts are arranged along the fuel assembly axis. Fuel holes filled with compacts are closed at the ends by graphite plugs. A gap between fuel compacts and plugs minimizes axial stresses during operation.

TABLE 4.5. FUEL ASSEMBLY MAIN TECHNICAL DATA

Name of Characteristic	Value
Width Across Flats, mm	360
Height, mm	800
Control Rod & RSS Hole Diameter, mm	130
Number of Coolant Channels	
- type-1 fuel assembly	108
- type-2 fuel assembly	89
Number of Compacts in One Hole of a Fuel Assembly	15
Number of Compacts in:	
- type-1 fuel assembly	3030
- type-2 fuel assembly	2460
Number of Compacts in the Core	2 919 600
Compact Diameter, mm	12.5
Number of Holes for Burnable Poison Compacts	
- type-1 fuel assembly	14
- type-2 fuel assembly	10
Burnable Poison compact diameter, mm	12.5
Number of Burnable Poison Compacts in One Hole of Fuel Assembly:	
- type-1 fuel assembly	15
- type-2 fuel assembly	15

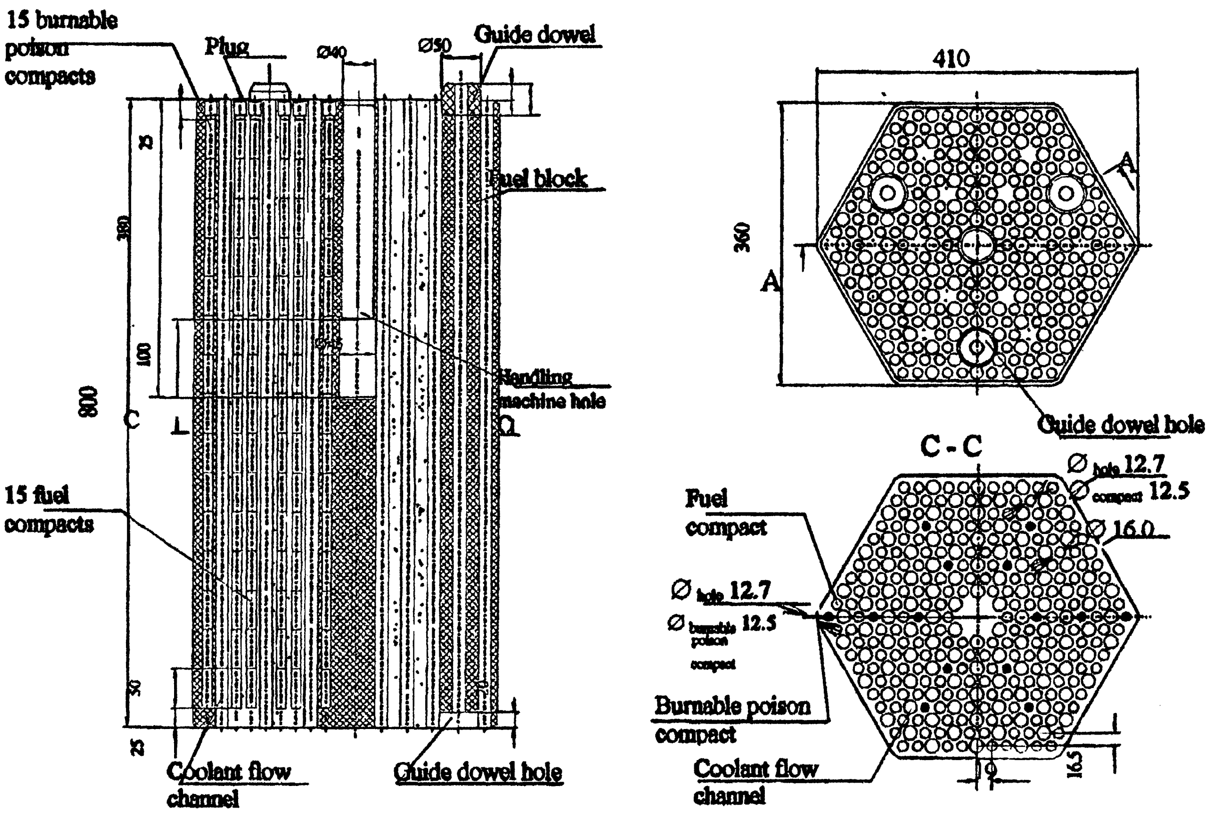


FIG. 4.8. Fuel assembly without control rod guide hole (Type 1).

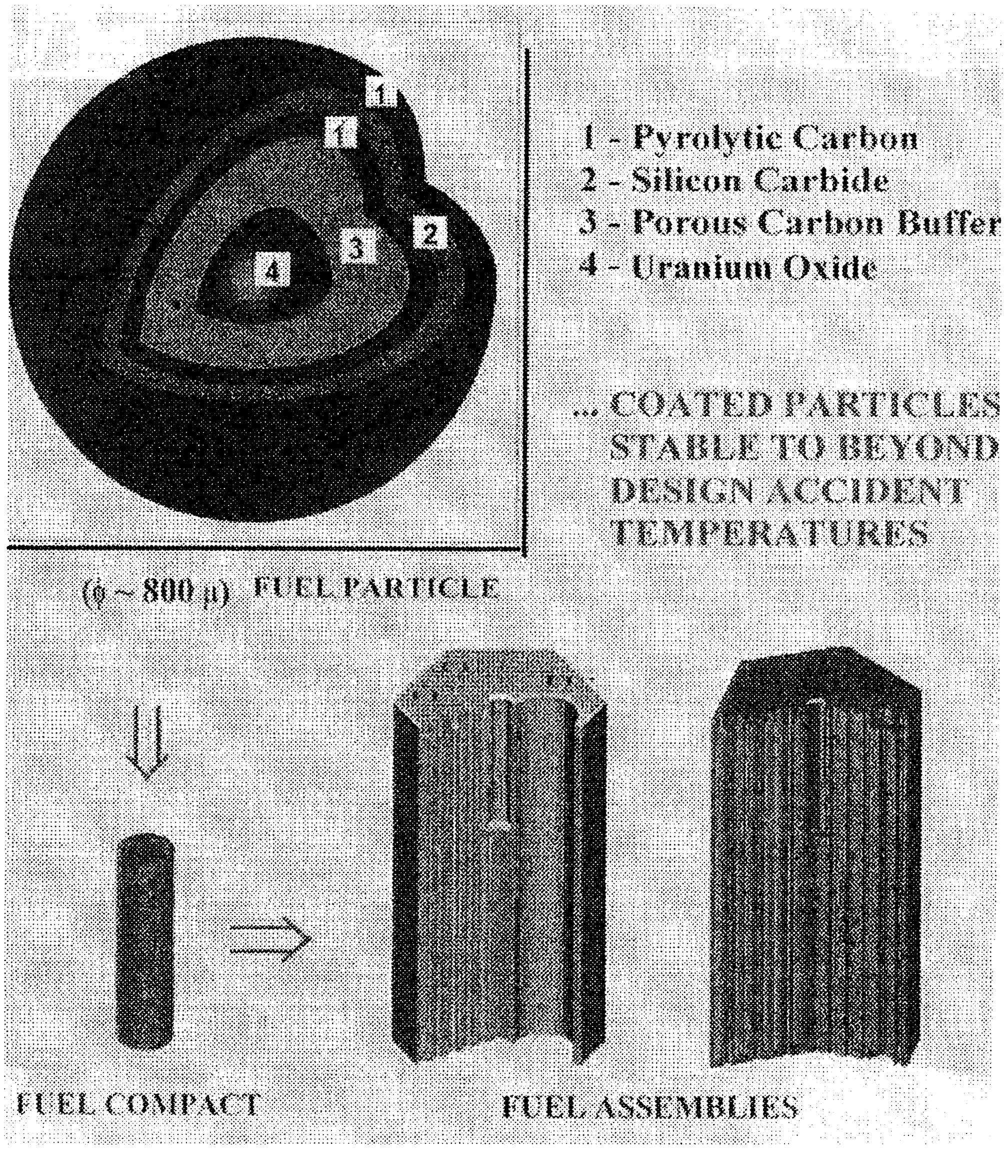


FIG. 4.9. GT-MHR fuel assembly.

The fuel kernels are coated by pyrocarbon and silicon carbide layers (TRISO- type), the main purpose of which is to reduce fission product release. TRISO coating contains three types of materials: low density pyrocarbon (PyC), silicon carbide (SiC) and high density PyC. Low density PyC serves as a fission product accumulator and reduces the load upon the next layer of dense PyC. The SiC coating is laid between the two layers of highly dense PyC and is intended to confine the volatile fission products (I, Cs), metals and provides the mechanical strength of the particles. Fig. 4.9 illustrates the GT-MHR fuel assembly.

The GT-MHR commercial reactor with uranium fuel is identical to the GT-MHR reactor with plutonium fuel with the exception of the fuel particle size and coatings. The basic characteristics of the GT-MHR reactor with a uranium core are given in Table 4.6.

Refueling of the GT-MHR includes shuffling of the fuel elements in order to optimize burn-up. Two fuel reload patterns were considered in the conceptual design including options where ½ and 1/3 of the core are reloaded. The comparison of the results for both these reloading modes has shown that there are no significant differences in terms of fuel life duration, discharge burn-up value and discharged fuel composition. Therefore, the GT-MHR reactor with uranium core can function with any of these fuel reload patterns. The ½ fuel reload pattern allows for a greater capacity factor to be obtained and from this standpoint it is preferable. However, the 1/3 fuel reload pattern appears to be more attractive regarding the experience, which would be obtained at designing and operation of the GT-MHR with plutonium fuel. The final decision on a fueling strategy for the uranium core will be made in the up-coming design stages.

In order to shape the power distribution in the core, boron-carbide absorbers are arranged in a layer of graphite elements of the internal reflector adjacent to the core (their negative reactivity worth is ~4 to 5% Δk/k, which is similar for both the plutonium and uranium cores). The radial power peaking factor for the uranium core obtained does not exceed Kr = 1.3. The axial peaking factor does not exceed Kz = 1.5.

While using the fuel reload pattern with a large fraction of discharged fuel for the GT-MHR with a uranium core, it will be necessary to use a burnable poison to compensate the reactivity margin for the extended fuel cycle. Materials such as Er, B, and Gd are considered as burnable poison candidates and final selection of a burnable poison material will be made in the following design stage.

The reactivity margin for fuel burn-up during a cycle was taken as 2.5% Δk/k. The reactivity coefficient for temperature is negative over the entire temperature range and equals ~ -9×10^{-5}/°C in the working area thereby promoting safety assurance and internal self protection capability of this reactor.

TABLE 4.6. CHARACTERISTICS OF THE GT-MHR LOW-ENRICHED URANIUM FUELED CORE

Parameter	Value
Thermal output, MW	600
Capacity factor	0.85
Core equivalent diameter inner/outer, m	2.96/4.84
Core height, m	8
Top/bottom reflector thickness, m	0.8/0.8
Core coolant pressure, MPa	7
Inlet/outlet helium temperature, °C	490/850
Number of control rods: - in core - in side reflector	 12 36
Number of reserve shutdown system channels	18
Number of fuel blocks in the core	1020
Number of compacts in the core	3 100 000
Full fresh uranium inventory in the core, kg	4 570
U-235 enrichment of uranium, %	14
Parameters of UO_2 fuel particles: UO_2 density in kernel - γ; g/cm^3 Kernel diameter, μm	 10.2 500

Other reactor internals consist of the permanent side reflector, core graphite support structure, metallic support structure, top collector plenum, upper core restraint and hot duct. The reactor internals are located within the reactor vessel and are arranged to maintain a stable core position within the RPV, thus ensuring its cooling and control in all normal and emergency operating modes. Limitation of by-pass helium leakage along the core shell is ensured by installation of special graphite inserts with a capability to displace in the vertical direction. Leakage via gaps is limited between the columns of FAs and reflector as they are overlapped by the bottom reflector blocks which are larger than blocks in the columns. Stable position of the core and reflector columns during the RS warming up is ensured by virtue of relative displacement of graphite and metallic support elements (due to the difference in their linear expansion coefficients).

The permanent side reflector is intended (except for the neutron reflector function) to arrange the core, limit neutron exposure to the core barrel shell and RPV, and transfer radial loads between the core barrel and the core itself. The permanent side reflector is a compound annular graphite structure consisting of twelve rings of 800 mm height, one upper ring of 500 mm height and one lower ring of 400 mm height, which are assembled from five different sized blocks.

Absorber rods of B_4C are installed near the external side of all permanent side reflector blocks. There are also three holes for guide dowels and a hole for coupling with the handling machine grip.

The core graphite support structure serves as an immediate support for both the core and the permanent side reflector, and as a channel for helium passage from the core to both the bottom collector header and SCS header. The graphite support structure includes the bottom permanent reflector, bottom collector header, bottom support blocks and bottom support plate. Axially, the permanent bottom reflector is built up of two rows of hexahedral blocks. The blocks positioned under the core have channels for the passage of helium coolant. Absorber rods made of B_4C are arranged in the lower row of blocks and serve as radiological shielding beneath the core metallic support structure.

The bottom collector header is built of columns, the upper and lower parts of which are made as hexahedral prisms of 360 mm width across the flats. The central portion of the columns represent a cylinder. The columns positioned under the core arrangement and a portion of columns under the replaceable reflector have internal longitudinal channels for helium flow from the core to the collector header. Eighteen columns in the central part of the collector header have similar channels for helium passage which communicate the bottom collector plenum with the SCS header. The peripheral columns are made of blocks in the form of hexahedral prisms of 360 mm width across the flats. All collector header blocks have holes for the handing machine grip and for guide dowels. The lower support blocks of the bottom collector header columns and side support blocks are made as hexahedral prisms of 624 mm width across the flats. Absorber rods made of B_4C are located in all support blocks. A portion of the central blocks have channels providing coolant flow passage from the bottom collector header to the SCS header. The lower graphite support plate is made in the form of individual sectors fixed to the upper metallic support plate.

The core metallic support structure is intended to hold the core structure and reactor internals in the required configuration within the RPV. The metallic support also provides channels for cooling helium flow from the PCS to the top core collector plenum. The metallic support structure is installed into the reactor vessel and includes the upper support plate, core barrel shell, gas duct shell, bottom collector header shroud, hot duct nozzle and SCS header.

The top collector plenum is intended to distribute the helium coolant flow inlet into the core and provide thermal and radiological shielding for the RPV upper head. The top collector plenum also ensures the capability for the core fuel reloading by means of the handling machine. The collector is located above the core and installed on the core metallic support structure. The structure is made as a hemispherical dome on which thermal and radiological shielding is installed. There are penetrations in the dome for guide tubes of the control rods and RSS absorbers and for core reloading.

The upper core restraint provides spacing of the core and reflector graphite columns, organizes channels for coolant passage from the upper collector plenum to the core, distributes coolant over cooling channels in the core columns and holds the control rods and absorber elements guide channels in a proper position to ensure their normal motion. All the elements are provided with the means for their fixation and replacement.

The hot duct is intended to remove the hot primary coolant from the reactor system to PCS with minimum thermal losses caused by the hot-to-cold helium recuperation effect. The hot duct is located within the cross vessel and consists of a load bearing metallic tube, thermal insulation, expansion joint and supports. One end of the load bearing tube is welded tightly to the core metallic support. The other end of the hot duct is connected to the PCS inlet nozzle through an expansion joint. The thermal insulation made of an aluminia-silica fiber is fixed on the inner surface of the load bearing tube and is lined with a metallic casing made of CrNi alloy. The thermal expansion joint is a set of self-sealed metallic semi-rings pressed against a sealed surface due to the pressure difference of the hot and cold helium.

4.3.3.3. The power conversion system (PCS)

The function of the PCS is to convert the thermal energy generated in the reactor core into electrical energy by a direct gas turbine cycle (see Fig. 4.10, for PCS arrangement). In addition, the PCS and its components ensure fulfillment the following functions:

(1) power unit start up;
(2) operation in the house load mode;
(3) power unit operation at any steady power level in the automatic power control range with electricity supply to an external grid;
(4) power unit output changes;
(5) scheduled shutdown of the power unit;
(6) power unit cool down, including prior to a refueling outage.

The primary components of the PCS include the turbo-machine (consisting of the turbo-compressor and generator), recuperator, precooler, intercooler and associated supports and ducts. The system equipment items are arranged in a single vessel installed side-by-side with the reactor vessel as shown in Fig. 4.4.

The turbo-machine is a vertical assembly located in a central part of the PCS. The PCS related heat exchange components, (i.e. recuperator, intercooler and precooler) are arranged around the turbo-compressor in order to provide a compact system configuration and minimize the vessel dimensions. To minimize hydraulic losses in the helium flow path and make the entire system arrangement more compact, the recuperator heat exchange surface is divided into two equal sections operating in parallel and located above and below the hot duct.

The precooler and intercooler are arranged below the recuperator in the annular gap between the PCS vessel and duct shell at the low pressure compressor exit. The turbo-machine generator is located in a cavity separated from the main helium circulation path by a buffer gas seal. There is also a repair seal for sealing the primary circuit during generator repair and replacement operations. Gas is supplied to the seals from the helium services system.

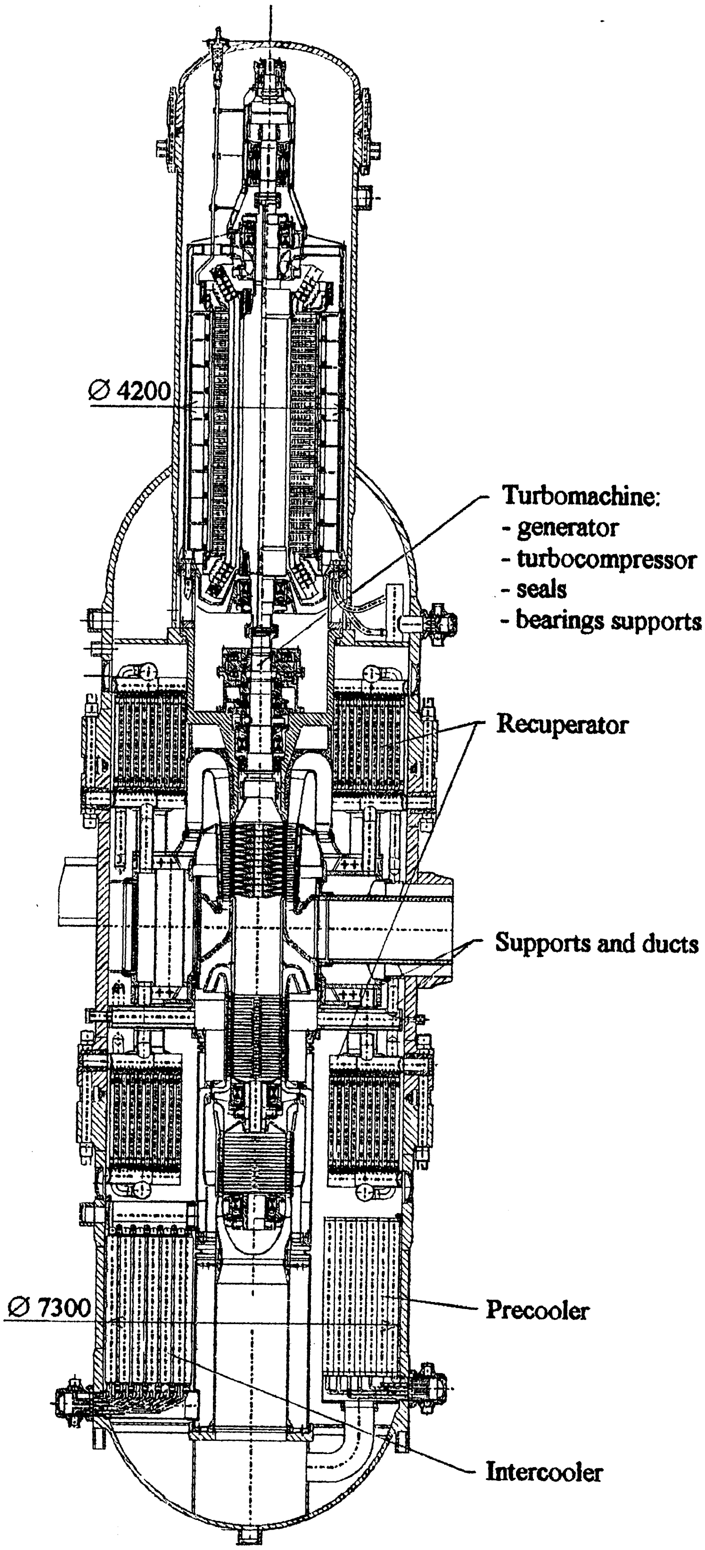


FIG. 4.10. PCS arrangement.

TABLE 4.7. PCS TECHNICAL CHARACTERISTICS AT 100% POWER

Characteristic	Value
Thermal energy transferred to PCS	595 MW*
Turbine power	561 MW
Generator output	285 MW(e)
Frequency	50 Hz
Voltage at generator terminals	20 kV
Electricity generation efficiency	47.5%
Helium flowrate through turbine	317 kg/s
Total bypass leakage over PSC train	~2.4% of nom. flow

* corresponds to full power of the reactor (600 MW) with allowance for heat losses in RCCS and SCS.

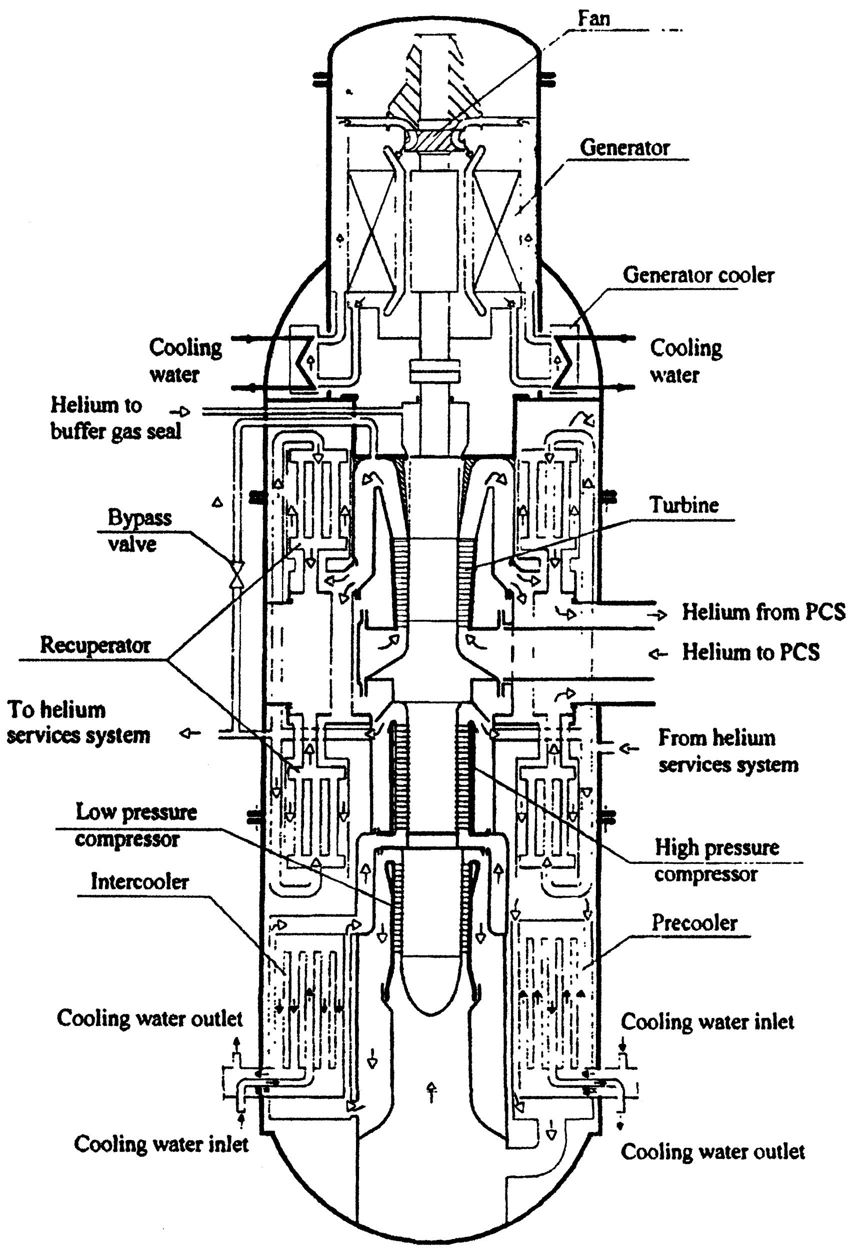


FIG. 4.11. PCS helium flow path.

Primary helium and water of the heat removal system are the working fluids in the PCS. Fig. 4.11 shows the primary helium circulation path within the PCS during power operation and in start-up, shutdown and normal PCS cool-down modes.

Helium is circulated in the primary circuit by the compressors which operate during plant power operation, start-up, shutdown and cool-down (when turbine power is insufficient and the generator is utilized in the motoring mode from the electrical grid). For generator cooling, helium is circulated by a fan installed on the generator rotor shaft. Heat from the generator plenum is then removed by special water coolers. Water in a cooling system for PCS coolers (including generator coolers) is forced circulated within the tubes of the coolers in a counter-current flow scheme.

Bypass leakage in the PCS component interfacing seals represents ~2.4% of nominal flow in the main helium path. Also, during plant operation a portion of the helium flow can bypassed the turbine via pipelines and bypass valves of the turbo-machine control and protection system (Fig. 4.11). The entire PCS and its components are controlled and monitored by means of the power unit APCS, i.e. by the I&C systems of both the turbo-machine and PCS.

The PCS is regulated by the following means:

(1) Bypass valves opening in the turbo-machine control and protection system pipelines (during a rapid load rejection event);
(2) By changes in the helium inventory through transfer between the primary circuit and the helium system storage (during slow decrease or rapid increase of electric load). Changing of the helium inventory in the primary circuit at partial load operation ensures maintaining the high efficiency of the electricity generation process;
(3) Turbo-machine speed variation by the frequency converter (in the start-up and cool-down modes).

The optimal method for regulating the PCS is to be determined in the following design stage.

The PCS and its associated components have been designed on the basis of relevant Russian Codes and Standards, also taking into account the US regulatory requirements presented by GA. According to established classification, the PCS is a system of normal operation important to safety. Physical boundaries of the PCS are locations of interface between PCS components and the vessel system, hot duct and pipelines of interfacing systems, as well as electric leads and terminals of external power and instrumentation circuits.

The turbo-machine is a vertical assembly consisting of a generator and turbo-compressor. The turbo-compressor and generator rotors are connected rigidly by a coupling that provides transmission of torque between these rotors. Active electromagnetic bearings (EMB) are utilized to support the turbo-machine rotor. In addition to the EMB, reserve catcher bearings are provided which take up the load of a turbo-machine coast-down in the event that the EMBs fail to operate. These catcher bearings are integrated with EMBs in the structural units. The turbo-machine is supported against a load-bearing partition of the vessel that separates the generator plenum from the main plenum of the PCS. The size characteristics of the turbo-machine at 100% power and main engineering parameters and the are given in Table 4.8 and 4.9, respectively.

TABLE 4.8. TURBO-MACHINE MASS AND SIZE CHARACTERISTICS

Name of Parameter	Value
1. Maximum diameter (for generator), mm	3920
2. Length (with generator terminal leads), mm	30380
3. Mass, t	620
4. Rotor length, mm	28975
5. Maximum rotor diameter (generator rotor barrel), mm	1075
6. Rotor mass, t	105.4

TABLE 4.9. TURBO-MACHINE ENGINEERING PARAMETERS

Name of Parameter	Value
Turbine	
1. Power, MW(t)	560.3
2. Speed, rpm	3000
3. Helium flow rate, kg/s	317.5
4. Helium inlet parameters:	
- temperature,°C	848
- pressure, MPa	7.07
- pressure losses, MPa	0.007
5. Helium outlet parameters	
- temperature,°C	510
- pressure, MPa	2.63
- pressure losses MPa	0.02
6. Blade system efficiency (adiabatic.), %, not less	93
7. Number of stages	12
Low /high pressure compressors:	
1. Capacity, kg/s	319.4/323
2. Helium inlet parameters	
- temperature,°C	26.1/27.4
- pressure, MPa	2.55/4.28
- pressure losses, MPa	0.002/0.003
3. Helium outlet parameters	
- temperature,°C	107.5/110.3
- pressure, MPa	4.34/7.27
- pressure losses MPa	0.025/0.025
4. Blade system efficiency (adiabatic), %, not less	88/87
5. Number of stages	16/24

The turbo-compressor consists of a turbine and high and low pressure compressors mounted on the same rotor shaft. Both the turbine and compressor stators form a single load-bearing cylindrical structure protecting the other PCS components and the vessel against failures from turbo-compressor accidents such as deblading. The turbine and compressors are multistage-type axial machines with non-cooled blades and disks. The turbo-compressor design allows for mounting/dismounting operations to be performed through the flanged joint of the generator plenum (following removal of the generator). The turbo-compressor design provides end seals for its rotor and multi-stage seals for the stator to minimize coolant by-pass leakage. The end seal is a labyrinth ridge-type seal to reduce leakage between the turbo-compressor plena with different pressures. Each stage of the stator seal consists of two rings that are multi-segment elements enclosed in casings. The stator seals limit helium by-pass leakage where the turbo-compressor interfaces with other PCS components (Fig. 4.12).

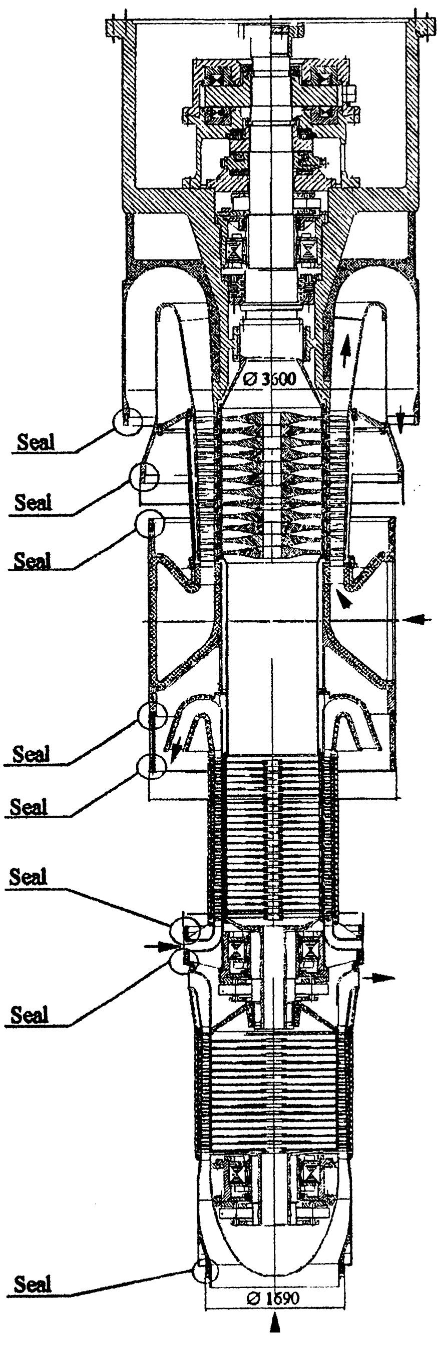


FIG. 4.12. Stator seal arrangement.

The turbo-machine utilizes a synchronous generator (Fig. 4.13). This unit consists of a main generator and a brushless exciter. Rotors of both the generator and exciter are joined rigidly. The electric terminal leads consist of leak-tight penetrations in the PCS vessel and connecting buses or cables within the vessel. The generator stator is enclosed in a vessel which is a load-bearing structure and is joined through an intermediate support to the turbo-compressor which, in turn, is supported by a support flange against the load-bearing PCS vessel partition. The temperature of the generator is controlled by forced helium circulation in the generator plenum with subsequent release of heat to the PCS cooler water cooling system. For this purpose a fan is provided which is mounted on the generator rotor shaft and special coolers are located in the generator plenum.

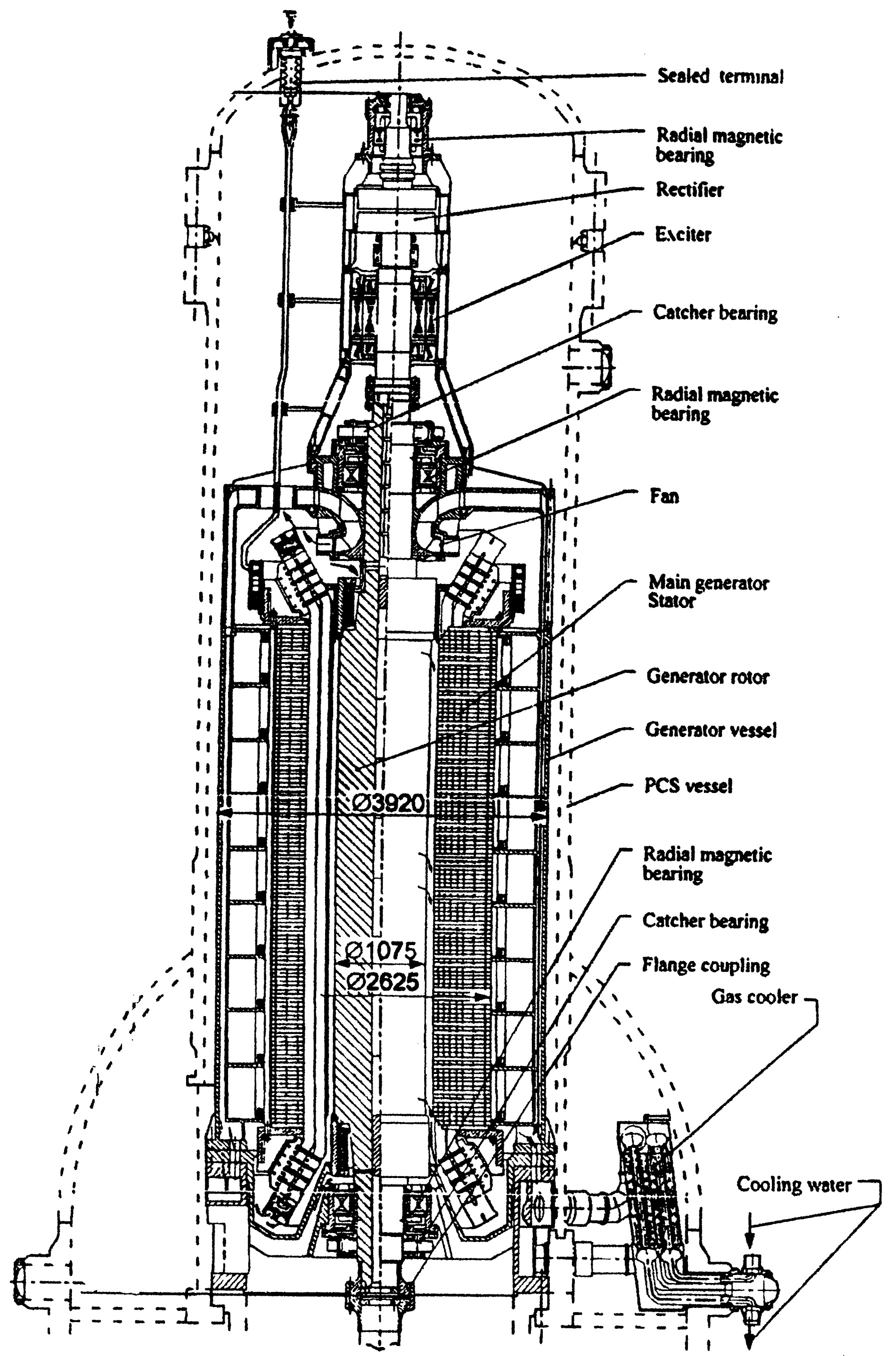


FIG. 4.13. Generator arrangement.

The recuperator is a modular heat exchanger with plate-corrugated heat exchange surfaces operating in a counter-current flow scheme. The recuperator consists of 20 identical modules operating in parallel which are arranged as ten modules on the recuperator support above and ten below the hot duct axis. A recuperator module consists of 207 heat exchange elements enclosed in a common casing (Fig. 4.14). Each heat exchange module consists of a tubular case in which the plate heat exchange surface is arranged. The surface is made of zigzag-wise strips, one section which is straight and the other corrugated. Straight and corrugated sections alternate, so that the heat exchange element is assembled with the tops of the corrugation supporting on the straight sections. The low pressure loop is arranged on one

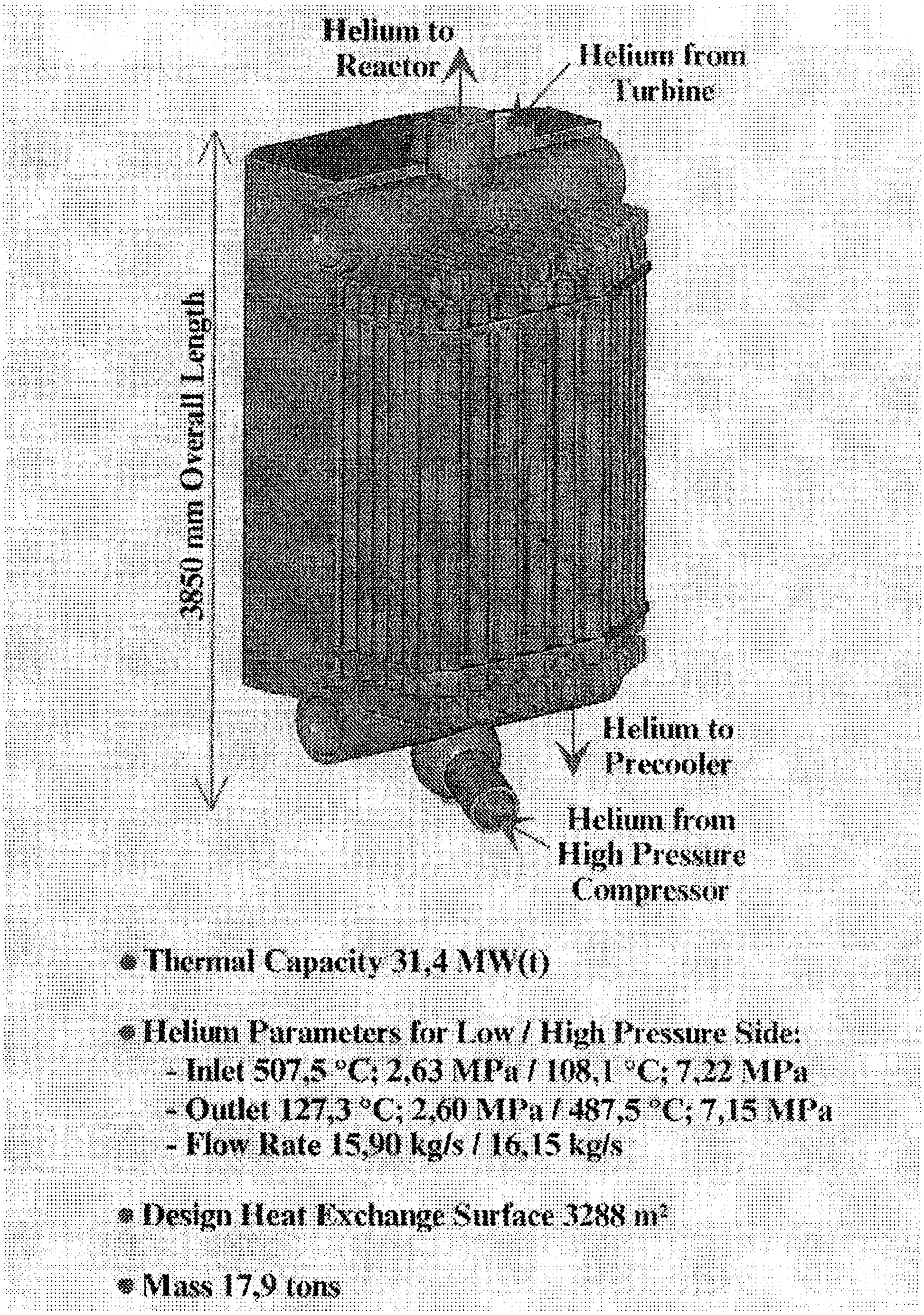


FIG. 4.14. Recuperator module.

side of the corrugations, with the high pressure loop on the reverse side. Parameters of the corrugation ensure appropriate strength of the heat exchange surface at pressure differential up to 5.0 MPa. The heat exchange element is assembled using a welding process.

The recuperator module structure allows for the detection of heat exchange surface loss-of-integrity occurrence and its repair is by means of plugging the affected heat exchange elements. The current recuperator module design permits mounting and removal from the

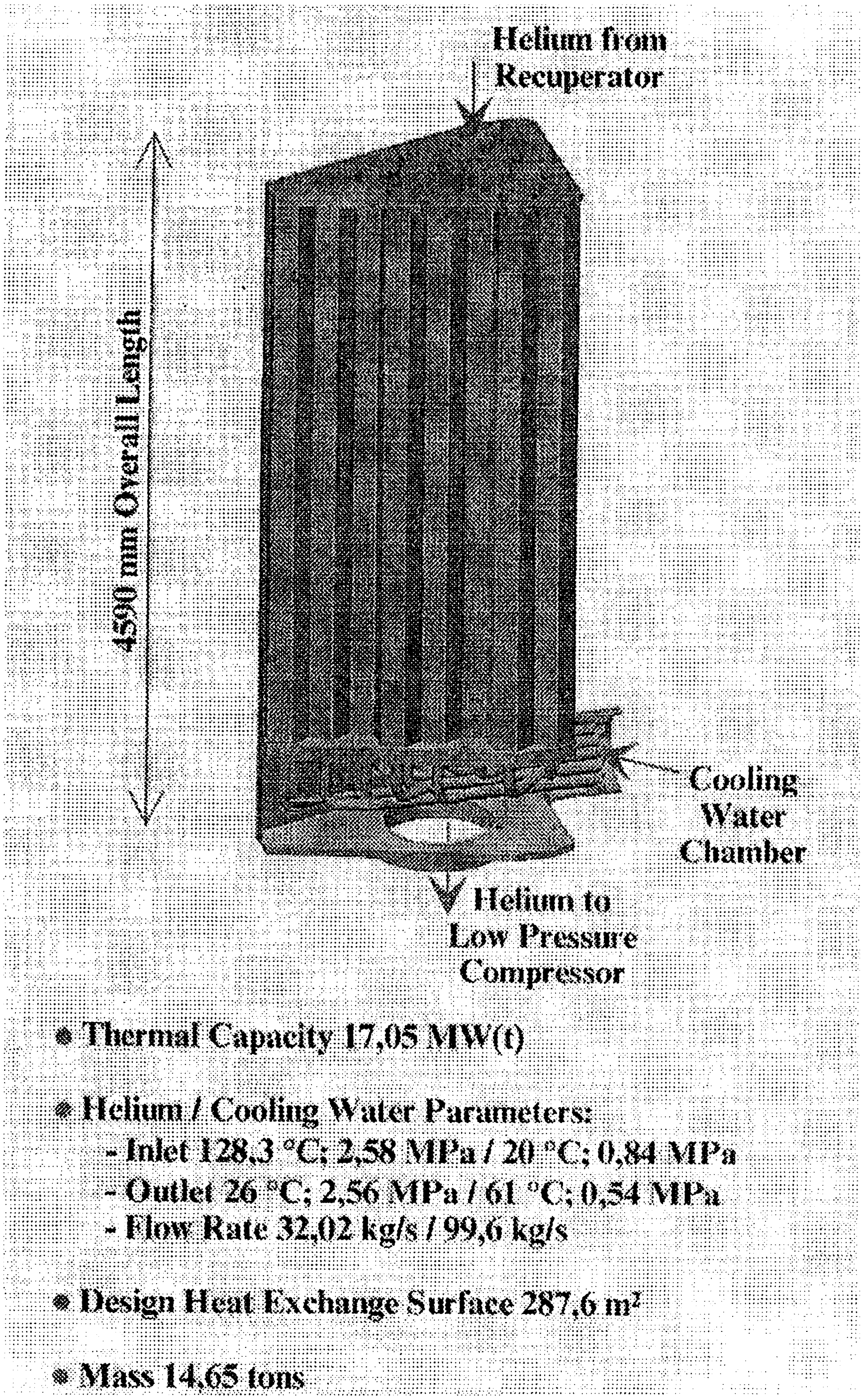


FIG. 4.15. Precooler module arrangement.

PCS vessel through the flanged joint at the generator plenum. Austenitic stainless steel 08Crl6NillMo3 has been selected as the recuperator structural material (including its heat exchange surface).

The precooler and intercooler are modular heat exchangers with heat exchange surface composed of straight tubes with outer fins arranged in triangular cells in cassettes (assemblies). Cooling water circulates within the tubes, with helium in the space surrounding the outside of the tubes. The coolants move in a counter-current flow scheme. Each cooler consists of 10 identical modules. The precooler and intercooler modules differ in configuration in order to provide the maximum compact arrangement the PCS vessel. A precooler module cassette (Fig. 4.15) consists of 252 finned tubes enclosed in a hexahedral casing. The cooling water supply tube is located in the center of the tube bundle.

The intercooler module cassette is assembled to the same principle and from the same elements as the precooler. However the number of heat exchange tubes in the intercooler cassette is 192 and the cassette casing is of cylindrical form.

The module design allows for the detection of heat exchange surface loss-of- integrity and subsequent repair by plugging of a leaking cassette. The cooler module design allows installation/removal from the PCS vessel through the flanged joint at the generator plenum. Stainless steel has been selected as the structural material of the coolers (including their heat exchange surface).

A heat removal system from the PCS coolers to an ultimate heat sink is shown in Fig. 4.16. This system provides for heat removal from the power unit during start-up, scheduled shutdown and cool-down (when through the PCS), and to confine radioactivity in order to prevent release to the environment in the event there is a loss of intercircuit integrity in the PCS coolers.

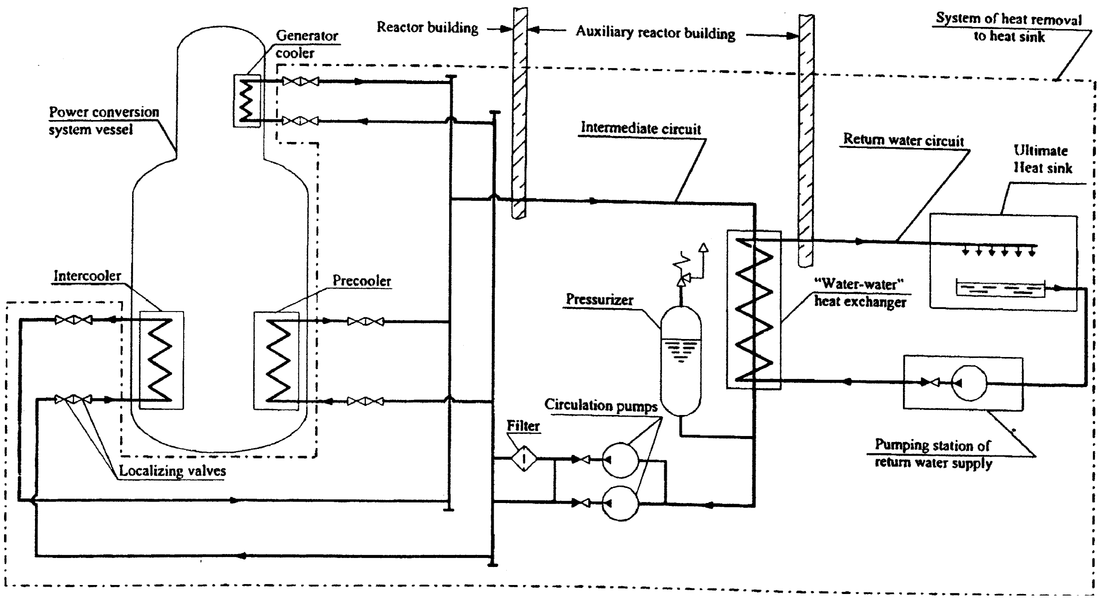


FIG. 4.16. Heat removal system for PCS coolers.

The supports and gas ducts for the PCS components include three basic groups i.e. turbo-machine and recuperator supports, pipelines, high and low pressure gas ducts. The supports and gas ducts also include the components for attachment of the PCS cooler modules. The main functions and requirements for these supports are as follows:

(1) Attachment of the individual components (i.e. turbo-machine, recuperator and coolers) in the PCS vessel taking into account the following;
   - Combined effect of the coolant temperature and pressure
   - Mass of the components
   - Anticipated seismic, acoustic and vibration loads

(2) The minimization of axial effects on the turbo-machine stator in order to eliminate its deformation in all operating modes;
(3) Organization of the primary helium circulation path between the system components within the PCS vessel to provide for a uniform flow pattern and the minimization of pressure losses;
(4) Minimization of heat losses, heat recuperation and by-pass leakage throughout the helium circulation path.

The lifetime of supports and gas duct components apart from the sliding seals is 60 years.

4.3.3.4. The shutdown cooling system

The reactor shutdown cooling system (SCS) is designed to remove residual heat from both the reactor and PCS, thereby maintaining their temperatures to allow for fuel reloading or repair operations at a helium pressure near atmospheric. The SCS removes heat from the shutdown reactor by forced circulation of coolant in heat transport loops. Heat from the SCS intermediate water circuit is removed to the circulating water circuit. The SCS is constantly capable of performing its functions (via a "hot" stand-by state) during reactor power operation. In this state, the SCS gas circulator is idle and its shut-off valve is closed.

The SCS and its components are designed in conformity with requirements of Codes and Standards of Russia with allowance for the requirements of the USA presented by GA. The SCS consists of:

(1) The helium loop including a helium/water HX, gas circulator, shut-off valve and associated drive and instrumentation;
(2) The cooling water loop including a water/water HX, circulating pumps, pressurizer, valves, pipelines and instrumentation.

The SCS helium loop components are arranged at a bottom part of the RPV. The cooling water loop is located beyond the reactor cavity. The SCS physical boundaries are defined by interfaces with the RPV, reactor internals, reactor building, power supply system, recirculation water system, make-up system, instrumentation and control system.

The SCS functions in the following operating modes:

(1) Standby mode, where the SCS is maintained in a hot state during reactor power operation with the gas circulator idle and cooling water circulating on both the intermediate and recirculation circuits;
(2) Cooling mode 1 cooling with the reactor shutdown while the primary circuit is pressurized;
(3) Cooling mode 2 cooling with the reactor shutdown and maintaining a prescribed temperature state during refueling or repair operations while pressure in the primary circuit is near atmospheric.

The helium flow rate through the helium HX is minimal in the stand-by mode, when the gas circulator is idle, and is defined by leakage through the shut-off valve and seals. In cooling modes 1 and 2 with the gas circulator operating and the shut-off valve open, helium circulates in the SCS as follows: From the reactor collector plenum, the hot helium enters the intertube space of the HX and then flows to the entrance of the gas circulator. From the circulator exit, the majority of the helium flows to the core pressure header, passes through the core and returns to the SCS collector header. A small portion of helium flows through the PCS and also returns to the core collector plenum.

Water in the cooling water loop circulates in modes 1 and 2 as follows: Circulating pumps supply water to the helium/water HX where the heat from the helium circuit is transferred to the water. This water then passes through the water/water HX and upon transferring the heat to a final cooling water system it then flows back to the helium/water HX.

4.3.3.5. Reactor cavity cooling system

The RCCS is designed to perform the following functions:

(1) Heat removal from the reactor cavity during normal plant operation with the capability to ensure the required temperature conditions for the reactor cavity concrete;
(2) Removal of residual heat and heat accumulated in the core and structural elements, as well as maintaining the reactor vessel, fuel, reactor internals and reactor cavity concrete temperatures within allowable limits during emergency cool-down;
(3) Confining of radioactivity released into the reactor cavity during normal operation and in emergencies

The RCCS consists of two independent heat removal trains. Each train includes a surface cooler, HX evaporator, RCCS headers, pipelines and instrumentation sensors. The surface cooler is located on a vertical wall of the reactor cavity along its entire height and forms a heat exchange surface around the reactor vessel. The surface cooler is connected to RCCS headers that are located in the neutron control assemblies room**.** Each RCCS train contains two headers that are connected by means of pipelines to the HX evaporator, which is located beyond the reactor building but within the protective concrete structure. This description is the conceptual design version of the RCCS and is subject to change in the up-coming design phase of the GT-MHR.

The RCCS is a normally operating system which also performs the protective and isolation functions. Isolation functions are performed only by the portion of RCCS that is located within the reactor containment building. The RCCS provides heat removal from the reactor under seismic impacts of magnitude up to 7 points on MSK-64 earthquake scale (maximum design basis earthquake). Water in the RCCS loop circulates due to natural connection as follows: The water heated in the pipes of the surface cooler sections located around the reactor vessel rises to the upper RCCS header. The heated water then flows via tubes into the HX evaporator where its heat is transferred to the re-circulating water. Cooled water enters the lower RCCS header and is distributed over the surface cooler tube sections positioned along the reactor cavity wall. In a loss of re-circulation water flow event in the HX evaporator, the cooling water circulates in the same manner as during normal operation. However, heat will be removed by water evaporation in the HX evaporator and the steam is released to the atmosphere.

Permanent functioning during operation and the principle of passive operation of the RCCS based on natural processes exclude the need of operator or control systems actions in the transient from normal operation to the emergency cool-down mode. Hydrogen generated due to water radiolysis is released into the HX evaporator's gas plenum and then removed to the atmosphere through the steam discharge nozzle, and excludes formation of explosive concentrations of hydrogen. The design provides monitoring of cooling water activity and quality. Hydrogen concentration is monitored in gas plenum of the HX evaporator.

The RCCS has the following performance characteristics during reactor power operation with two trains available:

(1) Heat removal capacity is ~4 MW
(2) Water temperature at surface cooler inlet is ~27°C
(3) Water temperature at surface cooler outlet is not more than 90°C
(4) Water temperature in the HX evaporator tank is not more than 90°C
(5) Pressure in the HX evaporator gas plenum is atmospheric
(6) Reactor cavity concrete temperature is not more than 80°C.

In emergency modes, the RCCS ensures maintaining the temperature of fuel, reactor vessel and reactor cavity concrete at admissible levels of not more than 1600°C, 540°C and 150°C, respectively. In a loss of power event, the RCCS ensures reactor cool down by evaporation of water stored in the HX evaporator. The RCCS water capacity is sufficient to ensure system operation for 48 hours. In order to assure that system performance is optimized, it is envisaged that alternative RCCS designs including one utilizing heat pipes, will also be evaluated during the following design stage.

4.3.3.6. Fuel handling system

The fuel handling system is comprised of a set of equipment which provides delivery of nuclear fuel and reflector blocks into the core and removal of spent fuel and reflector blocks from the reactor core to storage. The system operation is provided for the period of initial core loading, during scheduled core reloading in the course of the plant's operational lifetime, and for any required core defueling, such as decommissioning. The scope of the fuel handling system functions range from loading of fresh fuel blocks into the transfer cask in the temporary storage area to spent fuel disposition into storage. A review of the spent fuel management is provided in Chapter 8 of this document.

The main components of the fuel handling system include the refueling machine, fuel transfer cask, handling system support structures and positioner, and associated controls and instrumentation. The GT-MHR project provides for the maximum possible application of fuel handling equipment items that have been developed by GA for the FSV HTGR. Adaptation of this equipment is envisaged for the subsequent design stages addressing the GT-MHR plant specific core composition, dimensions and configuration as they differ from the FSV core.

A specified fuel cycle is composed of the following general characteristics:

- A core life of 840 effective full power days,
- The number of fuel blocks in the core is 1020,
- The number of fresh fuel blocks loaded into the core for one refueling cycle is 340,
- The number of totally spent fuel blocks unloaded from the core during one refueling cycle is 340.

The core refueling outage duration is presently estimated as ~21 days. Fuel blocks are reloaded by layers within the core sectors (1/6 core fraction) through NCA stand-pipes at the top of the RPV. Fuel and reflector blocks are reloaded within the reactor by the handling machine and transfer cask which both work jointly. The handling machine and cask are located on a handling equipment support structure. This structure is installed by the handling equipment positioner above the NCA stand- pipes after removing a protective plug provided in the reactor hall floor (reactor containment building ceiling).

Before fuel reloading starts, the in-reactor instrumentation arranged in the central penetration of the reactor vessel and NCAs located in the core sector to be reloaded are withdrawn from the reactor by a special maintenance tool using the auxiliary service cask. A guide sleeve of the transfer cask is then installed into the reactor vessel central penetration by means of the same cask. The refueling machine located over the inner row of NCA standpipes in the core sector to be reloaded ensures the ability for grappling each block being reloaded and its subsequent installation into the guide sleeve of the transfer cask. The cask is intended to transfer fuel blocks between the core and temporary storage.

The handling machine, transfer cask and support structure are equipped with gate valves which are intended to eliminate primary helium leakage into the ambient air and to ensure radiological protection for personnel. The guide sleeve is intended to guide the transfer cask lifting mechanism grapple. There is a shelf platform in the lower part of the sleeve which is installed on the upper core reflector where fuel blocks are placed by the handling machine manipulator.

Two transfer casks are used to reload the blocks. As the first cask is completely filled it is transported to the temporary storage socket for fuel block unloading. Simultaneously with the first cask unloading, the second cask is placed above the reactor to take the next portion of fuel. The operations with the transfer casks are repeated until the entire core sector is free.

Fresh fuel blocks are transported from the fresh fuel storage to the temporary storage facility during preparation for refueling. Spent fuel elements are transported to sealing and fuel inspection equipment for packing and shipping. These operations are performed during the period between reactor refueling outages.

4.3.3.7. Helium services system

The helium services system (HSS) is intended to maintain needed primary coolant quality and perform a number of functions in the plant, including:

(1) Maintaining the primary coolant qualitative composition to provide necessary conditions for long-term and reliable operation of the primary circuit components at normal operation and in accidents associated with water or air ingress into the primary circuit;
(2) Primary circuit filling and draining;
(3) Primary helium pressure (mass) control during unit power regulation;
(4) Helium receipt from the primary circuit overpressure protection devices;
(5) Helium supply to the PCS turbo-machine buffer seal;
(6) Tests of the primary circuit leak-tightness and strength parameters following completion of erection, repair or technical inspection operations;
(7) Clean and "contaminated" helium storage;
(8) Filling/purging of the fuel handling system or casks for equipment replacement during their connection to the primary circuit (to prevent air ingress into the circuit).

The HSS includes the helium purification system (HPS) which consists of the equipment for helium cleaning up of chemical and radioactive impurities, heat exchange equipment, pipelines, valves, instrumentation sensors and the means for regeneration. The HSS also includes the helium storage system consisting of storage tanks, helium transport means (compressors), heat exchange equipment, filters, pipelines, valves and sensors. The HPS is located in confines of the reactor containment building. The helium storage system equipment is located in a separate building beyond the reactor building.

The helium purification system ensures maintaining the following levels of impurities in the helium under normal operation conditions: $H_2O < 2$ vpm; $CO_2 + CO < 6$ vpm; $H_2 < 5$ vpm. Primary helium flow rate through the HPS is 0.5 kg/s. Radioactive impurities retention time (noble gases and iodines) is 3 days for short-lived nuclides and 40 days for long-lived I^{131} and Xe^{133}.

Helium flow through the filter for purification from long-lived radioactive substances is 10% of the total flow through the purification system. Primary helium is taken to the HPS from a delivery plenum of the high pressure compressor. Purified helium is then returned to the low pressure compressor suction cavity. The HSS also supplies helium to other primary circuit sections, as necessary, such as the buffer seal of the turbo-machine.

The GT-MHR also has many other systems standard to an NPP such as plant control and monitoring. However, the PCS in the GT-MHR is the substitute for all heat conversion systems and respective steam/water systems (steam generators, steam turbine plant, condensate/feedwater system) which are characteristic of NPPs with the steam cycle.

4.4. PLANT SAFETY

The general safety of the public and staff living near the GT-MHR is ensured through the implementation of the defense-in-depth application of both a set of barriers preventing propagation of radioactive substances and provisions which ensure maintaining the effectiveness of the barriers in any anticipated accident situations. Maintaining the effectiveness of the protective barriers under emergency conditions is ensured through the high quality of systems and equipment design, fabrication, operation, maintenance, and the developed properties or the reactor's safety attributes and the use of safety systems. The combination of passive and active safety system principles provides significant time for personnel to implement decisions and actions for accident prevention and control.

4.4.1. Safety features

The high level of GT-MHR safety is ensured through the core specific structural features and neutron/physical properties as well as technical solutions applied in the design of the plant including the following:

(1) Use of helium as the primary coolant: The helium coolant is chemically inert and does not influence the neutron balance in the reactor due to essentially "zero" neutron absorption and scattering cross sections. This excludes the possibility of uncontrollable increase in the reactor power for coolant density variation or loss of coolant in emergency situations. The single phase state of the helium coolant (no liquid to gas transformation) excludes a potential for abrupt changes in heat removal conditions in the core. Low induced activity of helium also causes a low level of radiological consequence with leakage during normal operation and in primary circuit loss of integrity events.

(2) Design for a negative temperature coefficient of reactivity. Therefore, a core temperature increase caused by an unbalance between heat generation and removal from the reactor automatically leads to a reactor power decrease. This property ensures the self limitation of a fuel temperature rise in the event of an emergency such as loss of heat removal from the reactor or an emergency condition where the reactor shutdown mechanism(s) fail to actuate.

(3) Use of graphite as a core structural material which is capable of withstanding high temperatures (graphite sublimation temperature is ~3000°C): The core graphite structures retain strength at temperatures which significantly exceed the level which is possible under emergency conditions. This property provides the assurance of core configuration stability and the impossibility for both nuclear fuel relocation in the core under emergency conditions and uncontrollable reactor power excursion (due to nuclear fuel reconfiguration into a so-called "secondary" critical mass).

(4) Core and reactor structural features (annular geometry of the core and low power density): These design features in combination with the high thermal capacity of the core ensure the cool-down of the shutdown reactor under emergency conditions by passive removal of heat from the reactor vessel using heat emission, conductivity and convection, thus maintaining the fuel and core temperature within allowable (for safe operation) limits in all accidents including loss of coolant.

(5) Use of fuel in the form of small particles with multi-layer coatings made of PyC and silicon carbide which are capable of effectively retaining fission products at high temperatures (up to 1600°C) and fuel burn-up.

(6) Assurance of vessel integrity with an in-service diagnostic system to ensure the implementation of the "leak-before break" concept, thus eliminating large scale loss of integrity occurrences in the vessel system components.

The allowable level of fuel temperature to 1600°C is not exceeded in any accident with loss of heat removal from the reactor including those with failure of all “active” means for the reactor shutdown and cool-down. This is due to the above indicated structural features and neutron/physical properties of the core, and application of passive systems for heat removal from the reactor. The effectiveness of the main protective barriers, i.e. fuel kernel claddings is maintained thus preventing fission product release beyond the confines of fuel particles.

4.4.1.1. Safety barriers

There are five main protective barriers (Fig. 4.17) in the design of the GT-MHR to confine fission products. These include:

— The fuel kernel: The fuel is fairly effective in confining solid fission products and partially retains gaseous and volatile fission products. The fuel temperature level and burn-up are limited in the design in such a way as to ensure the most effective confinement of fission products immediately within the fuel kernel.

— Fuel kernel coatings: The next and most important barrier for fission product confinement are the protective coatings on each fuel kernel. The GT-MHR design uses fuel with coatings made of three layers of PyC of different density and one layer of SiC (see Chapter 2). Numerous experimental investigations into the coated particles of this type have validated their high capability to confine fission products up to the temperature of 1600°C which corresponds to the maximum possible temperature level under emergency conditions.

— Graphite: The graphite of the fuel compacts and fuel blocks where the coated particles are located is the next barrier to confine fission products. The graphite confines solid fission products (Se, Rb, Cs, Ba, rare-earth elements) fairly effectively, however it is practically "transparent" for noble gases.

— Primary coolant pressure boundaries (vessel system): The vessel system is the fourth barrier to confine fission products. The vessel system components (reactor vessel, PCS vessel and cross duct) have been developed in conformity with the requirements of

"Rules for Design and Safe Operation of NPP Equipment and Pipelines" and other codes and standards for NPP equipment design. The vessel svstem maintains its service characteristics (including leak-tightness) in all anticipated operating modes including emergencies.

— Containment: The weapons plutonium burning GT-MHR reactor module is housed within a reinforced concrete leak-tight containment which is located below ground level. Although the decision of a leak-tight containment for the commercial uranium burning GT-MHR has not, as yet, been made, the plutonium burning plant containment is designed to confine the helium-air medium at pressures and temperatures which could develop in the course of a primary circuit loss-of-integrity accident and also to withstand external loads caused by seismic impacts, air-crash, etc. Activity release from the containment into the environment is determined by the degree of its non-integrity (about 1 vol.% per day). During the gradual release of the helium-air medium from the containment the natural drop of its activity takes place caused by the decay of short-lived fission products.

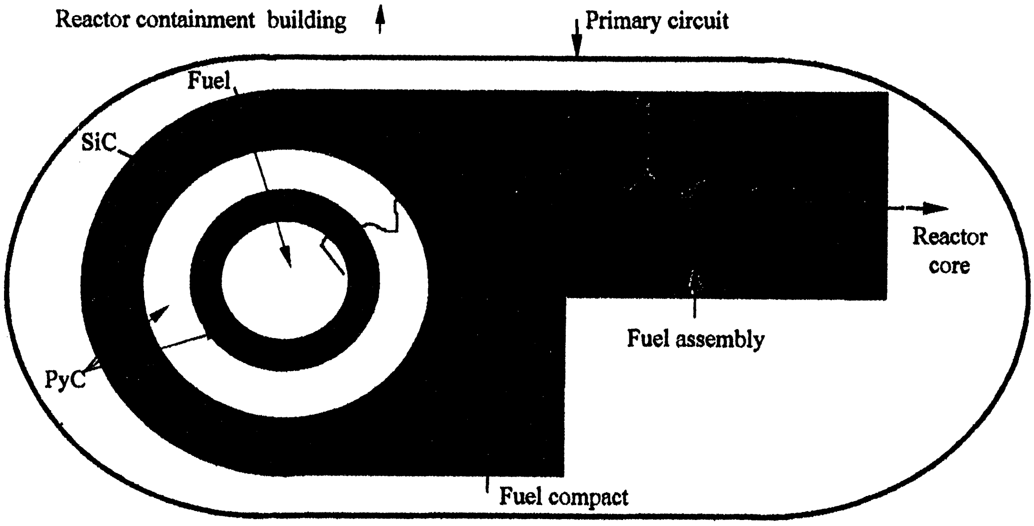


FIG. 4.17. Fission product barriers for the plutonium burning GT-MHR.

4.4.1.2. Accident prevention and mitigation

Maintaining effectiveness of the protective barriers and limiting their damage in all anticipated operating states represent the main task for ensuring safety during plant operation.

Reliability improvement has been achieved by reducing the number of station systems and their subsequent simplification, the use of standard equipment, and the performance of continuous monitoring and predicting system and equipment status, as well as by the use of systems based on the passive principle of operation. For example, the continuous functioning of the RCCS during operation and the passive principle of its operation based on natural processes, exclude the need of actions by operating staff or protection safety systems during NPP transition from normal operation to emergency cool-down mode.

Two independent systems are provided in the design to shutdown the reactor and maintain it in a subcritical state. The first system uses control rods being moved within channels in both the core and side reflector. The rods are inserted into the core during accidents under gravity. The second shutdown system (RSS) includes hoppers with boron carbide-base absorber balls and related channels in the core fuel columns. As the RSS hopper gates are opened the absorber balls flow into the core channels under gravity. The effectiveness of each shutdown system is sufficient to transfer the reactor and maintain it in a subcritical state. In a hypothetical situation with failure of both independent reactor shutdown systems, in the course of core heating the reactor power reduces down to full shutdown state due to the negative power reactivity effect. As this happens, the fuel temperature in the core does not exceed the allowable limit of 1600°C, which assure preservation of the fuel coating integrity. Maintaining the reactor in this state is possible for 24 to 30 hours (before sufficient loss of reactor poison following the termination of Xenon build up). This provides ample time for measures to be taken to insert the control rods or absorber elements into the core.

There are three systems for heat removal from the core in the GT-MHR: two normal operation systems (PCS. SCS) and one dual-train system (RCCS) which perform the functions of both normal operation and protection. Characteristics of the PCS and SCS (individually) ensure the possibility to cool-down the reactor to the level of repair temperature and to maintain it in the required temperature state in the shutdown mode. These systems remove heat by forced circulation of coolant in their loops. The turbo-generator operation in the motoring mode from an outside power source is provided to ensure helium circulation in the primary circuit during reactor cool-down by the PCS. The SCS is used to cool down the reactor when the PCS is inoperable (or switched off). The SCS can also be used for the cool-down of the plant in a loss-of-power event.

If it becomes impossible to use neither the PCS nor the SCS, the GT-MHR cool-down is performed by the RCCS. The heat released in the core is removed by heat transport of conduction through the reactor vessel and emission to the RCCS surface cooler, which is placed on the reactor cavity walls. The water circulates in the RCCS due to natural convection. The heat in the heat exchanger-evaporator is transmitted to the re-circulation water system which is used for cooldown and in loss-of-power supply events. In the event of a loss of water flow, the heat is removed by means of water evaporation in the heat exchanger and generated steam is discharged into the atmosphere. In all modes of the RCCS operation, the core temperature in the course of cool-down does not exceed the allowable limit of 1600°C in all accidents including loss of primary coolant. Continuous functioning of the RCCS during normal operation and in accidents and its passive principle of operation based on natural processes ensure the system continuous availability and actions are not required by the operator or control systems during transition from normal operation to the emergency cool-down mode.

One of the potential methods for degradation of the primary (essential) protective barriers, i.e. fuel coatings and graphite matrix of' the fuel compacts and fuel assemblies, is their corrosion during an accidental ingress of air or water (steam) into the reactor. Taking account the potential hazard of this phenomenon, technical measures are provided in the design that make the ingress of air or water into the core less likely and also allow for decreasing the consequences should this event occur. These measures include:

(1) The working pressure of the water in the PCS coolers and SCS helium/water heat exchanger is considerably less than helium working pressure in the primary circuit (about 7.0 MPa in the primary circuit versus 0.35 to 0.8 MPa in the intermediate water circuits of PCS and SCS coolers). This nearly eliminates the possibility of water ingress into the core during reactor power operation or cool down, if pressurized

conditions are maintained in the primary circuit. When starting up or cooling the reactor down under depressurized (close to atmospheric) conditions, the most likely water source within the reactor is the SCS cooler from which water (steam) could enter the reactor vessel bottom head. However in this case direct contact of water with the core graphite structures is excluded.

(2) Implementation of the "leak before break" concept limits the scope of a potential vessel system loss-of-integrity event, that in combination with the leak-tight containment minimizes accidental ingress of air into the reactor. Experimental investigation and calculations show that during a loss-of-primary circuit integrity accident within the potential range of realism, the helium is replaced by air due to the mass exchange and diffusion processes over a long duration and the amount of air flow is small through the place of leakage. Gases generated by the graphite oxidation process prevent the air coming to the core graphite structures and moderate the process of their oxidation. The graphite temperature decreases in the course of the reactor emergency cool-down to about 600°C which leads to shutting down of the oxidation process. According to the most conservative assessments made for the modular HTGR design, not more than 1% of the total graphite mass would be subjected to corrosion. Concerning energy release during oxidation, this would not exceed the residual heat release caused by fission products decaying in the shutdown reactor.

4.4.1.3. Safety related systems, structures and components

In the course of preliminary assessment of the GT-MHR safety, systems and components have been identified which exert the greatest influence on the plant safety during operation. These systems and components are:

(1) The reactor, including the core and internals;
(2) The vessel system and external primary circuit-related systems forming the high helium pressure boundary;
(3) The RCCS and its components;
(4) The reactor protection system, including sensors and logic instrumentation;
(5) The containment.

According to the safety concept of the GT-MHR, only a relatively small number of the station systems, structures and equipment are important to its safety. This is due to the utilization of passive safety systems and the safety characteristics designed into the GT-MHR.

4.5. GT-MHR SCHEDULE AND PLANS FOR DEPLOYMENT

The schedule for having the prototype single unit GT-MHR plant in operation at the site in the Tomsk region of the Russian Federation, is 2009. The first four module GT-MHR plant scheduled for operation, approximately five years thereafter. The overall strategy for the GT-MHR is:

— Use existing worldwide gas cooled reactor technology to the maximum extent possible
— Design and develop the remaining GT-MHR technology in the Russian Federation
— Construct and demonstrate the GT-MHR by building and operating the prototype unit at Seversk
— Build and operate several four module plants to burn weapons grade plutonium
— Use the GT-MHR design and technology for commercial applications with uranium fuel.

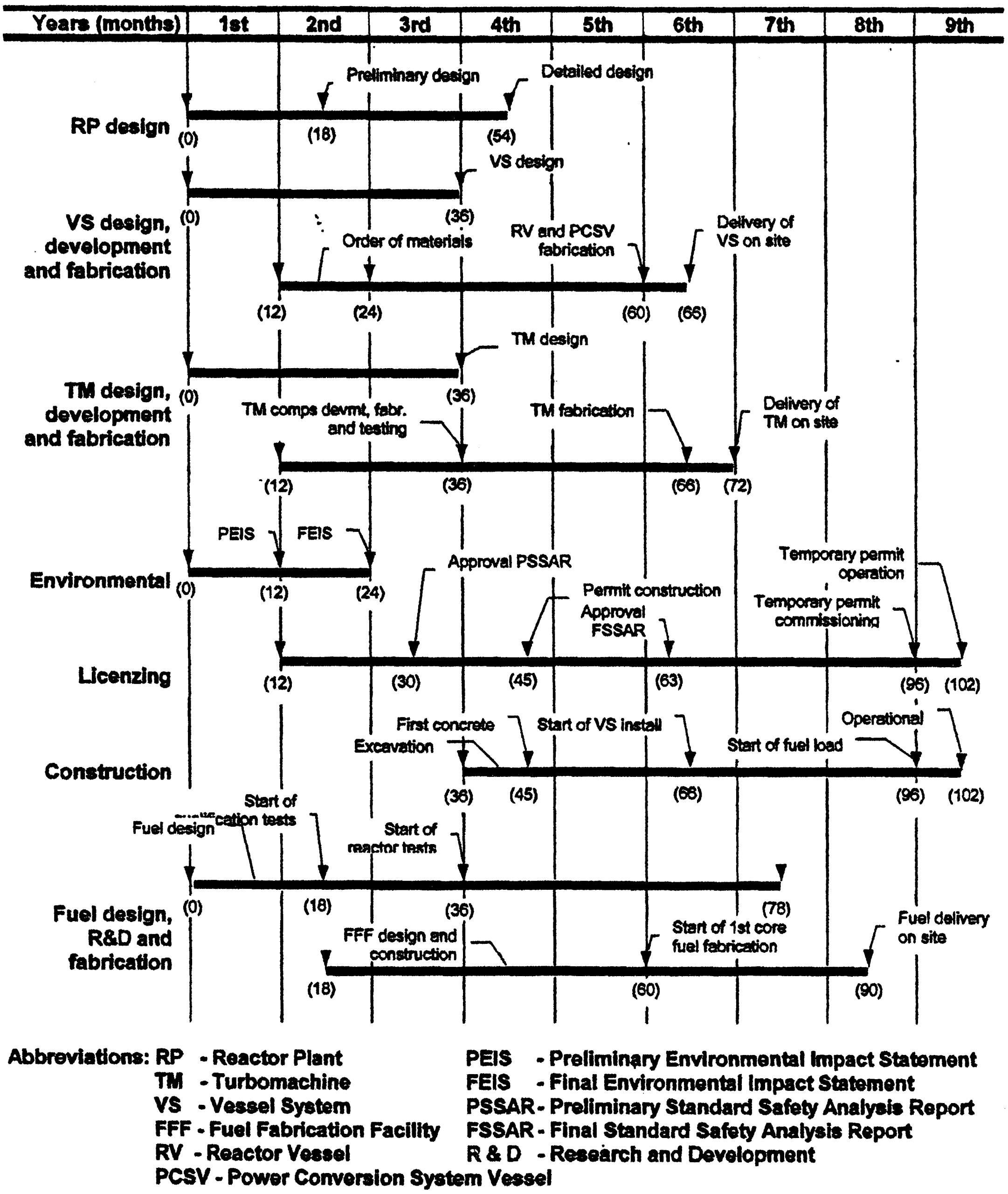


FIG. 4.18. GT-MHR project summary schedule [4-7].

4.5.1. Schedule for the prototype GT-MHR NPP

Principal work schedules were developed in the conceptual design both for the prolotype NPP and related plutonium fuel. All of the required stages of work stipulated by the Russian Codes and Standards were addressed in the schedules, including design, R&ED performance, construction and startup adjustment work. Terms of the fulfillment of this work correspond to the most favorable conditions, particularly: timely and sufficient financing, no negative impacts on the process of the NPP creation, etc.

The total duration of the prototype GT-MHR NPP creation is estimated as 100 months. The main stages of the schedule are given below as Fig. 4.18 (with terms in months from the beginning of work):

Total time required for performance of the work to create both the plutonium fuel and the pilot industrial fabrication facility is not more than 7 years from the beginning of the prototype NPP preliminary design development.

4.5.2. Priority near term activities

The following near term high priority GT-MHR development activities include:

- Review and update of the top level plant design requirements
- Advance with the GT-MHR preliminary design, including development of design documents for major systems and components
- Establish a bench-scale facility in VNIINM and prepare for the manufacturing of experimental batches of plutonium fuel
- Develop a preliminary design of a pilot production facility for manufacturing plutonium fuel
- Develop and prepare for testing of PCS turbo-machine components (separately), such as magnetic bearings, seals, turbine blades, etc.
- Develop and test the high temperature, high effectiveness recuperator
- Evaluate plant performance, plant transients, seismic response, availability, maintainability, PRA and safety studies of the GT-MHR
- Develop the preliminary safety information document (similar to the psar) and the preliminary environmental impact report
- Assess the technical and economic characteristics of the GT-MHR NPP on plutonium fuel as it applies to the site
- Develop a manufacturing, construction and start-up plan, including identification of potential fabrication vendors [4-7].

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Chapter 5

THE HTTR AND HTR-10 TEST REACTORS

Interest in the HTGR as an advanced nuclear power source for co-generation applications of electricity production and high temperature heat for industrial processes has resulted in the construction of the HTTR by the Japan Atomic Energy Research Institute (JAERI) and the HTR-10 by the Institute of Nuclear Energy Technology (INET) of Tsinghua University in Beijing, China. These nuclear test facilities have the capability of achieving core outlet temperatures to 950°C and 900°C, respectively, and will be utilized to support research and development activities including validation of HTGR safety and general performance characteristics, electricity generation via the gas turbine and validation of high temperature process heat applications [5-1].

5.1. HIGH TEMPERATURE ENGINEERING TEST REACTOR (HTTR)

5.1.1. HTTR background

In June 1987, the Japanese Atomic Energy Commission (JAEC) issued a revised "Long Term Program for Development and Utilization of Nuclear Energy" stressing that Japan should proceed to develop more advanced reactor technologies in parallel with the upgrading of existing nuclear reactors. It was recognized in this programme that the HTGR was not to be incorporated into the present existing power plant system, but the benefits that could be derived, such as its inherent safety and production of high temperature heat, are remarkable and should be pursued. Therefore, the promotion of R&D on the HTGR in Japan is quite significant from the viewpoint of a new nuclear technology frontier.

Within this program, the early construction of a test reactor in place of the experimental very high temperature reactor (VHTR) was recommended based on the estimated length of time (~10 years) to construct the plant and perform associated testing. Based on the conclusions by a special committee investigating the HTGR R&D plan for the JAEC, demand of nuclear heat applications to ~1000°C is expected to become strong in the early part of the 21st century. The committee requested that a test reactor making the most use of Japanese technology be designed and built to test and study advanced HTGRs for the future. Accordingly, the test reactor should have an in-core irradiation region equipped with the capability to test the threshold of fuel failure and for the irradiation of various materials.

The committee's report also stated that the reactor outlet coolant temperature should be 950°C, which is the highest temperature attainable considering the current technology level. For irradiation tests, a prismatic block type core structure was proposed, with a thermal power rating of 30 MW for securing adequate regions for irradiation tests at high temperature [5-6].

Based on these suggestions, JAERI suspended the previous programme where the experimental VHTR was defined as an initial step toward nuclear heat application development and proceed with the design and R&D specifically necessary to prepare the safety analysis report of the HTTR, with the objectives to:

— Establish and upgrade the technology base of the HTGR
— Perform innovative basic research in the field of high temperature engineering
— Demonstrate high temperature heat applications and utilization achieved from nuclear heat.

The Japanese government approved proceeding with the HTTR in its 1989 fiscal year budget. Construction began in March 1991 following submittal by JAERI of the HTTR safety analysis report and subsequent review by the Science and Technology Agency, and then, by the Nuclear Safety Commission.

5.1.2. General design features of the HTTR

The reactor core is designed to keep all specific safety features within the graphite blocks. The intermediate heat exchanger (IHX) is equipped to supply high temperature clean helium for process heat application systems, and the instrumentation and control system is designed to allow operations that simulate accidents and anticipated operational occurrences (AOOs). As the HTTR is the first HTGR in Japan and a test reactor with various purposes, it incorporates specific aspects regarding the safety design. JAERI established the following safety design principles for the HTTR in reference to the "Guidelines for Safety Design of LWR Power Plants", but taking into account the significant safety characteristics of the HTGR and the corresponding design requirements as a test reactor:

- Coated fuel particles shall not fail during normal operation and AOOs. To satisfy this principle, the maximum fuel temperature, including systematic and random uncertainties, shall not exceed 1600°C for any AOO.
- The reactor shall be shut down safely and reliably during operation using the control rod system. Furthermore, a reserved shutdown system (RSS) which is independent of the control rod system shall be provided.
- A severe accident resulting from control rod ejection must be avoided.
- The residual heat after reactor shutdown shall be removed safely and reliably for any AOO or accident.
- A containment vessel (C/V) shall be provided to prevent fission product release and excessive air ingress into the core in case of a depressurization accident.
- The pressure in the pressurized water cooling system (PWCS) shall be controlled so as to be lower than that of the primary helium gas to prevent a large water ingress into the core in case of rupture of a heat transfer tube in the primary pressurized water cooler (PPWC).
- The helium gas pressure in the secondary helium cooling system (SHCS) shall be controlled to be slightly higher than that of the primary helium gas to prevent fission product leakage from the primary cooling system (PCS) to the secondary due to a crack in a heat transfer tube in the IHX.
- The pressure and heat resisting functions of the structures, where the high pressure and high temperature coolant is contained, are separated to reduce mechanical loads on the high temperature metal structures [5-6].

Safe and reliable shutdown of the reactor from any operational condition is achieved with the control rod system. Furthermore, a reserved shutdown system composed of B_4C/C pellets is provided. The power control and normal reactor shutdown of the HTTR are achieved with 16 pairs of control rods or 15 pairs when the center column of the core is used for an irradiation test. The control rod system can achieve subcriticality from any operational condition and maintain subcriticality under cold core conditions including the postulated event of a pair of control rods stuck in the operational position. The major design specifications of the HTTR are shown in Table 5.1. The reactor outlet coolant temperature at the full power is set at both 850° and 950°C. The reactor operational mode at 850°C is defined as "rated operation" and at 950°C is "high temperature test operation" because the HTTR is not allowed to be operated at 950°C for the full life of the initial core. Tests such as the safety demonstration tests and irradiation tests are allowed only in the rated operation mode. The high temperature nuclear process heat utilization system will be operated at the high temperature test operational mode. The design life of permanent structural components in the HTTR plant is based on 20 years with a load factor of 60% of full power operation.

The HTGR has excellent safety capabilities with respect to the accidental release of fission products. Nevertheless, the HTTR is required to have a containment vessel to meet Japanese safety design guidelines for the light water nuclear power plants.

TABLE 5.1. MAJOR SPECIFICATIONS OF THE HTTR

Thermal power	30 MW
Coolant	Helium gas
Outlet coolant temperature	max. 950°C
Inlet coolant temperature	395°C
Fuel	Low-enriched UO_2
Fuel element type	Prismatic block
Direction of coolant flow	Downward
Pressure vessel	Steel
Number of main cooling loop	1
Heat removal system	Pressurized water cooler Intermediate heat exchanger
Primary coolant pressure	4 MPa
Containment vessel	Steel containment
Plant lifetime	20 years

5.1.3. HTTR plant layout and cooling system

The HTTR is located on JAERI's Oarai Research Establishment site which is approximately 100 kilometers north of the Tokyo metropolitan area and is near the Pacific Ocean. The plant area is 200 m × 300 m in size. The shortest distance between the HTTR reactor core and site boundary is about 280 m in the southwest direction. As illustrated in Fig. 5.1, the HTTR plant arrangement is comprised of the reactor building, spent fuel storage building, a machinery building, cooling towers, exhaust stack, a high temperature process heat utilization system and other auxiliary facilities. The reactor building of 48 m × 50 m in size is situated in the central area of the plant. The exhaust stack of 80 m in height is north of the reactor building for the air ventilated from the reactor building to be released to the atmosphere. The heat utilization system will be constructed south of the reactor building.

The reactor building includes five levels with three floors underground (Fig. 5.2). A steel reactor containment vessel of 18.5 m in diameter and 30 m in height is installed in the center of the reactor building. A refueling hatch is attached to the C/V above the reactor pressure vessel (RPV). Functions of the C/V are to:

(1) Contain fission products (FPs) as one of multiple barriers against FP release into the atmosphere, and
(2) Limit the amount of air ingress into the core in a primary pipe rupture accident.

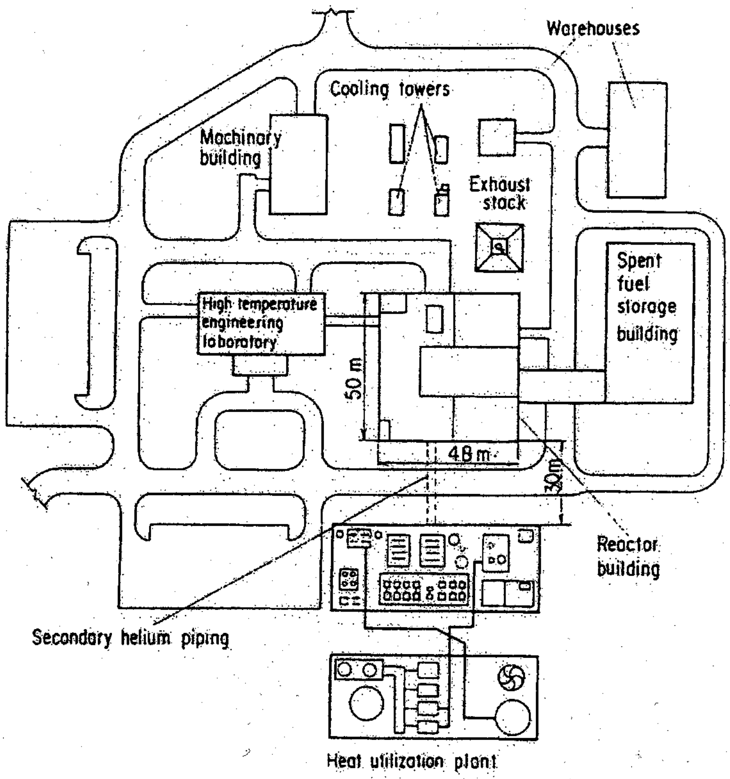

FIG. 5.1. HTTR plant arrangement.

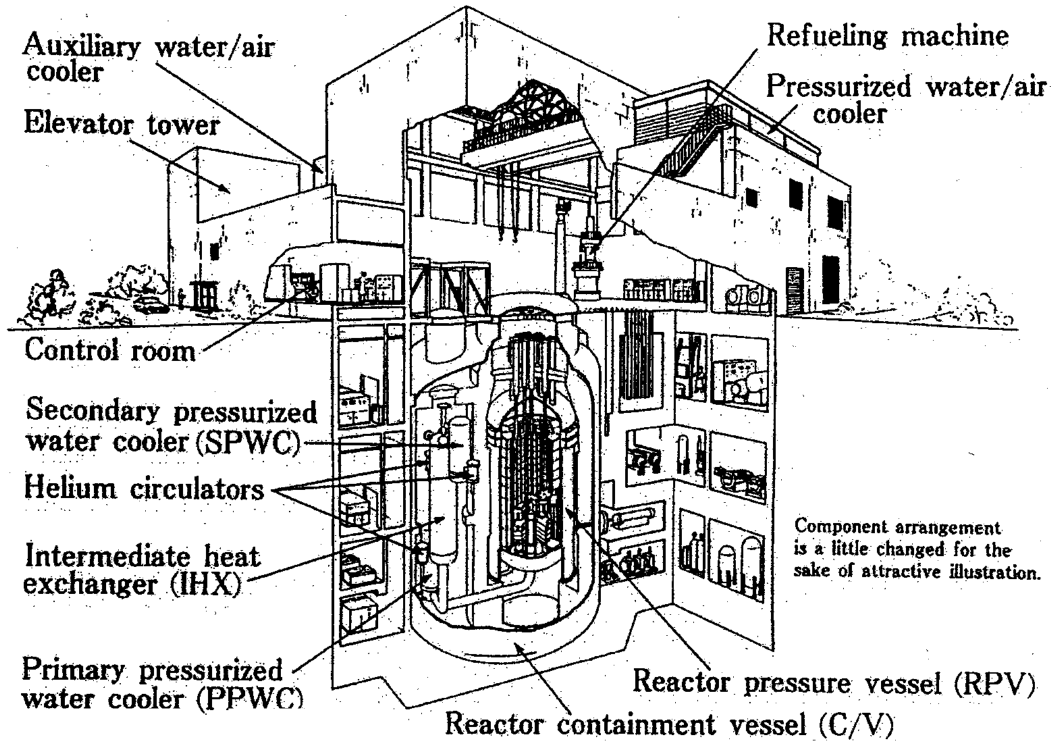

FIG. 5.2. HTTR reactor building.

The reactor pressure vessel (RPV) is formed as a vertical cylinder, with a hemispherical top and bottom head closures and 31 standpipes. The top head closure is bolted to a flange of the vessel cylinder. The standpipes include "control rod (CR) stand-pipes", "irradiation stand-pipes", and standpipes for instrumentation. The irradiation standpipes are utilized to introduce specimens and experimental equipment into the core. A thermal shield is attached to the inner surface of the top head closure to prevent the closure from overheating in a depressurization accident such as a primary helium pipe rupture. The RPV is of 2-1/4Cr1Mo steel normalized and tempered.

The double containment concept (RPV + C/V) was applied to the HTTR, because the safety features of the HTGR have not been developed in Japan. The HTTR has multiple barriers to fission product release, namely, the fuel coatings, the RPV boundary, the C/V and the reactor building. Most HTGRs being designed in other countries also include these barriers except the C/V. Some compartments surrounding the C/V in the reactor building serve as confinement, or service area. The service area is maintained at a slightly negative pressure to atmosphere by a ventilation and an air conditioning system during normal operation and accident conditions. The barriers of the C/V and the service area significantly reduce the off-site radiation dose in an accident condition such as a primary helium pipe rupture. Major components such as primary cooling system components as well as the RPV (Fig. 5.3) are contained within the C/V.

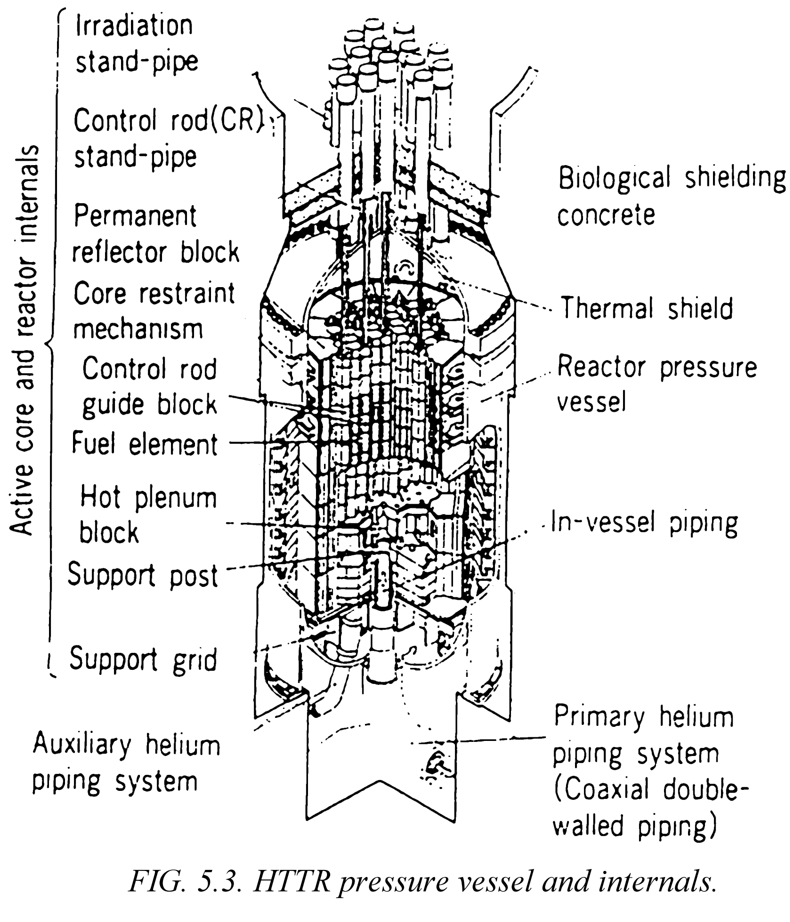

FIG. 5.3. HTTR pressure vessel and internals.

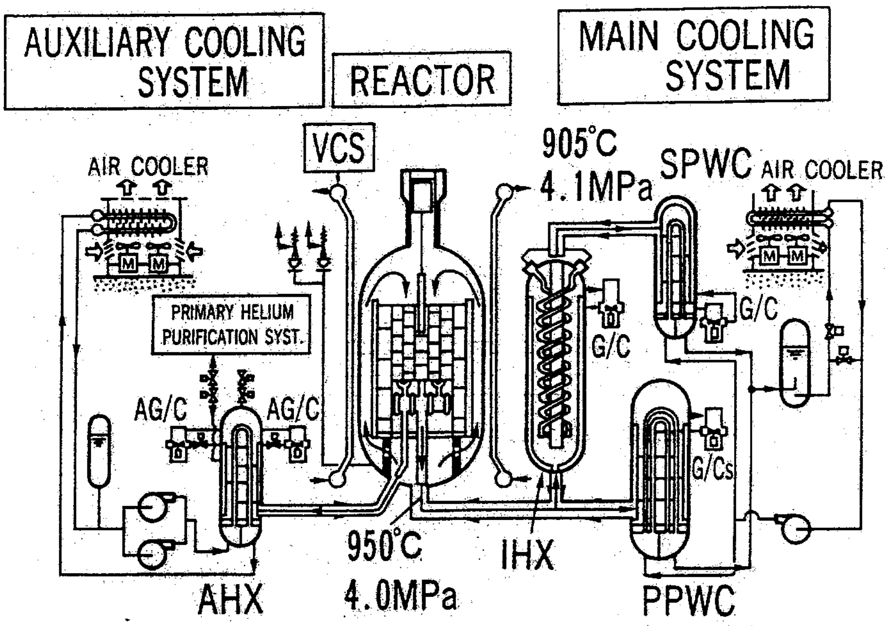

FIG. 5.4. HTTR cooling system.

The flow diagram of the reactor cooling system is shown in Fig. 5.4. This system is composed of a main cooling system (MCS), an auxiliary cooling system (ACS) and two reactor vessel cooling systems (VCSs). The MCS removes the heat energy from the reactor core during the normal operation, while the ACS and VCSs are functioned as engineered safety features and remove the residual heat energy after a reactor scram. As the core restraint mechanism requires protection against thermal damage from reactor heat during normal operation and anticipated operational occurrences, the ACS functions as protection by forced-cooling the restraint mechanism.

During a reactor scram, gas circulators of the MCS are shut to protect the heat transfer tubes of the two pressurized water coolers against overheating. In an AOO and accident condition when forced cooling of the core is available, the ACS automatically starts in responce to the reactor scram signal. The VCS functions as a residual heat removal system when forced circulation in the primary cooling system is no longer available due to a rupture of its piping system. It also operates during normal operation to cool the reactor shielding concrete wall.

The MCS consists of a IHX, a primary pressurized water cooler (PPWC), a secondary pressurized water cooler (SPWC) and pressurized water/air cooler. The MCS has two operational modes; “single loaded operation”, and “parallel loaded operation”. The PPWC functions to remove the reactor heat of 30 MW during the single loaded operation, while during parallel load operation the IHX removes 10 MW and the PPWC removes 20 MW. The SPWC serves the function of removing the heat from the IHX. The heat removed by the PPWC and the SPWC is transported through the pressurized water at 3.5 MPa. The pressurized water is then cooled down by the air cooler. In the HTTR reactor plant, the reactor heat of 30 MW is eventually transferred to the atmosphere by the pressurized water and the air cooler. During normal operation, the pressure of the secondary helium is controlled to always be 0.1 MPa higher than that of the primary helium at the IHX heat transfer tubes in order to reduce the pressure load on the tubes and to protect for accidental leakage of radioactive materials into the secondary helium. The water pressure is controlled so that a large amount of water can not ingress into the core with a PPWC tube rupture accident.

The auxiliary helium transfers a small fraction of the reactor heat to pressurized water. Eventually, the reactor heat is dissipated to the atmosphere at the auxiliary water/air cooler. The ACS consists of an auxiliary heat exchanger (AHX), two auxiliary helium circulators and an air cooler. At the AHX, the auxiliary helium is cooled by water. During normal operation, a small flow of auxiliary helium (~200 kg/h) passes through the AHX to the primary helium purification system so as to remove impurities contained in the reactor coolant. With a reactor scram, while the reactor coolant pressure boundary remains intact, the auxiliary helium cooling system automatically starts and transfers the residual heat from the core to the auxiliary air cooler. The AHX has heat transfer capacity of approximately 3.5 MW.

Two vessel cooling systems are provided as protection of the reactor core and the RPV against thermal damage by residual heat after a reactor scram when the ACS cannot or fails to cool the core. Each of these systems is capable of controlling temperatures of the core and RPV within safe limits and consists of water-cooled panels surrounding the RPV with two cooling water systems. Cooling tubes with fins form the panels and are arranged so that adjacent tubes do not belong to the same system and a tube failure will not danger the RPV and core. The heat removal rate from the RPV to the panels is designed as 0.6 MW so as to effectively remove heat to meet the requirement for the maximum allowable normal fuel temperature of 1495°C and also 0.3 MW or more with an accident condition where the reactor core is not cooled by the ACS. The VCS is also an engineered safety feature equipped with two independent complete sets which are backed up with an emergency power supply. It is operated even during normal operation in order to cool the biological shielding concrete wall.

The IHX is a helically coiled counter flow type heat exchanger. To minimize constraints of axial and radial thermal expansion of the helically coiled heat transfer tubes, a floating hot header with a combination of a central hot gas duct passes through the central space inside the helix bundle. An assembled tube support allows free thermal expansion of a helix in the radial direction. The primary helium enters the IHX through the inner pipe of the primary concentric hot gas duct attached to the bottom of the IHX. It flows up outside the tubes thereby transferring nuclear heat of 10 MW to the secondary helium and flows back to the annular space between the inner and outer shells, The secondary helium flows down inside the heat transfer tubes and flows up through the center as hot gas. A double-walled shell with thermal insulation attached to the inside surface of the inner shell provides reliable separation of the heat resisting and pressure retaining functions. Cold helium flowing through the annulus brings uniform temperature distribution throughout the outer shell which serves the function of being the pressure retaining member.

5.1.4. HTTR fuel and core design

The active core is 2.9 m in height and 2.3 m in equivalent diameter, and consists of fuel elements of hexagonal blocks and control rod (CR) guide hexagonal blocks. In defining the core configuration, the term “column” is used for an axial row of prismatic hexagonal blocks. The active core is comprised of 30 fuel columns and 7 CR guide columns. Twelve replaceable reflector columns, 9 CR guide columns and 3 irradiation test columns surround the core, as shown in Fig. 5.5.

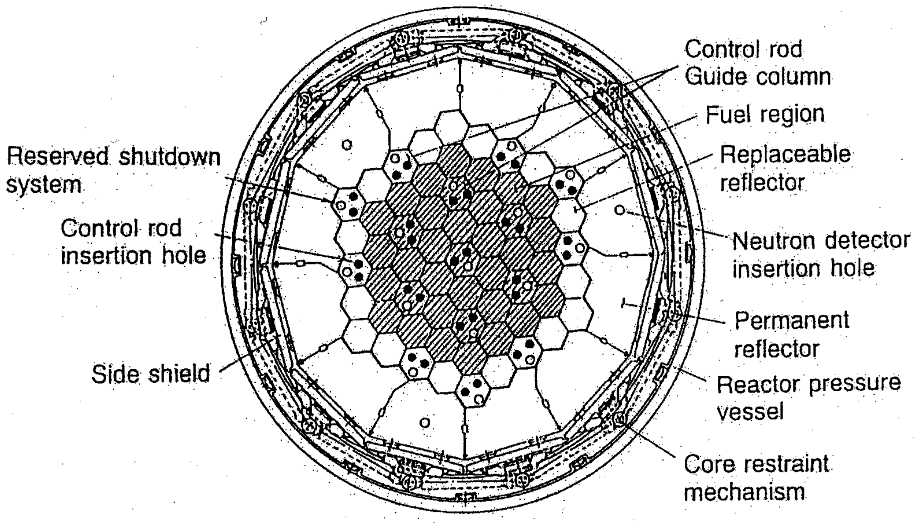

FIG. 5.5. Horizontal arrangement of the HTTR core.

Permanent reflector blocks are tightened by the core restraint mechanism. The fuel elements of the initial HTTR core are pin-in-block type and each element consists of fuel rods and a hexagonal graphite block, 360 mm in width between flats and 580 mm in length, as shown in Fig. 5.6. The fuel element has three dowels on the top and three mating sockets at the bottom to align the fuel elements. TRISO coated fuel particles with $U0_2$ kernels, approximately 6 wt% in averaged enrichment and 600 µm in diameter, are dispersed in a graphite matrix and sintered to form a fuel compact. Fuel compacts are contained in fuel rods of 34 mm in outer diameter and 577 mm in length. Fuel rods are inserted into vertical holes in the graphite block. The reactor coolant flows downward through annulus gaps between the hole and rod .

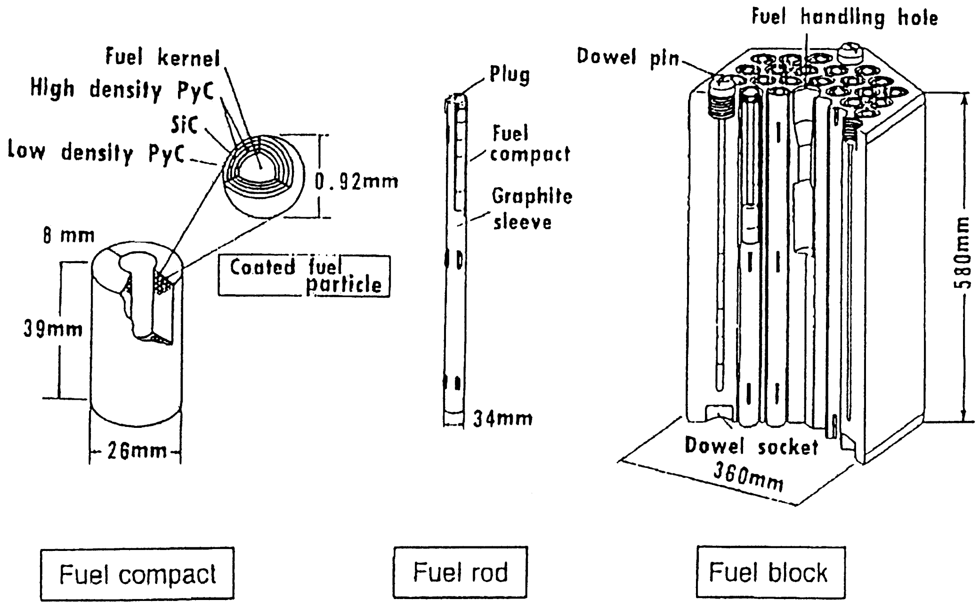

FIG. 5.6. HTTR fuel element.

The block type fuel has been adopted for the HTTR core due to the advantages of fuel zoning, control and coolant flow rate in each column, easy insertion of control rods, and irradiation flexibility in the core. The HTTR has 16 pairs of CRs which are driven by control rod drive mechanisms (CRDMs) contained in the stand pipes at the top head closure of the reactor pressure vessel. The core is cooled by helium gas of 4 MPa flowing downward. A control rod guide block at the center of the core can be replaced with graphite baskets with irradiation holes [5-6].

The basic design criteria for the fuel are to minimize the failure fraction of as fabricated fuel coating layers and to avoid additional fuel failures during operation. To meet the latter criterion, the fuel temperature is limited below 1495°C during normal operation and below 1600°C during AOOs, and the fuel burn-up is limited to 33 000 MW•d/t based on the results of irradiation tests [5-2]. On these basic considerations, the safety design requirements for the HTTR fuel were defined as follows:

— The initial failure fraction in the coating layers of the coated fuel particles shall be less than 0.2% in terms of the sum of heavy metal contamination and SiC defects, while the expected fraction is less than 5×10^{-4}. The value of 0.2% was determined from the viewpoint of limit of off-site exposure during normal operations.
— The coated fuel particles shall not fail systematically under normal operating condition, i.e. the penetration depth of the palladium-SiC interaction shall not exceed the thickness of the SiC layer of 25 μm. The kernel migration shall not exceed the thickness of 90 μm, which is the sum of the first and second layers.
— The coated particles shall be designed so as to avoid, in principle, failure considering irradiation induced damage and chemical attack through the full service period, i.e. the additional failure fraction in the coating layers of the coated fuel particles shall be less than 0.2% through the full service period. The value of 0.2% was determined in the same manner as that for the initial failure fraction.
— The maximum fuel temperature shall not exceed 1600°C at any AOO to avoid fuel failure.

The coated fuel particles consist of a spherical fuel kernel of low enriched UO_2 with TRISO coating. The TRISO coatings consist of a low density, porous PyC buffer layer adjacent to the fuel kernel, followed by a high density PyC layer, a SiC layer and an outer PyC coating. The coated fuel particles are incorporated into the fuel compact with graphite matrix. The fuel rod, which is composed of fuel compacts and the graphite sleeve, is contained within a vertical hole of a graphite block.

Reactivity is controlled through CRs, which are inserted into the CR guide columns in the active core and the replaceable reflector columns. A reserved shutdown system (RSS) is provided as a back-up system for reactor shutdown by means of inserting boron carbide/graphite pellets into the third channel in each CR guide column. The neutron absorber is made of B_4C/C, and is sheathed with a cylindrical clad of Alloy 800H. All pairs of CRs are individually supported by CRDMs which are housed inside the stand-pipes. The CRDM withdraws and inserts a pair of the CRs during normal operation. The CRs drop into the active core and reflectors by gravity during scram. A shock absorbing mechanism provides reduction in speed of CRs at the last stage of scram. During a scram, sheath temperature reaches approximately 900°C. To prevent thermal damage of the sheaths, nine pairs of CRs are first inserted into the replaceable reflector column holes. The remaining seven pairs are inserted into the active core column holes after the core is cooled down (40 minutes later) when the reactor outlet temperature has decreased below 750°C.

5.1.5. Operation of the HTTR

The HTTR has two operational modes; one is "single loaded operation", the other "parallel loaded operation". Furthermore, the HTTR has a rated operation and a high temperature test operation. A simplified process flow diagram of the MCS is shown in Fig. 5.7. This figure indicates the coolant flow path and the values of flow rate, including pressure and temperature at key stations at rated power in the parallel loaded high temperature test operation. For the primary helium cooling system, a reactor coolant of 36.7 t/h (10.2 kg/s) in total mass flow rate is heated up to 950°C through the active core in the high temperature test operation mode. The reactor coolant of 36.5 t/h or 99.5 per cent of the total flow rate, flows into the MCS. The remaining 0.5 per cent of the reactor coolant, i.e. 200 kg/h, flows into the ACS. Two-thirds of the primary helium enters the PPWC and the remaining one-third goes to the IHX.

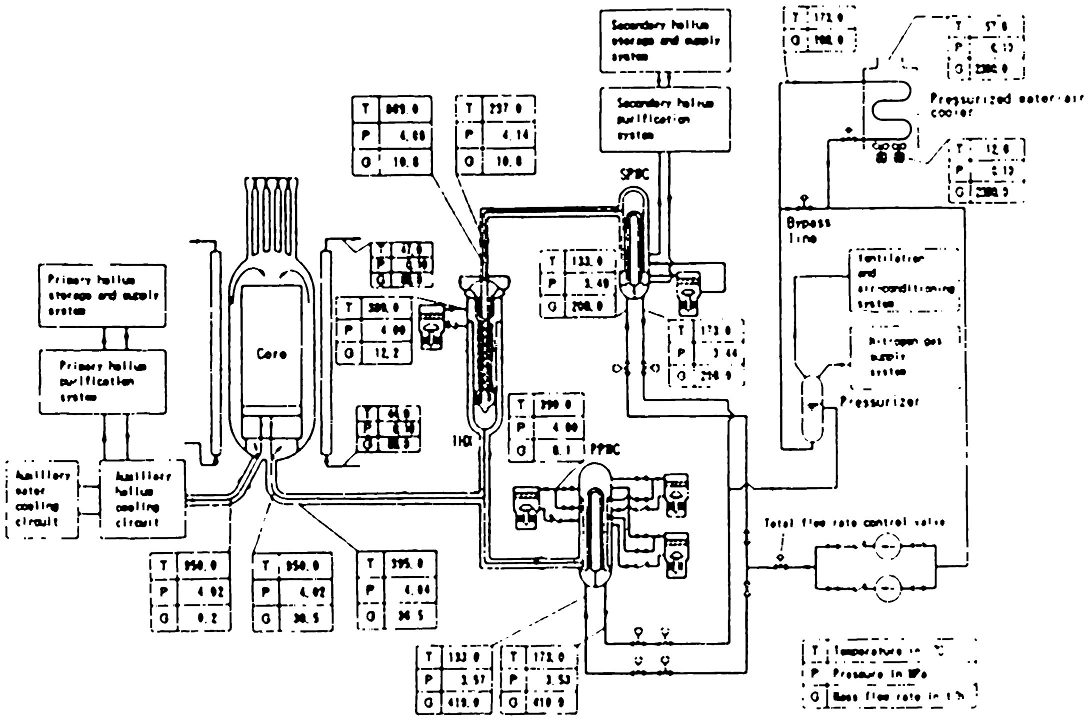

FIG 5.7. Flow diagram of the main cooling system in the parallel loaded, high temperature test operation mode.

Each helium path returning to the reactor is cooled to approximately 390°C. In the concentric hot gas duct, the cooler helium is heated up by hot helium flowing internal to the duct, which results in a temperature of 395°C at the inlet of the RPV The primary helium pressure is maintained at 4.04 MPa at the inlet of the RPV, and adequate mass flow is maintained by controlling circulator speed. The secondary helium temperature is strongly dependent upon the operating conditions of the pressurized water cooling system.

Under nominal conditions, the secondary helium is heated up to approximately 870°C, with the return secondary helium cooled to approximately 237°C by the SPWC. The secondary helium pressure is maintained at ~0.074 MPa higher than the primary helium pressure. The pressure control is achieved by the secondary helium storage and supply system via the secondary helium purification system. At the rated power operation, a temperature increase of 40°C is produced in the pressurized water through the PPWC and SPWC. After mixing, the water enters the air cooler that is mounted on the roof of the reactor building. Water flow rate through the air cooler is controlled to maintain the heat transfer rate of

30 MW with a flow rate control valve at the inlet of the air cooler as the atmospheric temperature varies.

5.1.6. HTTR safety requirements

The HTTR will be the first nuclear reactor to achieve a reactor outlet coolant temperature of 950°C. At this temperature, due to the possibility of damage to the reactor internals, strict operational limits and conditions are specified in the operation of the HTTR reactor plant (Table 5.2). These are for the parallel loaded high temperature test operational mode. The safety design acceptance criteria for AOOs and accident conditions are shown in Table 5.3. The fuel under normal operation condition is set to meet the design fuel temperature of 1600°C at the most severe AOO of quasi-steady overpower operation.

TABLE 5.2. NORMAL OPERATION LIMITS

Maximum fuel temperature	1,495°C
Maximum RPV temperature	440°C
Maximum pressure of reactor coolant	4.05 MPa
Maximum temperature of IHX heat transfer tube	955°C

TABLE 5.3. ACCIDENT SAFETY DESIGN CRITERIA

Core integrity	The reactor shall not be seriously damaged and sufficient cooling capacity for residual heat removal shall be maintained
Maximum RPV temperature	550°C
Maximum reactor coolant pressure	5.75 MPa
Maximum IHX heat transfer tube temperature	1000°C

The reactor coolant pressure boundary dimensions for the structures are specified primarily based on design temperature, design pressure and other associated mechanical loads. These design requirements have been determined to be not less than the maximum values over their time histories which exist under the most severe loadings for normal operation in a given zone of component. These design parameter values include allowances of control system error, system configuration effects and measurement inaccuracy. Moreover, service temperature and pressure limits are imposed on the reactor coolant pressure boundary as safety design acceptance criteria for AOOs and accidents as well as normal operation. Service pressure limits were stipulated as follows so that requirements for acceptance of the mechanical design of the reactor coolant pressure boundary structures are satisfied:

(1) At AOOs, the pressure shall be less than 1.1 times the design pressure or the maximum pressure in service.

(2) At accidents, the pressure shall be less than 1.2 times the design pressure except for the IHX heat transfer tubes and central hot gas dust. The IHX structures forming the boundary separating the primary and secondary helium shall withstand a creep buckling load for a pipe rupture accident on the secondary helium piping system. This accident is the most severe for these structures .

The maximum allowable rate of temperature heatup and cooldown for the reactor coolant under normal operation depends upon the coolant temperature and, therefore, upon the metal temperature of the structural component in contact with the coolant. Based on parametric analysis of the structural integrity of the IHX hot header and reducer, the maximum allowable reactor coolant rate of temperature change has been limited to 15°C/h at a temperature of 650°C or above.

5.1.7. HTTR tests and schedule

Fuel loading of the HTTR reactor and the subsequent initial criticality testing occurred in late 1998. Criticality was first demonstrated in an annular core configuration on 10 November, and with the first full core on 16 December 1998. The plant is now engaged in a comprehensive testing programme associated with evaluation of the HTGR. This programme is divided into six major categories; safety, thermal hydraulics, fuel, high temperature components, core physics and control/instrumentation. These tests basically focus on evaluation and confirmation of the safety of the HTGR through actual demonstration on the HTTR. The testing will also demonstrate the safety concept of the (Japanese) severe accident free HTR (SFHTR) for licensing of the next generation of HTGRs. These programmes will be submitted to the IAEA Co-ordinated Research Program on Evaluation of HTGR Performance, so that all participants can use HTTR measured data for their future generation HTGRS.

By using the HTTR for safety demonstration tests, the two most severe postulated events for the SFHTR are simulated. These include a loss of forced cooling accident and control rod withdrawal accident simulation. These tests will be performed in order to confirm the safety concept of the SFHTR and the other future generation HTGRs. The loss of forced cooling accident test has been partially licensed and simulates conduction cooldown behavior during depressurization accidents. In this test, the primary coolant system is shutdown at rated power operation, without initiation of an active direct core cooling system. During the depressurization accident in the SFHTR, natural circulation occurs in the RPV and transfers heat from the core to the outside. This test will not simulate natural circulation because a guillotine break of the primary piping cannot be simulated. However, the flow rate of the natural circulation is low in this low pressure condition and heat transfer by natural circulation is limited. It is confirmed in the HTTR safety analysis that the maximum fuel temperature between the depressurization accident with and without the natural circulation is not significantly different.

Control rod withdrawal accident simulation has been partially licensed and this test stimulates fuel temperature rise by reactivity addition. In a preliminary test, a pair of control rods is withdrawn from the core by a CRDM. In a secondary test (not licensed now), a capsule containing a reactivity absorption material such as B_4C balls is installed in the center of the core and then ejected from the core. The maximum fuel temperatures during both simulation tests will be measured by temperature monitoring elements installed in the fuel blocks.

In HTGRs, even in a depressurization accident and no direct core cooling, the core temperature is not expected to exceed the temperature limit. In the HTTR, the VCS passive heat removal system installed around the RPV removes the heat from the RPV by radiation and natural convection and keeps the core below the temperature limit. The amount of heat transfer from the RPV to the VCS will be measured at 0, 30, 50 and 100% of rated power conditions and also after the transient of a reactor scram. RPV and VCS temperatures will also be measured to confirm that hot spots do not exist on the RPV and to evaluate the amount of heat being transferred by radiation or by natural convection. In addition to these tests, a plan to change the water cooling VCS to a complete passive air cooling VCS has been studied for tests in the future testing. The best system to cool the reactor core and keep the fuel temperature lower than the limit will be selected after these tests.

During start-up and normal operation, the following tests are planned concerning investigation of fuel and fission product behavior:

(1) Initial fuel performance test. Performance of the first fuel loading will be examined by fractional releases of gaseous fission products.
(2) Fuel temperature measurement test. Fuel temperature will be measured by melt wires. Validation of the temperature calculation codes is carried out by comparison of the measured and calculated temperatures.
(3) Fission product plateout measurement test. The release fractions and plateout distribution of condensable fission products will be investigated by the measurement of plateout distribution in the primary cooling system of the HTTR.
(4) Organic iodine measurement test. Fractions of organic iodine in the HTTR will be measured by sampling of the primary coolant as a function of coolant temperatures during the power-up test. Based on measured data, the mechanism of organic iodine formation in the HTGR system will then be investigated.
(5) Dust sampling test. Dust in the primary coolant will be a source term in depressurization accidents of the HTGR. Amount, composition and size of dusts will be measured and basic data will be accumulated during the start up test.

Future tests of ZrC coated fuel specimens will be conducted after testing of the initial HTTR fuel loads.

In the area of thermal hydraulics, a total code system to evaluate the fuel and internal core structure temperatures will be developed and validated. It includes steady-state fuel temperature evaluation code (FLOWNET, TEMDIM) in-core transient heat transfer analysis code (TAC-NC, SSPHEAT), and the plant dynamic code (ACCORD). This code system will be used to design future generation HTGRs. During operation, direct comparison between experimental and analytical fuel temperatures cannot be conducted because the fuel temperature is too high to be measured. However, the inside and outside temperatures of the permanent reflector blocks around the reactor core, inlet and outlet helium gas temperature, helium gas temperatures in hot plenums, etc. will be measured and these experimental data will be utilized for validation of fuel temperature evaluation codes. After operation, temperature monitors installed in fuel blocks will be removed, and analytical and experimental fuel temperatures can then be directly compared. Also dynamic tests such as step and ramp change of thermal power and reactor scram will be carried out to validate plant dynamics and in-core transient analysis codes.

Control and instrumentation tests concerning an adaptive neural network methodology will be conducted to investigate its applicability to HTGRs and for reliability. In this test, real time operational data in the HTTR are sent to a neural network system. Researchers for this system are to then locate abnormal events in the HTTR without any information from the plant operators.

Tests concerning high temperature components will focus on the IHX, which was developed for the HTTR. Thermal performance tests under steady state conditions will be carried out to obtain the heat transfer coefficient in the tube side, heat transfer coefficient on the shell side, over-all heat transfer characteristics, temperature of the manifold, heat transfer tube, radiation promoter effect, etc. Also, the temperature of manifold, heat transfer tube, etc. will be measured under transient conditions, and stress and strain evaluations using actual temperature conditions will be conducted to investigate deterioration of thermal performance after the first high temperature operation to prolong life time [5-3].

The programme to test high temperature industrial heat applications coupled to the HTTR is planned upon completion of the safety demonstration programme. Details of this future activity are provided in Chapter 7.

5.2. THE HIGH TEMPERATURE GAS COOLED REACTOR TEST MODULE (HTR-10)

5.2.1. China's energy situation

China, with its very large population, is ranked third in commercial energy consumption in the world. It is also on the threshold of major increases in economic growth and corresponding industrial development resulting in very large additional energy requirements for the future. This is particularly acute in the area of electricity demand which is expected to more than double in the next two decades. However, within China's primary energy mix, coal is dominant and provides about ¾ of the total energy needs (e.g. in 1990, coal was 75.6%, oil 17%, natural gas 2% and hydropower at 5%). This energy mix based on coal consumption is leading to serious environmental issues and has other disadvantages including very high CO_2 emissions and corresponding serious air pollution and a low energy utilization efficiency. Shortage of commercial energy supply has also been a long standing problem in China, particularly with electricity. It is estimated that the gap in electricity supply has reached about 80 TW•h/year.

The Chinese economy has been developing very rapidly. The national economic development goal is that the GDP value is to reach 24 590 billion RMB Yuan by the middle of this century in contrast to 1769 billion RMB Yuan in the year 1990. To realize this, a corresponding increase in the energy supply is required. Table. 5.4 provides projections of the increase in population, GDP value, and primary energy and electricity demand. It can be seen that the primary energy demand will increase by a factor of 3.2 to 3.7 from 1990 to 2050 and the electricity demand by a factor of 7.2 to 7.7.

TABLE. 5.4. PROJECTIONS OF ECONOMIC DEVELOPMENT AND ENERGY DEMAND

Year	1990	2000	2020	2050
Population (Million)	1134	1294	1455	1498
Annual GDP growth rate (%)*		8.0	5.0	3.0
GDP (Billion, RMB Yuan)	1769	3818	10131	24590
Annual growth rate of energy consumption*		4.02–4.20	2.10–2.48	1.18–1.39
Total demand for primary energy (Mtce)	987	1464–1490	2218–2434	3157–3682
Annual growth rate of electricity demand*		6.31–6.43	4.03–4.20	1.94–2.02
Electricity demand (TWh)	623	1149–1162	2532–2646	4508–4814

* Referring to the average % figure over the previous period before that year.

It will be a significant challenge to meet the projected demands on electricity generation and primary energy supply. China is richly endowed with coal resources that are estimated at about 967 billion ton. But the projected coal production capacity around 2050 is about 2000 Mtce. In this projection, account is not taken for the possible lmitation on coal consumption due to its environmental effects. The argument is commonly shared that coal has to be the overwhelming energy supplier throughout most of the next century. But this could come into serious question if the greenhouse effect would eventually prove to be unacceptable [5-5].

5.2.2. Nuclear energy development for the future

Nuclear power technology represents an excellent solution to help meet China's demands for electricity generation and primary energy supply. China has an established nuclear industry system with plants now in operation, and has proven nuclear energy resource deposits that allow self sufficiency.

Within the framework of China's general energy substitution strategy, nuclear energy is recognized to provide the following specific attributes:

(1) Development of nuclear power within areas of energy shortage can increase electricity supply, and lighten the heavy burden of coal transportation on railway systems, since nuclear power does not require significant fuel transportation. This is particularly important in coastal regions and northeast industrial zones.
(2) Providing large cities and industrial centers with nuclear heating can effectively mitigate local environmental pollution, as nuclear heating will not release the harmful pollutants that are discharged by conventional coal fired boilers. It is also of particularly significance that China does not have enough oil and gas fuel to provide industrial process heat and space heating. In addition, thermal energy cannot be transmitted for long distances in a manner comparable to electricity. However, today's progress of nuclear heating technology has made it possible for it to be a good source in large city and industrial sectors.
(3) Providing high temperature nuclear heat for energy/coal conversion to produce liquid fuel can alleviate shortages of this fuel supply with the perspective of raising the living standard due to the large automotive market potential.

China's intent is to continue to expand nuclear power capacity in order to ease its energy shortage problem and the negative environmental effects of fossil fuel consumption. Initially, HTGR technology is to be developed as an important supplement of the PWR to generate electricity. Despite the vast territory of China, the population is relatively concentrated, especially in the industrial centers, where large increments of electricity are needed. Therefore, the current development of nuclear energy would be limited by plant siting. The HTGR poses greater freedom in the selection of plant siting due to its significant safety characteristics. In addition, direct or indirect gas turbine cycles or the gas/steam combined cycle can be utilized with the HTGR. The conversion efficiency for this plant is in the range of 47% in contrast to 33% by the present LWR plants. Together with LWRs, the HTGR is expected to be a good contribution for China's nuclear electric generation programme of the future. Also under investigation is the use of the HTGR as a future supply of high temperature heat for industrial processes including coal gasification or liquefaction to produce liquid fuels to alleviate shortfalls in these fuels [5-5].

5.2.3. HTR-10 background and general design

China has a substantial programme for the development of advanced reactors which have favorable safety features, economic competitiveness and uranium resource availability. Their assessment is that the HTGR, with its unique capability to provide coolant temperatures to 950°C, has two significant benefits; efficient electrical power generation, and it can supply process heat for a variety of industrial applications. China recognizes the advantages of the modular HTGR, particularly in the area of safety and has decided to develop this technology. The HTR-10 is the first important step of modular HTGR development. It is projected as part of the framework of China's High Technology R&D Programme. The HTR-10 project was approved by the State Council in March 1992, and is being constructed on the site of INET, located in the northwest suburb of Beijing. INET is responsible for design, license applications, construction and operation of this test reactor. Construction completion and initial criticality of the HTR-10 is scheduled for the year 2000.

The objective of the HTR-10 is to verify and demonstrate the technical and safety features of the modular HTGR and to establish an experimental base for developing nuclear process heat applications and the gas turbine cycle for electricity production. The specific goals of the HTR-10 have been defined as follows:

- Acquiring expertise in the design, construction and operation of HTGRs.
- Establishing an irradiation and experimental facility for fuel elements.
- Demonstrating the inherent safety features of the modular HTGR.
- Testing electricity/heat co-generation and gas turbine technology.
- Carrying out R&D on high temperature process heat application.

The design criteria and the safety analysis report for the HTR-10 were approved by the NNSA in August 1992 and March 1993, respectively. The basic design and budget estimate were carried out and subsequently approved by the State Education Commission and the State Science and Technology Commission in 1994. The detailed design of the components, systems and buildings was then carried out by INET under cooperation with sub-contractors responsible for the helium purification system and other helium auxiliary systems including the turbine generator system and its building. For the detailed design of the main components e.g. the reactor pressure vessel, the steam generator and the helium circulator, design engineers of INET closely interfaced with manufacturing engineers to evaluated and finalize these designs.

The major technical features incorporated into the design of the HTR-10 include the following:

- Use of spherical fuel elements, which are formed with coated particles.
- A reactor core design which ensures that the maximum fuel element temperature of 1600°C cannot be exceeded in any accident.
- The reactor and the steam generator are housed in two separate steel pressure vessels. They are connected by the hot gas duct pressure vessel and arranged side by side.
- An active core cooling system is not required for residual heat removal in case of accident. Residual heat can be dissipated by means of passive heat transfer mechanism to the surrounding atmosphere
- The reactor core is entirely constructed by graphite materials, no metallic component are used in the region of the core.

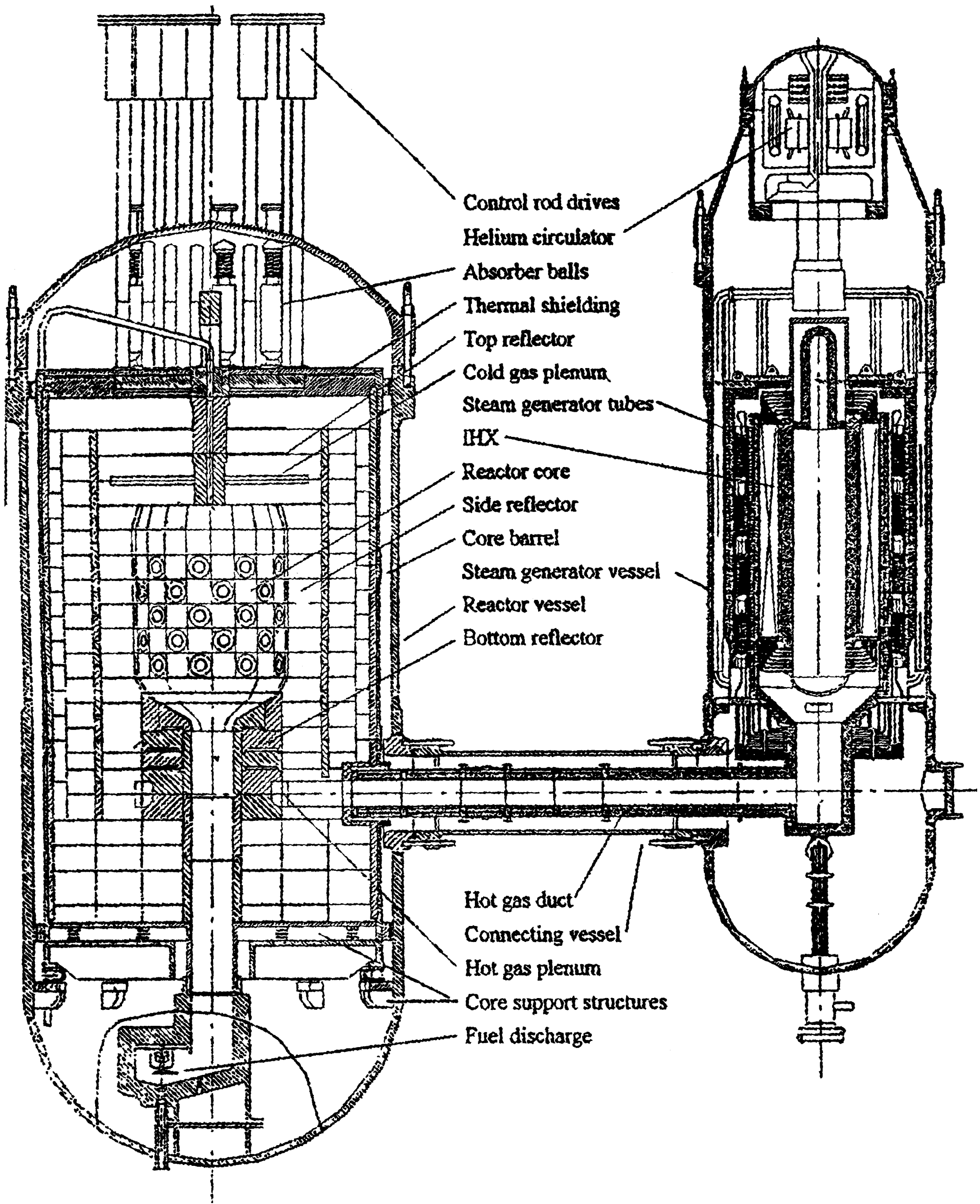

FIG. 5.8. Cross section of the HTR-10 primary circuit.

— The two reactor shut down systems, i.e. ten control rods and seven small absorber ball systems, are all positioned in the side reflector. They can drop into borings by means of gravity. In-core control rods are not needed.
— Spherical fuel elements go through the reactor core in a “multi-pass" pattern. Thus all fuel elements attain a relatively uniform burn up distribution in the core. Fuel pebbles are continuously discharged via a pneumatic pulse single-exit gate which is placed inside the reactor pressure vessel.
— A design incorporating an integrated steam generator and intermediate heat exchanger (IHX). The steam generator is a once through, modular small helical tube type. The helium circulator is installed in the steam generator pressure vessel and positioned above the steam generator.

— A ventilated primary cavity is designed as a confinement to restrict the radioactivity release into the environment, it does not serve the function of gas-tight and pressure retaining containment.
— A digital reactor protection system is utilized.
— Use of a domestic standard turbine-generator unit in the secondary circuit provides electrical power [5-5].

The HTR-10 is a pebble bed HTGR utilizing spherical fuel elements with ceramic coated fuel particles. The reactor core has a diameter of 1.8 m, a mean height of 1.97 m and the volume of 5.0 m^3, and is surrounded by graphite reflectors. The core is composed of 27 000 fuel elements. The fuel elements use low enriched uranium with a design mean burn-up of 80 000 MW•d/t. The pressure of the primary helium circuit is 3.0 MPa. The reactor core and steam generator are housed in two pressure vessels which are arranged in a “side by side” manner, as depicted in Fig. 5.8.

In the general reactor design, graphite serves as the primary core structural material, which is primarily in the top, bottom and side reflectors. The ceramic core structures are housed in a metallic core vessel that is supported on the steel pressure vessel. The thickness of the side reflector is 100 cm. Cold helium channels are designed within the side reflector for the primary coolant (helium) to flow upward after entering the RPV from the annular space between the connecting vessel and the hot gas duct. Helium flow reverses at the top of reactor core to enter the pebble bed, so that a downward flow pattern takes place. After being heated in the pebble bed, helium then enters into a hot gas chamber in the bottom reflector, and from there it flows through the hot gas duct and then on to the heat exchanging components. Spherical fuel elements (6 cm in diameter) with TRISO coated fuel particles make up the core. These elements go through the reactor core in a “multi-pass” pattern with a pulse pneumatic fuel handling system utilized for continually charging and discharging the fuel elements. The burn-up of the discharged fuel elements is measured individually and those elements which have not reached the limit are sent back pneumatically to the top of the reactor core.

Decay heat removal of the HTR-10 is designed on a completely passive basis. With a loss of pressure accident, where no core cooling is foreseen, decay power will dissipate through the core structures by means of heat conduction and radiation to the outside of the RPV. This heat will then be dissipated into a surface cooling system located on the wall of the concrete housing. This system works on the principle of natural circulation of water and discharges the decay heat via air coolers to the atmosphere. This surface cooling system is designed to protect the vessel and concrete structures more than the ceramic reactor core from being overheated by decay power.

There are two reactor shutdown systems, a control rod system and a small absorber ball system. These are located in the side reflector. Both systems are capable of bringing the reactor to cold shutdown conditions. Since the reactor has a strong negative temperature coefficient and decay heat removal does not require any circulation of the helium coolant, turning off the helium circulator can also shut down the reactor from power operating conditions [5-4].

The pressure vessel unit consists of the RPV, the steam pressure vessel and the hot gas duct pressure vessel. The upper part of the reactor pressure vessel is a cover which is connected via eighty bolts, and its lower part is a cylindrical shelf with a lower closure head.

A metallic 0-ring and an Ω-ring are used for sealing between the upper and lower parts. The tube nozzle for irradiation channels and the control rods driving system are mounted on the cover.

At the level of the hot gas duct there is a 900 mm diameter opening in the wall of the RPV. This opening is connected to the steam generator pressure vessel by the hot gas duct pressure vessel. The fuel element discharge tube penetrates through the lower closure head of the RPV. The reactor pressure vessel is supported in the reactor cavity by four brackets at the same level as the hot gas duct pressure vessel. The inner diameter is 4.2 m, and its height is 11.1 m. The net weight of the RPV is approximate 142 tons. The vessel material is steel SA516-70.

The steam generator pressure vessel contains the steam generator, the IHX and the helium blower. It is approximately 11.3 m in height, 2.5 m inner diameter and weighs 70 tons. The steam generator pressure vessel is supported in the steam generator cavity at the same level as that of the hot gas duct pressure vessel.

The hot gas duct pressure vessel and the hot gas duct form the primary gas passage between the RPV and the steam generator pressure vessel. Both the steam generator pressure vessel and the hot gas duct pressure vessel are made of SA516-70 steel.

The steam generator (S/G) is a once through, modular helical tube type. Hot helium from the hot gas duct flows through its center tube to the top part of the S/G and then is fed in above the S/G heating tubes. While flowing around the tubes, the helium releases its heat to the water/steam side, thereby cooling down from 700° to 250°C. The cool helium flow is then deflected to the inlet of the helium blower and returns to the reactor along the wall of the pressure vessel. The water flows through the helical tubes from the bottom to the top. The feed water temperature is 104°C and the steam temperature is 435°C. The S/G mainly consists of the pressure vessel, the steam generator tube bundle modules and the internals.

Because the arrangement of an integrated IHX and S/G is used, the S/G adopts modular structure. There are 30 helical tube bundle modules arranged in the outer space around the IHX. Each module consists of the heat transfer tube, central pipe, fixed support structure, outer case and leak preventer. The material of the heat transfer tubes is 2-1/4Cr1Mo. The tube diameter is changed in the different heat transfer areas to improve hydrodynamic stability of steam/water two phase flow. Also, throttle orifices are installed in the inlet of the feed water connection tubes. The tube bundle is formed by 4 series connected helical coils wound around a circular central pipe. There are 92 coils in each heat transfer tube for a total heat transfer area of ~53 m^2.

The primary circuit blower is a single stage centrifugal unit with an impeller at the end of the shaft. The drive motor is assembled on the blower shaft. The blower with its drive motor is installed on the top of the S/G pressure vessel and connected to the S/G through the connecting tube. There is an isolation valve in the connecting tube to avoid natural circulation during blower hot standby. Oil-lubricated ball bearings are utilized. The blower has the flow rate of 4.3 kg/s at a pressure of 3.0 MPa and temperature of 250°C and the pressure head is 0.06 MPa. The flow rate can be regulated down to 30% by means of changing the drive motor speed (a squirrel-case induction motor) through a frequency converter [5-5].

5.2.4. HTR-10 test phases

There are two operational phases for the HTR-10. In the first phase, the plant will be operated at a core outlet temperature of 700°C and inlet of 250°C. The secondary circuit will include a steam turbine cycle for electricity generation with the capability for district heating. The steam generator will produce steam at temperature of 440°C and pressure of 4.0 MPa to supply a standard turbine-generator unit. The process flow diagram for this phase and the main design data of the HTR-10 are provided in Fig.5.9 and Table 5.5, respectively.

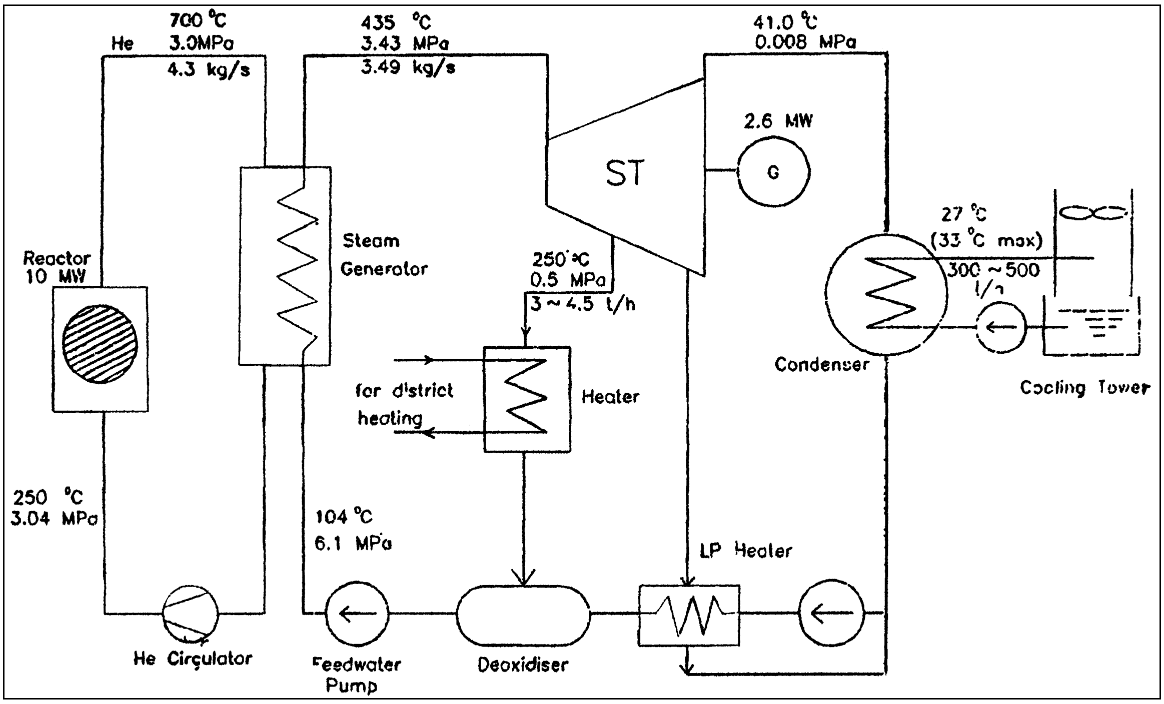

FIG. 5.9. Flow diagram for the first phase of HTR-10 operation.

TABLE 5.5. MAIN DESIGN DATA OF THE HTR-10

Reactor thermal power	10 MW
Primary helium pressure	3.0 MPa
Core outlet temperature	700°C
Core inlet temperature	250°C
Primary helium mass flow	4.3 kg/s
Outlet pressure of steam generator	4.0 MPa
Outlet temperature of steam generator	440°C
Secondary steam flow	3.47 kg/s
Power output (max.)	–2.6 MW(e)

In the second phase, the HTR-10 will be operated with a core outlet temperature of 900°C and an inlet of 300°C. A gas turbine (GT) and steam turbine (ST) combined cycle (Fig. 5.10) for electricity generation is in preliminary design. The intermediate heat exchanger (IHX), with a thermal power of 5 MW, provides high temperature nitrogen gas of 850°C for the GT cycle. The steam generator (SG), with the remaining thermal power of 5 MW, produces the steam at a temperature of 435°C for the ST cycle. The design data for this phase are shown in Table 5.6.

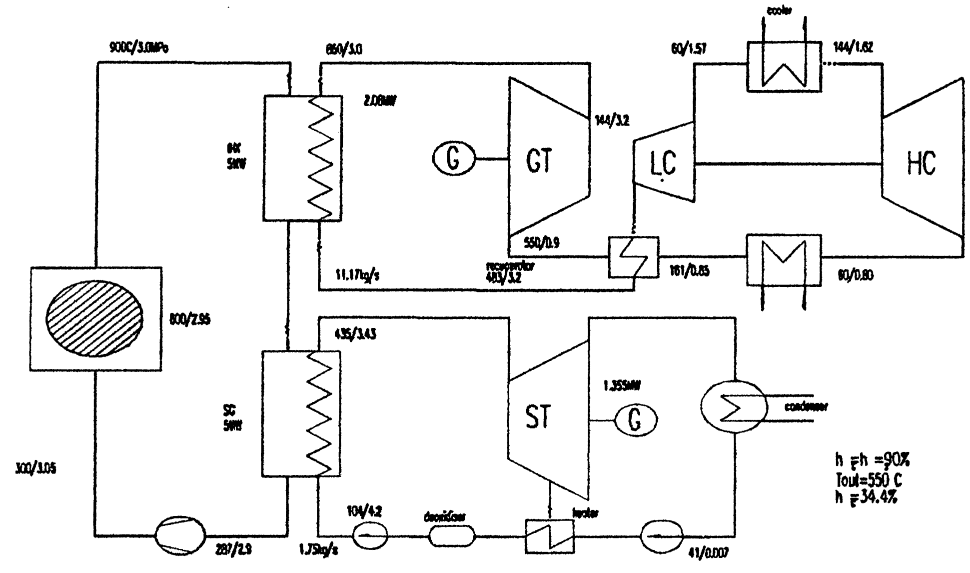

FIG. 5.10. Flow scheme of the HTR-10 gas turbine/steam turbine combined cycle.

5.2.4.1. HTR-10 testing programme

The primary goals to be achieved by the testing programme for the GT-ST combined cycle are as following:

(1) Demonstration of the gas turbine coupled with a nuclear reactor, particularly safety related testing of the GT-ST system for development of the commercial HTGR-GT plant.
(2) Thermodynamic simulation of the commercial HTGR-GT-ST plant gas turbine cycle including selection of optimum pressures and working fluids to meet the thermodynamic simulation conditions, such as, Mach number, Reynolds number, specific velocity, etc.).
(3) Verification of protection and control of HTGR-GT-ST.
(4) Optimization of the GT cycle.
(5) Verification of control coordination between the GT and ST cycles.
(6) Component development (seals, bearings, etc.) .

The HTR-10 performance test will be conducted to verify reactor and system performance at design temperature. The test items will include:

TABLE 5.6. DESIGN DATA FOR THE GT-ST COMBINED CYCLE CONFIGURATION

Reactor		
Core thermal power	MW	10
Core outlet temperature	°C	900
Core inlet temperature	°C	300
Primary pressure	Mpa	3.0
IHX		
Thermal power	MW	5
Primary helium inlet temperature	°C	900
Primary helium outlet temperature	°C	600
Primary pressure	MPa	3.0
Secondary nitrogen inlet temperature	°C	483
Second nitrogen outlet temperature	°C	850
Secondary pressure	MPa	3.2
Nitrogen flow rate	kg/s	11.17
SG		
Thermal power	MW	5
Temperature at helium side	°C	600/287
Temperature at water side	°C	435/104
Pressure at water side	Mpa	3.43/4.2
Power		
Power for gas turbine	MW(e)	2.08
Power for steam turbine	MW(e)	1.36
Total efficiency	%	34.4

(1) Graphite de-gasification

(2) Measurement of reactor physical parameters and distribution of neutron flux and gamma dose

(3) Test for instrumentation, control and power supply system including:
- Inspection of alarm and protection performance at the low power region
- Examination of detectors' sensitivity to shielding temperature
- Calibration of irradiation monitors
- Measurement of control rods reactivity worth at hot condition

(4) Primary circuit and fuel handling test including:
- Inspection/measurement of displacement, vibration and thermal expansion
- Leakage test
- Circulator test at hot condition and performance test for its cooling system
- Test of primary relief valves
- Inspection for vibration of the reactor internals and the active system
- Inspection of instruments for the monitor and protection systems
- Performance test at operating condition for the fuel handling system
- Test of isolation equipment (valves), including closure time

(5) Helium auxiliary system test
- Performance test of pressure control capability
- Test of operating conditions
- Validation of operational capability for the helium purification system

(6) Cavity cooling system test
- Performance test for the residual heat removal system
- Performance test for the shielding cooling system
- Performance test for the cavity HVAC system, including determination of minimum flow rate

(7) Verification of operational capability for the cooling system components

(8) Verification of the performance of the sampling system and the associated alarm system

(9) Verification of operational capability for the power conversion system, start up and shut down system.

The test conditions for the performance test will be 100 to 200 KWt power, with temperatures ranging from room to 250°C and system pressures ranging from .1 to 3.0 MPa, in a helium atmosphere.

The testing programme on the HTR-10 is divided into two power levels, 0 to 30% and 30 to 100% rated power. The 0 to 30% tests include:

— Determination of the HTR-10 response characteristics
— Performance test at operating temperature for the power conversion system, start up and shut down system
— Measurement of reactor physical and thermal hydraulic parameters
— Helium circulator test.

The 30 to 100% rated power testing programme includes:

— Determination of the plants' response characteristics for this power range
— Measurement of irradiation dose distribution
— Measurement of the main parameters for the entire plant [5-7].

5.2.5. HTR-10 safety design criteria

The HTR-10 reactor incorporates the design features of the modular HTGR and is characterized by the passive safety characteristics which dominate current small HTGRs. This safety design philosophy deviates significantly from the traditional LWR approach which relies on highly reliable redundant and diversified active components and systems as well as their power supplies. The amount of credit given to this new safety design approach is a substantial factor in achieving the plant economics required for the future competitiveness of nuclear power. The HTR-10 test reactor is expected to serve as a test facility in demonstrating this passive safety approach and to help in proving the capabilities and attributes of the HTGR to the regulatory bodies, utilities and the public.

In the current nuclear culture, the area of design criteria is generally the first issue encountered in beginning the design of a new type of nuclear reactor. However, China does not have an established set of nuclear standards, codes and guides specifically compiled for the HTGR to utilize in the licensing of the HTR-10 reactor. There are general standards for the design of nuclear power plants including specifying the limits on off-site doses and radioactive release rates as well as doses for the operating staff. These serve only as the top-level standards to follow. Before the basic design of the reactor could begin, the engineers had to develop the design criteria for the HTR-10 for review and approval by the licensing authority. Significant aspects of these criteria include the identification of systems important to safety and safety classification, accident identification and analysis, determination of the radioactive source term and containment design.

5.2.5.1. Systems important to safety

Systems important to safety are those that perform safety functions including reactor shutdown, decay heat removal and limitations on radioactivity release. For the HTR-10 reactor, these systems are primarily the following:

- Reactor protection system and its related instrumentation and associated power supplies
- Reactor shutdown systems (the control rod system and the small neutron absorber reserve shutdown system)
- Decay heat removal system
- Primary coolant pressure boundary and its pressure relief system.

Definition of the functions and configuration of the systems important to safety which take proper credit for the safety features of the modular reactor design are key issues in terms of maintaining nuclear safety and cost effectiveness. The safety classification and related Quality Assurance requirements and seismic categorization of components is an associated important issue. For the HTR-10 test reactor, safety classification of components departs significantly from the LWR. For example, the primary pressure boundary is defined to the first isolation valve. Steam generator tubes are classified as Safety Class 2 components. Diesel generators are not required to be as highly qualified as those used for the LWRs, since no systems or components with large power demands would require an immediate start of the diesel engines in a plant black-out accident.

5.2.5.2. Accident identification

The design basis accidents (DBA) are classified into several categories for the HTR-10. These categories include situations that result in:

- Increase of the heat removal capacity in the primary circuit
- Decrease of the heat removal capacity in the primary circuit
- Decrease of the primary flow rate

— Abnormality of core reactivity and power distribution
— Rupture of the primary pressure boundary tubes
— Anticipated transients without scram (ATWS).

The reactor has been designed against these accidents with conservative analysis. The results of these analyses has indicated excellent safe response of the reactor to these events. Within the framework of the DBAs, no accident will lead to the relevant release of fission products from the fuel elements.

The issue of severe accidents was also addressed for the HTR-10. A number of postulated accidents were selected to be analyzed including:

— Simultaneous withdrawal of all control rods at power operation and at reactor start-up
— Long term failure of the decay heat removal system
— Failure and shut down of the helium circulator
— Simultaneous rupture of all steam generator tubes
— Rupture of the connecting vessel and the hot gas duct.

These accidents were chosen for the HTR-10 based on input from licensing authorities and on the reference to practices in Germany and USA For the HTR-10 reactor, as long as the reactor protection system operates, most of the above accidents lead to no damage of the fuel elements. The rupture of the connecting vessel and the hot gas duct is seen as the most severe accident leading to some fuel damage, but non-permitted release of radioactivity into the environment is not anticipated. A required action, within a relatively long time scale of a few days, is the need to block the rupture area in order to prevent air from continuously entering into the reactor core.

5.2.5.3. Source term

A mechanistic approach is adopted for determining the radioactive source term. Severe core damage is not arbitrarily postulated in the siting evaluation, as is common for large LWRs. The release of radioactivity is calculated specifically for those individually demanding accidents that lead to the most release of radioactive nuclides from the fuel elements. The calculational results serve as the basis for off-site dose evaluation. This mechanistic approach is taken primarily based on the plants' safety features, and it is directly related to the quality of the fuel elements and to the knowledge of fission product release behavior during normal operation and accident conditions. The HTR-10 reactor will serve as a test bed for fuel elements and as a facility to study fission product behavior.

5.2.5.4. Containment design

The HTR-10 does not require a pressure containing and leak tight containment system. The concrete compartments which house the reactor and the steam generator as well as other parts of the primary pressure boundary are categorized as confinement, and together with the accident ventilation system, serve as the last barrier to the radioactivity release into the environment (Fig. 5.11). During normal operation, the confinement is ventilated and kept sub-atmospheric in pressure. If the integrity of the primary pressure boundary is lost, the primary helium coolant is allowed to be released into the environment without filtering because of its

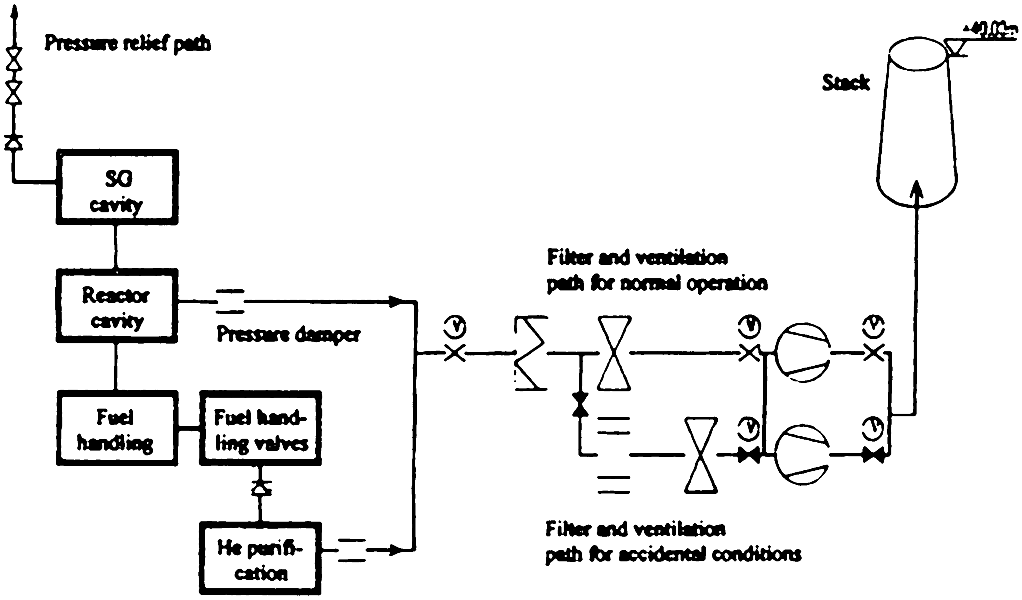

FIG. 5.11. Schematic diagram of the HTR-10 confinement design.

low radioactivity content. Afterwards, the confinement is again ventilated and the gases are filtered before they reach the environment [5-5].

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Chapter 6

CURRENT STATUS OF NATIONAL HTGR PROGRAMMES

Significant activities are occurring in the development of the HTGR particularly with regard to the utilization of this advanced source of nuclear power to achieve high efficiency in the generation of electricity and in high temperature process heat applications. Technological advances in component design and processes such as heat exchangers, hydrogen production techniques, turbo machinery, reformers and magnetic bearings coupled with the international capability to fabricate, test and procure these components and processes provides an excellent opportunity for achieving economic commercialization of the HTGR [6-1].

IAEA Member States involved with the HTGR programme are investing substantial resources in the development of this plant type because of the projected benefits to be realized upon its commercialization. Substantial interaction and sharing of resources and data between Member States is a key aspect of the HTGR development process.

6.1. CHINA

The HTGR programme in China centers primarily with the R&D, licensing, construction and subsequent operation of the HTR-10. A review of this plant is provided herein in Chapter 5. The center for HTGR R&D and the HTR-10 within China is Tsinghua University's INET in Beijing.

6.1.1. System/component development testing

An extensive programme of engineering experiments for HTR-10 key technology has been implemented at INET. The main goals of these engineering experiments are to verify the design characteristics and performance of the components and systems, to give feedback on the design and to obtain operational experience. Various experimental facilities have either been completed or are currently in development at INET and include:

— Test of the hot gas duct on the high temperature helium loop. The performance test of the hot gas duct was carried out in the helium test loop. The effective thermal conductivity of the insulation layer at the average temperatures of 200° to 420°C and helium pressures of 3.0, 2.5 and 1.5 MPa was measured. A hot gas duct test section was operated through the following sequence of conditions; 258 hours at the temperature of over 700°C and pressure of 3.0 MPa, for 98 hours at the temperature of 900°C and the pressure of 3.0 MPa, tested over 18 temperature cycles between 300° and 900°C, and 28 pressure cycles between atmospheric pressure and 3.0 MPa. The depressurization test of the hot gas duct, which is a simulation of the depressurization accident of the reactor primary coolant system, was also carried out at the helium test loop.
— Performance test of the pulse pneumatic fuel handling system. The initial test at room temperature was carried out to prove the design concept for fuel discharging. More than 150 000 balls were discharged by the pulse pneumatic discharging method. A full scale test facility of the fuel handling system was installed, and tested at helium temperatures of 150° to180°C and a pressure of 0.5 MPa. More than 35 000 balls were discharged from the simulated "core" without any breakage.
— Test of the control rods driving apparatus. The performance of the control rod driving apparatus was tested in a full scale test facility at operating temperature and with low

helium pressure . Normal control rod movement, maximum fall speed, indicator performance and life time tests have been carried out in order to verify performance.

— Test of the small absorber ball simulating system: The test of the small absorber ball system at conditions of room temperature and atmosphere has been carried out. The test facility at operating temperature and helium conditions is now being installed.

— Two phase flow stability test for the once through steam generator: The two phase flow stability test facility of the steam generator consists of the helium loop, electric heater, primary water loop and secondary water loop. The test section includes two full scale helical heating tubes. The main purposes of this test were to study the stability of the steam generator at HTR-10 operating conditions, to determine maximum throat diameter of the throttle apparatus, and to measure flow resistance on the water side and the average heat transfer coefficient on the helium side of the helical tubes [6-9].

6.1.2. HTGR plant designs

6.1.2.1. 200 MW•t MHTGR with an indirect gas turbine cycle (MHTGR-IGT)

This design features an indirect gas turbine system coupled to a 200 MW•t pebble bed HTGR. Although the HTGR can provide heat at 950°C with the attributes of outstanding safety and gas turbine cycle efficiency in the range of 47%, the possible radioactivity deposition on the turbine blades and thus the increase in maintenance difficulties suggests that the indirect gas turbine cycle should be applied initially in the development process to help solve these problems. The major issue that limits the higher thermal efficiency of the indirect cycle is the lower core inlet temperature. For the direct cycle, the cold gas leaving the precooler can be extracted to cool the RPV and the other steel structures. Therefore the core inlet temperature can be as high as 500° to 600°C. The indirect cycles previously proposed maintained the core inlet temperature as low as 310°C in order to cool the RPV.

In this design, the helium out of the IHX is extracted to a RPV cooling system. The gas flows through a small RPV recuperator and is cooled. It then is used to cool the RPV. The whole primary circuit is integrated in a pressure vessel (Fig. 6.1) with the core inlet/outlet temperatures 550°C/900°C, which can supply heat of ~850°C on the secondary side. The heat source would be used to drive a nitrogen gas turbine cycle with a busbar electricity generation efficiency of about 48%. As shown in this figure, a 200 MW•t pebble bed reactor core is located at the lower position of RPV. The core geometry is similar to that of the Siemens 200 MW HTR-Modul. The straight tube IHX is located at the upper position of the RPV and connected with the core through a gas duct. Similar to the AVR, the control rod system is installed at the bottom of the RPV and all rods are inserted upward into the side reflector. The main helium blower and an auxiliary blower for shutdown cooling are located at the top of the RPV. A reactor vessel cooling system recuperator and cooler are installed respectively in annular regions outside of the IHX and blowers.

The cold primary coolant helium from the main blower flows downward through vertical channels in the side reflector and enters into rector vessel cooling system a plenum at the bottom reflector. From the bottom plenum, the cold helium flows upward into the pebble bed core and is heated up from 550° to 900°C. The hot helium then converges into a plenum at the top reflector and flows upward through the IHX shell side and is cooled down by the secondary nitrogen. When it leaves the IHX, the 550°C cold helium flows into the main blower. Brief technical data of the integrated MHTGR-IGT are given in Table 6.1. The GT

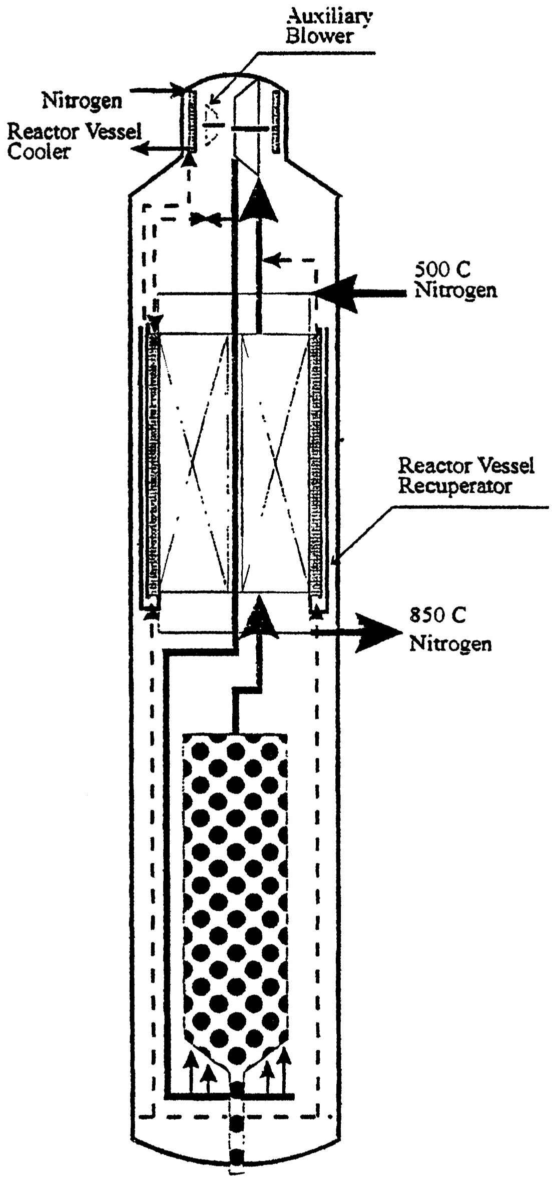


FIG. 6.1. Layout of the MHTGR-IGT.

cycle of MHTGR-IGT is similar to the design given by GA. In order to gain high plant efficiency, three stage compression and two stage intercooling are used. A busbar efficiency of about 48% is estimated [6-10] for this plant.

6.1.2.2. HTGR co-generation plant

There is a large potential in China to use the HTGR for co-generation and non-electric applications in the future. These applications can cover a wide range of process temperatures as described in the following chapter. Seawater desalination, heating, heavy oil recovery, process steam for petrochemical industry, high temperature heat for coal conversion and other processes are possible utilization areas [6-11]. China, as well as other parts of the world, is facing an increased shortage of potable water as well as electric power. Using the HTGR for seawater desalination to satisfy the potable water demand in these areas would be an excellent co-generation application.

TABLE. 6-1. TECHNICAL DATA OF THE MHTGR-IGT

Contents	*Parameters*
Reactor thermal power (MW)	200.
Core power density (MW•t/m^3	3.0
Core diameter (m)	3.0
Core average height (m)	9.43
Pressure of the primary loop (MPa)	6.0
Primary coolant	helium
Core inlet temperature (°C)	550
Core outlet temperature (°C)	900
Helium Flow Rate (Kg/s)	110
Working Fluid of GT cycle	Nitrogen (N_2)
Pressure in the secondary side of IHX (MPa)	6.0
Inlet temperature of N2 in IHX (°C)	500
Outlet temperature of N2 in IHX (°C)	850
N_2 Dow rate (Kg/s)	529
RPV height (m)	~30
RPV diameter (m)	6.0

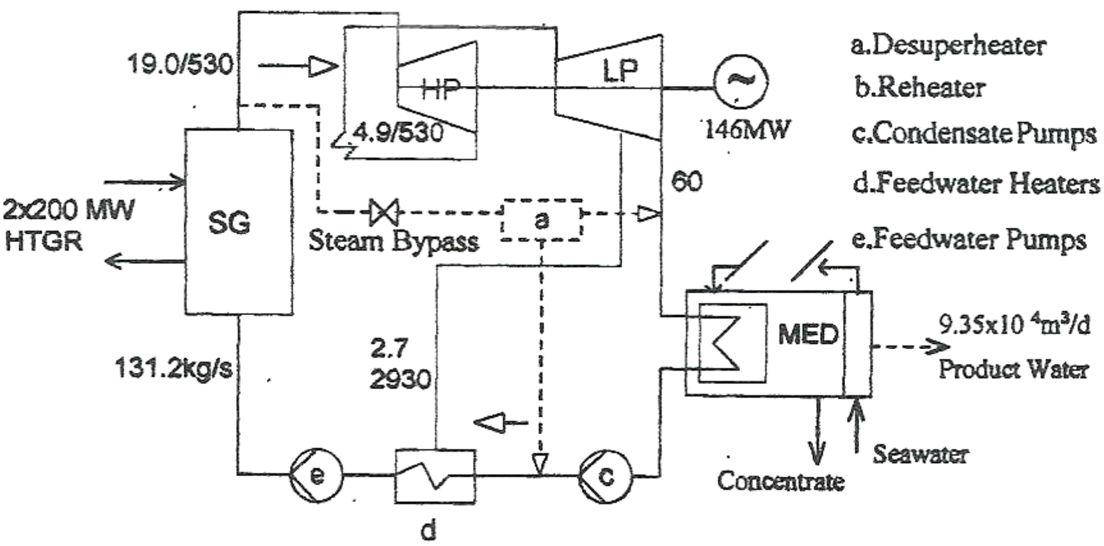


FIG. 6.2. ST-MED steam side flow diagram for the 2 × 200 MW•t HTGR plant.

Due to the safety attributes and the quality vapor produced with the HTGR, a plant consisting of 2 × 200 MW•t was chosen to evaluate technical and economic feasibility as an energy source for desalination in the dual production of clean water and power. The optimized design characteristics result in a final product of 93,500 m^3/d and 146 MW of electricity (Fig. 6.2) [6-12].

6.2. FRANCE

The major French organizations associated with the HTGR are the research organization, CEA, and the industrial corporation, Framatome. A summary review of the HTGR activities of these organizations is included in the following sections.

6.2.1. R&D Activities

Within CEA, the primary HTGR activities have centered on scientific investigation associated with the following three IAEA's CRPs and have included:

— Investigation on the PROTEUS experiments at the Paul Scherrer Institute, Switzerland, in relation to the CRP on "Validation of safety related physics calculations in low enriched gas-cooled reactors".
— For the CRP on heat transport and after heat removal for gas-cooled reactors under accident conditions, CEA provided benchmark problems associated with the code to experiment evaluation of the HTTR start up core physics testing. These calculations were made with a finite elements method 3D code (TRIO-EF) which is a general code. This code made it possible to have conduction, convection and radiation heat transfer coupled together. The results of the calculations were in good agreement with the HTTR experiment.
— In the CRP on "Validation of predictive methods for fuel and fission product behaviour in gas-cooled reactors", the CEA undertook an experimental study in the COMEDIE loop of the SILOE research reactor at the Grenoble centre on behalf of the US DOE [6-13]. This experiment was carried out to obtain integral test data to validate the methods and transport models used to predict fission product release from the core and plate-out in the primary coolant circuit of the MHTGR during normal operation and liftoff, and during rapid depressurization transients. The loop consists of an in-pile section with the fuel element, deposition section (heat exchanger), filters for collecting condensable fission products (FP) during depressurization tests and an out-of-pile section devoted to chemical composition control of the gas and on line analysis of gaseous FP. After steady state irradiation, the loop was subjected to a series of in-situ blow-downs at shear ratios (ratio of the wall shear stress during blowdown to that during steady state operation) ranging from 0.7 to 5.6. The results regarding the FP profiles on the plate-out section, before and after blowdowns were obtained with the general conclusions that; 1) the plate-out profiles depend on the FP chemistry, and 2) the depressurization phases have led to significant desorption of I-131, but they have almost no effect for the other FP such as Ag-110m, Cs-134, Cs-137 and Te-132 [6-14].

Besides contributions for review of specific documents, Framatome and CEA have initiated an analytical research programme to determine the best utilization of a GT-MHR for burning of fissile materials including plutonium and uranium 235, with or without thorium, in order to maximize fuel and core efficiency and minimize the stock of actinides and long life fission products. Coupled to these studies, the fuel management of the core is also being investigated starting with the traditional 18 month fuel cycle and extending to the feasibility of a 3 year fuel cycle.

Framatome is a partner in the development of the GT-MHR. Its contribution to this project lies in areas which are directly linked to its experience and work in pressurized water reactor design and construction. These include heavy component design, fabrication and installation, site erection and validation of seismic conditions and cost estimation. Two large vessels, one containing the reactor and one containing the PCS, and one cross duct, represent the main heavy components of the GT-MHR. The expertise in designing such vessels, adapted with respect to the material for specific temperature conditions and fabrication experience is being utilized for optimizing technological solutions. Similarly, the experience in building 65 PWR vessels is being applied to seismic evaluation, building arrangement and dimensioning criteria as well as the cost estimation of the GT-MHR [6-15].

France takes part, with partners from Germany, the Netherlands, United Kingdom and Italy, in the European Concerted Action (CA) on Innovative HTR to define cooperative developments in the HTR field for the European Union's Fifth Framework Programme.

6.3. GERMANY

6.3.1. Background

Germany's interest in gas cooled reactors developed as much for its potential applications to meet industrial process heat requirements, primarily coal gasification processes, as for high efficiency electricity generation. Initial work was carried out in 1956 on concepts utilizing spherical graphite fuel elements with embedded uranium fuel. The construction of the first HTGR began in Germany in 1961, with the 15 MW(e) AVR, began in 1961 at the KFA (now FZJ) National Laboratory in Juelich. The operation of this plant was subsequently started in 1967. In addition to continuous component testing and the collection of operating experience and results, the plant was also used during its operating period for the implementation of several experiments. In particular, these included the testing of various types of fuel elements, investigations of fission product behaviour in the circuit, experiments on the chemistry of coolant gas impurities and the performance of several tests related to HTGR safety. Most noteworthy in its operation was the sustained operation with core outlet temperatures as high as 950°C. During its last years, the operating mode of AVR was increasingly oriented to performing test programmes related to HTGR performance and safety. During 1988, operations were centered almost exclusively on tests related to the safety of HTGRs and, in particular, to the performance of simulated loss of coolant accidents. In completing 21 years of service in December 1988, the plant had accumulated more than 122 000 hours of operation with a 66.4% overall average availability and had generated 1.67 billion kWh of electricity. AVR was shut down on 31 December 1988 and is being decommissioned (see Chapter 8 for decommissioning details).

The construction of the second HTGR in Germany, the 300 MW(e) THTR-300 plant, at a site near Hamm-Uentrop, began in 1971. As a result of delays caused by continually increasing licensing requirements, construction was not completed until late in 1984. Full power during the commissioning phase was achieved in September 1986, and commercial operation began in June 1987. Some control and mechanical problems were experienced with the fueling equipment during the commissioning phase and continued to persist afterwards. The main problem centered on the reduced pebble discharge rate as the power level was increased, most noticeably above 40% power. The reduced discharge rate was caused by the increasing opposing force on the pebbles due to the increasing countercurrent flow of the cooling gas in the discharge pipe as power was increased. The interim solution of refueling the plant on weekends while running at 40% power and operating at higher powers during the working week, without refueling, was adopted. A permanent solution was devised and tested, and in October 1987, when the plant was shut down for a scheduled maintenance, the permanent solution was implemented. In autumn 1988, during a planned shutdown for scheduled maintenance and in-service inspection, several bolt heads holding down the cover plates on the insulation in the hot ducts were detected in the six hot gas ducts which take hot helium from below the core to the steam generators. A few graphite dowels used to align the graphite blocks near the hot gas duct entrances were also dislodged. A subsequent investigation showed that continued operation, without safety concerns, was possible.

The THTR-300 was operated under a risk sharing contract between the Federal Government, the State Government of North Rhine-Westphalia (NRW) and the Hochtemperatur-Kernkraftwerk GmbH (HKG), the utility owner group. The contract stipulated that, during the demonstration phase which terminated in 1991, the operating losses plus the decommissioning costs of the plant are to be shared on the basis of 60% by the Federal Government, 30% by the State Government and 10% by the HKG. A funding limit of DM 450 million had been established to cover these costs. In late 1988, the HKG submitted a request for the early decommissioning of the THTR on the basis that the funding limit was insufficient to cover the estimated increase in decommissioning costs, the additional costs due to increased licensing requirements and potential downtime losses due to the possible interruption of the fuel supply for the plant. The request was submitted as a precautionary measure and was required by the contract when the HKG foresaw the possible early attainment of the limit due to the cost increases and not as a result of any dissatisfaction with the operation of the THTR. The decision was subsequently made to permanently shut the THTR-300 plant.

Other development programmes on HTGR power plants in the 1980s included the HTR-500 (550 MW(e)) concept, developed by the ASEA Brown Boveri/ Hochtemperatur-Reaktorbau (ABB/HRB) organization, as the follow-on to the THTR. The HTR-500 was evaluated by a German utility group in 1983–1984, with the positive conclusions that the concept was technically feasible, licensable and competitive in power costs with the much larger LWRs in Germany. The feasibility of the technology and of the safety concepts incorporated in the HTR-500 were also confirmed by an expert committee set up by the German Ministry of the Interior. In July 1988, a consortium of German and Swiss utilities placed the order for an HTR-500 safety analysis report. The objective of this order was to prepare the groundwork for a safety assessment and a licensing procedure. This activity was regarded as the first step in eventually obtaining the complete construction license and preparing a detailed tender for the building of an HTR-500 plant. A project related R&D programme sponsored from public funds was conducted in parallel with the design of the first HTR-500 power plant. In addition to support from German industrial companies, a substantial contribution to the R&D activities was furnished by KFA as well as by industrial and research partners of the Swiss Interessengemeinschaft IGNT.

The 80 MW(e) HTR module concept, developed by Siemens/Interatom, was the first small, modular type HTGR concept to be proposed. Although initially developed in the early 1980s for industrial process heat applications, the passive safety features of the side by side concept, coupled with the other attractive characteristics of the modular concept, soon led to the proposed electricity generation application [6-16]. The design and licensing of this plant has formed the basis for the PBMR currently under development by ESKOM in South Africa (see Chapter 3).

6.3.2. Current status of the German HTGR programme

The industrial activities towards the market introduction of HTGRs as well as the related R&D programmes in Germany were stopped in the early 1990s in conjunction with the decision not to continue the commissioning of the THTR-300 demonstration reactor. The 15 MW(e) test reactor AVR at KFA Juelich had already been shut down in expectation of further successful operation of the THTR-300.

HTGR-related activities in Germany are now mainly oriented to the decommissioning or safe closure of both plants and to the disposal of their irradiated fuel elements. Complementary to the commercial projects on evolutionary LWRs, additional generic R&D is being performed in Germany with the aim of identifying the fundamentals of an approach towards 'catastrophe-free' innovative nuclear reactor technologies as being requested by the German Atomic Act of 1994, even for beyond design basis accidents. The proven safety characteristics and principles of HTGRs form a sound start-up basis for these considerations aiming to self-acting (passive) nuclear, thermal, chemical and mechanical stability of the primary circuit and an independence of the fission product barriers.

Basic technologies and safety principles originating from HTGR related R&D are also partially relevant for the prevention and mitigation of severe accidents of other reactor types and vice versa as those scenarios are also characterized by high temperatures and highly corrosive conditions offering new application potentials e.g. for high-tech ceramic materials. The technology of coated particles might offer significant advantages for safety and operational aspects as well as for an effective burning of plutonium and for direct disposal even beyond the HTGR frame. The use of thorium is also being re-discussed as an effective tool to burn plutonium and to reduce the generation of actinides.

HTGR related technologies continue to be vital within this broader perspective and fertilize the orientation and perspectives of future nuclear reactor technologies. Within the European Union there is a growing interest to assess the benefits and status of recent HTGR techniques and the innovative HTGR concepts being under discussion in different parts of the world [6-17]. Germany takes part, with partners from France, the Netherlands, United Kingdom and Italy, in the European Concerted Action (CA) on Innovative HTR to define cooperative developments in the HTR field for the European Union’s Fifth Framework Programme.

6.3.3. HTGR research and development activities

The ongoing German R&D activities on the HTGR are governed by the above mentioned aspects as well as by the technologically innovation potential of HTGRs and by the associated spin-off for other nuclear and non-nuclear applications. The HTGR safety philosophy, based on passive safety characteristics is beneficially influential to new reactor designs giving more attention to self-acting safety principals. Severe accidents in other reactor types are often accompanied by high temperatures and highly corrosive atmospheres. It is evident that the specific HTGR knowledge on the high temperature behaviour of materials and systems as well as the use of ceramics is beneficial for the prevention and mitigation of those accident scenarios [6-18]. The application of coated particle fuel may be of interest also for other reactor types especially concerning re-feeding/burning of actinides without intermediate reprocessing and for the direct disposal of the spent fuel.

The qualification of HTGR technology elements for different conditions or applications may produce significant innovations to be applied to the HTGR. The development of corrosion resistant ceramic structures under water/steam conditions or the further consideration of prestressed cast iron pressure vessels for innovative LWRs are possible illustrative examples for improved mitigation of water ingress accidents in HTGRs. This more generic R&D approach provides a broader basis for innovations of nuclear technology and a cross-fertilization between the different nuclear reactor lines or concepts instead of continuing the previous competition and dedicated R&D approach [6-17].

6.3.3.1. Safety related R&D on core physics and plant dynamics

R&D on the principles of nuclear reactors with self acting main safety functions has continued for different fuel types, moderator materials/ratios and coolants. The results show that HTGR comparable safety characteristics can also be achieved with the combination of HTGR type fuel and heavy water as the moderator and coolant. Investigation of these heavy water or helium cooled reactor systems also illustrate the potential for effective plutonium or actinide burning far beyond the existing MOX technology in LWRs. The disposal characteristics of the coated particle fuel would enable direct disposal and offers improved protection against misuse of fissile residua comparable or even better than vitrification. Extremely high burn-up eliminates the need for intermediate reprocessing of such a fuel and thus offers a great motivation to consider this technique beyond dedicated HTGR applications.

The use of thorium is being re-analyzed with respect to the minimization of plutonium and actinide generation. International benchmark calculations show good agreement in the reduction rates for plutonium, reactivity coefficients, residual inventories and radiotoxicities being relevant for the final disposal. The importance of thorium in this context has also been underlined by the proposals of Prof. Rubbia for accelerator driven systems burning plutonium and actinides.

The improvement and validation of analytical tools is also continuing including other innovative core configurations. This allows for reliable code calculations of different reactor types covering extreme assumptions on accident scenarios. The codes are also used for safety analyses of new HTGR concepts within international collaborations under the IAEA umbrella.

The analyses show that self acting stability characteristics of nuclear reactors are feasible within given limits and configurations, that the generation of long lived radiotoxic wastes can be significantly reduced and that weapon grade fissile materials can be effectively destroyed even without intermediate reprocessing. This provides sufficient innovation potential to respond to the open issues and problems that nuclear power is facing today in many countries of the world [6-17].

6.3.3.2. R&D on improved process techniques and safety systems

The above mentioned analyses have to be based on verification experiments and be accompanied by system and component improvements that incorporate on-going technical progress. This is performed in complementary collaboration with different national and international partners to take benefit from other know how, data and new ideas as well as to enhance general safety standards. The contribution of nuclear energy to reduce the environmental impact of fossil fuel can also be enhanced by opening new market sectors like heating, process heat and combined generation of heat and power beyond dedicated electricity production and by improving the economics against low-priced, but polluting fossil energy carriers.

Self-acting decay heat removal characteristics have been demonstrated in the SANA facility at FZJ and offered for international benchmark calculations. The effective heat transport is of special importance for the prevention or mitigation of core damage and must be investigated for different fuel geometries and material combinations. Additional experiments showed that core melt sequences could even be stopped by inserting materials with high heat conductivity into fuel rod lattices as an emergency measure. Follow-up problems like steam explosions, hydrogen production, fission product releases etc. would be completely avoided in the case that still has to be analyzed in more detail.

The experiments on natural convection and corrosion (NACOC) under hypothetical air ingress accident scenarios have been performed up to 800°C with special air outlet configurations. The results are consistent with the fluid dynamic calculations and offer a good basis for the prediction and assessment of comparable safety related phenomena even in other reactor types.

The transport of water droplets in gas circuits/ducts has been studied in the SEAT facility. It could be shown that an effective separation can be realized by sedimentation, deflections and different separation devices. The experiments delivered essential data for the calculations that are now able to describe the influence of the main parameters. These activities also have generic importance for both non-nuclear and nuclear applications. The further development of magnetic bearings and tests on relevant configurations for future HTGR concepts have been in progress since 1993 at the Technical High school in Zittau in the Institute for Process Automation (IPM). Innovative solutions for the catcher bearings will be investigated experimentally with a 1.5 t vertical rotor test facility being capable of running to 7200 rpm. Bearings with permanent magnets are being further developed at FZJ which also offers interesting options for diverse improvements. The recent progress in this field was reported in the 3rd Workshop on Magnetic Bearings [6-19] and supports the use of this technology including for nuclear applications.

The technology of prestressed cast iron vessels (PCIV) has been considered for innovative BWR designs as the temperatures, pressures and dimensions are nearly the same as for modular HTGRs. The Siempelkarnp foundry has provided new ideas to simplify the structure and to lower the cost of a ‘burst-free’ RPV. This also opens the way to overcome the restrictions of limited dimensions and transport capabilities. RPV decommissioning could be considerably simplified due to the segmented configuration.

Improvements in the chemical stability of ceramic fuel elements can be achieved by SiC coatings. First results of coated pebble fuel irradiated in Petten is available now and show that the coatings can withstand the required doses without damage. SiC encapsulations also have been manufactured in the form of pellets for LWR applications and have been successfully tested under mechanical, thermal and corrosive loads. Such fuel could increase the grace periods during core melt initiating accident sequences and would have significant advantages for the final disposal of spent fuel due to their resistance against corrosive attack.

Safety considerations have also been performed for transmutation processes. The influence of long-lived actinides on the safety of final disposals is higher than formerly expected when new assessment rules on radiotoxicities are assumed. These results foster enhanced R&D activities for effective burning of actinides, increased efforts for the minimization of actinide generation and a new look on the use of ceramics and coatings for fuel and disposal techniques.

FZJ continues to perform R&D on the generic approach for safety related innovative nuclear reactor technology according to the requirements of the German Atomic Law. Additional generic R&D activities being investigated include:

— Reliability of passive/self acting safety systems
— Reliability of human interactions
— Transmutation/reduction of long lived radiotoxic isotopes
— Safety of waste disposal
— System analyses on the innovation potential of nuclear technologies.

The performance of these activities within the frame of broader international collaboration and a synergistic cross-fertilization between different nuclear reactor lines allows for significant and sustainable innovations, for reconciling nuclear energy with public opinion and for responding to the evolution of legal and industrial requirements [6-17].

6.3.4. Additional support activities and test facilities

FZJ is prominent in the support of international development of safety of nuclear power. This is particularly evident with its participation in the European nuclear programmes of the MICHELANGELO initiative and the European Union funded Concerted Action on Safety Related Nuclear Reactor Technology Elements-R&D (SINTER) network which aim for the identification of priority items for sustainable innovations of nuclear technologies and work shared European collaboration structures. FZJ takes part, with partners from France, the Netherlands, the United Kingdom and Italy, in the European Concerted Action (CA) on Innovative HTR to define cooperative developments in the HTR field for the European Union’s Fifth Framework Programme. Areas under consideration in this framework include physics and the fuel cycle, system analysis, materials (including irradiation and PIE), components and economics.

A similar approach is under investigation for future HTGR related R&D under the umbrella of the IAEA as initially proposed by the IWGGCR in 1996 and subsequently termed the “Global HTR Network”(GHTRN). Development activities continue on the GHTRN including funding from the (Japanese) organization, RAHP, to develop the GHTRN information web site in a manner similar to the SINTER web site configuration. The IAEA is also considering the establishment of a new CRP on Conservation and Application of HTGR Technology in support of the needs expressed by the IWGGCR.

Significant HTGR support facilities existed within Germany as of the early 1990s. Many of these have since been deactivated, but some may be brought into active use for the HTGR should the specific needs require. The facilities include:

(1) Self-Acting Removal of After (decay) Heat (SANA) at the Research Center Juelich; This facility serves to investigate the thermic properties of a HTR pebble bed and other fuel arrangements of different type like block or rod geometry for advanced, innovative reactor concepts. The special objective is the investigation of the self acting removal of decay heat from simulated core arrangements. The heat transport can be examined in a wide range of different conditions. The effective heat conductivity of the arrangements can be measured.

(2) Experiment Natural Convection in the Core with Corrosion (NACOK) at the Research Center Juelich is for simulation of an air ingress accident into the hot core of a HTR with side by side arrangement of core and heat exchanger like the HTR-MODULE after a complete rupture of the coaxial hot gas duct and depressurization. In this large scale experiment, the course of the accident and related phenomena are investigated starting from the rupture up to a continuous penetration of air through the core caused by the natural draught. The onset of the natural convection, the resulting mass flow of air as well as the caused corrosion dependent from location and temperature profile are examined. The results are used to validate existing computer models used for predictions related to course and consequences of a hypothetical accident with a massive air ingress.

(3) Self Acting Removal of Water Droplets (SEAT) at the Research Center Juelich; Depending on the design of a HTGR with a water/ steam secondary circuit, the ingress of water droplets or steam into the reactor core can cause an increase of reactivity and the corrosion of the graphite. Assurance must exist that in all cases of an ingress accident no uncontrollable power excursion of the reactor can occur, nor will intolerable damage of the core structures take place as the result of corrosion of the graphite. Sources for an ingress could be the leakage or rupture of water filled tubes in the region of the steam generator. The objective of the experiment SEAT is to show that in the case of water ingress the mass of penetrating water into the core can be limited to non-critical values by suitable system design and to examine the processes which govern a self acting removal of water droplets from the primary coolant gas.

(4) Experiment Corrosion Test Apparatus (KORA) at the Research Center Juelich is used to examine fuel specimens under oxidizing atmosphere, i.e. both in helium with defined moisture content and gas mixtures of both helium and air. The influence of moisture on the release of fission gas and thus of iodine from defective fuel particles is measured in the temperature range from 600° to 1200°C. The tests with air are directed to examine the stability of SiC coatings and the corrosion of graphite up to temperatures of 1600°C.

(5) The Test Facility for Magnetic Bearings at the Tech High School of Zilidu is utilized for the development and testing of catcher bearing and thrust bearing technology for aggregates with magnetic bearings in conventional and nuclear applications. Also for the development and testing of controlling systems for magnetic bearings and for development and testing of concepts for error diagnostics of mechanical, electrical and electronic components.

(6) Test Container for Prestressed Cast Iron Pressure Vessels (PCIV) at Stempelkamp GmbH is used for demonstration of the feasibility and successful operation of the PCIV concept for pressure vessels of HTRs. Testing and examination includes the behaviour of design, tightness, tension system and liner of the total system under operational conditions and relevant overload conditions [6-8].

Other substantial testing facilities were established in support of the Germany HTGR development programme throughout the previous three decades. Significant to the development of the gas turbine was the programme to test the performance of high temperature helium turbomachinery. This programme involved two experimental facilities. The first was an experimental co-generation power plant constructed and operated by the municipal utility, Energieversorgung Oberhausen. This facility consisted of a fossil fired heater, helium turbines, compressors and related equipment. The second facility was the High Temperature Helium Test plant for developing helium turbomachinery and components at FZJ-Juelich. The heat source for this plant was derived from an electric motor driven helium compressor. The experiences gained from these facilities were both positive and negative. However, the research and development programmes can be judged successful and supportive of the feasibility to use high temperature helium as a Brayton Cycle working fluid for direct power conversion from an HTGR [6-28].

6.4. INDONESIA

The National Atomic Energy Agency of Indonesia (Batan) established the HTGR programme in 1993 to conduct studies of HTR technology and its application. The team was initially divided into two groups including reactor technology and safety and applications. This programme was subsequently expanded in 1997 into five general areas of HTGR development including reactor technology and optimization of electricity and steam co-generation, safety and environmental, coal liquefaction and desalination, instrumentation and control and the HTR fuel cycle.

6.4.1. R&D activities

The energy diversification policy of the Indonesian government has the objective of reducing domestic oil consumption and promoting other energy resources. Batan initiated the HTGR programme to investigate the utilization of the HTR because of its capabilities to produce heat for electricity generation and a wide variety of industrial processes. Of specific interest is the investigation of coal liquefaction, desalination, enhanced heavy oil recovery, hydrogen production and $C0_2$ conversion.

The growth of Indonesian industry and the standard of living throughout the past decade has been significant. Although there are large oil resources, this growth will result in Indonesia going from an oil exporter, with the attendant financial influx into the country, to an oil importer, with a loss of revenue. Therefore, the HTR team study is concentrated in the coal liquefaction programme. That programme is based on the reality that Sumatera Island has 55% of Indonesia's coal and also has oil resources, which are distributed in several regions. The team proposed the HTR for heat utilization, not only for the electricity generation but also for coal liquefaction and enhanced heavy oil recovery.

Research of coal conversion to synthetic crude oil is to be conducted in cooperation with the Mineral Technology Research and Development Centre of the Indonesia Mines and Energy Ministry. A laboratory scale liquefaction plant has been in operation, and several experiments have been completed. The extraction of the crude oil, which is produced by the coal liquefaction process, is to be performed by the government oil company (Pertamina).

The vast Natuna gas field has also been a recent area of evaluation for the application of HTGRs. The Natuna gas field represents one of the largest remaining gas resources in the world, but has the drawback that the gas contains ~70% CO_2. A prominent high temperature application of the HTGR is for the production of hydrogen from methane with subsequent synthesis to methanol through the CO_2 conversion process (see Chapters 5 and 7). This use of the HTGR is anticipated to lessen the environmental (and cost) issues associated with the present need to re-inject the CO_2 gas into the earth upon extraction of the methane [6-21].

Major R&D activities associated with HTGR technology and safety have been in the areas of nuclear design codes for fuel management and safety analysis. The Batan-MPASS code, an in-core fuel management code, has been developed and verified to simulate the continuous flow of the fuel elements in a pebble bed type HTGR core for both once-through-then-out (OTTO) as well as multi-pass fueling schemes. One important feature of the code is that it can search directly for equilibrium core conditions without simulating the transient cores. A similar code, Batan-PEU, has also been built to simulate the peu-a-peu fueling scheme. These codes are equipped with in-core thermal hydraulic modules for estimating the pebble fuel temperature.

For the prismatic/block type HTGR, the Batan-FUEL code has been compiled where originally the code was developed for in core fuel management of a research reactor with an ordinary batch refueling scheme. The thermal hydraulic modules for Batan-FUEL are planned for development in the future. The diffusion calculation module within the Batan-FUEL code has been successfully used for the benchmark problems of the Japanese HTTR start up core physics experiments. These codes are based on 2-D, 3-D few group diffusion theory and the required cross section libraries were compiled using VSOP and the DELIGHT-7 cell calculation codes, developed by KFA and JAERI, respectively. The accomplishment of these codes are expected to contribute to the assessment of the technical and economic aspects of small and medium scale modular HTR designs presently being conducted by Batan, especially for providing fuel cost estimation.

For safety and accident analysis, two codes have been developed to simulate the depressurization and reactivity accidents in a pebble bed type HTGR. For the depressurization accident, the decay heat generated in the core is calculated based on the core composition prior to the accident using the JNDC fission product and yield data compiled by JAERI, taking into account also the contribution of decay heat of heavy metals. For the reactivity accidents, the neutron dynamics is simulated in the code with the improved quasi-static model under 2-D, RZ geometry, neutron diffusion theory. The code application is utilized for to assess of the modular HTR-M 200 MW•t plant safety under these accident conditions [6-22].

Facilities for the HTGR include the research reactor in Serpong and the facilities and hot cells at Yogyakarta for fuel and metallic materials irradiation and post irradiation testing, and various test stands for evaluation of reactor components and thermal hydraulic performance at Batan Research Centre [6-8].

6.5. JAPAN

Japan has an extensive HTGR research and development programme including investigation into the eventual commercialization of the gas turbine plant for electricity production, co-generation for electricity and industrial heat processes and for high temperature heat utilization applications. Many of these HTGR activities involve both JAERI and Japanese private industry working together. This section provides a review of some of these activities. Information associated with the HTTR project and the heat utilization development programme are provided in Chapters 5 and 7, respectively.

6.5.1. Gas turbine plant designs

Plant designs featuring the HTGR coupled to a closed cycle gas turbine power conversion system (HTGR-GT) are under evaluation within Japan. A feasibility study of the HTGR-GT system was initiated in 1997 by JAERI as an assigned activity by the Science and Technology Agency. The goal of the study is to obtain a promising concept of the HTGR-GT system which has a high thermal efficiency and is economically competitive. Designs of three candidate plants were developed and their power generation costs are presently being evaluated. In parallel with these design efforts, experimental work such as fabrication of the plate-fin type heat exchanger core, fabrication of carbon/carbon (C/C) composite discs and fission product (FP) filter tests are being carried out. A significant part of the study is anticipated for completion in FY 2000, with supplemental design improvement efforts completed in FY 2001 [6-2].

The initial design, beginning in FY 1997, was of a 600 MW•t direct cycle plant which was carried out by JAERI in cooperation with Fuji Electric Co. Ltd, Hitachi Ltd, Mitsubishi Heavy Industries Ltd and Toshiba Corporation. Progress has also proceeded with the design of a direct cycle plant with the smaller capacity of 300 MW•t. Conceptual design work on HTGR-GTs of 600 MW•t and 400 MW•t was started in FY1999 and is continuing to the present. In parallel with these designs, R&D on related components has progressed including trial fabrication of the core section of a plate-fin recuperator and measurement of its heat transfer coefficient, fabrication of a C/C composite disc and preparation of a high temperature filter test facility. FY 2000 and 2001 will include cost evaluation of 600 MW•t, 300 MW•t and 400 MW•t plants and work associated with improving these design concepts. A technical summary (with emphasis on the 600 MW•t due to the level of design) of the Japanese HTGR-GT developmental work is described in the proceeding sub-sections. The GTHTR 300 is also described in some detail as JAERI is seeking to optimize the closed cycle gas turbine plant to reduce technical and cost requirements for near term development of the technology

6.5.1.1. 600 MW•t HTGR-GT plant [6-2]

This design utilizes a block type fuel core to achieve a higher power level than could be generated with a pebble bed core. In this case, the maximum achievable power is expected to be 600 MW•t. According to the preliminary design considerations, the following basic design parameters were selected:

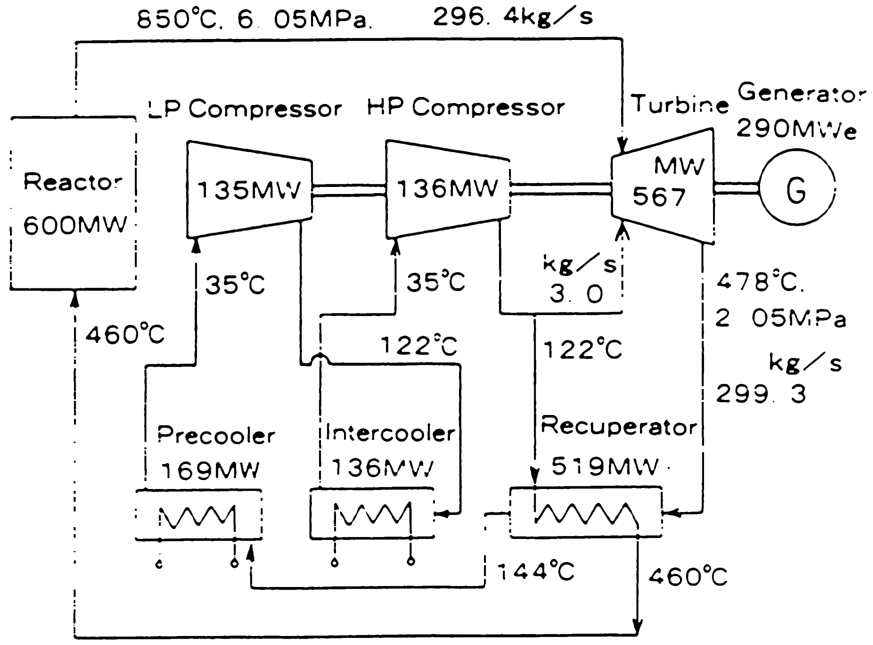
— Reactor outlet gas temperature of 850°C, which is considered to be optimum from the trade off between the lower fuel temperature and higher cycle thermal efficiency,
— Reactor inlet gas temperature of 460°C. This relatively moderate value is required to ease the RPV structural design by establishing a sufficient material strength of 9Cr-IMo-V steel.
— Helium gas pressure of 6 MPa, which is the highest value within the limit of RPV weight of 1000 ton, and makes the compact design of components possible.

Based on these main parameters, the reactor and turbine system component design were carried out. The flow scheme and the mass and heat balance obtained are shown in Fig. 6.3. The heat and mass balance was calculated with the following conditions:

— Turbine and compressor adiabatic efficiencies of 93.34%, 89.33% (for the low pressure compressor), and 89.21% (for the high pressure compressor), respectively,
— High pressure side recuperator effectiveness of 95%,

— Pressure loss ratios of .902% for the reactor, .339% for the intercooler, 1.16% for the precooler, and .587% for the high pressure side and 1.429% for the low pressure side of the recuperator, respectively,

— Generator efficiency of 98%.



$\eta_{G\ net} = 46.7\%$

FIG. 6.3. Mass and heat balance of the 600 MW•t HTGR-GT plant.

Although the actual recuperator effectiveness was not known, the value of 95% for the high pressure side was assumed as it was judged to be achievable from the results of related research. The cycle thermal efficiency then becomes 49.3%. With a generator efficiency of 98%, heat loss of 5 MW•t from RPV surface and the house load power needed of 7.5 MW(e), the net power generating efficiency becomes 46.7%.

Aerodynamic designs were carried out on the turbine for three cases of differing stage numbers (5, 8 and 11), with selection of the eleven stage turbine considering a trade off between rotor length and performance. In this case, the design limit comes from the stress in the first rotor blade as shown in Fig. 6.4 and the adiabatic efficiency becomes 93.34%. In the low pressure and high pressure compressors (LPC and HPC), the efficiency was calculated by changing the number of stages one by one. The maximum efficiency was reached in 19 and 21 stages for LPC and HPC, respectively. However, the rotor length becomes quite long. The final number of stages selected were 16 (LPC) and 17 (HPC), which results in adiabatic efficiencies of 89.33% and 89.21% for the LPC and HPC, respectively. The arrangement of compressor rotor is shown in Figure 6.5.

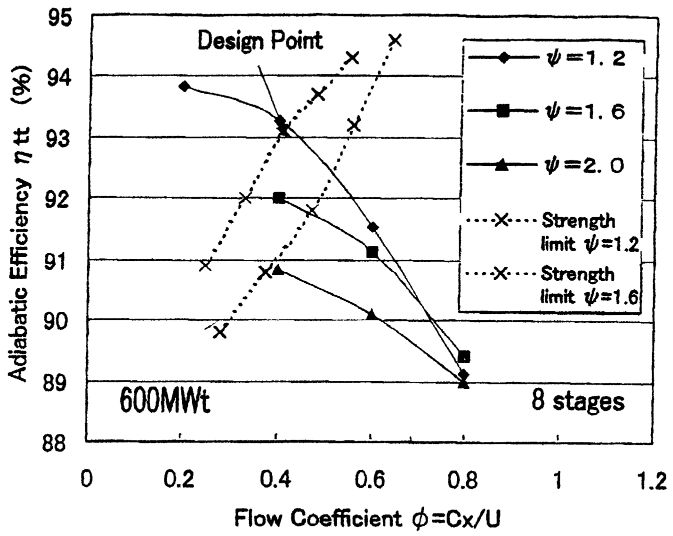


FIG. 6.4. Turbine design point for the 600 MW•t plant.

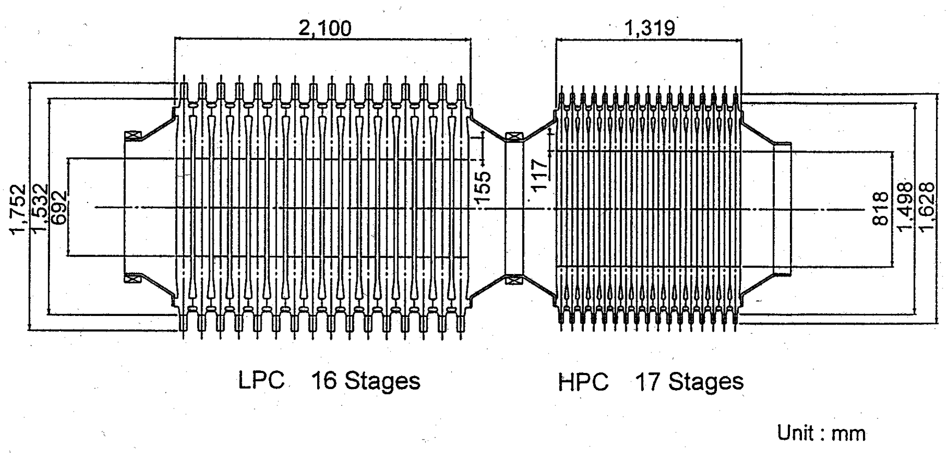


FIG. 6.5. Compressor rotor for the 600 MW•t plant.

The overall rotor is divided to three members, that is, LPC rotor, HPC plus turbine rotor and generator rotor. These rotors are connected by means of diaphragm couplings so as to ease the rotor dynamics design as shown in Fig. 6.6.

The cooling media for the generator is helium, which is different from the usual cooling (hydrogen). However, the design is possible based on the state of the art technology of hydrogen cooled generators. The main features are a submerged arrangement to prevent helium gas leakage to the outside and a means of supplying rotor current to use the generator as a motor for plant startup. A maximum achievable rotor diameter of 1.2 m is employed to reduce the rotor length. Even in this design, the total length becomes 13.0 m, which is equivalent to that of the entire gas turbine rotor.

The gas turbine components are placed in a single horizontal pressure vessel. By utilizing the horizontal arrangement, the support structures for the bearings becomes sufficiently stiff and stable operation of the rotor is realized. The thrust load is nearly cancelled by the balance pistons. According to the critical speed and vibration analyses, the

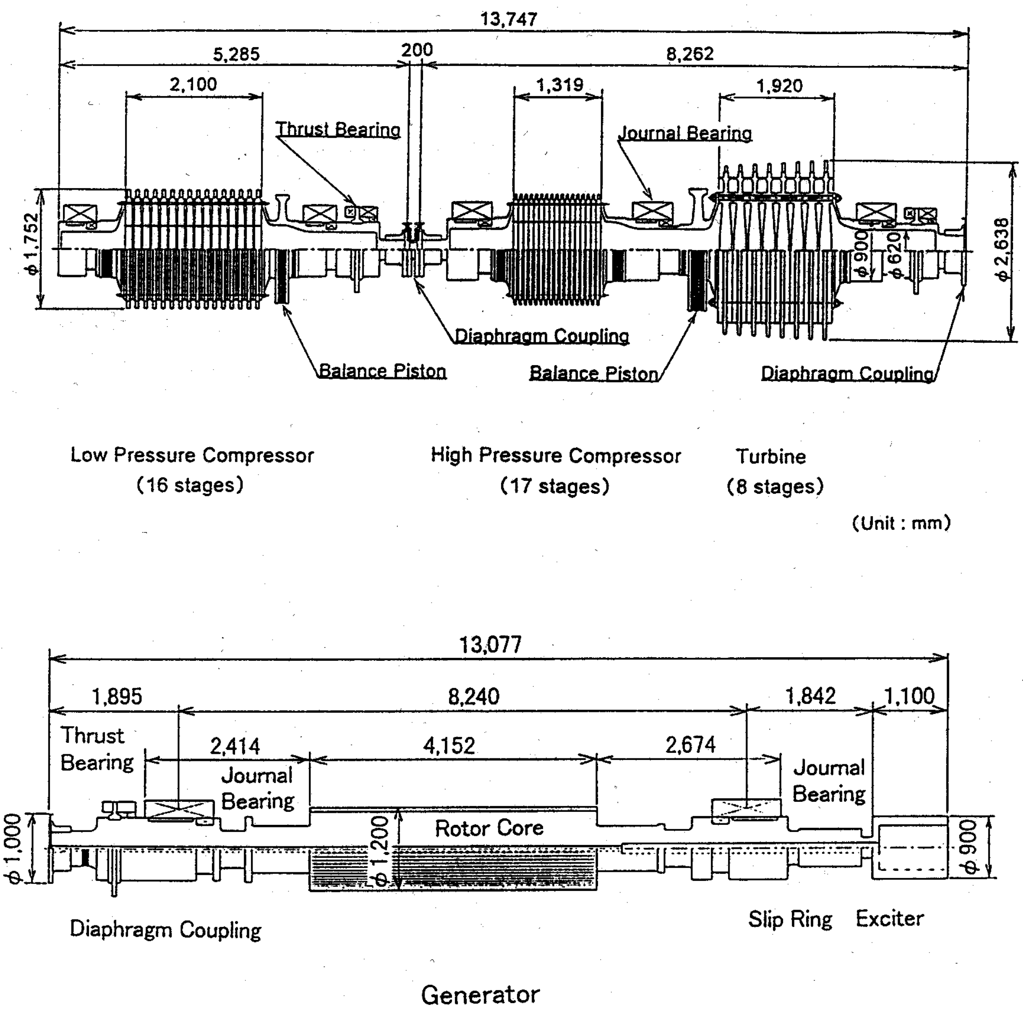


Fig.6.6. Turbine and generator for the 600 MW•t plant.

operation was judged to be possible. Vertical and horizontal views of the 600 MW•t plant vessel arrangement are shown in Fig. 6.7. Vertical orientation is employed for the heat exchange vessel to provide ease of maintenance. The RPV is fixed and the turbine and heat exchanger vessels are designed to move horizontally due to thermal expansion. Slide joints and clank joints are used for the turbine and heat exchanger vessels, respectively [6-2].

6.5.1.2. The 300 MW•t HTGR-GT plan

A reactor utilizing the pebble bed core was selected to be appropriate for the 300 MW•t plant because it is generally believed that this type is superior to the block type for small capacity. As the output of 300 MW•t is larger than the standard pebble bed core of 200 MW•t, the annulus core was employed. With the pebble bed core, an outlet temperature of 900°C could be achieved compared with the block type. Further increase of core outlet temperature was determined to not be advantageous because of the requirement of significant cooling for the turbine blades. As a result of a preliminary survey, by providing a thin thermal insulation within the RPV, a reduction of RPV temperature is realized for the modest power density without sacrificing the passive cooling ability [6-2]. Co-generation of electricity production (148MW(e)) and desalination (product water of 283 t/h) are a significant feature for this plant. This application provides for a net cycle thermal efficiency of 48.2% (Fig. 6.8).

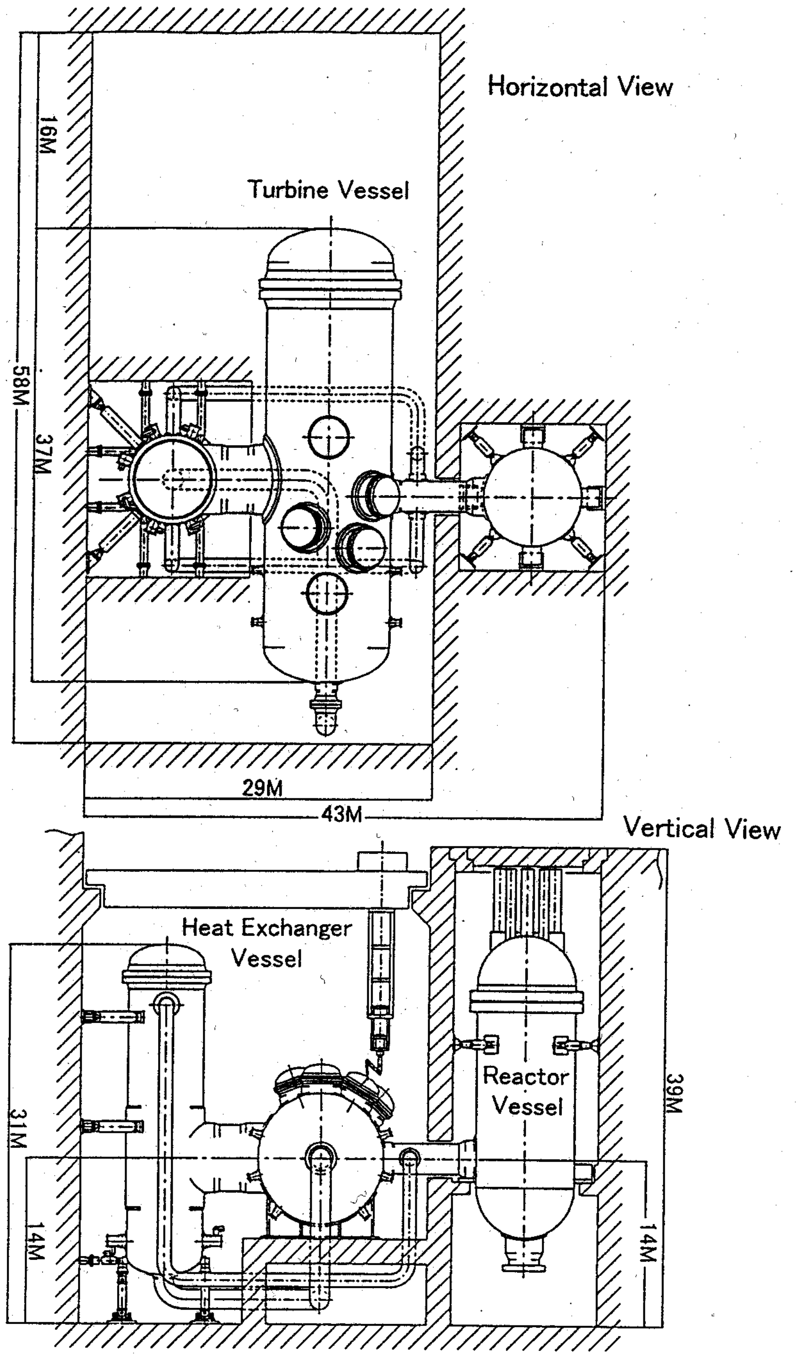


FIG. 6.7. Vertical and horizontal views of the 600 MW•t vessel arrangement.

As a result of preliminary consideration, the increase of reactor power from 300 MW•t to 400 MW•t is judged to be possible for the pebble bed reactor. Therefore, this value was adopted to improve the economic aspects of the closed cycle gas turbine plant. Other specifications are quite similar to those for the 300 MW•t design. However, a fission product trap filter made of sintered Hastelloy X powder and fabricated to cylindrical form is provided in the primary circuit to help in the removal of fission product isotopes prior to their deposit on the components (primarily the turbine blades) of the power conversion system. This would ease maintenance considerations to achieve ALARA and help in the material selection process for these components.

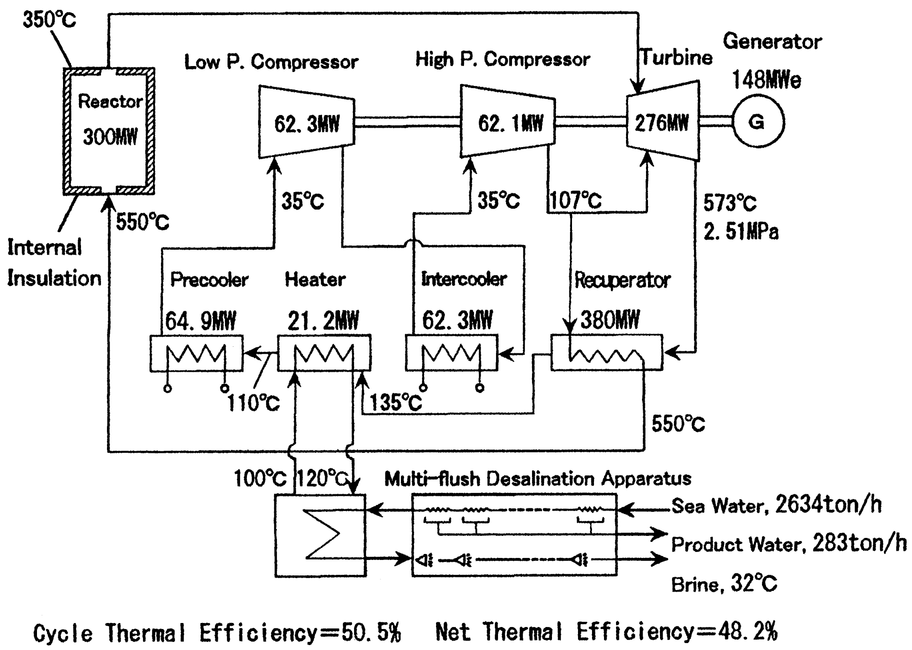


FIG. 6.8. Flow diagram of the 300 MW•t plant.

TABLE 6.2. GTHTR 300 PERFORMANCE DATA

Reactor Power	600 MW•t
Reactor Inlet/Outlet Temperature	561/850°C
Turbine Inlet Pressure	6.83 MPa
Turbine Mass Flowrate	398.3 kg/s
Turbine Pressure Ratio	2.06
Number of Turbine Stages	7
Turbine Blading Polytropic Efficiency	0.925
Turbine Cooling Flow	1.1%
Compressor Inlet Temperature	25.8°C
Compressor Pressure Ratio	2.20
Number of Compressor Stages	21
Compressor Blading Polytropic Efficiency	0.905
Generator Efficiency (electric & windage)	0.985
Recuperator Effectiveness	0.950
Precooler LMTD	33.3°C
Cycle Pressure Loss	6.6%
Parasite Heat Loss	210 kW
Cycle Thermal Efficiency	47.3%
Gross Power Generation	279.6 MW(e)
Net Plant Output	272.6 MW(e)
Net Plant Generating Efficiency	45.4%

6.5.1.3. The GTHTR 300 plant [6-3]

Research by JAERI to optimize the closed cycle gas turbine HTGR plant has resulted in the design of the GTHTR 300. The central objective of the study to arrive at this simplified design was to substantially reduce technical and cost requirements for near-term development of the technology. Of the major simplified design features achieved with the GTHTR 300 based on innovative, yet present day technical solutions, are a reactor pressure vessel constructed of a proven steel, a stand-alone horizontal gas turbine generator module that conforms to general industrial practice, and a conventional layout of plant subsystems of a size that enhances modular construction and maintainability [6-3]. Table 6.2 provides the performance data for this optimized plant.

The rating of the GTHTR 300 modular plant was maximized by attaining the highest thermal output possible per reactor module, to the extent that neither safety nor modularity of the design is essentially penalized. This design philosophy has dictated the selection of a prismatic over pebble-bed core reactor, with the former approaching an output rating up to 300 MW(e) and having the potential cost advantage in the range of 10 to 20% over a 200 MW(e) plant of similar design. The design targets a high fuel burn up averaging 120 GWD/ton with a 10 to 15% fuel enrichment. This should allow for a 2 to 3 year interval between refuelings under an appropriate design scheme by optimizing the axial and horizontal distribution of the fuel enrichment in the core. The long refueling interval along with reliable off-line operation should warrant comparable plant availability between the prismatic and the pebble bed core which does not require off line refueling.

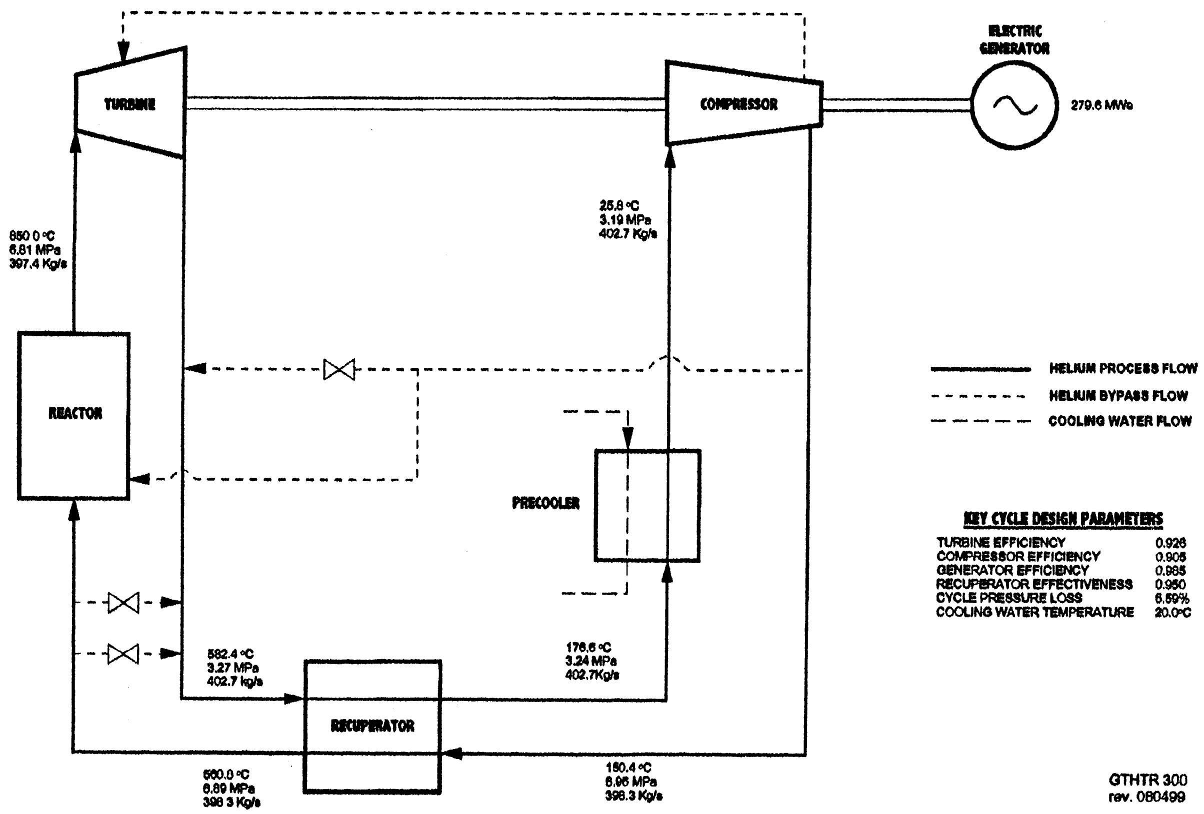


FIG. 6.9. GTHTR 300 process cycle design.

The cycle design is a direct Brayton cycle with effective turbine exhaust heat recovery. The cycle process, as shown in Fig. 6.9, begins with the helium being heated in the reactor to 850°C at 6.81 MPa. The hot helium expands in the turbine to convert thermal energy into the shaft power needed to drive the compressor and a synchronous generator at 3,600 rpm. The helium exhausted from the turbine at 582.4°C and 3.27 MPa enters the recuperator, wherein heat recovery takes place by using the sensible heat remaining in the turbine exhaust flow to preheat the reactor inlet coolant. An effective recovery attainable at this stage contributes significantly to high overall plant efficiency. The helium is further cooled from 176.6° to 25.8°C in the precooler by cooling water, carrying out the waste heat rejection of the cycle. The helium is then guided into the compressor to be compressed to 6.94 MPa. It is then discharged from the compressor at 150.4°C, and is first preheated in the recuperator to 560.8°C and finally heated in the reactor core to the final temperature of 850°C.

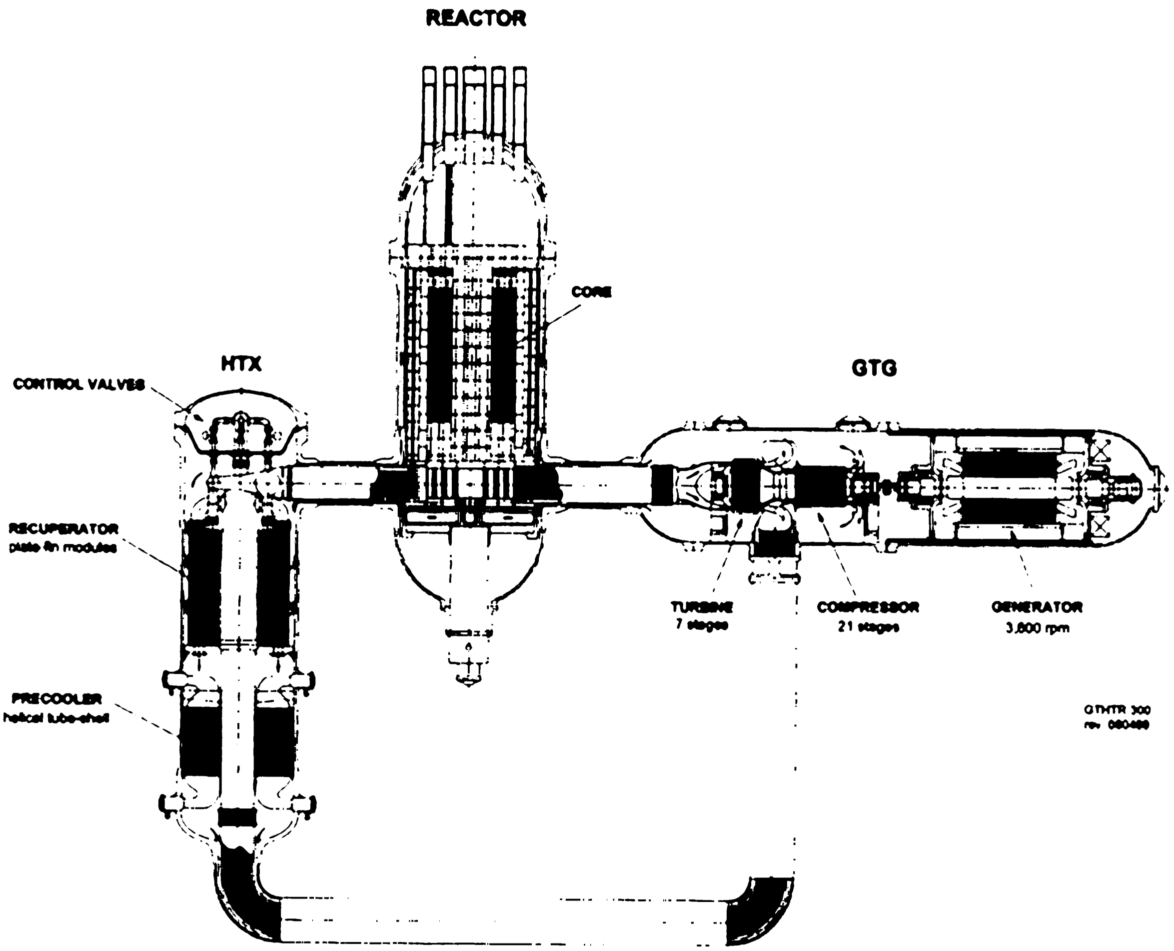


FIG. 6.10. Flow path of the GTHTR 300.

A key technology simplification made with the GTHTR 300 is a RPV that can be fabricated of code-certified, low-cost steel SA533/SA508. This is made possible by a newly conceived plant flow scheme (Figure 6.10) in which coolant is circulated through the reactor via a pair of leveled coaxial crossducts, thereby exposing the RPV, and all other vessels, to temperatures below the design limit and at which the behavior of the materials under irradiation is well understood. Use of SA533/SA508 should also drive down the cost per vessel by a large margin. As a result, despite the need for three vessels, the overall vessel system is still estimated to cost less than potential vessel system designs of less vessel count but having to use high-temperature, more costly steels such as 2 1/4Cr-1Mo and 9Cr-1Mo-V. The latter has not yet been certified for reactor vessel construction. The large saving in material and ease of fabrication, transportation and erection of the vessels of reduced size contribute to the cost advantage of the present vessel system design.

The new flow scheme has additional benefit for cycle design. It allows the core inlet temperature to be chosen so as to maximize cycle efficiency at a low cycle pressure ratio for the 850°C core outlet temperature. This is the optimum cycle design point, as low cycle pressure ratio mitigates peak fuel operating temperatures due to larger core coolant flow rate, and also simplifies the design of the gas turbine.

The present cycle design forgoes cycle intercooling, but still yields a 45.4% net plant efficiency. The intent of the non-intercooled cycle is to simplify the system design to the fullest extent which also results in further lowering the cycle pressure ratio by excluding the compressor intercooler and associated piping,. This also results in simplifying the primary pressure boundaries. However, cycle intercooling may be more appropriately considered as a longer term growth option. The optimum cycle becomes more adaptive to technology growth in that the cycle and resulting structural design would experience no large changes necessary to take advantage of higher core outlet temperatures in order to significantly increase efficiency for the future.

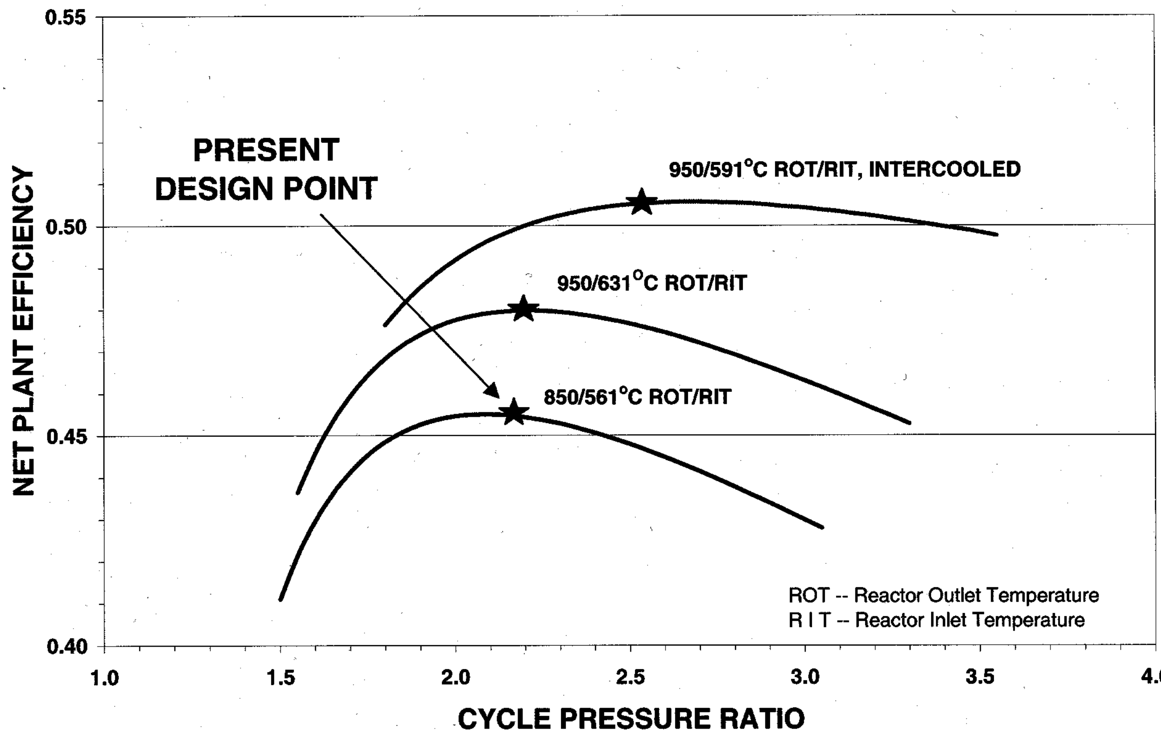
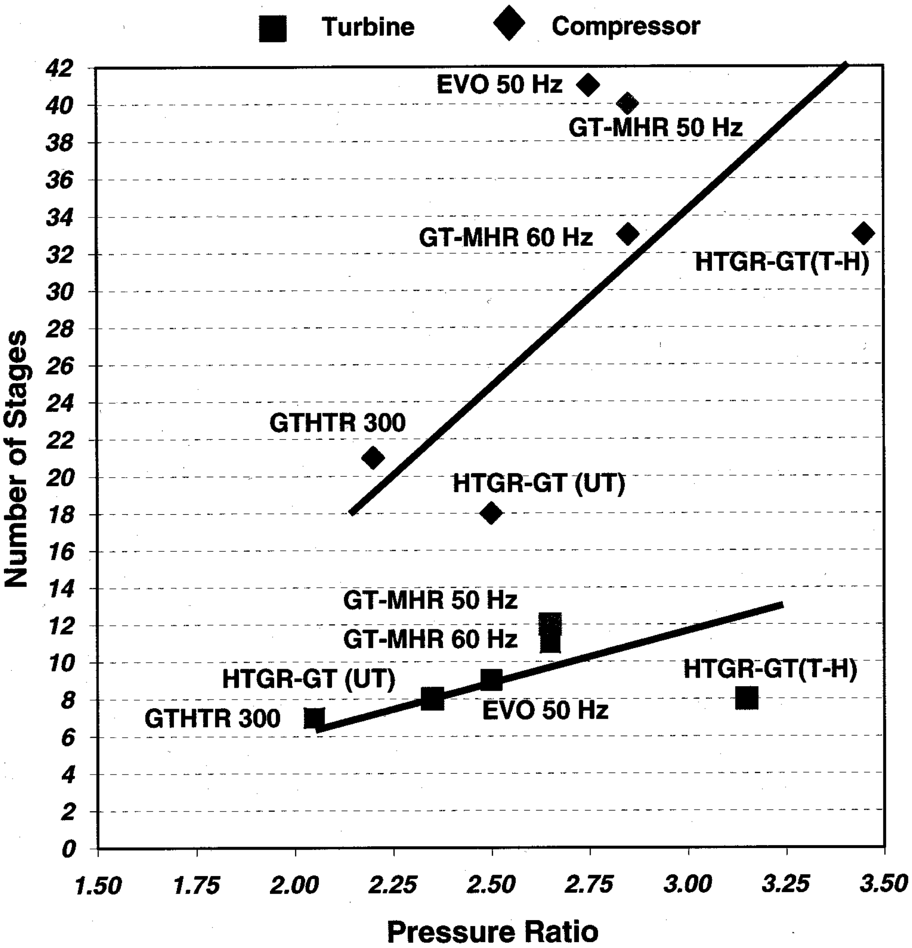


FIG. 6.11. Cycle characteristics and growth potential.

The gas turbine generator (GTG) is made into a stand alone module enclosed in a pressure vessel. By excluding major heat exchange equipment, the size and weight of the gas turbine and the generator contained in the two respective vessel sections falls within the range where site erection of fully shop assembled modules appears practical. The horizontal rather than vertical orientation of the GTG places less demand on the bearings and conforms to general industrial experience and practice. Also owing to the stand alone horizontal arrangement, the gas turbine, being radioactively contaminated in operation, can be more easily serviced and maintained.

The low cycle pressure ratio simplifies the gas turbine mechanical design, which is further helped by improved aerodynamic design conditions of the optimum cycle. Fig. 6.12 is the result of a survey of reported large helium gas turbine designs including the present one, comparing the number of turbine and compressor stages versus their respective design pressure ratios. Since part count is the chief metric to costing out the gas turbine, a reduced number of rotor groups and stages is preferred. Also, fewer rotor groups can result in fewer required bearings; and smaller number of stages means a reduction in the number of vanes and blades. As shown in Fig. 6.12, the present gas turbine design reflects on all these cost benefits.



Sources:

GTHTR 300:	This Presentation
GT-MHR (60 Hz):	GA-A21740, 1994
GT-MHR (50 Hz):	Proc. of 14th IAEA Meeting of IWGGCR, 1997
HTGR-GT (UT):	J. of Engineering for Power, Jan., 1981
HTGR-GT (T-H):	Proc. of IGTC, 1999
EVO (50 HZ):	J. of VGB Kraftwerkstechik, July 1988

FIG. 6.12. Number of stages vs. design pressure ratio.

The plant design consists of three subsystem modules, the 600 MW•t prismatic core reactor, the stand-alone GTG, and a heat exchanger unit (HTX). The three modules are contained in individual vessels and allocated to separate silos. Partitioning the large plant into properly sized subsystems and then arranging them separately is essential to effectuating rapid modular construction and maintenance. In the present design the three modules can be factory built and site erected essentially in parallel. Maintenance can be similarly performed with better simplicity. The ability to not only construct but also service the plant in a truly modular manner will positively impact the economy of the technology.

Two attractive growth options for the GTHTR 300 would be to further increase reactor outlet temperature and to exercise the cycle intercooling option, with the former already being planned in the HTTR demonstration from 850° to 950°C. An increase in efficiency by 2.5% is attainable when reactor outlet temperature is raised from 850°C to 950°C. This is better appreciated because the thermal and aerodynamic designs of the cycle and gas turbine as well as the system structural design remains largely unchanged with the exception that the front stages of the compressor need to be slightly modified, for reduced blade span, to preserve similar aerodynamic conditions. Addition of a single stage cycle of intercooling would have the potential to increase efficiency by another 2 to 3%, depending on the extent of design changes desired.

TABLE 6.3. GTHTR GROWTH POTENTIAL

	Present 850°C ROT*	Near-Term 950°C ROT	Longer-Term Intercooling
Reactor Power	600 MW•t	600 MW•t	600 MW•t
Reactor Inlet/Outlet Temperature	560/850°C	631/950 °C	631/950 °C
Turbine Inlet Pressure	6.83 MPa	6.71 MPa	6.71 MPa
Turbine Pressure Ratio	2.06	2.06	2.04
Turbine Polytropic Efficiency	0.925	0.925	0.925
Turbine Cooling Flow	1.1%	1.6%	1.6%
Compressor Polytropic Efficiency	0.905	0.905	0.905
Generator Efficiency	0.985	0.985	0.985
Recuperator Effectiveness	0.950	0.950	0.950
Cycle Pressure Loss	6.6%	6.6%	7.2%
Cycle Thermal Efficiency	47.3%	49.9%	52.1%
Gross Power Generation	279.6 MW(e)	295.3 MW(e)	307.7 MW(e)
Net Plant Output	272.6 MW(e)	287.9 MW(e)	300.0 MW(e)
Net Plant Generating Efficiency	45.4%	48.0%	50.0%

* ROT: reactor outlet temperature.

The growth potential of the GTHTR 300 is shown in Fig. 6.11. The design points were marked for potential evolutionary systems, whose main design parameters are listed in Table 6.3. It is noted that the paramenters of the last intercooled design included in Table 6.3 are selected such that the changes in design from its non-intercooled predecessors would be limited mainly to the high-pressure compressor section. The efficiency potential of the fully optimized intercooled cycle would approach 50.6%, as marked in Fig. 6.11, but it would require additional design modifications in the turbine and other cycle equipment [6-3].

6.5.2. Research association of HTGR plant (RAHP)

Established in 1985, RAHP is the HTGR plant research association of the private and industrial sector of Japan. The theme of research has consistently been plant concept of the htgr and its commercial feasibility. Membership in RAHP is currently composed of 5 specialists from academia, and representatives from member companies, including all eleven electric power companies, and seven fabricators in the Japanese nuclear industry. Representatives of the New Energy and Industrial Technology Development Organization (NEDO), as well as JAERI participate as observers. The principle activities of RAHP include; (a) Periodic research meetings for investigation on global trends, including special invited lectures as well as exchange of mutual views, (b) Working group meetings for the study of specific problems, (c) Activities related to the public and industrial acceptance of the HTGR in Japan and abroad, including presentations at domestic or international symposiums, and (d) Publishing documents, such as specific reports, executive summaries or annual reports.

RAHP's membership recognizes that the development of nuclear power is essential and supports the HTGR because of its safety attributes and the feasibility for heat energy utilization at temperatures approaching 1000°C, as well as for electricity production. The role of the HTGR is an inter-complementary relationship with the roles of the LWR and the fast breeder reactor, which is the basic line of nuclear development currently adopted by Japan. RAHP considers the following subjects important to the future; (a) Technical demonstration of HTGR safety and high temperature technology, (b) Authoritative, industrial and public acceptance of these safety attributes, (c) The economic prospects towards HTGR commercialization, (d) R&D through international cooperation and effective sharing, and (e) Successful construction and operation of demonstration plant(s) for initial commercialization.

Future activities being addressed by RAHP include. (a) Continue the survey and analysis of global trends of HTGR development, refinement of the national and international HTGR development scenario, identification of subjects remaining to be cleared in the fields of technology and economics, investigation of the fuel cycle, and activities for acceptance of the HTGR and its safety, (b) Support test reactor programmes such as the HTTR and HTR-10, and global activities including the global HTR network, the PBMR and the GT-MHR [6-4].

6.5.3. HTGR research activities and testing facilities

As part of the feasibility study associated with HTGR-GT development, a significant research programme has been initiated in Japan on component design and testing. This includes the recuperator, ceramic turbine rotor, fission product filter and a study of a power conversion system testing facility.

The effectiveness of the recuperator has a strong influence on cycle thermal efficiency. Though a value of 95% is designated in the design criteria, it has not, as yet, been realized. The following research items were selected for the development of the recuperator:

- Fabrication of a model of the heat transfer section (core) for the plate-fin heat exchanger with small fins,
- Analysis of flow unbalance through the core,
- Measurement of the heat transfer coefficient for the small fin,
- Stress analysis of the component.

Fabrication of a model of the recuperator core (200 mm × 200 mm × 200 mm) has been successful and the analysis of flow unbalance for the cold flow side of the core for 95% effectiveness was carried out. Even assuming the poorest flow distribution in the hot side, the reduction of effectiveness was determined to be only 1.7%. The measurement of heat transfer coefficient and the stress analysis are currently in process.

In the area of ceramic turbine rotor development, it is understood that achieving a rotor of light weight and high strength can facilitate significant improvement in the design of the turbomachine and eliminate the need for seal gas on the turbine disc. In the inert helium environment, C/C composite material is the most suitable material. At present, the strength and Young's modulus of the C/C composite is roughly 1/4 that of Inconel 706, and many technical items need resolution and will require long term research. In FY 1998, a disc was fabricated and subsequently tested with the determination that the rotating burst strength was not satisfactory and further evaluation of this problem is under investigation. Currently, three additional discs are being fabricated for additional testing.

R&D associated with the fission product filter has resulted in the determination that the sintered metal comprising the filter will effectively trap micron size impurity particles in the primary system gas. The filters are of stainless steel and Hastelloy X and are commercially available. Also, the pressure drop through the unit is less than 1%, and although the strength of the filter is not high due to its porous media, the expected pressure difference is so low that the structural integrity of filter in an accident should not pose a problem.

An evaporation test facility for silver is also under construction. In this facility, the evaporated silver vapor (or a mixture with the graphite powder) is transferred by the flow of nitrogen gas to the test section, where the filter specimens are currently being tested [6-2].

In the past, it was considered that tests with a full scale turbine are indispensable for the demonstration of the helium turbomachines. According to this policy, the Oberhausen power station and HHV test facility were constructed in Germany [6-28]. To realize a full scale test, a very large non-nuclear high temperature heat source was need. However, research into this area by JAERI has determined that this very large experimental facility is not required. In accordance with the similarity design method, the same performance of the turbomachinery can be demonstrated by appropriate modeling, but with significantly reduced energy requirements. In this arrangement, a non-nuclear energy source of 30 MW is sufficient to validate the turbomachine for the 600 MW•t HTGR-GT plant. It is anticipated that a single test facility of this nature (possibly located at JAERI's Oarai Research site) could support the testing of the power conversion system components being developed throughout the world.

Along with the HTTR, Japan has extensive facilities to support R&D associated with the HTGR. These include facilities for the fabrication of fuel, the Helium Engineering Demonstration Loop (HENDEL), the Japan Material Test Reactor (JMTR), the Oarai Gas Loop (OLG-1), the Japan Research Reactor –2 (JRR-2), the Vessel Panel test facility and the Very High Temperature Reactor Critical Assembly (VHTRC).

HENDEL is a test facility for performing demonstration tests of core components, reactor internals, and high-temperature components in the HTTR. The facility provides the same conditions for the helium gas regarding temperature, pressure and flow rate as in the HTTR. Full scale models of components are used to confirm performance and safety characteristics before the final units are fabricated for the HTTR. The test data from the HENDEL were used for the licensing and detailed design of the HTTR (Fig. 6.13) The fuel stack test section (T_1 test section) is used mainly to investigate the heat transfer and flow characteristics of the fuel elements. The in core structure test section (T_2 test section) is a full-scale model of the core bottom structure of the HTTR, with seven regions of core bottom structure.

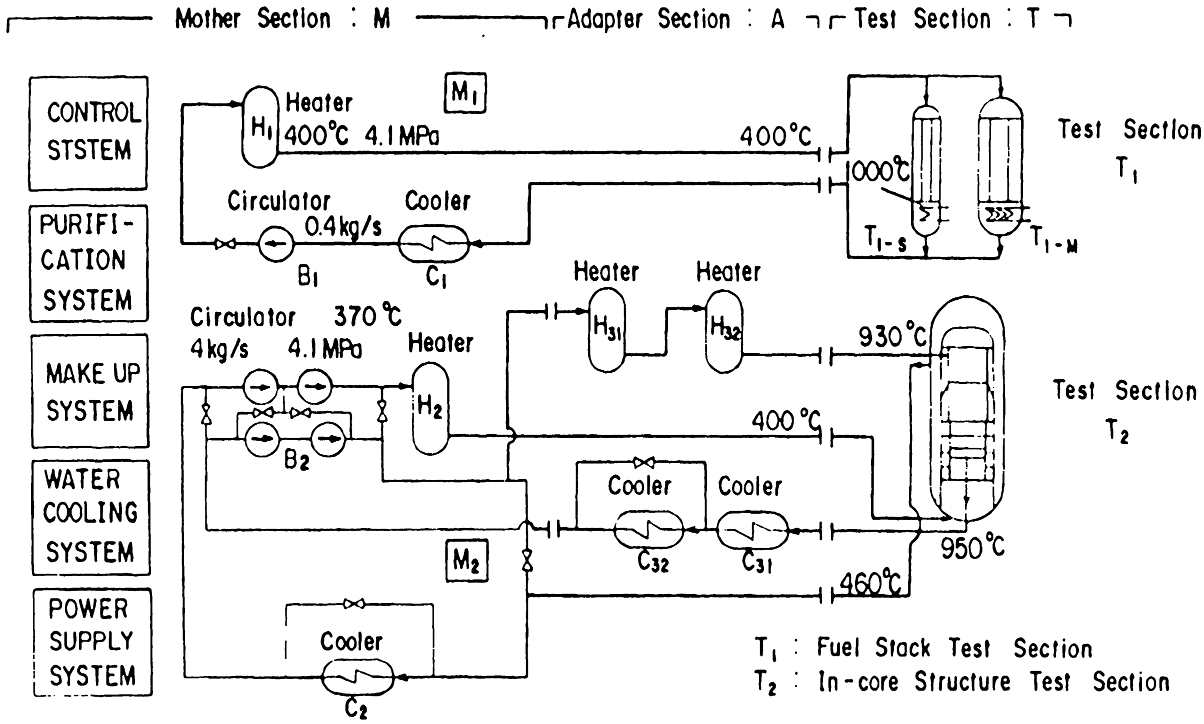


Fig. 6.13. Schematic flow diagram of HENDEL.

The fuel fabrication facility produces HTGR coated fuel particles and compacts. This plant includes processes for kernel, coated particle and fuel compact production in support of the HTTR and for international HTGR fuel fabrication needs.

The JMTR is a tank-in-pool type reactor cooled and moderated by light water with a thermal power of 50 MW. The OGL-1 was installed in the JMTR for irradiation testing of coated fuel particles, heat resisting materials and graphite materials for the HTTR.

The vessel cooling panel test facility is used to investigate the heat removal performance and temperature distribution of the pressure vessel when utilizing an outside arranged water cooling panel (a surface cooler) system. The test apparatus is composed of a pressure vessel 1 m in diameter and 3 m in height containing an electric heater of 100 kW and 600°C, which simulates a reactor core and the water cooling panels surrounding the pressure vessel. The VHTRC is a critical assembly used to study the neutronic characteristics of the HTTR core [6-8].

6.6. NETHERLANDS

The Netherlands is a significant developer of the small HTGR for co-generation applications due to the plants' favorable characteristics of safety and simplicity. ECN Nuclear Research (NRG, an ECN/KEMA company) is developing a conceptual design of an HTR for the combined generation of heat and power for industry within and outside the Netherlands. The design of this small plant for industrial applications is mainly based on a pre-feasibility study in 1996, performed by a joint working group of five Dutch organizations, in which its technical feasibility was determined. The concept that was the subject of this study, INCOGEN, used a 40 MW thermal pebble bed HTR and produced a maximum amount of electricity plus low temperature heat. The system has been improved to produce industrial quality heat, and has been renamed the Advanced Atomic Co-generator for Industrial Applications (ACACIA). The output of this installation is 14 MW of electricity and 17 tonnes of steam per hour, at a pressure of 10 bar and a temperature of 220°C.

The research work for this installation is embedded in a programme that has links to the major HTGR projects in the world. Accordingly, ECN participates in several IAEA Coordinated Research Programmes. ECN is also involved in the South African PBMR project and participates in the European Concerted Action on Innovative HTR [6-5].

6.6.1. The ACACIA plant

6.6.1.1. Background

Safety studies have indicated that an HTR with spherical fuel elements has very favorable safety characteristics: the loss of coolant as well as graphite oxidation will not result in any significant fuel damage. Because of the relatively high temperatures of the system, the HTR has been determined to be very suitable as a heat source for combined generation of heat and power (CHP). Forecasts on the future consumption and production of energy indicate an expanding world market for the CHP. This market for energy efficient CHP with overall capacity of 10 to 150 MW is particularly well developed in the Netherlands. In 1995, approximately 20 per cent of the total electricity supply in the Netherlands was generated by decentralized units and auto-producers. Another 33 per cent was generated by large natural gas fired power plants. Given the expected further depletion of the indigenous resources of natural gas (the current fuel for CHP), a market could emerge for an alternate primary energy source within the next two decades, with nuclear energy as one of the substitutes if competitive prices and public acceptance for this new nuclear application is achieved.

6.6.1.2. Plant design and configuration

The basic configuration of the ACACIA unit includes a helium cooled, graphite moderated nuclear reactor with a thermal power of 40 MW, and a core exit temperature of 800°C. The power is generated in the reactor vessel which contains standard pebble bed HTR spherical fuel elements. The design features of the ACACIA unit include:

- An HTR based versatile heat source combined with a closed (Brayton) cycle energy conversion system,
- A design thermal power of the reactor core of 40 MW with spherical fuel elements 60 mm in diameter containing small (0.9 mm diameter) fuel particles according to the German TRISO design,
- A fueling mechanism during plant operation referred to as the ‘peu-a-peu’ concept
- Resistance against accidents characterized by the two scenarios, loss of flow and loss of coolant, both combined with an ATWS condition,

— Economic and simplified design by reduction of the systems, structures and components that need to meet nuclear qualification requirements.

The helium is first compressed in a compressor (1) and flows subsequently through a recuperator (2), in which it is heated by the exhaust flow from the turbine. In this design, the recuperator could be omitted with very little loss of efficiency. However, the optimum pressure ratio would be much larger, leading to a more expensive turbomachine. After the recuperator, the helium passes through the reactor-core (3), where it is heated. The helium is then expanded in the turbine (4), which drives the compressor and the generator. High speeds are allowed by application of a power electronic converter, which modulate the frequency to 50 Hz. Because of the converter, the shaft speed can be allowed to vary. An alternative for this single shaft arrangement would be a twin shaft system. Then a high pressure turbine would be used to drive the compressor, while a low pressure turbine drives the generator. A synchronous generator could be used, but the low pressure turbine would be much longer. The single shaft system is thus chosen for reasons of simplicity. The helium leaving the turbine is cooled, first in the recuperator and later in the precooler (5). In the precooler, heat is transmitted to the intermediate helium circuit. In the steam cycle, the intermediate helium first flows partly through a superheater (6). The bulk of the flow bypasses this heat exchanger, because the steam has to only be slightly superheated. Then the helium flows through a natural convection evaporator (7) and an economiser (8). The flow is subsequently cooled in the final cooler (9) and compressed in the blower (10). The main design parameters and the flow diagram of the ACACIA unit are presented in Table 6.4 and Fig. 6.14, respectively.

TABLE 6.4. MAIN DESIGN PARAMETERS OF THE ACACIA PLANT

Characteristics of Main Components		**Design Values**
Reactor		
Thermal Power	MW	40
Operating Pressure	MPa	2.3
Helium Inlet/Outlet Temperatures	°C	494/800
Helium Mass Flow	kg/s	25
Heat Cogeneration		
Steam Inlet/ Outlet Temperatures	°C	80/220
Steam Pressure	MPa	1
Steam Mass Flow	kg/s	4.7
Transferred Heat	MW•t	12
Generator		
Electrical Power	MW(e)	13.6
Overall Performance		
Electric Efficiency ($P_{electric}$ /$P_{reactor}$)	-	0.34
Heat Efficiency (P_{heat} /$P_{reactor}$)	-	0.30
Power to Heat Ratio	-	0.88

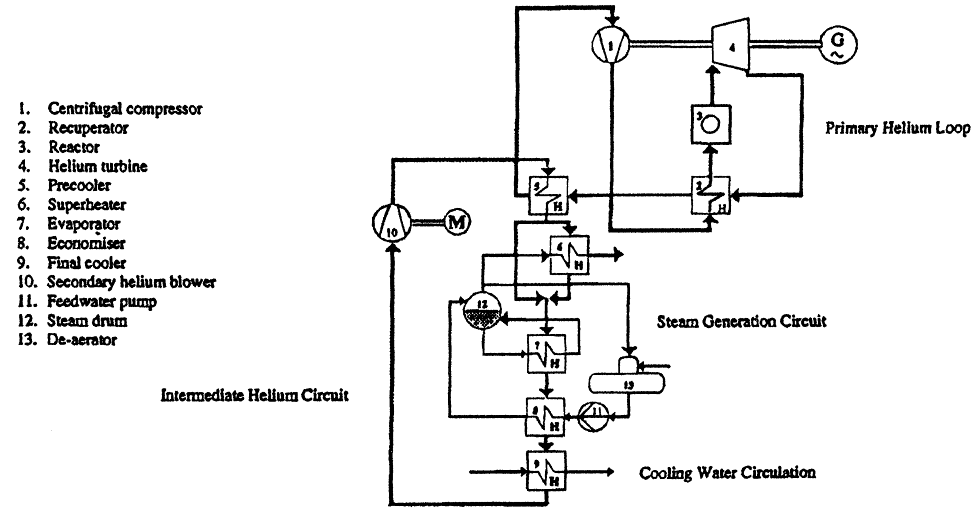


FIG. 6.14. Flow diagram of the ACACIA co-generation plant.

6.6.1.3. Reactor design

The fuel will be loaded into the core, as long as space allows, during operation in a manner that keeps the core only marginally critical. Void core volume can accommodate added fuel until defueling. Because of the foreseen period for inspection and maintenance of the power conversion system and the reactor pressure vessel a defueling interval of three years has been chosen for the ACACIA system (versus 10 years in the INCOGEN[2] study).

This reduction of the defueling interval has advantages for the core dimensions, in particular, a significant decrease is expected in the core height. Safety relevant parameters like temperature coefficients and fuel temperatures for the most adverse heat-up conditions (LOCA, LOFA, ATWS) have been determined for the INCOGEN configuration. Since ACACIA uses the same core configuration, the ACACIA core safety behaviour is expected to be similar.

6.6.1.4. Energy conversion system and plant layout

A major change of the ACACIA installation relative to INCOGEN is the energy conversion system. The INCOGEN concept is optimized for the production of electricity, while industrial quality heat is the primary output of the ACACIA system. This results in a more suitable output of superheated steam for industrial applications, which is mainly driven by reasons of market potential. A consequence of this optimization to the production of heat is less electrical power and a lower overall performance. But, these disadvantages are lessened against the much higher market potential of the co-generation unit for industrial applications, and therefore the configuration becomes a more realistic alternative to the conventional (natural gas fired) industrial CHP systems. Figure 6.15 is a cross section of the reactor and turbine building of the ACACIA unit.

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[2] INCOGEN, which is an acronym for Inherently safe Nuclear COGENeration, is the name of the analysed configuration of the pre-feasibility study, in 1996. The primary output of this installation was 16.5 MW electricity. The remaining heat to be used for co-generation applications, which heats an external pressurized water circuit from 40°C to 150°C, useable for low temperature industrial processes or district heating networks.

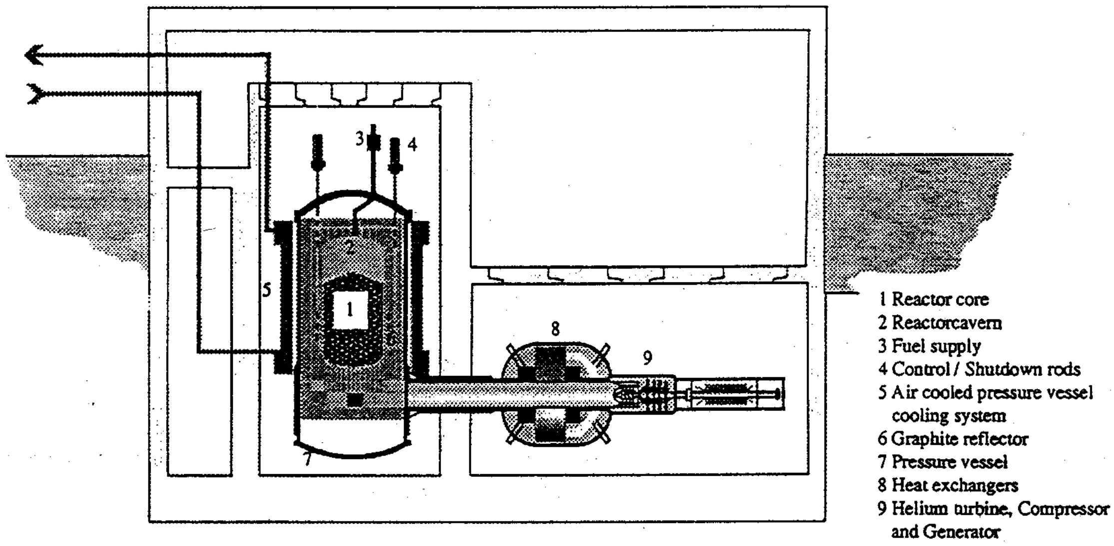

FIG. 6.15. Cross section of the ACACIA reactor and turbine building.

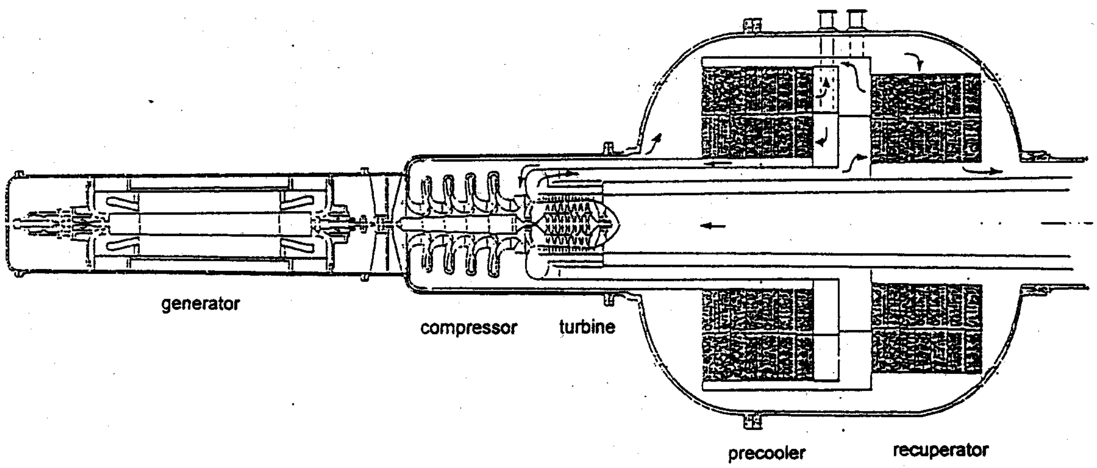

FIG. 6.16. Schematic of the primary cycle of the energy conversion system.

The primary cycle of the energy conversion system is completely contained in a pressure vessel as shown in Figure 6.16. The steam drum and final cooler are not shown. This equipment is located on the same level as the gas to gas heat exchangers and the turbomachine. The design of the secondary cycle is currently not integrated, but it is a simple series connection of tube and shell heat exchangers.

6.6.1.5. ACACIA economic assessment

A reassessment of the economics was highly recommended in the INCOGEN pre-feasibility reports[3,4], because the cost figures for INCOGEN were based on down-scaled data from

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[3] van Heek, A.I. (Ed.), INCOGEN Pre-feasibility Study; Nuclear Cogeneration, INCOGEN Working Group, ECN, Netherlands (1997).

[4] van Heek, A.I. (Ed), INCOGEN Pre-feasibility Study; Nuclear Cogeneration, INCOGEN Working Group, ECN, Netherlands (1997).

Siemens for the reactor and General Atomics for the closed cycle helium turbine. This approach lead to the conclusion that the costs were too high compared to the natural gas fired CHP plant. A second review of the economics was then carried out. This focuses on these new cost estimates for an Nth-Of-A-Kind (NOAK) ACACIA system, which is based on recent cost data for the South African conceptual PBMR[5].

In the INCOGEN pre-feasibility study, the investment cost and production cost have been estimated at 8909 Netherlands Guilders (NLG) per produced kW(e) and 0.158 NLG per produced electrical kWh, respectively. A credit for the co-generated heat (based on the Dutch industry price for natural gas), and an annual discount rate of 10% for the capital costs have been taken into account. The new assessment shows that for the ACACIA unit, by scaling down recent South African cost data translated to the Dutch situation, these important key figures are reduced by 33% to 5961 NLG/kW(e) and 0.106 NLG/kWh, respectively[6] [7]**.** Due to the simplified design of the smaller ACACIA system versus the PBMR, an additional price reduction of at least 20% are envisaged. The production cost of an equally sized natural gas fired CHP unit is 0.057 NLG per kWh. Comparing the new economic figures with this amount indicates that a 40 MW thermal CHP unit will be entering the competitive area [6-5].

6.6.2. R&D facilities in the Netherlands

The principle focus of HTR related research at NRG is divided into the following specific areas:

- Reactor physics, including core neutronics, thermal hydraulics, shielding associated with the PBMR, dynamics of the core and power conversion unit and plutonium burning in the HTR,
- Waste behavior during long term storage,
- CFD analyses on the PBMR power conversion unit,
- Thermal hydraulic safety analyses on the PBMR,
- HTR co-generation energy conversion [6-7].

Within this research programme, a concerted effort is devoted to computer code development. Although the pebble bed neutronics/thermal hydraulics code VSOP of FZJ Juelich is available through the NEA Databank, ECN decided to develop its own code system. The dynamic neutronics code PANTHER has been acquired from AEA, UK This code is coupled to the thermal hydraulics code THERMIX-DIREKT, delivered to ECN by FZJ Juelich in a cooperation framework. Considerable attention is paid to the generation of nuclear data. For the PANTHER-THERMIX system, neutron cross sections are generated by the WIMS-E and SCALE-4 codes. The Monte-Carlo code, MCNP, is used for this as well, and to check certain reactor calculations. In order to gain expertise in calculations on gas cooled graphite moderated reactors, bench mark calculations were performed with these codes

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[5] Nicholls, D.R., Pebble Bed Modular Reactor (PBMR) - Programme Status, Proceedings of the IAEA Technical Conmittee Meeting on High Temperature Gas Cooled Reactor Applications and Future Prospects, Petten, Netherlands(103–109).

[6] To facilitate a comparison with the existing cost estimates, which was part of the INCOGEN pre-feasibility study, the same methodology is used in the current economic assessment. The scaling exponents are chosen in accordance to the publication "Nuclear Energy Cost Data Base" of the US Department of Energy, September 1988.

[7] van der Laag, P.C. Mozaffarian, H., Cost estimates for the 40 MW•th ACACIA system, ECN Fuels, Conversion and Environmental internal memo K.1605-GR23.

in parallel with international partners on similar systems: the PROTEUS experiment at PSI Villigen, Switzerland and the 450 MW•th MHTGR of General Atomics [6-6].

In order to establish the ACACIA plant, it is essential that the economic and technological research work be embedded in an international HTGR network. ECN participates in many IAEA activities, and is involved with the HTTR, PBMR and GT-MHR designs. ECN takes part, with partners from Germany, France, United Kingdom and Italy, in the European Concerted Action (CA) on Innovative HTR to define cooperative developments in the HTR field for the European Union's Fifth Framework Programme [6-5]. In this regard, NRG is involved in the areas of core physics and the fuel cycle, fuel and material irradiation and PIE, system analysis, component development and economics [6-7].

6.7. RUSSIAN FEDERATION

Development of the HTGR was initiated in the Russian Federation approximately 30 years ago. During this period designs were carried out for the VGR-50, VG-400, VGM and VGW reactor plants. Although the principle focus of the Russian Federation's current HTGR programme is with the development of the GT-MHR plant (Chapter 4), extensive design capability and testing facilities are available for the international development of this advanced nuclear power reactor

6.7.1. Background

The VG series of HTGR plants and their final phase of development is provided in Table 6.5. Basically, the VG-50 consisted of a helium cooled pebble bed reactor of 136 MW•t (50 MW(e)) for the generation of electricity and irradiation of polyethylene tubes.

TABLE 6.5. HTGR DESIGNS IN THE RUSSIAN FEDERATION (PRIOR TO THE GT-MHR)

Name of the project	Dates of beginning and completion of the project	General Designer of the reactor	Phase of development
VGR-50	1963–1985	Research Institute of Machinery for Atomics industry, Moscow	Detailed design
VG-400	1974–1987	OKBM Nizhny Novgorod	Detailed design
VGM	1986–1991	OKBM	Detailed design
VGMP	1991–1992	OKBM	Feasibility study

The VG-400 was a much larger plant (4 loops with 1060 MW•t/300 MW(e)) intended for both electricity generation and process heat production for steam reforming of methane. The reactor outlet helium temperature for the VG-400 was designed at 950°C. During the preliminary phases of the design development both the pebble bed and prismatic fuel block variants of the core were analyzed. As a result of the design and engineering analysis, the pebble bed core was chosen for further development due to the following considerations:

— The simplified technology of fuel element manufacturing and possibility of their full scale testing in experimental reactors,
— The utilization of a simplified core refueling mechanism,
— The possibility of core refueling during on-load reactor operation.

The VG-400 reactor with a closed gas turbine cycle was also evaluated [6-8].

An interest in smaller, more passively safe reactors became apparent after the Chernobyl accident. A closer working relationship developed between Germany and the Russian Federation and, in 1988, agreements were signed between the State Committee of the USSR for the Utilization of Nuclear Energy and the Federal German HTGR industry, ABB/HRB and Siemens/Interatom. Under the agreements, the Russian Federation intended to build an experimental modular HTGR plant with the support of German industry.

The conceptual design of the modular plant (VGM) that evolved was very similar to the Siemens/Interatom side by side HTR module. A notable difference was the incorporation of both an intermediate heat exchanger (for process heat experiments) and a steam generator in the main heat transport system (see Fig. 6.17 and Table 6.6 for the primary system arrangement and basic parameters of the VGM, respectively).

TABLE 6.6. VGM BASIC PARAMETERS

Thermal power (MW)	200
Primary circuit parameters:	
– pressure (MPa)	3.9
– helium outlet temperature (°C)	750 (950)
Power density (MW/m^3)	3
Fuel element type and diameter (mm)	spherical, 60
Number of fuel elements in the core	3×10^5
Fuel enrichment (%)	6.5–10.0
Mean burn-up (MWd/t)	$(6\text{-}9) \times 10^4$
Cycle length (effective days)	600–900
Number of main cooling loops	1
Steam pressure (MPa)	17.0
Steam temperature (°C)	540
Reactor vessel	steel

The intent was to operate initially with an outlet temperature of 750°C by mixing an adjustable core bypass helium flow with the higher temperature core outlet helium flow and generate steam only. Later, the mixed outlet temperature would be adjusted to 950°C by changing the core bypass flow, to generate high temperature helium for process heat used in the intermediate heat exchanger followed by steam generation.

The primary goals of the pilot plant VGM project were:

(1) The solution of research and engineering tasks associated with mastering a new type of reactor, i.e.:
- Validation of neutron and thermal hydraulic parameters of the pebble bed core, analysis and optimization of methods for gas quality monitoring and control of helium coolant and its interaction with primary circuit materials.
- Operability testing of high temperature structural and heat insulating materials, design verification during reactor operation of fuel elements, graphite structures, steam generator, gas circulator, control rod drive mechanisms, fueling/defueling facility, instrumentation, control systems, etc.
- Study of core heat removal in normal operation and accidental conditions.

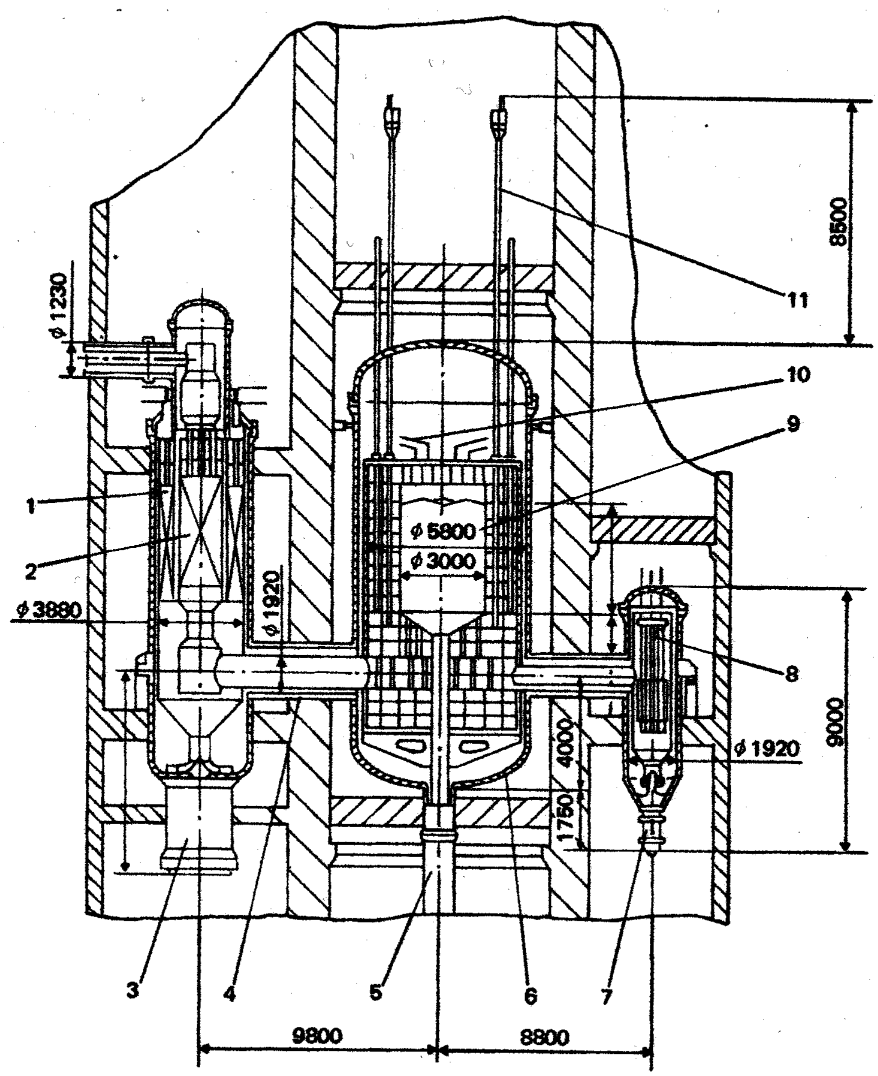

FIG. 6.17. VGM primary system arrangement.

(2) Serve as the pilot plant for VG-400 plant components in reactor conditions.
(3) Test stage by stage reactor plant operation in the 750° to 950°C coolant temperature range using a core bypass system.

The following safety principles underly the VGM concept:

(1) In all emergencies, reactor shutdown is reliably effected by free falling control rods and the reactivity compensation system, the latter based on insertion of small diameter absorber balls into the side reflector channels. For 'hot' shutdown, actuation of the control rods is sufficient. Control rods immersible in the pebble bed are not required.
(2) In all emergencies, including primary circuit depressurization, core residual heat removal can be effected by a surface cooling system consisting of three independent channels. Heat through the reactor vessel is transferred to a surface cooling system that can remove the core residual heat by the natural circulation of water. To reduce the number of actuations of the surface cooling system, the plant is provided with one ~7 MW auxiliary heat removal loop, assuring prolonged residual heat removal should the main loop be switched off or fail.
(3) Multiple passage of fuel elements through the core allows reactor operation without a sizeable margin of reactivity.
(4) The arrangement adopted in the VGM favors localization of hot helium in the graphite internals region and minimization of heat transfer to supporting metal structures when primary circuit heat removal fails and limitation of steam/water mixture ingress into the core if the heat exchangers develop a leak.
(5) For assuring residual heat removal after primary circuit depressurization it is not necessary to have a pressure tight containment. The containment functions are: protection from external impacts, retention of radioactive substances escaping from the core using filters and prevention of intensive core oxidation.

The main criterion characterizing the plant safety was adoption of 1600°C as the maximum fuel temperature for all possible accidents, a temperature that ensures retention of fission products in the coated fuel particles to within required limits [6-16].

6.7.2. Technical resources and facilities

Along with nearly 35 different research institutes, numerous experimental facilities have been developed in the Russian Federation for R&D associated with the HTGR including an irradiation loop for spherical fuel elements, two critical facilities and facilities to investigate pebble bed core characteristics, graphite friction and wear, and helium coolant technology at temperatures up to 1000°C [6-16]. Prominent among these are the GROG and ASTRA critical facilities placed in RRC "KI".

The "cold" GROG facility has a maximum space of 4500 × 4500 × 4500 mm. The packing is formed by graphite square blocks with cross sections of 250 × 250 mm consisting of 9 channels with a diameter of 55 mm arranged in the nodes of square cells of 83.3 mm spacing. A portion of these can be filled with cylindrical universal physical imitators that simulate fuel rods and consisting of uranium and a fluorumplastic mixture. A portion of the holes can be used for the arrangement of reactivity compensation elements. A wide variety of universal physical imitators allows the simulation of an initial HTGR loading with main physical characteristics which adequately describe the reactor. A rig is equipped with modern instrumentation which allows the performance of physical experiments in both the sub-critical and critical states.

The “cold” ASTRA facility has a maximum critical assembly of 460 mm height and a maximum diameter of 3800 mm. This facility permits simulation of a core comprised of pebble fuel elements with different fuel element packings, including point-on-point packing with a porosity of .26, which is closely spaced to the porosity of a block type core. The rig has the possibility for up to 50 000 spherical fuel elements of 60 mm diameter with a U-235 loading of .51 g/element and of 21% enrichment, and also of graphite spherical elements of the same diameter and poison balls with B_4C containing 0.1 g natural boron per ball.

Other facilities associated with the development of the GT-MHR are addressed in Chapter 4. Along with GROG and ASTRA, other significant HTGR support facilities existed within the Russian Federation as of the early 1990s. Many of these have since been deactivated, but some may be brought into active use for the HTGR should a specific need develop. A listing of these by subject and location include:

- Fuel and graphite irradiation; The helium circulation loop, PG-100 at the MR reactor, RRC “KI”, the “Kashtan” and “Karat” channels of the reactor MR, the ampoule channels of the RBT at the RBT-6 reactor (NIIAR, Melekes) and the IVV-2M reactor (Zarachy, Ekaterinburg region), the channel “Udar-III of the VVR-C reactor (Obninsk)
- Facilities for testing fuel behaviour under accident conditions; The helium loop PG-100 and the hot helium CGS test facility and the pulse reactor GIDRA at the RRC-“KI”
- The OKBM has the following experimental facilities for testing HTGR components:

  (1) The ST-1312 High temperature Gas Test Facility is used for full scale testing of steam generators and high temperature heat exchangers under operating conditions. This facility is also used for development of control and instrumentation systems for reactors

  (2) The main circulator test facility (ST-1383) is used for testing of full scale prototypes of the primary circuit gas circulators and valves as well are other equipment

  (3) The ST-1565 high temperature helium test facility is used for testing of fittings, thermal insulation, relief valve, helium mixer models; study of helium coolant technology, study of thermal and hydraulic parameters of steam generators and heat exchangers models. Also, for simulation of residual heat removal from the core in emergency cases. Study of heat and mass transfer under depressurization of the primary circuit and the study of thermal dynamic processes for water ingress into the primary circuit

  (4) The control rod test facility is for the evaluation of control rod drive systems

  (5) Test facilily for tribology investigation is used for testing of tribotechnical data of different materials in units with radial and axial friction as well as for rolling bearings serviceability in inert gases.

  (6) Air test facility ST-1654 is for investigation of characteristics of non-isothermal flows and validation of computer codes. Non-isothermal flows are simulated with the help of a propane gas tracer. This facility is also used for investigation of the core outlet plenum and mixing devices.

  (7) Masex test facility is for investigation of heat and mass transfer through penetrations, orifices and tubes. This facility is also used for computer code validations.

(8) TIGR test facility for investigation of the change of primary circuit parameters in the course of a coolant outflow through the orifice upon loss of tightness. It is also used for the determination of the pressure drops at structural elements upon a large section rupture. [6-8].

6.8. SOUTH AFRICA

Details associated with the South African HTGR programme are provided in Chapter 3. Significant utilization of international expertise and R&D facilities is a key aspect of the South African PBMR development programme (see Table 3-16 for a listing of technical areas and associated suppliers). However, ESKOM has substantial nuclear power expertise primarily due to its Koeberg PWR plant near Cape Town.

6.8.1. R&D resources and facilities

The nuclear facilities at the South African Atomic Energy Corporation (AEC) include a test stand for evaluation of small turbo-machines with magnetic bearings, the SAFARI material test reactor, hot cell facilities and a fuel fabrication facility which previously provided fuel for the PWRs at Koeberg. This facility is being made available for the transfer of fuel development technology for fabrication of coated fuel particles and spherical fuel elements for the PBMR. AEC has extensive expertise in nuclear research and component development.

A dynamic engineering simulator of the PBMR reactor and power conversion system has also been developed. The major areas for study with this simulator include:

- Simulation of predefined normal, incident and accident operating scenarios
- Demonstration and verification of the plant engineering model
- Defining the control system requirements
- Analyze dynamic plant behaviour [6-8].

6.9. SWITZERLAND

6.9.1. Background

Although Switzerland is presently not an active participant in development of the HTGR, it has an extensive history in R&D associated with this technology.

Starting in the early 1960s, several Swiss industrial companies and the Swiss Federal Institute for Reactor Research (EIR), now the Paul Scherrer Institute, participated in development programmes and provided engineering and components for several gas cooled reactor projects. Switzerland maintained a primary interest in the application of the HTGR to closed cycle gas turbine (direct cycle) technology and, with the termination of the German gas turbine HTGR programme (HHT) in 1979, Swiss activity in HTR development decreased significantly.

However, in 1983, the German company Hochtemperatur-Reaktorbau (HRB) and the Schweizerische Interessengemeinschaft zur Wahrnehmung gemeinsamer Interessen an der Entwicklung nuklearer Technologien (IGNT), an industrial group formed initially to support the closed cycle gas turbine work, signed a letter of intent to cooperate on the HTR-500 in order to enable Swiss industry to maintain its knowledge of HTGR technology and to be able to participate in future projects. The IGNT request to the Swiss Ministry of Energy for funding for research and development support work for this reactor power plant was approved by the government with the proviso that industry share the cost on a 50/50 basis. Work performed by the industrial group included:

— Steam generator and auxiliary heat exchanger design development and advanced methods for thermal and mechanical analysis, investigation in the fields of material behaviour and surface protection,
— Contributions to development and design calculations on prestressed concrete pressure vessels, including tests on the behaviour of concrete at temperatures of up to 120°C and tests for the determination of anchoring characteristics and on the behaviour of the liner at singularities,
— Reactor core and ceramic components including investigation of the dynamic behaviour of the pebble bed and of the side reflector under seismic excitation, and development of computer programs for the structural analysis of the core bottom plate under normal and accident conditions.

In 1982, analyses of the district heating market and the possible application of nuclear energy from small heating reactors were conducted in Switzerland. To meet the specific Swiss requirements, a short distance heat supply from heating reactors rated between 10 and 50 MW•t power output was envisaged. This initiative had a positive response from the public as well as from industry. A gas cooled heating reactor (GHR), whose conceptual design was developed by ABB/HRB based on HTGR technology, was one of the concepts under consideration [6-16]. However, the developmental work on this plant was stopped in 1988.

The Paul Scherrer Institute, in addition to its other activities in directly supporting industrial firms, has also performed significant work in the fields of high temperature materials and reactor theory. Most recently, the staff at the institute has provided the use of the PROTEUS facility for a series of critical experiments on low enriched fuelled small MHTGR type cores within the IAEA CRP on Validation of Safety Related Physics Calculations for Low Enriched Gas Cooled Reactors”.

6.10. UNITED KINGDOM

The British were early pioneers in the field of gas cooled reactors and made significant contributions to HTGR development, especially in the areas of fuel, core and heat transfer technology. The 20 MW•t Dragon reactor, which was sponsored by the Organisation for Economic Co-operation and Development (OECD), Euratom and the UKAEA, was one of the principal HTGR test facilities until its decommissioning during the mid 1970s. The British HTGR programme was discontinued at that time because of a decision taken by the UK Department of Energy to concentrate on the development of the steam generating heavy water reactor concept which was, however, abandoned a few years later. The AGR became the mainstay of the British nuclear reactor programme in the 1970s but continuing controversy over the economics of the AGR system led to an extensive evaluation of the PWR and the decision to turn to this reactor for the next nuclear power programme.

6.10.1. Background

Some of the oldest Magnox reactors, the first generation of gas cooled reactors, have been in operation for more than 25 years. These reactors are fueled with natural uranium metal, clad with Magnox (a magnesium-aluminium alloy), and use $C0_2$ as the coolant. The early Magnox plants had steel pressure vessels while the later, higher output versions at Oldbury and Wylfa used PCRVs. In the United Kingdom, the higher power output versions had to be derated by approximately 20% following the discovery of a steel oxidation problem in 1969 and, except for the twin Hunterston-A station, these reactors continue to be operated in this manner. Other temporary operating restrictions on the Magnox plants have gradually been relieved through refurbishment of equipment and, in general, the Magnox reactors have shown a steady improvement in capacity factor over the years. Reviews were initiated in the early 1980s to investigate the potential of extending the operation of the older Magnox plants beyond the original design fife. In April 1988, the Central Electricity Generating Board (CEGB) decided to shut down the Berkeley station. One of the reactors was shut down in late 1988, the second reactor in early 1989. The South of Scotland Electricity Board (SSEB) safety review of Hunterston-A was delivered to the Nuclear Installations Inspectorate in 1988 and these two Magnox plants were subsequently shutdown in 1989 and 1990.

The AGR plants have also had problems with oxidation and have experienced other situations which precluded on line refueling in the early years of operation. Subsequent development work resulted in the ability to progressively raise the output of the AGRs to more than 97% of the original gross electric output and thermal ratings to greater than 100% of design. On line refueling was able to commence in 1982 but with the power output temporarily reduced. The CEGB anticipates that the output of the reactors will be able to be progressively raised to about 700 MW(e) each (about 40 MW(e) above design rating) with minor modifications [6-8]. The cumulative load and availability factors for the AGRs to 1996 was 62.9% and 66.7%, respectively [6-23].

6.10.2. Gas cooled reactor R&D

The Health and Safety Executive (HSE) are responsible for managing nuclear safety research related to the UK thermal reactor sites. The research programme is funded by the major nuclear licensees (Nuclear Electric, Scottish Nuclear, Magnox and British Nuclear Fuels). The HSEs Nuclear Installations Inspectorate (NII) identifies major safety issues, which are used for the basis of the research programme. These issues are arranged the following into 16 technical areas:

(1) Plant life management — steel components
(2) Plant life management — civil engineering
(3) Chemical processes
(4) Fuel and core
(5) Graphite
(6) Fission products
(7) Reactor physics, shielding and criticality
(8) Heat transfer and thermal hydraulics
(9) Severe accidents
(10) External events
(11) Control and instrumentation
(12) Human factors
(13) Probabilistic safety analysis
(14) Radiological protection
(15) Waste and decommissioning
(16) Nuclear systems and equipment.

The main driver for the gas cooled reactor research programme is the maintenance of a capability that will address any generic safety issue that arises. This is of particular importance as the gas cooled reactors age and their lives are extended to meet the industries need to maximize plant output [6-24].

In May 2000, BNFL became a significant joint venture partner with ESKOM and IDC of South Africa in the Holding Company of the PBMR project. As such, BNFL will participate in the PBMR including design and development, fuel manufacture and all subsequent plant manufacturing.

A significant area of research for the HTGR is nuclear graphite performance with plant life and associated structural changes due to the effects of irradiation and temperature. As an outgrowth of an IAEA sponsored specialists meeting held in 1995 at Bath University, a plan for the archiving of irradiated nuclear graphite data for the common use of all Member States was developed and subsequently presented to the IWGGCR. This recommendation was the result of a growing need to share in the scientific progress and operational experience on the influence of neutron irradiation of graphite; particularly now with nearly forty years of reactor operational history and growing concerns that the number of experts in this field is diminishing significantly [6-25].

The HSE, because of the safety implications associated with the structural and other property changes which occur with irradiation of graphite, initiated a follow on study concerning this archive and subsequently initiated its development. This irradiated nuclear graphite database is now being established at the IAEA. Also, the IAEA and HSE jointly supported the preparation of the document Irradiation Damage in Graphite due to Fast Neutrons in Fission and Fusion Systems (see Chapter 10, Section 10.1.2. and 10.3.3. for details). Major international support and development activities on nuclear graphite are focused at AEA Technologies plc. In Risley, UK.

6.11. UNITED STATES OF AMERICA

Gas cooled reactor development in the USA was originally initiated in the late 1940s. Subsequent development into the HTGR included the construction and operation of the Peach Bottom-1 and Fort St. Vrain plants for the generation of electricity. Further development activities to early in the 1990s included designs primarily incorporating the steam cycle. HTGR development activities were then re-addressed to the closed cycle gas turbine (GT-MHR) plant as discussed in Chapter 4.

6.11.1. Background

The present ongoing HTGR programme in the USA had its inception in 1957 when General Atomics (then the General Atomic Division of General Dynamics Corporation) evolved the first design concepts based on earlier work, primarily in the UK, on lower temperature, gas cooled concepts. Driven primarily by the incentive of higher efficiency electricity generation and supported by various utility organizations, the programme has been concentrated exclusively on the helium cooled, graphite moderated, coated particle fuelled HTGR.

The Peach Bottom-1 (40 MW(e)) developmental plant, which was constructed for and operated by Philadelphia Electric Company as part of the US government's Power Reactor Demonstration Program, was the first HTGR power plant built in the USA. Construction on the plant was started in 1962 and was completed in 1967. Operation of Peach Bottom-1 during the seven year period from 1967 to 1974 resulted in the generation of electricity for an equivalent of 1349 full power days with an overall average availability of 88% for the nuclear steam supply system exclusive of planned shutdowns for research and development programmes.

The overall operation of Peach Bottom-1 was a significant accomplishment in terms of demonstrating the capability of equipment in a high temperature helium environment. The reactor did undergo one complete core refueling after the first 150 effective full days of operation as a result of fuel failure. The first core contained the earliest version of coated fuel, utilizing single pyrocarbon coatings to prevent hydrolysis of the carbide particles during fuel element fabrication, shipment and storage. During operation, cracking of the coatings resulted in fuel compact swelling and the subsequent cracking of some of the unique fuel element sleeves used in Peach Bottom-1. The core was replaced with fuel elements containing the multiple barrier type of coatings which were being developed for follow-on plants, such as Fort St. Vrain, and operated successfully for the remainder of the plant operation period. Having completed its mission, the Peach Bottom-1 plant was shut down in 1974 and has been partially decommissioned.

The 330 MW(e) Fort St. Vrain (FSV) Nuclear Generating Station constructed for and operated by the Public Service Company (PSC) of Colorado was the second HTGR built in the US. Site construction was started in September 1968. Fuel was loaded in late 1973, and initial criticality was achieved in early 1974. Electric power generation commenced in late 1976, but the plant was limited to 70% power because of core region gas outlet temperature fluctuations. Mechanical restraints placed in existing holes on top of the top reflector tied the regions of the core together and stopped the fluctuations.

FSV contained many innovative features and first-of-a-kind components that, in part, contributed to its poor performance. Most noteworthy in this regard were the steam turbine driven helium circulators. To avoid the potential for oil ingress into the primary helium coolant system, the circulator design incorporated a water bearing/lubricating system and a labyrinth sealing system. The service system required to maintain proper operation under all reactor operating conditions was complex and the leakage of water from the bearing/seal system into the primary system was a consistent problem and the most significant unavailability factor for the plant. Considerable time and effort were required to lower the moisture content to the required levels to permit power operation after each water ingress incident.

In December 1988, PSC announced plans to discontinue nuclear operations at FSV by 30 June 90. The decision to begin defueling and decommissioning activities was based on economic considerations associated with the ongoing operating costs of the plant. FSV was subsequently prematurely shutdown in 1989 due to a generic structural problem associated with the steam generator ring headers. The plant was then decommissioned under the “early dismantlement” programme and the site was authorized for “unrestricted use” by the US Nuclear Regulatory Commission (NRC) in 1997 (see Section 8.2.2 for decommissioning details).

In the early 1970s, General Atomics entered the US commercial market, and ten units (five twin plant contracts) ranging in size from 770 MW(e) to 1160 MW(e) were sold in the period 1971 to 1974. However, six of these units were cancelled by the utilities in 1974–1975, because of the economic recession following the oil crisis in 1973, and the vendor terminated the remaining contracts and withdrew from the market in 1975.

The government sponsored programme was subsequently broadened in 1977 concurrent with the formation of Gas Cooled Reactor Associates (GCRA), a utility organization formed to provide support and user direction to the HTGR programme. The initial efforts were focused on the design and development of an 840 MW(e) HTGR based on evolving design improvements of the commercial plants and the once through, low enriched uranium/thorium fuel cycle. However, the continuing disenchantment with nuclear power in the USA, magnified by the Three Mile Island accident, and more recently by the events at Chernobyl, led to increasing requirements for the assurance of nuclear safety and to the redirection of the government programme in 1984 to consideration of small modular designs.

Following a detailed evaluation in the spring and summer of 1985, the side by side concept similar in configuration to the German module was chosen in September 1985, as the reference concept for further design and development by the US programme. The basic module was designed to deliver 350 MW•t of heat in the form of superheated steam with 17.3 MPa, 538°C conditions.

As a result of a US utility survey conducted by GCRA in 1985, the future market for electricity in the USA was identified to be concentrated predominantly in the 400 to 600 MW(e) size range. The reference plant concept was therefore identified as a four module plant with a total thermal output of 1400 MW and a net electrical output of approximately 540 MW(e). A preliminary safety information document (PSID) was submitted to the NRC in autumn 1986. Review meetings between the NRC staff and the programme participants were initiated in autumn 1986 and continued through 1987. NRC conclusions from the review were documented in a safety evaluation report, a draft of which was completed in June 1988 and reviewed by the Advisory Committee on Reactor Safeguards [6-16]. A draft version of the safety evaluation report was issued in 1989. This draft was generally favorable toward the MHTGR concept on major items including the containment approach and emergency planning requirements. However, formal issue of the safety evaluation report was subsequently delayed indefinitely due to NRC questions regarding design philosophy differences between the reviewed MHTGR design and the New Production Reactor variant of the MHTGR and due to the (1995) termination of the US DOE MHTGR programme [6-8].

In August 1988, the Defense Programs Office of the US DOE recommended, a dual reactor concept, dual site strategy for the acquisition of new production capacity for strategic nuclear materials, with a modified version of the four module MHTGR plant (MHTGR-NPR) as one of the reactor concepts. This programme was subsequently shutdown in late 1992 as the result of a reassessment for the near term needs by the US government for additional strategic nuclear material.

Irrespective of this programme, work continued on the civilian programme with the signing by the government, in 1988, of multi-year contracts with the team of General Atomics (GA), Combustion Engineering and Bechtel for the nuclear island design and Stone and Webster with support from Combustion Engineering for the balance of the MHTGR plant design. The contracts covered the preliminary and final designs of the reference plant and the related licensing activities required to obtain a final design approval (FDA) from the US NRC [6-16].

In the early 1990s, advances in technologies encouraged reconsideration of the direct cycle gas turbine. The small passively safe modular helium reactor size now better matched the sizes of available gas turbines. Moreover, the reduced core size of the MHTGR and the reactor's completely passive decay heat removal capability minimized the safety concerns which had previously been troublesome for the large HTGR gas turbine designs considered in the 1970s [6-8]. This, along with the advances made in the previous decade on small, modular recuperators and magnetic bearings, resulted in the re-direction of the US programme to the closed cycle gas turbine GT-MHR (see Chapter 4 for details of this programme).

Subsequently, in 1995, the government cancelled further support of the US gas cooled reactor programme. However, GA continued work on the GT-MHR and in 1993 signed an agreement with the Russian Federation for technical development (principally OKBM) of this concept with the initial deployment of the plant for the burning of plutonium.

6.11.2. Current HTGR activities in the USA

The interest in using the GT-MHR for the burning of weapons material was the catalyst for the decision by the US government to re-initiate support of the GT-MHR in 1999. The economic, safety, environmental and high efficiency aspects of the gas turbine HTGR plant has resulted in development work by the Massachusetts Institute of Technology (MIT) and the Idaho National Engineering & Environmental Laboratory (INEEL) on the Modular Pebble Bed Reactor project and initiation of a Helium Gas Reactor (HGR) R&D programme by the US Electric Power Research Institute (EPRI).

6.11.2.1. The Modular pebble bed reactor (MPBR) project of MIT and INEEL

The development of the MPBR is being addressed based on preliminary research into future energy options by MIT student work beginning in 1998. It was concluded that this technology provided the best opportunity to satisfy the safety, economic, proliferation, and waste disposal concerns that face all nuclear generating technologies. The areas of research for this project are aimed at addressing some of these fundamental concerns to determine whether the small 110 MW(e) modular gas-cooled pebble bed plant can become the next generation of nuclear technology for worldwide deployment [6-27].

MIT and INEEL have utilized the reference design from the ESKOM PBMR, but with a significantly different balance of plant. The HTGR is of pebble bed design with a power level of 250 MW•t. Primary coolant helium from the reactor flows through an intermediate heat exchange (IHX) providing a transfer of energy to the secondary coolant (air). The secondary loop consists of a high pressure turbine which drives three (high, medium and low pressure) compressors with two stages of intercooling. A second shaft incorporates the low pressure turbine and electric generator. High temperature coolant air leaves the IHX to drive the compressors via the high pressure turbine. After exiting this turbine, the coolant then enters the low pressure turbine to expand and drive the generator. From this turbine, the secondary coolant path includes the recuperator, precooler and compressors/intercoolers. The flow diagram for this plant is included as Figure 6.18. The MPBR utilizes conventional oil bearings rather then magnetic bearings on its turbomachines.

The specific areas of research being conducted by MIT and INEEL on the MPBR include improvements in fuel design, spent fuel disposal characteristics, methods to enhance non-proliferation, thermal hydraulics modeling and core neutronics modeling [6-27].

6.11.2.2. The EPRI HGR programme

EPRI's interest in the HGR is the enhancement of its technical capability to support member utilities in the development and potential deployment of this advanced nuclear power plant as a competitive electricity generator. As an initial step in fostering this capability, EPRI hosted a workshop on the HGR on December 7–8, 1999. Participation in this workshop included representatives from US utilities, EPRI, and HGR design and scientific personnel from France, Japan, Netherlands, South Africa, the United Kingdom and the USA. The objectives of EPRI's HGR programme include:

- Contributing to specific advancements in technology for HGR deployment potential and readiness
- Leveraging the results of the HGR programme by establishing collaboration with technical and business interests in the nuclear power field
- Choose specific advancements to pursue consistent with EPRI's experience, expertise and capabilities
- Maintain a balance between pursuing parallel HGR design alternatives and focusing available resources on the design closest to construction (the PBMR)
- Establish a comprehensive owner/operator involvement to maximize HGR deployment
- Develop an increasingly accurate evaluation of the HGR's economic competitiveness and deployability.

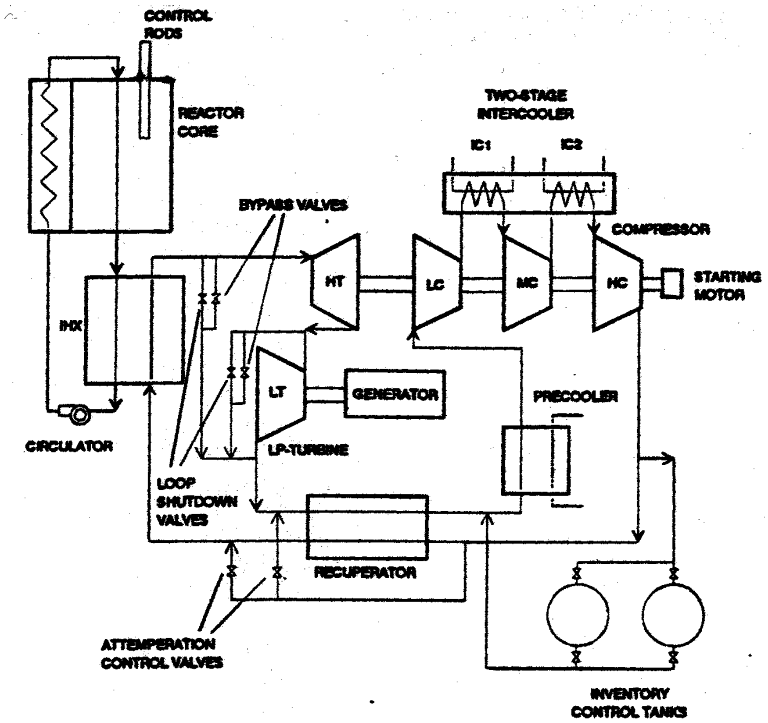

Fig. 6.18. MPBR reference design flow schematic.

A tentative action plan for future support of the HGR was presented by EPRI and discussed at the Workshop. As the PBMR appears to be the closest to prototype deployment, the intent is for this plant to be the primary focus. However, series/parallel attention is being placed on the alternative designs including the prismatic core HGR, single shaft turbo-machine configuration and the indirect cycle. Subject to final changes in scope, activities in the following general areas are being initiated by EPRI in 2000:

- Review of the HGR's economic competitiveness
- Development of owner/operator requirements for the HGR
- Evaluate issues of accessibility, inspectability and maintainability of HGR primary system components
- Confirmation of reliability on magnetic/catcher bearings for size, orientation and operating environment of the HGR

— Confirmation of the 60 year plant life operating requirement(s) of nuclear graphite in the operating conditions anticipated in the HGR
— Evaluation of air as the turbine driver in an alternative HGR design that utilizes an intermediate heat exchanger ahead of the power conversion system [6-26].

6.11.3. US HTGR R&D support facilities

Significant HTGR support facilities existed within the USA as of the early 1990s. Many of these have since been deactivated, but some may be brought into active use for the HTGR should the specific needs require. A listing of these by subject and location include:

— Irradiation facilities; INEEL (Advanced Test Reactor), the High Flux Isotope Reactor at Oak Ridge National Laboratory (ORNL)
— Post irradiation examination; INEEL and ORNL Hot Cell facilities
— Materials Testing; the Integral Air-Graphite Oxidation facility at Los Alamos National Laboratory (LANL), the Graphite Materials Test facility at ORNL
— Fuel fabrication facilities; Babcock and Wilcox
— Flow testing; Air Flow Test facility at ABB-CE, Chattanooga
— Physics test facilities; the Transient Reactor Test facility at Argonne National Laboratory-West, Critical Experiments Facility at LANL [6-8].

Other facilities associated with the development of the GT-MHR are addressed in Chapter 4.

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Chapter 7

CO-GENERATION AND NON-ELECTRIC CAPABILITIES OF THE HTGR

Over half of the world's primary energy consumption is used for the production of hot water, steam and heat for various industrial applications. This requirement is presently met almost exclusively through the use of fossil fuels. The emission of carbon dioxide and other gases resulting from the burning of fossil fuels poses serious challenges to our climate and environment. Nuclear energy can help contribute to the solution of these problems. This technology which, to date, has been used almost exclusively in the production of electricity, has the capability of being a significant, clean alternative to fossil fuels for the production of hot water, steam and industrial process heat applications [7-1].

Nuclear energy is now being used to produce about 17% of the world's electricity. As of 1998, this included 442 nuclear reactors, with a total capacity of about 354 gigawatts-electric (GW(e)). Yet, only a few of these plants are being used to supply hot water and steam. The total capacity of these plants is about 5 GW•t, and they are operating in just a few countries, mostly in Canada, China, Kazakhstan, the Russian Federation and the Ukraine.

For heat applications, specific temperature requirements vary greatly (Fig. 7.1). They range from low temperatures (~ room temperature), for applications such as hot water and steam for agro-industry, district heating, and seawater desalination, to ~1000°C for process steam and heat for the chemical industry and high pressure injection steam for enhanced oil recovery, oil refinery processes, and refinement of coal and lignite. The process of water splitting for the production of hydrogen is at the upper end. Up to about 550°C, the heat can be supplied by steam; above that, requirements must be served directly by process heat due to the need for high steam pressures. An upper limit of 1000°C for nuclear supplied process heat is set on the basis of the long term strength capabilities of metallic reactor materials.

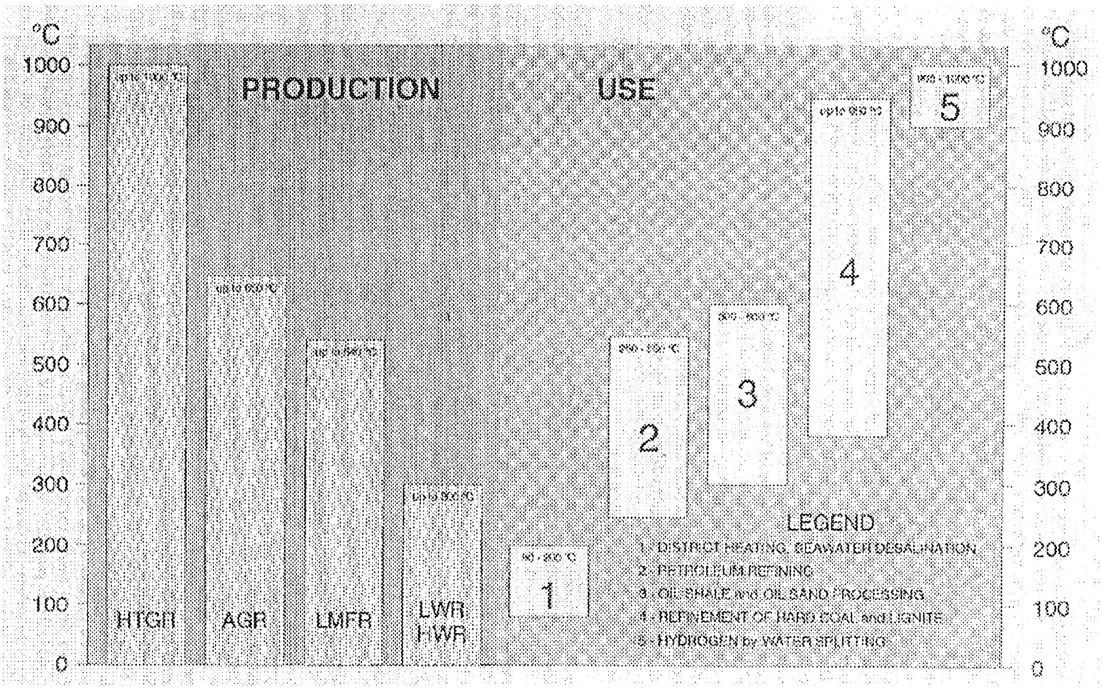

FIG. 7.1. Temperature ranges in production and use of nuclear energy.

Water cooled reactors offer heat up to 300°C. These reactor types include PWRs, BWRs, pressurized heavy water reactors, and light water cooled graphite moderated reactors. Organic cooled heavy water moderated reactors reach temperatures of about 400°C, while liquid metal fast breeder reactors produce heat up to 540°C. Gas cooled reactors reach even higher temperatures, about 650°C for the AGR, and 950°C for the HTGR [7-2].

The ultimate potential offered by HTGRs derives from their unique ability to provide heat at high-temperatures (e.g. in the range from about 550° to 1000°C) for endothermic chemical processes and, at 850°C and above, for highly efficient generation of electricity with gas turbine technology. Heat from HTGRs could be used for production of synthesis gas and/or hydrogen and methanol by steam-methane reforming, production of hydrogen by high temperature electrolysis of steam and by thermal-chemical splitting of water. The HTGR can also be the high temperature energy source for the production of methanol by steam or hydrogasification of coal, and for processes requiring lower temperatures, such as petroleum refining, seawater desalination, district heating and generation of steam for heavy oil recovery. If the heat demand is not in the immediate vicinity of the reactor, a chemical heat pipe could be developed as a high temperature heat transporter.

It is important to establish nuclear heat process application technology through research, development and demonstration [7-3]. In this regard, a significant field of investigation by Member States comprising the IWGGCR is in the co-generation and non-electric applications of the HTGR. This Chapter provides an overview of the multiple industrial process heat applications of the HTGR and on-going activities to achieve their potential.

7.1. CO-GENERATION APPLICATIONS OF THE HTGR

7.1.1. Co-generation/steam cycle

7.1.1.1. China's 2 × 200 MW•t plant

China's 2 × 200 MW•t plant offers the co-generation capabilities of providing electricity and desalination of seawater. This plant, under design at INET, was chosen based on being sited at locations in China where the dual need exists for both electricity and clean water. The optimized design characteristics of this plant result in a final product of 93 500 m^3/d and 147 MW(e) [7-4]. The flow diagram for this design is provided in Section 6.1.2.2.

Other co-generation applications of the 2 × 200 MW•t plant under investigation by INET include process steam for the petrochemical industry, heavy oil recovery, coal conversion and district heating. An example of the versatility provided by the HTGR for co-generation applications is with the petrochemical industry.

In the late 1980s, a Sino–German joint study was made on the application of the modular HTGR in the petrochemical industry. The complex of the Yan Shan Petrochemical General Corporation (YSPGC) was selected as the reference user candidate in the study.

The annual energy consumption for supplying steam, process heat and electricity in the YSPGC complex is in the range of 1.2 million tons of oil. The total requirement of steam in the different pressure and temperature ranges is ~730 t/h in the summer and 1650 t/h in winter. The steam parameters are 118 bar/500°C, 47–50 bar/450°C, 34–39 bar/350°C and 8–13 bar/280°C. The energy supply is mainly through steam-electricity co-generation.

Based on the above demand on steam and electricity, a HTGR 4-module plant was suggested with the following key parameters:

Thermal power output: 4 × 200 MW•t = 800 MW•t
Live steam mass flow: 4 × 250 t/h = 1000 t/h
Live steam pressure/temperature: 190 bar/530°C

The four modules supply the steam for operating three back pressure turbines and for providing the heat source to four process steam systems. The secondary water/steam circuit is separated from the process steam systems by heat exchanging components to avoid mixing. The industrial process water feeding the four process steam systems is taken from cold water storage and jointly preheated up to 170°C. Outputs of the overall plant are as follows:

Electricity output of generator: 139 MW(e)
Process steam of 118 bar/500°C at the rate of 30 t/h
Process steam of 48 bar/450°C at the rate of 73 t/h
Process steam of 36 bar/350°C at the rate of 310 t/h
Process steam of 10 bar/280°C at the rate of 500 t/h

In the joint study, another reference plant design with two ABB-HRB reactor modules was also made. The economic analysis performed within the framework of the joint study show that HTGR plants for the application in petrochemical industry are competitive with comparison to fossil fueled plants on the basis of the international market price for fossil fuels. With the energy market in China becoming more and more international and due to the large energy consumption by the chemical industry, the HTGR should have a large application potential in this field [7-13].

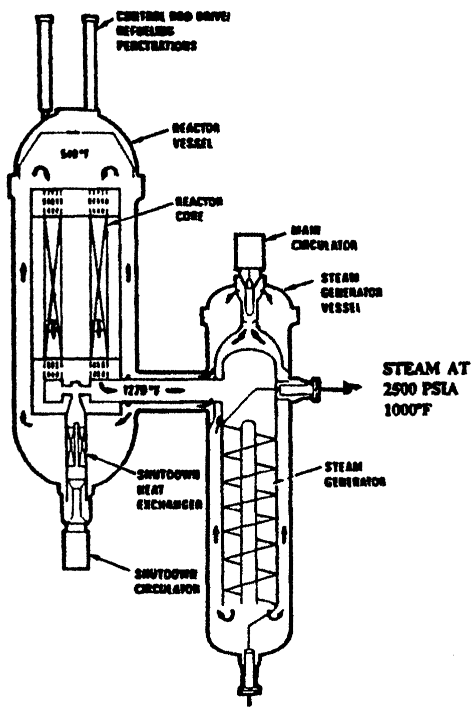

FIG. 7.2. PS/C-MHR plant module.

7.1.1.2. MHTGR applications by General Atomics

General Atomic is investigating the capabilities and flexibility of the MHTGR to provide superheated steam for a multitude of industrial applications. Energy requirements of industrial process complexes vary widely, according to varying steam conditions, capacity requirements, and the ratio of thermal to electric power. The high temperature and pressure steam at 17.3 MPa and 540°C produced by the process steam/co-generation modular helium reactor (PS/C-MHR) can provide energy for heat cycles in a wide range of process applications and industrial complex sizes and capacities. The PS/C-MHR is being designed to meet the rigorous requirements established by the US NRC and the electric utility-user industry for a second generation power source for the future.

The most economic PS/C-MHR plant configuration includes an arrangement of several identical modular reactor units, each located in a single reactor building. The plant is divided into two major areas: the nuclear island, containing the several reactor modules, and an energy conversion area, containing turbine generators and other balance of plant equipment. The basic layout for a single reactor module is shown in Figure 7.2. Each reactor module can be connected independently to a steam turbine and/or other steam utilizing systems. The nominal PS/C-MHR plant parameters are given in Table 7.1.

TABLE 7.1. PS/C-MHR PLANT PARAMETERS

Reactor Module Parameters	Recommended Design
Thermal Power, MW(t)	600
Fuel Columns	102
Fuel Cycle	LEU/Natural U
Average Power Density, W/cm³	6.6
Primary Side Pressure, MPa (psia)	7.07 (1025)
Induced Helium Flowrate	281 kg/s
Core Inlet Temperature, °C (°F)	288(550)
Core Outlet Temperature, °C (°F)	704(1300)
Steam Temperature, °C (°F)	541(1005)
Steam Pressure, MPa (psia)	17.3(2515)
Circulator Power, MW(e)	6.0

The reactor module components are contained within three steel pressure vessels; the reactor vessel, a steam generator vessel, and connecting cross vessel. The uninsulated steel reactor pressure vessel is approximately the same size as that of a large BWR and contains the core, reflector, and associated supports. The reactor core and the surrounding graphite reflectors are supported on a steel core support plate at the lower end of the reactor vessel. Top mounted penetrations house the control rod drive mechanisms and the hoppers containing boron carbide pellets for reserve shutdown.

The heat transport system (HTS) provides heat transfer during normal operation or under normal shutdown operation using high pressure, compressor driven helium that is heated as it flows down through the core. The coolant flows through the coaxial hot duct inside the cross vessel and downward over the once-through helical bundle steam generator. Helium then flows upward, in an annulus, between the steam generator vessel and a shroud leading to the main circulator inlet. The main circulator is a helium submerged, electric-motor-driven, two-stage axial compressor with active magnetic bearings. The circulator discharges helium through the annulus of the cross vessel and hot duct and then upward past the reactor vessel walls to the top plenum over the core.

For availability and maintenance requirements, a separate shutdown cooling system (SCS) is provided as a backup to the primary HTS. The shutdown heat exchanger and shutdown cooling circulator are mounted on the bottom of the reactor vessel. The heat removal systems allow hands-on module maintenance to begin within 24 hours after plant shutdown.

The reactor cavity cooling system (RCCS) is located in the concrete structure external to the reactor vessel to provide a passive heat sink for the removal of residual heat from the reactor cavity if the HTS and SCS are unavailable to perform their intended functions. The RCCS consists of above grade intake structures that naturally convect outside air down through enclosed ducts and panels that surround the below grade core cavity before returning the warmed air through above grade outflow structures. The core heat is transferred by conduction, convection, and radiation from the core to the RCCS. This system has no controls, valves, circulating fans, or other active components and operates continuously during normal operation and during shutdown conditions.

Major co-generation applications are highly energy intensive and diverse, including such processes as those associated with heavy oil recovery, tar sands oil recovery, coal liquification, coal gasification, steel mill and aluminum mill processes. Use of the MHR in each of these processes has been studied at General Atomics [7-5]. Figure 7.3. is typical of the tie between the 2 × 600 MW•t PS/C-MHR reactors and the co-generation applications of electricity production and heat for an industrial process, such as heavy oil recovery.

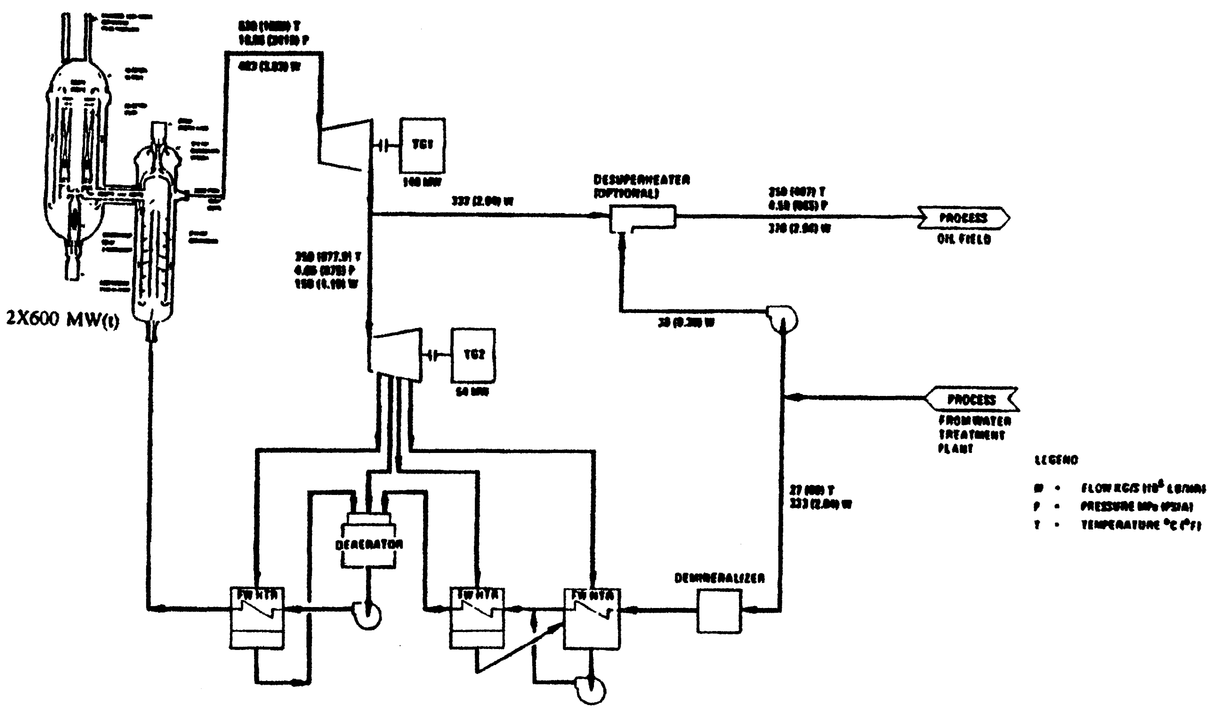

FIG. 7.3. PS/C-MHR plant for heavy oil recovery and electricity generation.

7.1.2. Co-generation/gas turbine cycle

Many IWGGCR Member States are actively investigating the capabilities of the HTGR for co-generation applications of electricity generation via the gas turbine combined with using the heat for industrial processes.

7.1.2.1. JAERI's 300 MW•t HTGR-GT

The 300 MW•t HTGR-GT design by JAERI (Section 6.5.1.2.) utilizes a 300 MW•t pebble bed HTGR to provide the energy to produce 148 MW(e) and 283 ton/h product water via the multi-flush desalination method. This efficient use of nuclear energy results in a cycle thermal efficiency of 50.5%.

7.1.2.2. ACACIA plant by NRG

At NRG, development of the ACACIA plant is proceeding for the combined output of 14 MW(e) and 17 tonnes/h steam at 10 bar and 220°C based on the nuclear thermal energy of 40 MW from a pebble bed reactor. The details and flow characteristics of the ACACIA plant are provided in Section 6.6.1. This design focuses on achieving superheated steam for industrial applications that are primarily driven by reasons of market potential.

7.1.2.3. HTR-10 test reactor

The HTR-10 is designed as a module for multipurpose research. In the first phase, the HTR-10 is planned for operation in connection with a steam turbine which works in a co-generation mode to provide electricity and heat (see Chapter 5 for details). In the second phase, the coolant outlet temperature will be raised from 700° to 900°C for operation with a gas turbine cycle.

7.2. NON-ELECTRIC APPLICATIONS OF THE HTGR

The HTGR is capable of providing the largest range of process heat temperatures of all nuclear power plant types and, therefore, can support a very wide range of industrial applications. At the Advisory Group Meeting held in Jakarta, Indonesia, to advise the IAEA on non-electric applications of nuclear energy, the question arose as to which of the industrial processes should receive the highest priority and attention for application of the HTGR.

Although unanimous agreement did not exist for international focus on a specific single heat utilization process, a majority consensus indicated that the future high temperature application of nuclear energy should be directed to the development of hydrogen production techniques including steam and CO_2 reforming of methane. The need to move away from the burning of fossil fuels and increase the use of hydrogen is felt by many to be a worldwide requirement for the future. The lower temperature application of desalination was also selected. Even though desalination via coupling to a LWR is now commercially available, it was generally felt that further development is necessary in the areas of safety, regulation and the synergism of treating as a single plant the coupling of different desalination processes with alternate nuclear heat sources. Other heat utilization processes such as heavy oil recovery, district heating and coal gasification and liquefaction should also receive international consideration [7-1].

It was a major determination by the Chief Scientific Investigators (CSIs) in the IAEA CRP on Design and Evaluation of Heat Utilization Systems for the HTTR that steam and CO_2 reforming of methane to produce hydrogen be the principle focus of study and testing for process heat applications with this plant.

7.2.1. Heat utilization applications with the HTTR

Production of hydrogen as an energy carrier for the future through the reforming of methane was selected as the highest priority heat utilization candidate. Reforming of methane with steam and carbon dioxide were investigated and, although the primary goal was the production of hydrogen, both processes have the proven ability to result in the final production of methanol (or syngas) through subsequent synthesis. Although the HTGR is currently not economically competitive with conventional plants due to existing low fossil fuel costs, this chemical conversion of natural gas with the HTGR offers the added benefits of a substantial decrease in CO_2 emissions and an increase in calorific value of the products with a greater fuel versatility.

Evaluation of the remaining high temperature heat utilization processes chosen for investigation by the CSIs resulted in the prioritized selection of hydrogen production through thermal-chemical water splitting, followed by the conversion of coal into higher quality fuels.

7.2.1.1. Steam reforming of methane

Steam reforming of methane for the production of hydrogen will be the initial heat utilization process demonstrated with the HTTR. This reforming process for hydrogen production is well known industrially and is technologically mature. The hydrogen production performance with a heat utilization ratio (i.e. ratio of the product energy to total input energy) of up to 78% in the reforming system is expected to be demonstrated in 2008 with the HTTR at a thermal power level of 10 MW. The integrated control system of the HTTR with the steam reforming system is determined to be technically feasible and will also be demonstrated in the HTTR (Figure 7.4).

Significant experience in out-of-pile tests and design studies associated with steam reforming of methane exist in Germany, China and the Russian Federation. The R&D activities to be conducted prior to demonstration of this process in the HTTR include out-of-pile testing, additional design studies and safety including the establishment of safety standards associated with the explosion of feed and product gases and determination of the tritium permeation rate.

7.2.1.2. CO_2 reforming of methane for hydrogen

Large resources consisting of a mixture of CO_2 and natural gas exist world-wide which have the capability to be converted into usable synthesis gas. Also, bio-resources (CO_2/CH_4) can be used for conversion into synthesis gas with no net generation of CO_2. Although not as highly developed as the steam reforming process, CO_2 reforming of methane has been proven experimentally.

As with steam reforming of methane, the R&D needs, design, and safety assessment requirements such as the tritium permeation rate and explosion of feed and product gasses and the goals of the carbon dioxide reforming process have many similarities. Therefore, the same facility can be used for the demonstration of both reforming systems. In this regard, the initial design work has been completed by JAERI for the HTTR heat application systems of steam and CO_2 reforming of methane and out-of-pile demonstrations of both processes will be performed prior to coupling to the HTTR.

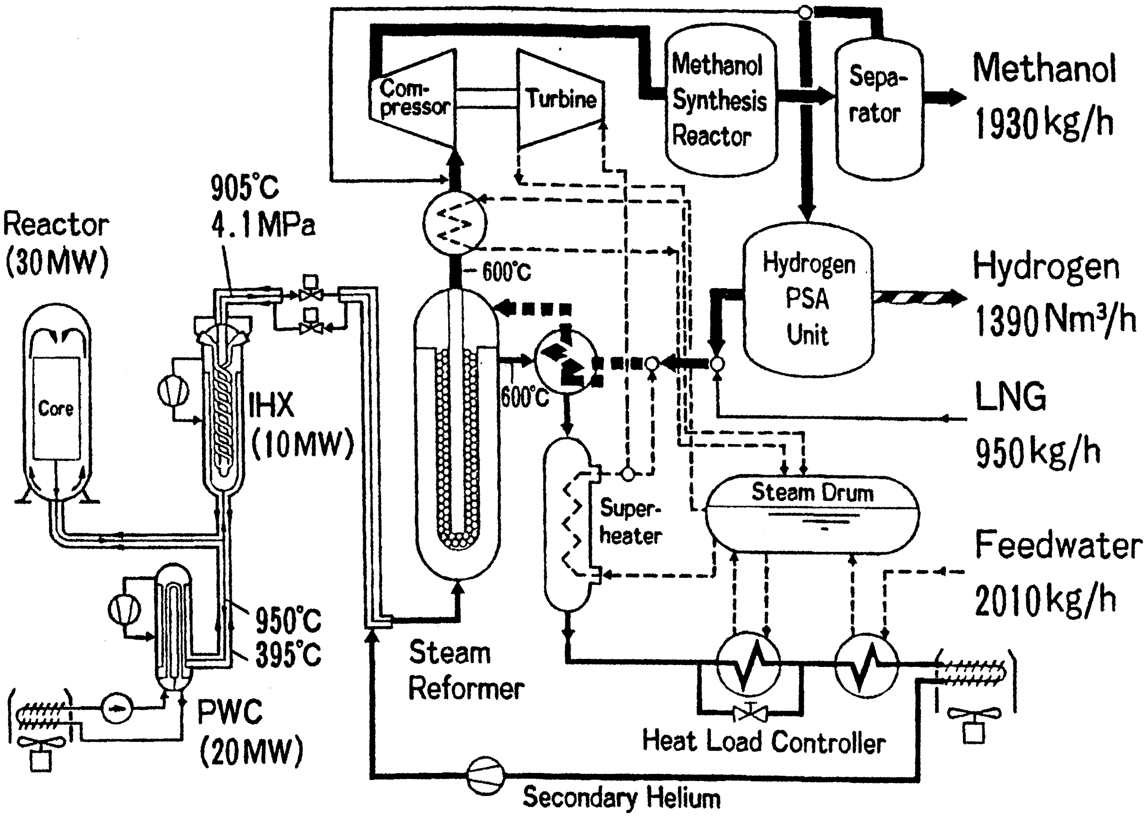

FIG. 7.4. HTTR for steam reforming with hydrogen/methanol co-production [7-6].

7.2.1.3. Thermochemical water splitting for hydrogen production

Of the many chemical reactions that have been evaluated utilizing the HTGR as the heat source, the iodine-sulfur (IS) process is considered one of the most attractive for thermal chemical water splitting to achieve hydrogen. This process has several excellent features, including the capability to produce hydrogen from naturally abundant water, freedom from carbon emission thereby helping prevent environmental issues such as global warming, direct conversion of nuclear heat into chemical energy, and it also exhibits a relatively (>40% has been evaluated) high thermal efficiency.

The basic concept of this process was developed by GA and demonstrated at JAERI in the course of the CRP on HTTR heat utilization systems in laboratory scale experiments attaining continuous and "closed-cycle" hydrogen production. Because of this achievement, a larger scale test was initiated in 1999 to develop closed cycle operation techniques with modified Bunsen reaction conditions. In addition to this test, studies are underway on the membrane technologies for establishing efficient processing of hydrogen-iodide and on the materials of construction for a bench scale plant [7-7]. See Chapter 5 for details associated with the HTTR.

7.2.2 Process heat applications with the HTR-10

A long-term test program for nuclear process heat applications utilizing the HTR-10 and the research facilities at INET is proposed to be accomplished as follows:

— A feasibility study,

— Simulation experiments on a laboratory scale for coal gasification by means of hot helium to acquire technology, experience, and to optimize operational parameters,

— Nuclear coal gasification test in a pilot facility where the HTR-10 is to be operated at 950°C and to demonstrate IHX operation (see Figure 7.5) [7-8].

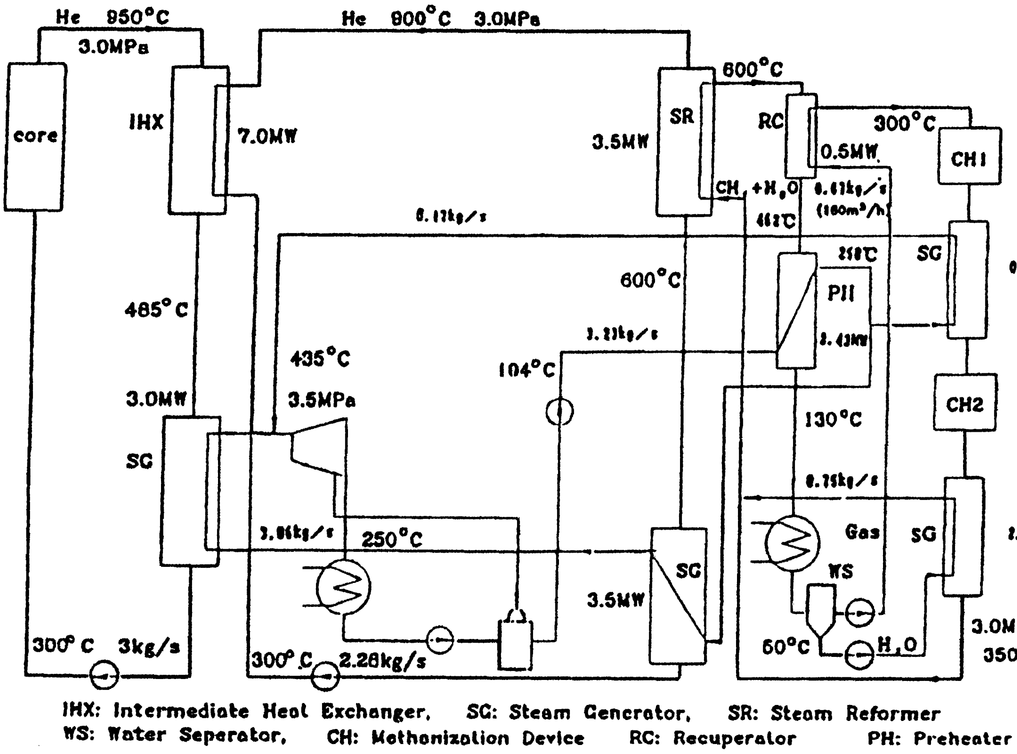

FIG. 7.5. HTR-10 circuit with process heat applications [7-9].

7.2.3. Additional HTGR process heat development activities

7.2.3.1. FZJ, Germany

The application of the HTGR to provide heat for industrial applications has long been a significant area of study in Germany at the Research Center Juelich (now FZJ), and considerable expertise resides at FZJ. The more recent projects at FZJ included:

— The prototype plant nuclear process heat project, from 1972–1992 with the goals of developing HTGRs for high gas outlet temperatures of 950°C as a source of process heat for coal gasification, developing and testing components for heat transfer to the process plant, developing and testing processes and experimental facilities for steam-coal gasification and hydro-gasification.

— The steam reforming process was experimentally investigated in the "nuclear long-distance energy" project. Two different bundles of steam reformer tubes were tested: (i) the "orifice baffle" design by Lurgi, and (ii) the "counter current concentric" design by Steinmueller. Convective helium heating was verified for both designs. The reaction rate for bundle (ii) was found to be dependent on pressure and fictive velocity in the empty tube, but independent of temperature indicating a heat-controlled reaction. Post operation inspections confirmed the integrity of all reforrner tubes. The complete EVA-H/ADAM-H system was operated for a total of 10 150 hours including steady state conditions at both full and partial load, and also under transient conditions, and special test situations such as tube blockage [7-10].

— The 10 MW component experimental loop, Komponenten-Versuchs-Kreislauf, (KVK) was a high temperature helium circuit operated by Siemens/Interatom in Bergisch Gladbach. It consisted of a primary and a secondary helium loop and a water steam system. Its flexible design, however, also allowed for single loop operation. Various components were tested under the operating conditions of coal gasification, including two concepts of a He-He intermediate heat exchanger for a heat rating of 125–170 MW were selected. For both, a 10 MW test plant was operated in the KVK loop verifying the operation of reformers with convective helium. A 10 MW decay heat removal system cooler, hot gas ducts including insulation and liner, hot gas valves, and a steam generator were other components of the KVK loop. Furthermore, a helium purification system was operated in a bypass mode of the main system. Starting in 1982, the KVK facility was operated for 18 400 h with ~7000 h above 900°C [7-8, 7-10]. The hot gas duct with internal insulation was operated at temperatures up to 950°C.

7.2.3.2. Exploitation of the Natuna gas field in Indonesia by heat from the HTGR

The off-shore Natuna natural gas field in Indonesia is estimated to have an energy content of 1.5 TWy which would allow a production of 38 million t/a of liquid natural gas (LNG) over 20 years. As indicated in Chapter 6, the Natuna gas features a very large content of $C0_2$ (~71 %). A feasibility study was initiated by the IAEA to investigate the application of the HTGR for co-generation purposes Of $C0_2$ conversion, desalination, and hydrogen production in terms of economy and technology. The prerequisite of minimal emission of $C0_2$ into the atmosphere has suggested the nuclear option in exploiting the Natuna gas field. Six alternatives were identified as being feasible and economically competitive, distinguished by the desired product methanol or methane (or hydrogen) and by the consumption of CO_2 [7-8, 7-11].

7.2.3.3. The VGM in the Russian Federation

The conceptual design of a 200 MW modular HTGR, the VGM, has been developed and is intended to be used for process heat applications in various industrial branches including oil refineries, petrochemical plants, and for the production of hydrogen, methanol, and synthetic fuels. To assure a high level of safety, the system is designed to have a helium-to-helium IHX and a helium-to-silicon oil process heat exchanger. A single-cycle variant, VGMP, with 215 MW•t is completely orientated towards production of process heat. A design including three units has been made for connection to a standard oil-refinery plant [7-8, 7-12].

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Chapter 8

HTGR SPENT FUEL DISPOSAL AND DECOMMISSIONING

In the early years of nuclear power development, little emphasis was placed on the issues of spent fuel disposal and decommissioning. However, over the past two decades, the importance of these areas has become increasingly dominant in the consideration of building new nuclear reactors and in the operation of existing plants.

Today, no industry, national agency or research institute would consider the installation of a NPP or a research reactor without extensive evaluation of the economic, environmental and political aspects of spent fuel disposal and the plant's eventual decommissioning. In some Member States with previous strong commitments to nuclear power (such as the USA), spent fuel disposal and decommissioning have become so dominant in the consideration of a new plant that they have become major reasons for discontinuing the development of this energy source. This chapter provides a review of the spent fuel disposal and decommissioning issues relative to the HTGR.

8.1 SPENT FUEL DISPOSAL

8.1.1. The PBMR [8-1]

Provision has been made at the PBMR to store fuel for the total life of the plant. The spent fuel storage facility consists of thirteen tanks with an approximate usable volume of 78 m^3 each. The spent fuel tanks are filled with fuel spheres via the fuel handling system. Due to the "multi-pass" fueling scheme of the PBMR, the fuel handling system is in continuous operation whenever the plant is running. This results in fuel spheres filling the spent fuel storage system on a one-by-one continuous basis either when the sphere has been determined to have reached its burn-up limit, or if the fuel sphere is judged defective (broken). Table 8-1 and Fig. 8.1 provide the defueling design characteristics and a schematic of the fuel handling/spent fuel storage scheme during normal operation.

TABLE 8-1. PBMR DEFUELING DESIGN DATA

Defueling	
Number of Spheres Handled per EFPD	4000 to 4880
Number of Spheres Handled by system during Defueling	10 000/hr.
Operating Pressure	<7.0 MPa
Anticipated Damaged/Failed Fuel Spheres at Discharge Tube/Year	~180

The fuel handling and storage system performs the following functions for filling of the spent fuel storage tanks:

- Determine if the fuel sphere has reached the level of burn-up to be deposited in storage as spent fuel
- Determine if the fuel sphere is defective and deposit it to spent fuel storage, as necessary

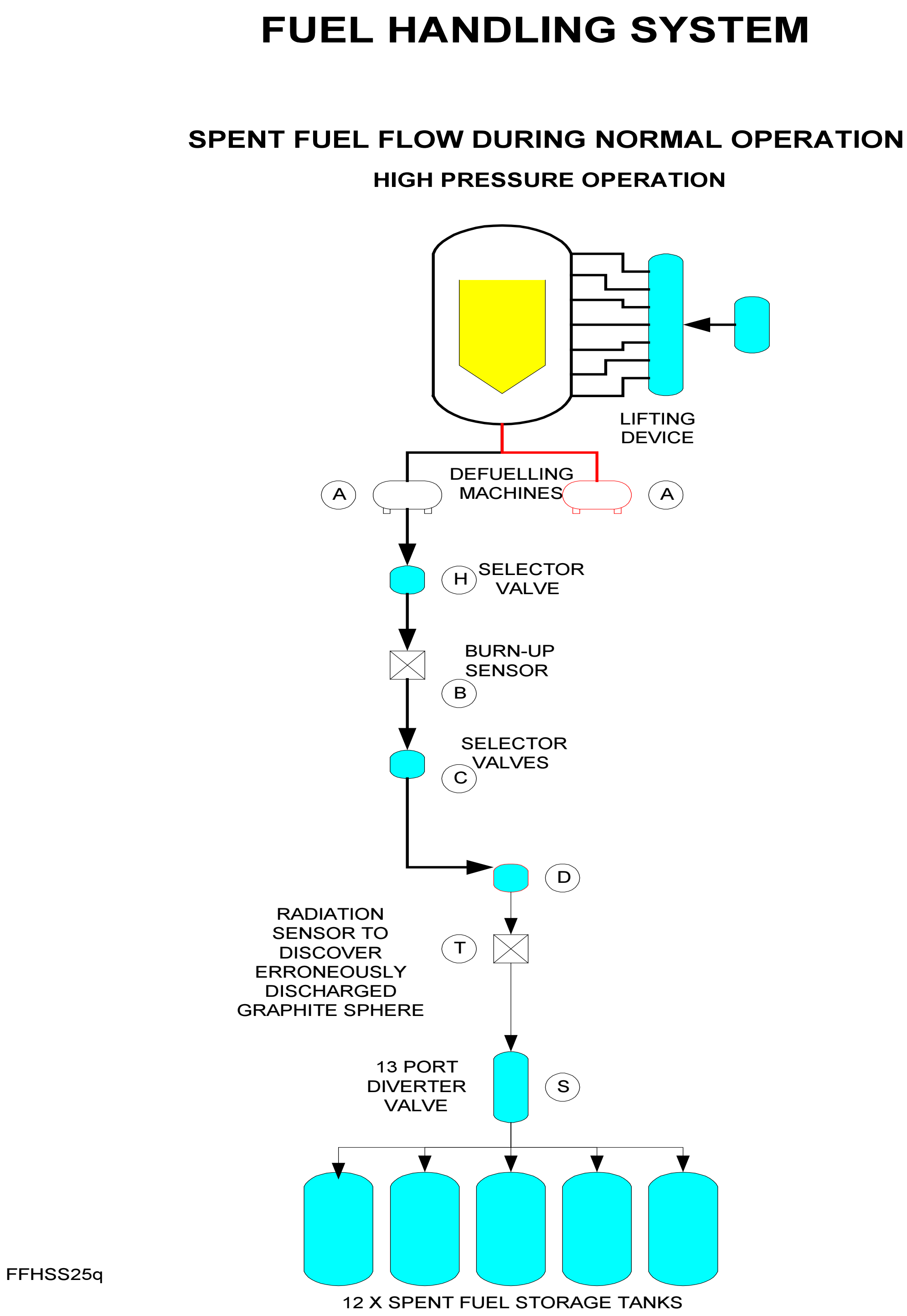

FIG. 8.1. Spent fuel flow during normal operation.

— Unloading of spent fuel from spent fuel storage tanks and defueling storage tanks to transport or intermediate casks.

After burn-up measurement of the spent fuel spheres is completed, they are collected in a buffer line provided for this purpose and then conveyed to a discharge lock. The fuel spheres are released one by one from this discharge lock and forwarded pneumatically by helium to the spent fuel tanks.

The loading procedure of the spent fuel tank is presently being evaluated and will be based on the method that is most advantageous in removing heat from the spent fuel tanks. There are two options being considered:

— In option 1, the fill level of the spent fuel tank is monitored by counters to supply information on the accumulated number of spheres placed into the tank over the lifetime of the plant. This is achieved by closing the valve to the full tank and opening the valve to the next empty spent fuel tank.

— Option 2, is to load more than one tank at a time. This method would reduce the heat loading per fuel tank and storage cell.

In case of a defective storage tank, it will be isolated and no longer used. Provision is made to transfer the contents of a defective tank system to another tank system or to an intermediate/permanent storage facility. Cooling of the tanks will be performed with a redundant passive system. The capability to transfer spent fuel to another storage site for intermediate or final storage has also been provided.

A spent fuel cooling system is provided to remove the time dependent decay heat of the PBMR spent fuel. The main purpose of this system is to protect the building concrete, and the system does not perform any nuclear safety functions. The heat to be removed from each spent fuel tank will depend on the filling scheme that is to be adopted. It is estimated that the total cumulative decay heat to be removed from all the storage tanks after thirty years of full power operating (no interruption assumed) is approximately 145 kW. It will take approximately three full power years to fill one tank (maximum discharge rate of ~370 fuel spheres per day), if series filling is assumed (see Table 8-2 for the spent fuel storage cooling system technical characteristics.

TABLE 8-2. SPENT FUEL COOLING TECHNICAL DATA

Heat load per storage tank	80 kW
Heat load at end of 30 full power years	150 kW
Heat flux density	1 kW/m^2
Panel wall temperature	70ºC
Storage vessel maximum wall temperature	167ºC
Heat transported - natural convection	21.1 kW
Radiation	60.5 kW
Total	81.6 kW
Mass flow of water	0.55 kg/s (natural convection)
Mass flow of air	~1.8 kg/s
Airflow in chimney	~1.73 m/s
Water-to-air heat exchanger capacity	150 kW each.

The spent fuel cooling system will be a low pressure, closed loop, natural convection driven, self-acting water based system with an external water-to-air heat exchanger. This system consists of:

- Stainless steel tubes with steel strips in between, installed vertically and joined by welding to form a closed tube wall around the storage tank,
- Two inlet headers that are located at the top,
- Two outlet headers that are located at the bottom.

A two-train configuration is achieved by joining every second tube in the wall to the same inlet and outlet headers. This enables heat removal from the entire circumference in the event of failure of one train. Each train can remove 100% of the heat load. The cooling panels are fixed only at the top to allow for free axial thermal expansion. The space between the cooling panels and the concrete walls, which is 350 mm, is closed at the top and bottom to prevent the circulation of hot air streams. Two 100% water-to-air heat exchangers eject the heat to the atmosphere. Airflow through these water-to-air heat exchangers is by natural convection and is enhanced by a chimney.

8.1.2. The GT-MHR

The GT-MHR employs the hexagonal fuel element, which is principally made of graphite with TRISO coated fuel particles in rods spaced within the block. Whole fuel element disposal is the preferred option for the GT-MHR spent fuel because of advantages related to ease of implementation, proliferation risks, safeguards requirements, cost and schedule. This recommendation corresponds with previous studies conducted by Oak Ridge National Laboratory and in Germany, which concluded that whole elements of HTGR spent fuel containing uranium and thorium coated fuel particles should perform better than un-reprocessed LWR spent fuel. Also, whole element disposal of spent GT-MHR fuel is directly analogous to options being developed for disposal of commercial LWR spent fuel in unprocessed, whole assembly form.

Detailed evaluations of whole elements were performed, including an assessment of the technical criteria for use of a multipurpose canister as used for LWR spent fuel storage to meet requirements for temporary dry on-site storage, transportation to the repository, and final disposal within the repository. It was concluded that spent fuel elements discharged from the GT-MHR are ideal waste forms for permanent disposal in a geologic repository. The graphite fuel elements and the ceramic coatings on the fuel particles are as-manufactured engineered barriers that provide excellent near-field containment of radionuclides and minimize reliance on the waste package and surrounding geologic media for long-term containment [8-2].

Due to the high level of plutonium destruction and degradation achieved by the plutonium burning GT-MHR, the plutonium in discharged fuel elements would be practically unusable for nuclear weapons. Dilution of plutonium within the relatively large volume of PC-MHR fuel elements provides additional resistance to diversion throughout the fuel cycle. This is accomplished without adversely impacting repository land requirements, since repository loading is determined by decay heat load and not by physical volume. It is estimated that the spent fuel from the GT-MHR would require about one-half of the repository land area needed for commercial LWR spent fuel, assuming the same quantity of electrical energy had been generated by the fuel discharged from each reactor type. Overall disposal costs for PC-MHR spent fuel is evaluated to be comparable to those for commercial LWR spent fuel, with the added benefit of a superior permanent waste form [8-11].

The following main stages of spent fuel management are envisioned for the GT-MHR:

(1) 1st stage; spent fuel assemblies (FAs) are removed from the reactor and placed in an interim storage located below the reactor maintenance building operating floor (see Fig. 4.2, item 2), where they are stored for about a year. In this storage, spent FAs are under helium environment conditions within leak-tight metallic multipurpose casks (MC) cooled externally by circulating water.
(2) 2nd stage; spent fuel in the MC is removed from the interim storage and placed into transport casks (TC) or concrete casks and relocated beyond the maintenance building onto a special area for long-term storage (see Fig. 4.2, item 15) within the NPP site boundaries. Spent fuel may be stored at this area up to 60 years (to be refined on the further design stage) and does not require forced cooling.
(3) 3rd stage; spent fuel in TCs are transported to a deep underground storage facility (in a geological formation) for final disposal.

The main design issues for the safe storage of this spent fuel are as follows:

— Residual heat removal,
— Designing and implementing provisions of radiological safety (protection against γ radiation and spontaneous neutrons);
— Providing for nuclear safety (meeting the requirements for sub-criticality);
— Designing and implementing provisions for fire safety (elimination of matrix graphite burning);
— Providing a confinement for radioactive and toxic aerosols.

The last two problems are practically removed on the initial stage of spent fuel storage by placing spent FAs in leak-tight MCs filled with noble gas, i.e. helium.

TABLE 8-3. RESIDUAL POWER REDUCTION WITH TIME

Time	Residual power, W/FA	Res. power in MC, kW/MC
24 h	2925	81.9
100 d	515	14.42
365 d	185	5.2
1000 d	61	1.7
5 yr.	28	0.8
10 yr.	18	0.5
35 yr.	15	0.45
100 yr.	11	0,3
350 yr.	7	0.2
1000 yr.	3	0.08
10000 yr.	0.3	0.01

Removal of residual heat (Table 8-3) on each of the indicated stages is performed by different methods. On the first stage, MCs are cooled in a forced manner by externally circulated water. During long-term storage with fuel in a special storage container or in concrete casks, as well as during spent fuel transportation in TCs, the FAs are air cooled by natural convection.

Radiological safety is ensured during long-term storage of the spent fuel by either the storage compartments or appropriate structure of the casks taking into account the thickness of the MC steel walls (about 20 mm).

The sub-criticality requirement ($k_{eff} < 0.95$) is met for FAs located in MCs and during all stages of their storage through selection of appropriate distance between the MCs, provided that concrete screens exist in respective compartments.

A multipurpose cask is intended for storing, shipping and final disposal of FAs with spent fuel. The MC provides integrity for FAs, fuel compacts and coated particles as well as reliable isolation of the environment from fission products, fissionable materials, γ radiation and very toxic substances in all stages of spent fuel storage or transportation. Robustness and leak-tightness of the MCs is also an additional protective barrier during final disposal of the spent fuel.

Figure 8.2 provides a schematic of the MC being considered for the G-MHR. The MC contains 28 FAs and provides for the application of the TU-6 transport cask that is used in Russia for shipping spent fuel from WWER-440 reactors. The MC consists of a load-bearing cylindrical vessel within which cells (canisters) are positioned for disposal of 28 (7 × 4) prismatic FAs.

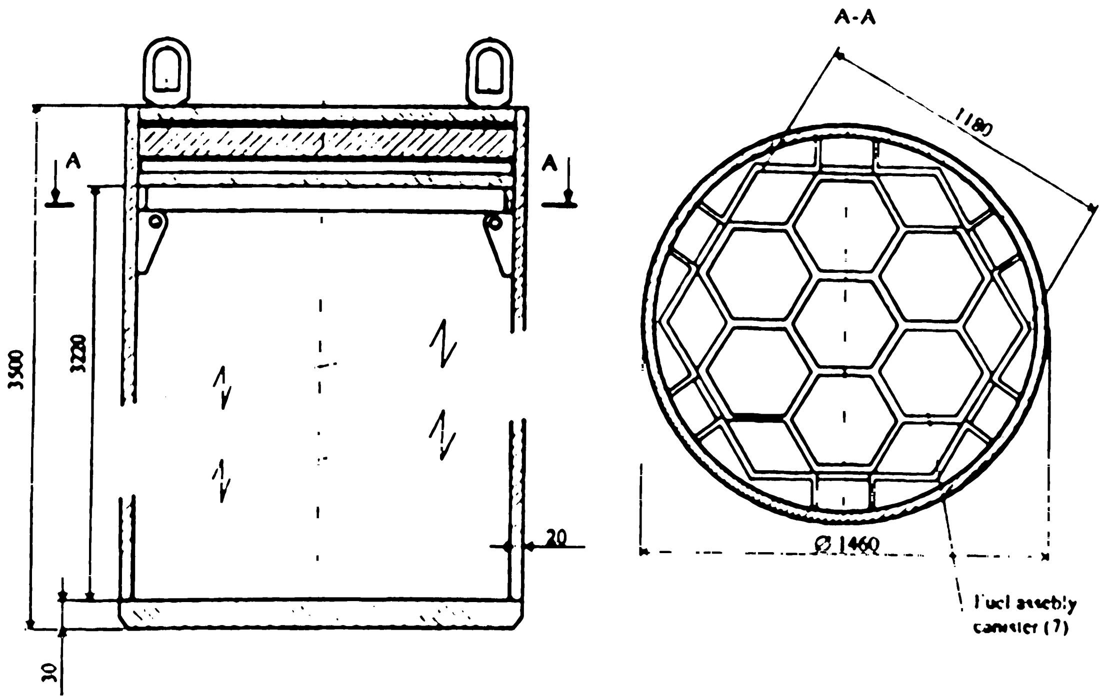

FIG. 8.2. Multipurpose cask.

The MC is closed from above by a shielding plug with two covers welded to it. There is a ring bolt in an upper cover for remote handling. An inner cover is provided by an air vent through which air is evacuated from the MC's internal plenum and helium is supplied with pressure of about 0.01 MPa following completion of the cover weld. Twelve MCs are needed to provide for one annular reloading of spent fuel (340 FAs). The annular volume of unloaded FAs with spent fuel is 30 m^3 , and the cost of a MC is estimated at ~$50 000 US. A less expensive concrete cask may be more appropriate for long term storage at the NPP site. Detailed design of the casks will be performed in the subsequent design stage following selection of the most economical method for spent fuel storage.

At the present, there is no proven technology in Russia concerning the final disposal of long-lived radioactive wastes. The Federal Program on management of radioactive wastes and spent nuclear materials, their utilization and disposal on the period 1996–2005 envisions the following work:

— Development of technology and equipment for containerizing spent nuclear fuel to be subjected to long-term storage;
— Construction of the first sites for high-level radwaste disposal in a number of Russia's regions.

In world practice (USA, Canada, France, etc.), underground disposal of radwastes in stable geological formations is being widely used. The concept of high-level radwaste disposal in deep underground storage is based on the combination of natural (geological) protective barriers and artificial (engineered) barriers, that minimizes a likelihood of radionuclides migration to the biosphere.

The GT-MHR fuel in the form of TRISO microparticles possesses a significant advantage in respect to the long-term retaining of radionuclides without affecting the spent fuel characteristics or geological environs. Quantitative assessments show that TRISO particles are capable of preserving their integrity for up to 10^6 years. Results of experimental investigation performed earlier by ORNL showed that corrosion damage of the protective coating on fuel particles made of PyC, SiC and nuclear-grade graphite is very low (as compared to the vitrified radwaste form), and they represent ideal components capable of serving as reliable engineered barriers for radwaste storage systems. The coating retention is especially good for nuclides such as cesium which is relatively simple to leach from vitrified forms.

With regard to plutonium release which is limited by the requirement for a permissible fraction of damaged policies (5.5×10^{-5}) it should be noted that this fraction in discharged fuel does not exceed 10^{-5} even after operation under accidental conditions. Conservative assessments made for the nuclide, C-14, (release and transport by ground water) showed that activity which is defined by this nuclide does not exceed ~1.5 Ci/m^3. This is more than 5 times less than the allowable limit value set by the US Environment Protection Agency [8-3].

8.1.3. Fort St. Vrain (FSV)

The FSV nuclear power station, owned and operated by the US utility, Public Service Company of Colorado (PSCo), was in commercial operation for over a decade, generating ~5.5 billion kW•h. This plant featured a HTGR coupled to a steam cycle electrical plant with an output of 330 MWe. The plant was permanently shut down in August, 1989, and subsequently decommissioned under the “early dismantlement” programme.

The first step in the decommissioning process required disposition of the spent nuclear fuel. This plant was originally built under the US “nuclear power demonstration programme”, and the spent fuel was, by contract, to be sent to a depository on the Idaho Nuclear Engineering Laboratory site. At the time of defueling, the access to this site was restricted due to political intervention by the Governor of the State of Idaho and the fuel could not be shipped. This situation left PSCo with no choice but to place the plant's fuel in an independent spent fuel storage installation (ISFSI). This passively cooled, stand-alone facility was licensed by the NRC per 10CFR Part 72 independent from the power reactor license (Fig. 8.3).

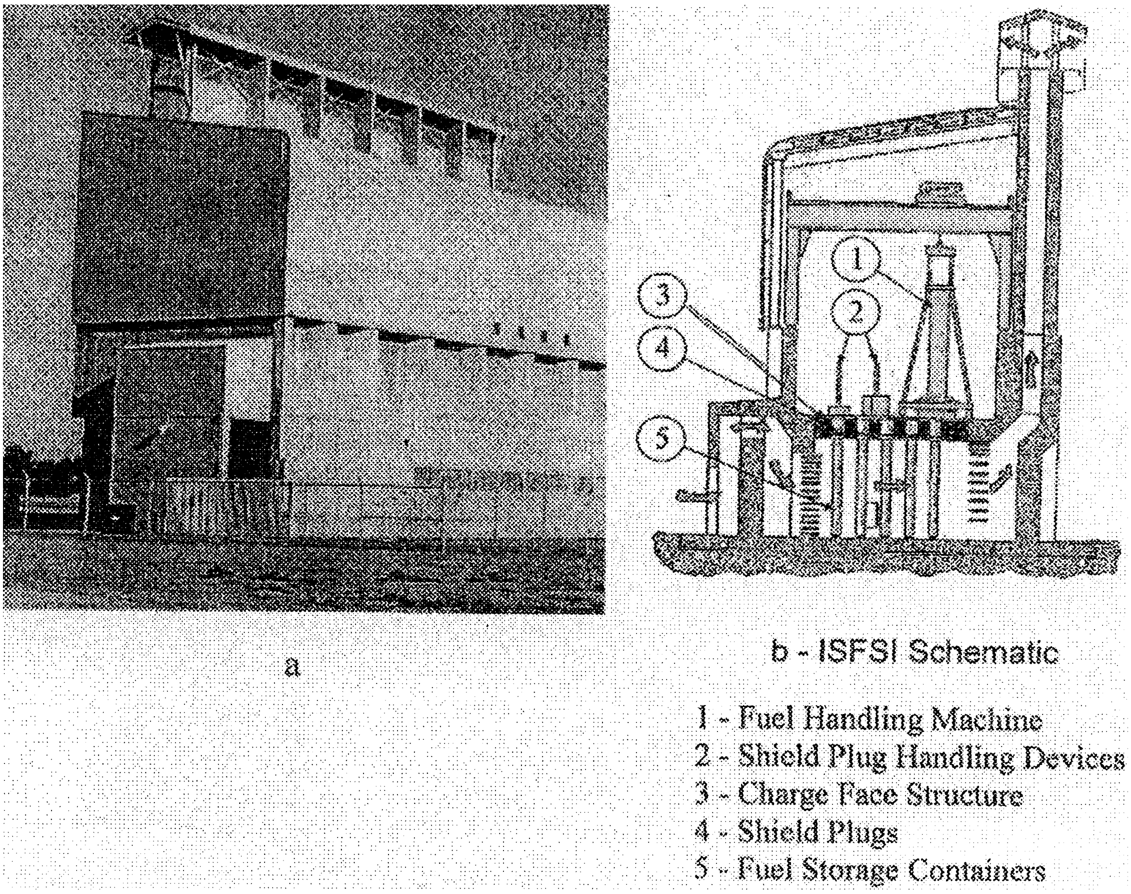

FIG. 8.3. Fort St. Vrain’s independent spent fuel storage facility.

The ISFSI was designed by GEC Alsthom Engineering Systems Ltd. The hexagonal graphite fuel elements (31" tall and 14" across the flats) are stored vertically in steel canisters with six fuel elements per canister. Each of the six vaults in the modular dry vault storage system contains 45 storage locations. Each storage location is closed by a removable shield plug allowing for easy access to load and eventually unload the ISFSI. The modular dry vault storage system is cooled by natural circulation. Cool air is drawn in from the outside, passes through each vault, is warmed and rises through the chimney structure for discharge into the environment. Since the air is only in contact with the outside of the storage containers, and not with the fuel, it remains free of any contamination. This simple design assisted PSCo in defueling the reactor to the ISFSI over a six month time frame which was approximately 10 weeks ahead of schedule [8-4].

8.1.4. The THTR 300 [8-5]

The THTR 300 prototype nuclear power plant in Hamm (Westphalia, Germany) featured a graphite moderated and helium cooled HTGR. This plant was shutdown in 1988 after an operation time equivalent to 423 days of full-load operation. Approximately one year later, the decision on decommissioning was taken by the federal and state authorities and the shareholders of HKG, the plant operator. Following granting of the license to proceed on October 22, 1993 and associated preparatory work, unloading of the THTR pebble bed reactor core was initiated on December 7, 1993.

The reactor core of the THTR 300 consists of a loose bed of spherical elements. At the beginning of the unloading operation, the core contained ~563 000 fuel elements, 76 000 graphite elements and 31 000 absorber elements. These so-called operating elements are spherical elements with a diameter of 60 mm and consist exclusively or in the main, of graphite. Unirradiated fuel elements of the THTR contain ~ one gram of highly enriched uranium (93% U-235) and ~ ten grams of thorium; the absorber elements and graphite elements do not contain fuel.

Figure 8.4 is a diagram the sphere charging system. During operation of the plant (September 1985 to September 1988), it was used for continuous charging of the reactor with fuel elements. During this period, the fuel elements were recirculated several times and damaged elements sorted out by the damaged sphere separator.

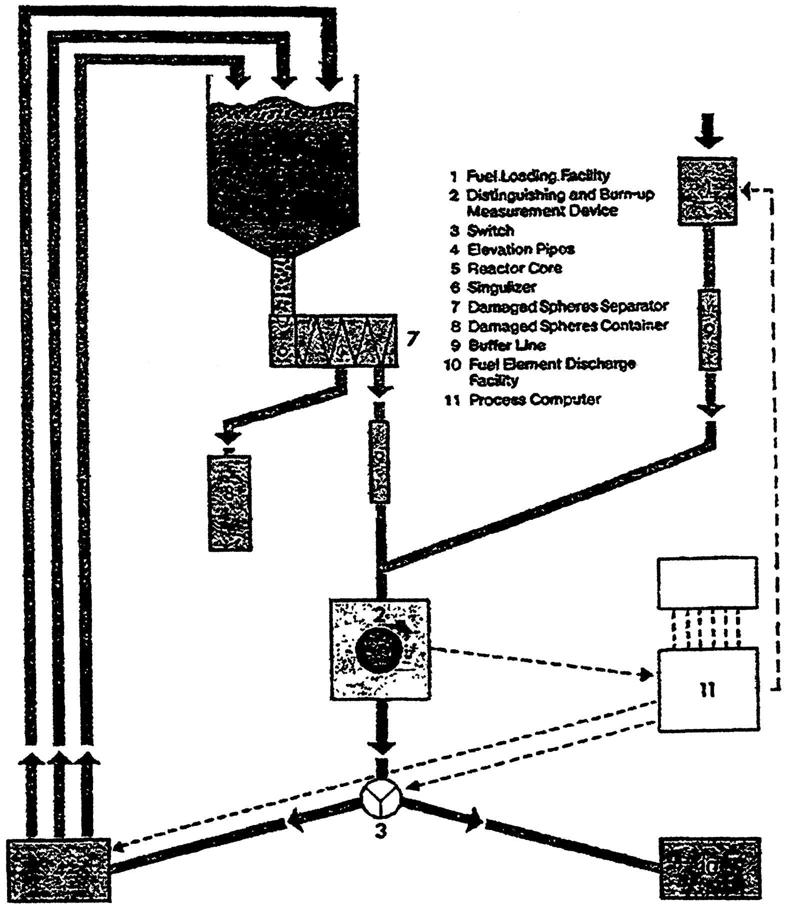

FIG. 8.4. THTR fuel sphere charging system.

Unloading of the core was implemented basically in the same manner as removal of the operating elements during operation. However, process engineering modifications to the charging system were required due to replacement of the primary helium gas with nitrogen and air and reduced temperature and pressure.

During the unloading operation (December 1993–October 1994), the operating elements were sorted by means of a burn-up measuring system (consisting of a graphite moderated reactor with a thermal output of 500 W and an evaluating process control computer) and transferred into operating element containers (steel cans), 2100 elements per container. Balancing of the removed fuel elements was carried out by means of the process control computer of the charging system and, independently from it, by means of the pebble counters of the charging and outward transfer installations. From the beginning, core unloading was organized as a three-shift operation with seven working days per week. Due to the fact that only a few interruptions occurred for short periods for equipment repair, it was possible to remove an average of ~2500 operating elements per day.

Fully inserted absorber rods and the addition of ~4200 unirradiated absorber elements at certain unloading steps ensured subcritical conditions at any moment during unloading of the core, which was confirmed by measured values of neutron flux density.

After an amendment to the license was issued on February 2, 1995, fuel elements that may have been placed into 20 containers during the first year of reactor operation (1985/1986), containing possibly a mix of different types of operating elements, were sorted and placed into the containers with damaged fuel elements.

The residual inventory of fissile material remaining in the reactor pressure vessel after completion of core unloading activities by December 1994 was 0.976 kg (equivalent to 2 198 irradiated fuel elements) and is thus significantly lower than the required value of 2.5 kg.

For outward transfer, the transport and storage cask CASTOR THTR/AVR (see Section 8.1.5 for details of the CASTOR cask) had to be prepared for loading. The cask was opened except for the primary lid and transferred into the loading station. Due to the high dose rate during loading, the shielding gate of the loading station was then closed. By means of a manipulator, the primary lid was removed and the operating element container inserted into the opened transport and storage cask through a ceiling hatch of the loading station. After re-inserting the primary lid into the transport and storage cask and screwing it into place initially by the manipulator, the loading station was opened and after radiological measurements, the cask was transferred from the loading station to the working platform. Here, tightening of the primary lid was completed and the leak-tightness of this first cask barrier was assured by achieving a leak rate $<1 \times 10^{-7}$ mbar/s.

After positioning and bolting of the secondary lid, again the leak-tightness of this second barrier was tested. For transport, the secondary lid was additionally provided with an electronic transport seal. Finally, the protective plate required for interim storage was fitted prior to loading the transport and storage cask onto the transport wagon. For the shipment of the transport and storage casks to an interim storage facility, HKG had four special six-axle railway wagons at its disposal, each capable of transporting three casks. By April 1995, ~620 000 spent fuel elements had been transported in 305 CASTOR casks from the THTR to the interim fuel element storage facility in Ahaus (BZA).

The total-body doses received during cask handling by personnel were monitored with operator owned digital dosimeters as well as with official dosimeters. The evaluation of the official dosimeters did not show any measured values by any employees. Measuring results are listed in Table 8.4.

TABLE 8.4. TOTAL BODY DOSES RECEIVED DURING THTR CASK HANDLING

Year	Number of Casks	Collective Dose [1]			
		Partial-Body Dose [2]		Total-Body Dose [3]	
		Operator's Measurements	Official Measurements	Operator's Measurements	Official Measurements
1992	14	-	-	0,10 mSv	< 0,0 mSv
1993	6	-	-	0,05 mSv	< 0,0 mSv
1994	278	-	38 mSv	0,99 mSv	< 0,0 mSv
1995	7	-	0 mSv	0,01 mSv	< 0,0 mSv

[1] Processing personnel : 10-20 persons
[2] Annual dose limit: 500 mSv/person (dose received during inserting the screws into the screw holes in the primary lid)
[3] Annual dose limit: 50 mSv/person

8.1.5. CASTOR spent fuel storage transfer cask [8-6]

The THTR and the AVR will manage their spent fuel via direct disposal without reprocessing after a period of interim storage. For this interval of interim disposal, the use of dry storage in casks was selected. The development of the CASTOR THTR/AVR cask began in 1982. The design of the cask was based on experience gained in the application of ductile cast iron (GGG 40) for the manufacture of CASTOR transport and storage casks for radioactive materials (Fig. 8.5).

On the basis of the safety analysis and the tests performed, the transport license was issued in 1987 for this cask. The storage license for BZA was issued in 1992 and for the AVR, in 1993.

The CASTOR THTR/AVR cask consists of a thick-walled cylindrical body which is closed with two lids, the primary and secondary lid, as well as a protection plate. For handling, the cask is equipped with two trunnions respectively at the top and bottom ends of the cask. The primary lid is made of forged carbon steel (TstE 355) and is bolted with 28 bolts to the cask body. The primary lid is sealed off with a metal and an elastomer gasket and a penetration is located in this lid to perform leak-tightness tests. The secondary lid is made of carbon steel plate (St 52-3) and is bolted with 28 bolts to the cask body. The secondary lid is sealed with a metal and an elastomer gasket. Two penetrations exist in the second lid, one for leak testing and setting of the monitoring pressure between the lids, and the other penetration is used for connecting the pressure monitoring system.

A carbon steel protection plate serves to protect the lid system from dust, moisture and mechanical influences. It is fastened over the lids with 20 bolts. The cask dimensions are provided in Fig. 8.5.

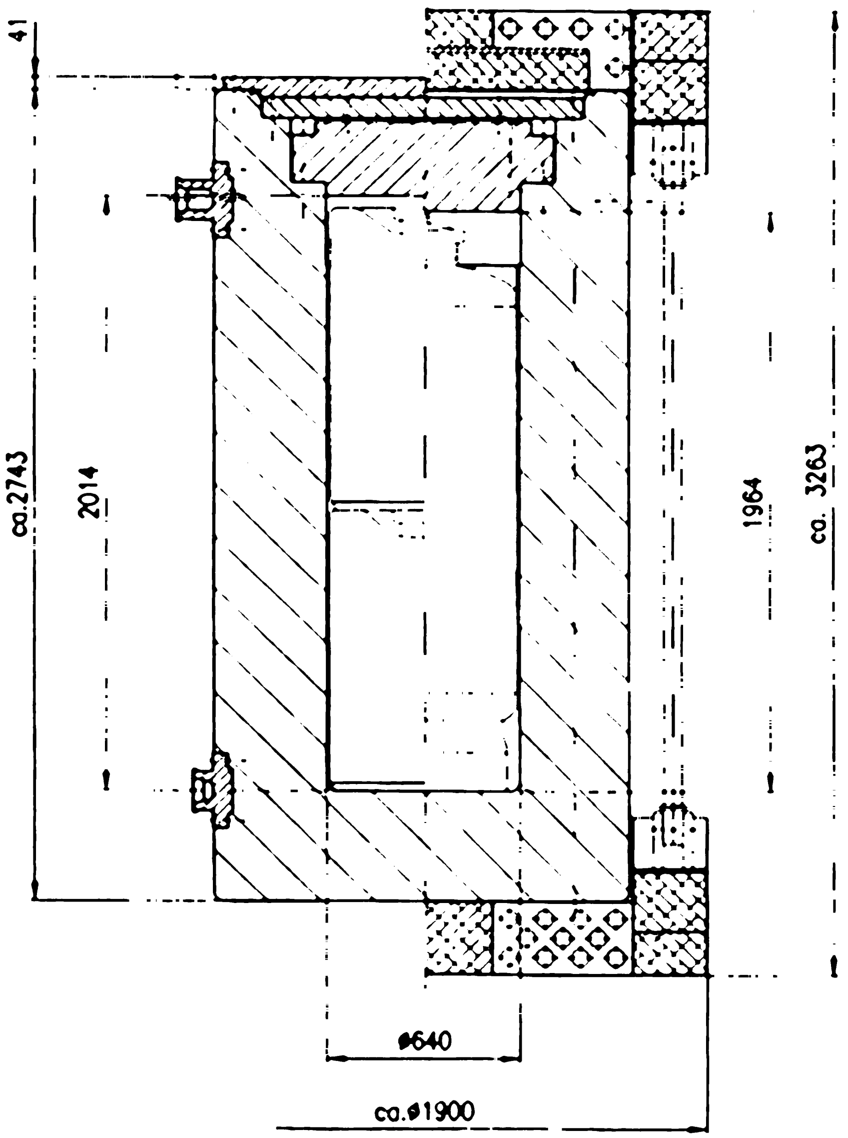

FIG. 8.5. CASTOR THTR/AVR.

Loading of the cask is performed differently between the THTR and the AVR plants. For the THTR 300, the cask is loaded with one fuel canister of ~2100 fuel elements. For the AVR, there are two fuel canisters, which together accommodate ~1900 fuel elements.

Special rail wagons (Fig. 8.6) were developed to transport the casks, both from the manufacturing plant to the THTR and from the THTR to the interim storage facility in Ahaus. Each rail unit can accommodate three casks in special transport frames with shock absorbers.

8.1.6. AVR spent fuel removal and storage [8-7]

The AVR pilot HTGR was shutdown at the end of 1988 for decommissioning and discharge of its core fuel inventory after operation for more than 20 years. Packaging of the fuel for interim storage in appropriate casks and facilities was started in 1994. Approximately 300 000 fuel elements were irradiated during operation of the AVR. The spent fuel from this plant was temporarily stored at different hot cell and pool facilities of the Forschungszenuum Juelich GmbH (FZJ).

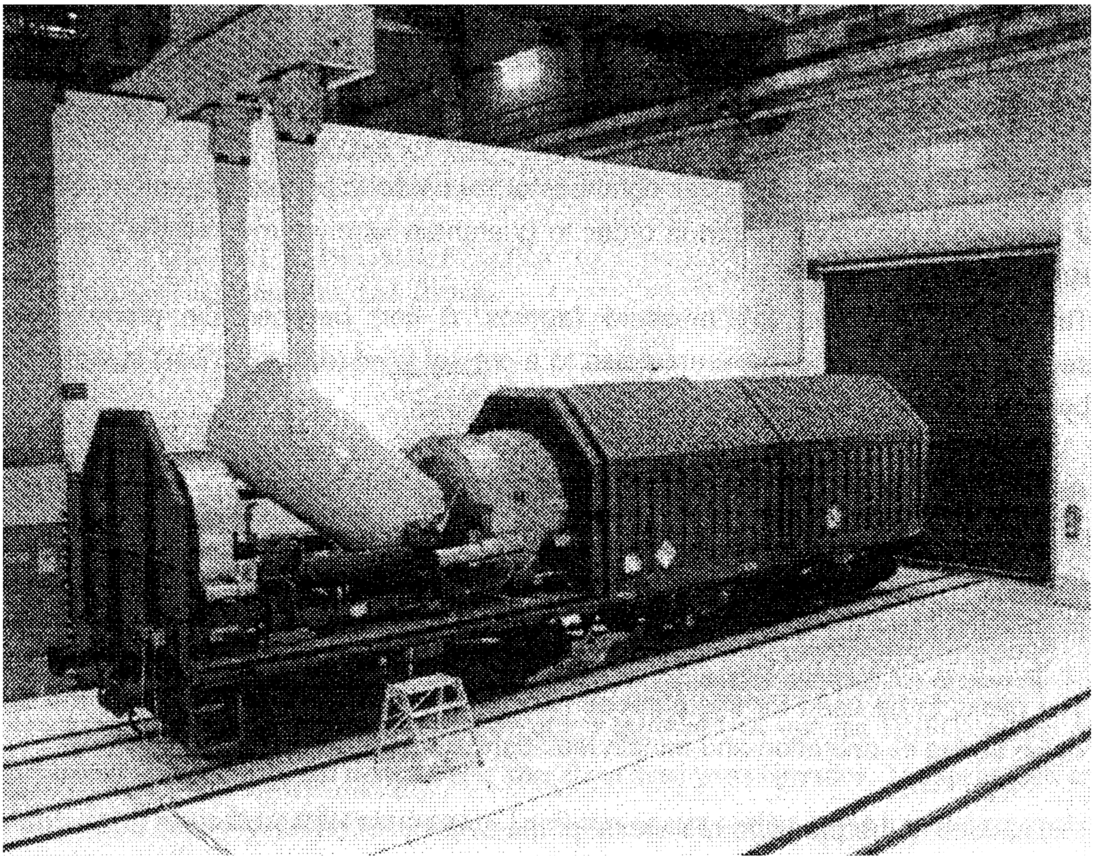

FIG. 8.6. Transport wagon for CASTOR THTR/AVR casks.

During the long operating period of the AVR reactor, extensive R&D was carried out by FZJ to characterize the different types of spent fuel elements for developing interim storage and final disposal concepts. As part of this work, experience has been accumulated by using spent fuel elements for experimental set-ups and by handling, shipping and temporarily storing fuel packages at different facilities of FZJ.

By the end of 1988, about 190 000 spent fuel elements had been discharged during the reactor operating period and were packaged and shipped to FZJ. Work began in August 1993 on discharging the AVR core inventory from the core after receipt of the licenses according to the Atomic Energy Act. This included discharging by using AVR cans (AVR-K), each containing 50 fuel elements, and from different FZJ facilities for fuel reloading from AVR-K cans into dry storage canisters (AVR-TLK) as well as charging of CASTOR casks for interim storage in the AVR interim storage facility (AVR-BL).

At that time, about 84 000 fuel elements packaged and sealed in AVR-K cans were stored in the water cooling facilities of the hot cell (HZ) and the research reactor (FR) departments. About 106 000 HEU fuel elements enclosed in AVR-TLK canisters were stored in the LZ storage cell of the AZ hot cell facilities, which is one portion of the waste treatment and storage building of the decontamination department. Also, ~110 000 fuel elements were still in the AVR reactor core.

From August 1993 up to September 1997, ~91 600 fuel elements were discharged from the reactor core, enclosed and shipped by means of AVR-K cans to the HZ hot cell and reloaded there into AVR-TLK canisters. About 65 AVR-TLK canisters were removed from the LZ, and ~50 000 fuel elements enclosed in AVR-K cans were removed from the above-mentioned water cooling facilities and reloaded into AVR-TLK canisters. In all, through September 1997, 100 CASTOR THTR/AVR casks were prepared and stored in the AVR- BL interim storage.

The AVR-BL interim storage facility was built and licensed according to the Atomic Energy Act for the interim storage of spent AVR fuel elements which are enclosed in CASTOR THTR/AVR shipping and storage casks. Layout of the storage area serves for the interim storage of 154 casks, which are stacked alternately on one and two levels (Fig. 8.7).

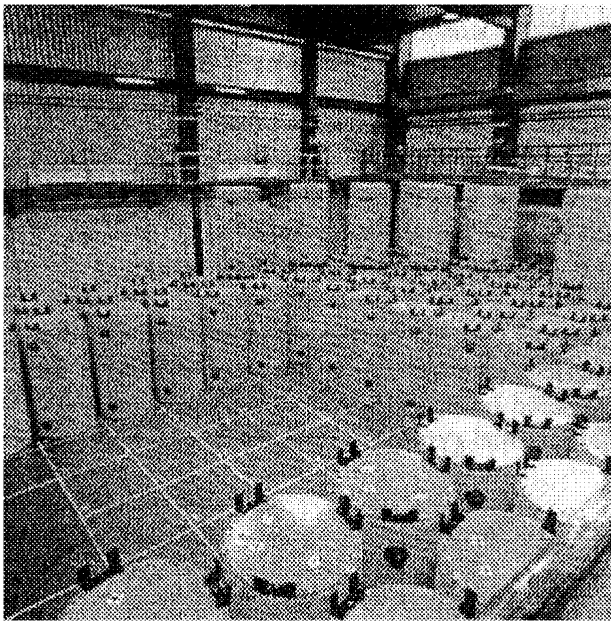

FIG. 8.7. View of the AVR-BL storage hall with CASTOR casks.

8.2. HTGR DECOMMISSIONING

8.2.1. Decommissioning of the PBMR

Although a decommissioning plan is yet to be developed for the PBMR, it is anticipated that the magnitude of this requirement will not be as extensive as the Koeberg Station PWRs. This is due to the relatively small area affected by the PBMR and its corresponding simplified primary coolant system.

A major part of the decommissioning process will be the removal of the spent fuel from the storage tanks. Provision has been made to empty the spent fuel tanks during decommissioning and to transfer the fuel spheres to intermediate or permanent storage tanks when required (see Fig. 8.8). The following equipment performs this operation:

- A fuel sphere extraction machine (movable).
- Two sets of valves in the line to the intermediate/permanent cast storage tanks.
- Connection lines (with valves) to the helium fuel lifting system.

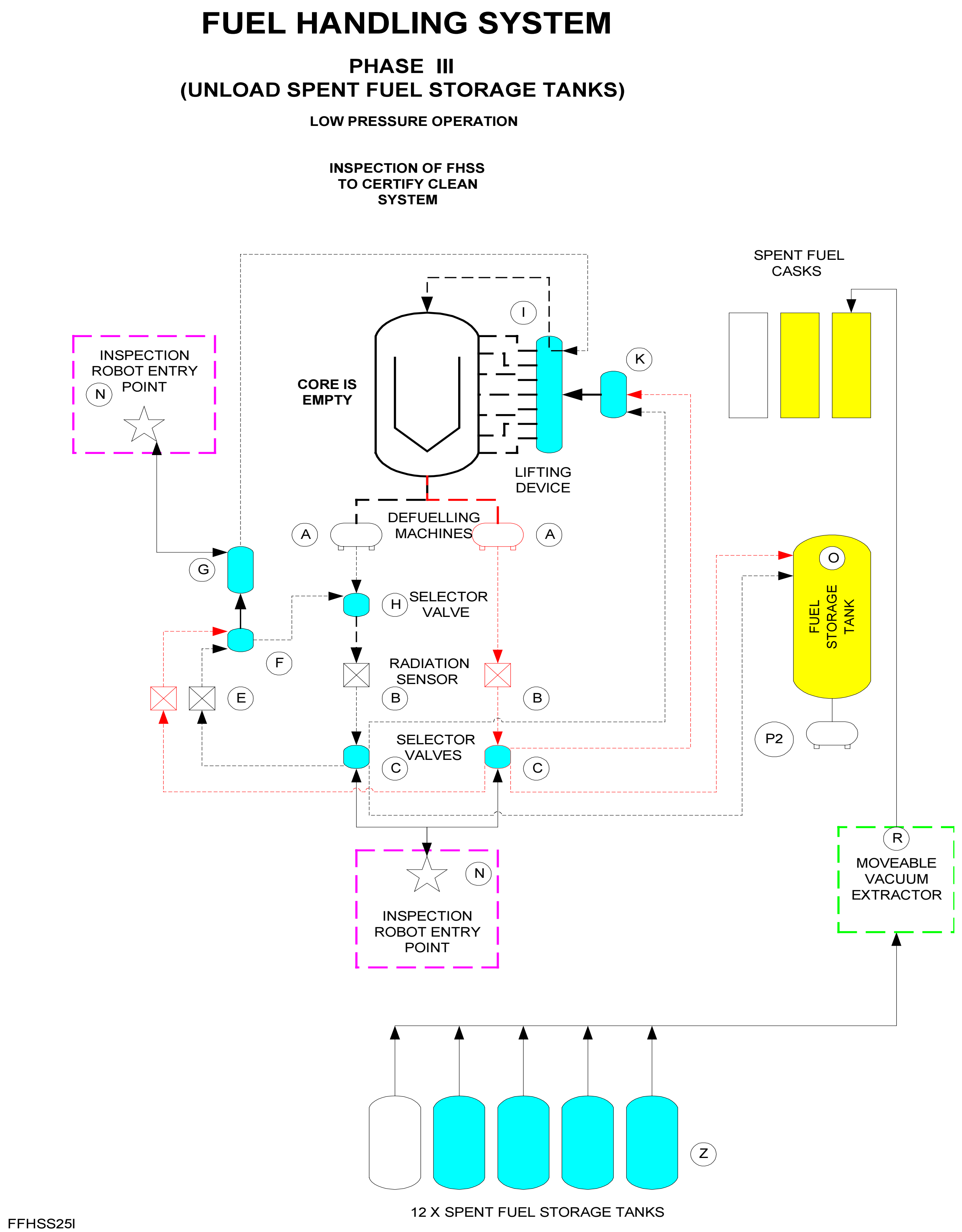

FIG. 8.8. Scheme for unloading of spent fuel storage tanks.

During this operation, the extraction pipe on the inlet side of the extraction machine is inserted into the spent fuel tank through a special valve connection. The buffer tank on the extraction machine is placed under vacuum. When the required conditions on the extraction machine have been reached, the inlet valve to the storage tank on the extraction machine is opened and fuel spheres are sucked into this buffer tank. The number of fuel spheres transferred is counted and when the required number of fuel spheres has been transferred, the inlet valve to the buffer storage tank on the extraction machine is closed, and the buffer tank is placed under helium pressure. The outlet valve on the extraction machine is then opened and the fuel spheres are transferred to the transport casks or storage casks [8-1].

8.2.2. Decommissioning of Fort St. Vrain [8-4]

Public Service Company of Colorado achieved its final decommissioning goal on August 5, 1997, when the US NRC terminated the Part 50 reactor license for the FSV HTGR plant, allowing the plant site to be allowed for "unrestricted use". This was the first successful decommissioning in the USA (and, possibly the world) of a commercial nuclear power plant that had been in operation for many years.

In August 1989, PSCo decided to permanently shutdown the FSV reactor and proceed with its decommissioning. The decision to proceed with the "early dismantlement" process as opposed to "Safstore" of the plant for ~60 years, as the chosen decommissioning method proved wise for all stakeholders in the plant by mitigating potential environmental impacts and reducing financial risks to company shareholders, customers, employees, neighboring communities and regulators.

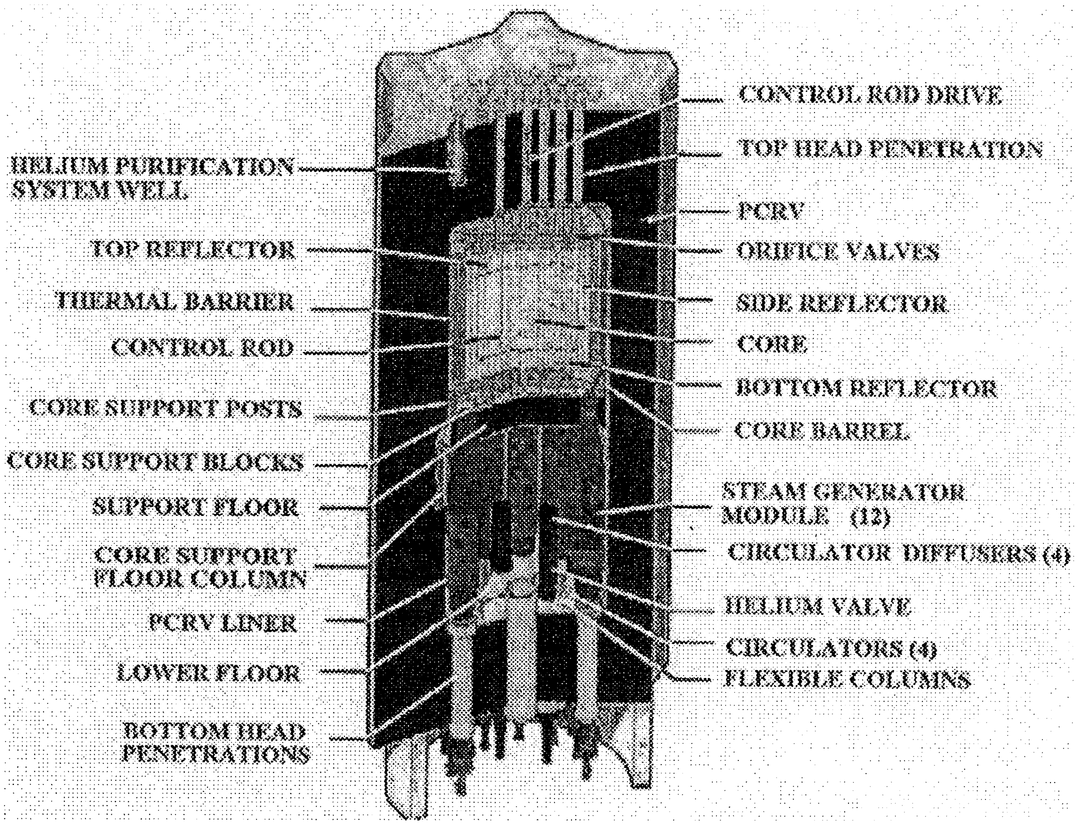

FIG. 8.9. FSV prestressed concrete reactor vessel.

Following defueling, the dismantlement process proceeded in accordance with the decommissioning plan approved by the US NRC. The next challenge entailed removing the radioactive components from the Prestressed Concrete Reactor Vessel (PCRV) (Fig. 8.9), which contained more than 95 percent of the radioactivity at FSV. To accomplish this task, the decommissioning team flooded the PCRV with water to shield the workers from radioactivity. Using two circulating loops of 500 gallons per minute each and a side stream demineralizer, the water was filtered and processed to ensure cleanliness and clarity.

Next, the 1320 ton, 15-foot-thick reinforced concrete top head was removed to provide access to the internal PCRV cavity. This step was accomplished by using diamond-wire cutting cables and cutting the top head concrete into 12 pie-shaped wedges. Each of these 110-ton wedges were radioactive due to neutron activation and read approximately 1.5 rem (15 mSv) per hour at the bottom of the wedge (Fig. 8.10). When removed from the PCRV the wedges were placed in a large segmenting tent, cut into three pieces, placed in special steel cans and shipped as low-level waste to Richland, Washington. The nearly one-inch-thick steel liner was then cut using oxilance cutting tools, removed and also shipped as low-level waste. Upon completion of the top head removal effort, which took approximately nine months, the upper plenum of the reactor was open and PCRV internals were accessible.

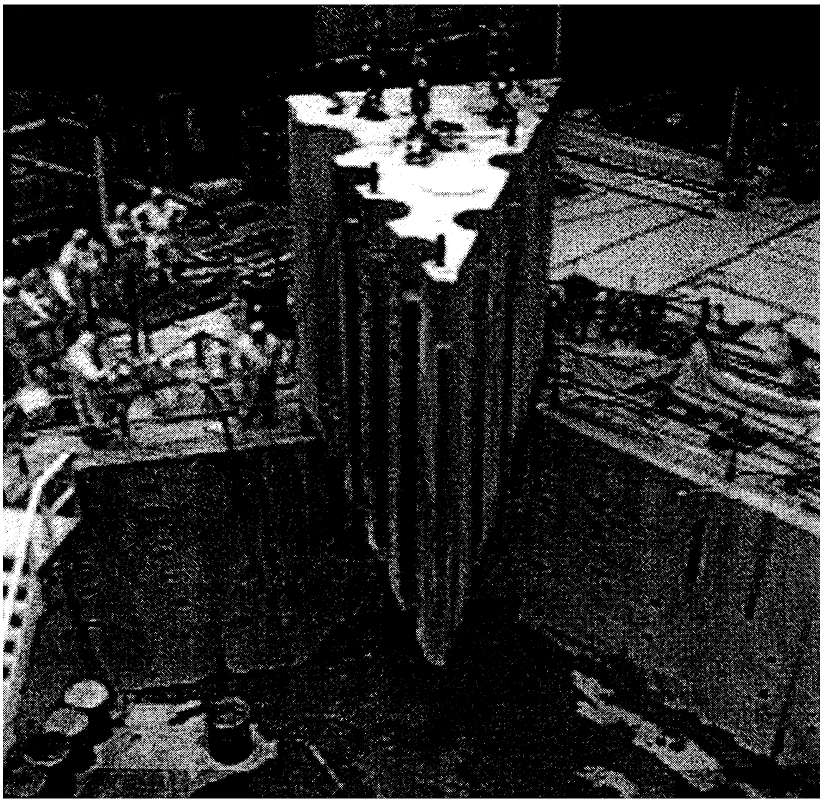

FIG. 8-10. A PCRV top head segment of concrete being removed.

A rotating work platform (Fig. 8.11) was then installed on the PCRV. Operating from this platform, the FSV decommissioning team removed more than 5000 graphite components from the upper plenum. These components, some of which read as high as 300 rem (3 Sv) per hour, were removed and placed into a transfer basket that had been lowered into the water. The basket was then drawn into a lead shield bell (Fig. 8.12a) and was subsequently taken to a hot cell. There the basket was lowered into a shipping cask for shipment as low-level waste (Fig. 8.12b).

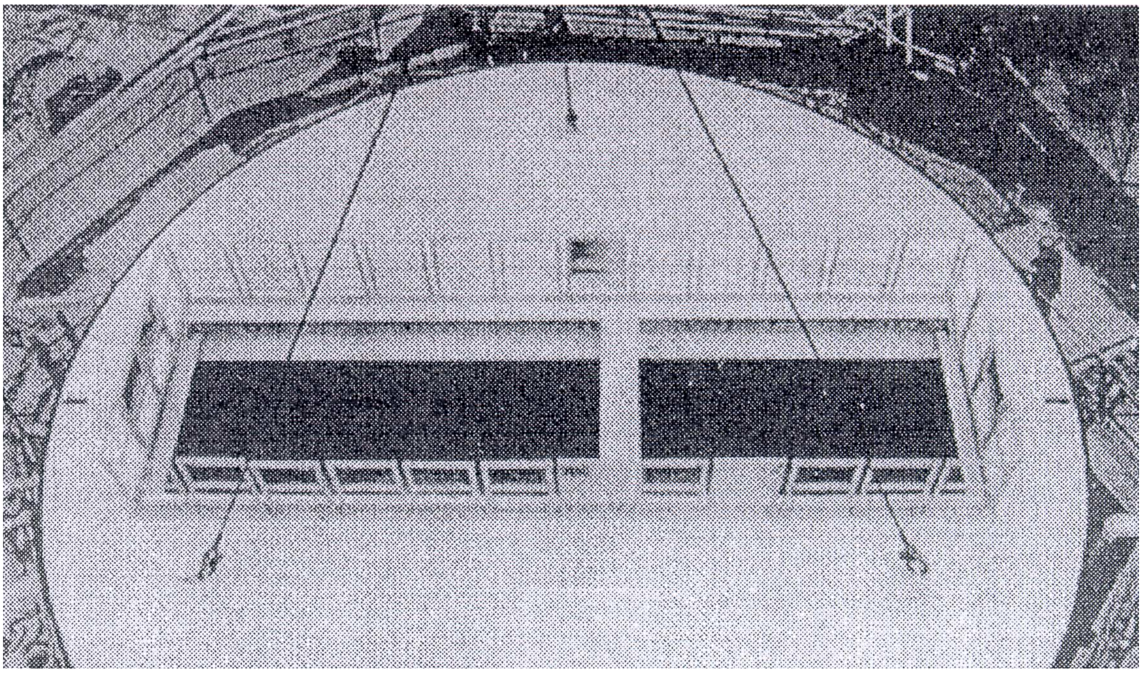

FIG. 8.11. Rotating work platform on PCRV.

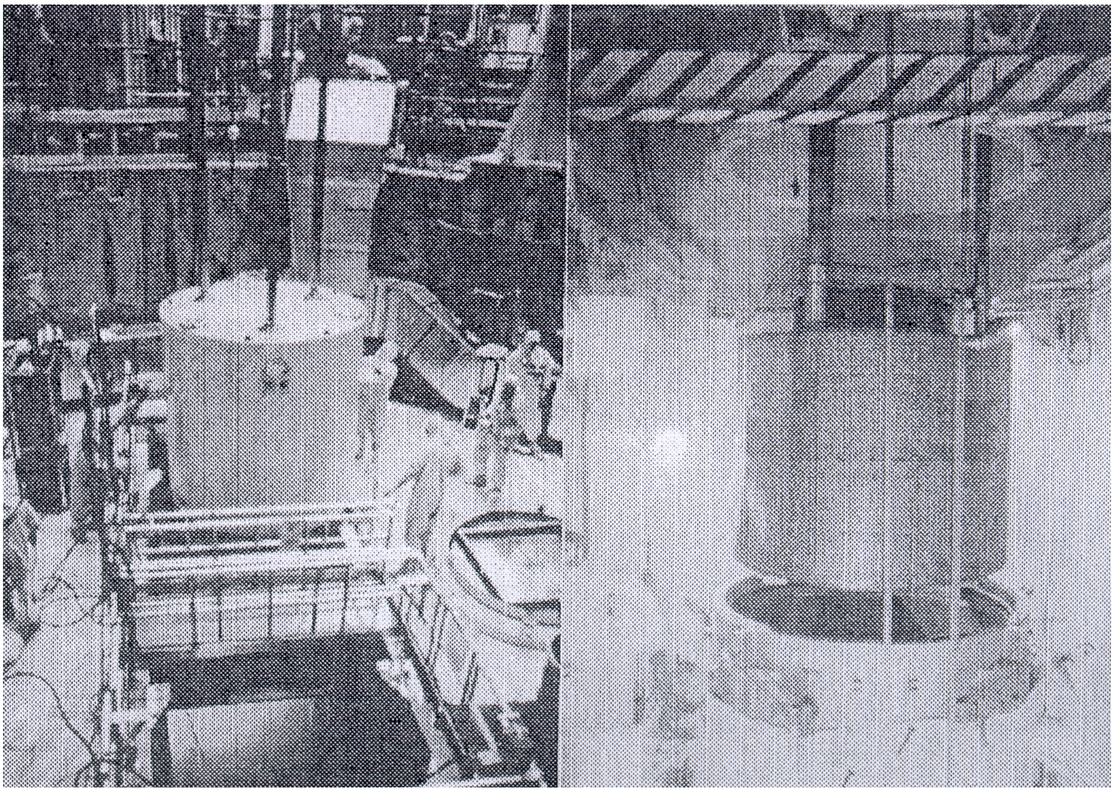

FIG. 8.12. Component shield bell (a) and basket arrangement (b).

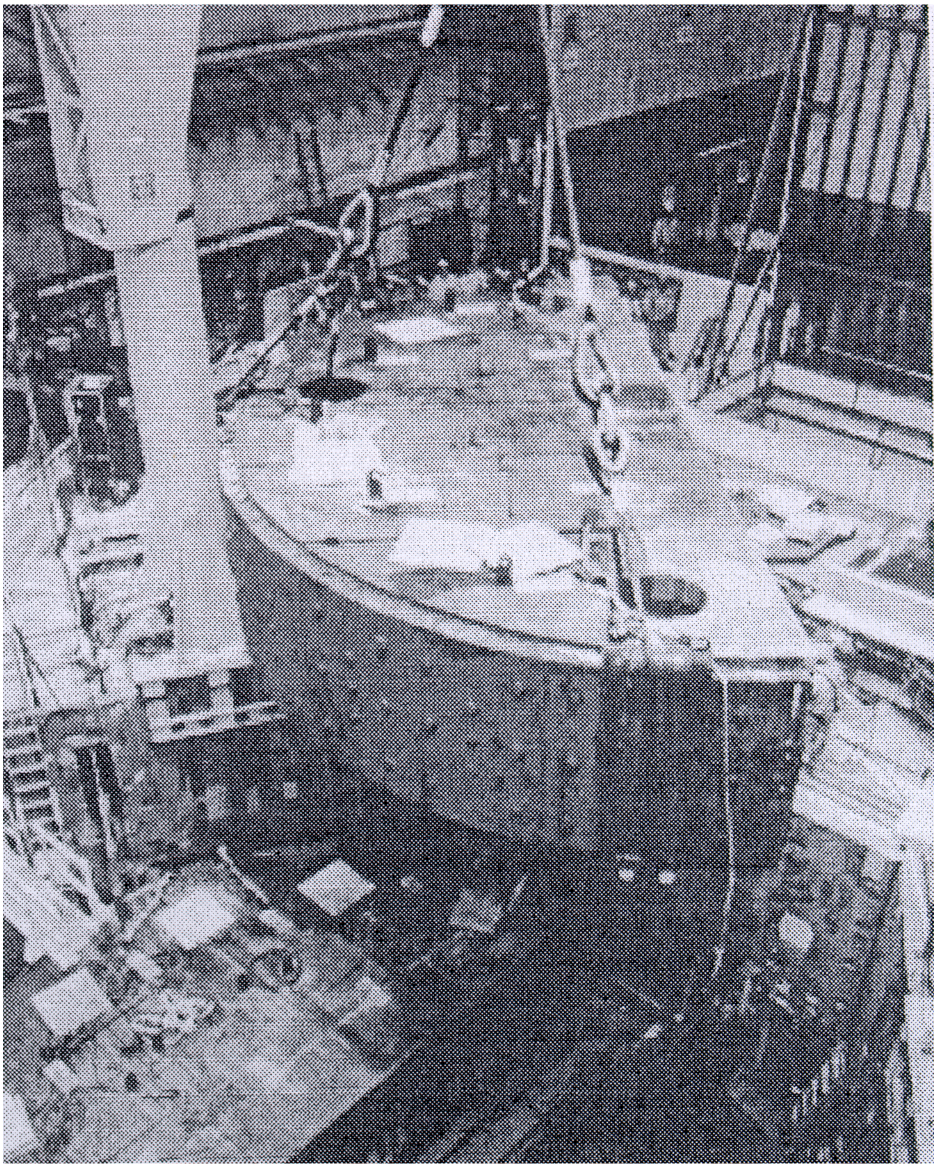

FIG. 8.13. Section of CSF being removed from the PCRV.

Finally, to expose the lower plenum, the core support floor (CSF) had to be raised and removed. The CSF, as installed, weighed 270-ton and is a five-foot-thick, 31-foot-diameter, concrete structure encased in a carbon steel liner. Since the CSF was radioactive, steel shielding plates were positioned on top of the floor prior to its removal with a hydraulic jacking system. Shield water in the PCRV protected underwater divers who entered into the steam generator ducts that went through the core support floor. Once inside these ducts the divers had to cut their way out to access the underside of the CSF. Once under the CSF, the divers severed all connections to the CSF so it could be raised and removed. The CSF, due to the added weight of the shield plates, attachments on the underside and any entrained water, weighed 345 tons during its removal. Again using diamond-wire cutting technology, the CSF was sectioned and removed from the building and shipped off site as low-level radioactive waste (Fig. 8.13).

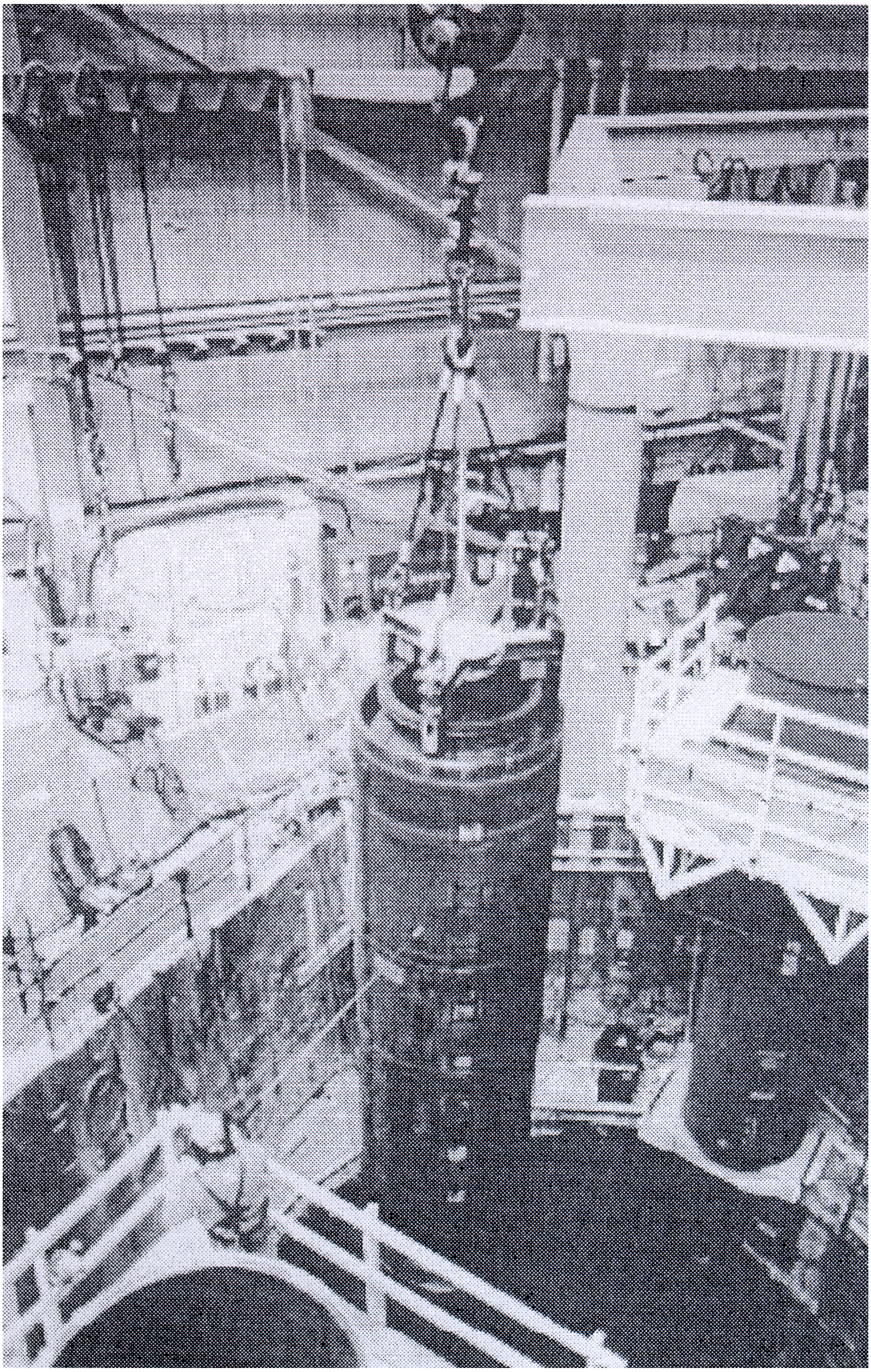

FIG. 8.14. Removal of a steam generator from the PCRV.

Once the CSF was extracted, all components within the PCRV were removed, including 12 steam generators and four helium circulators. Fig. 8.14 shows a steam generator being lifted out of the PCRV.

Finally, the activated concrete ringing the inside of the PCRV beltline and lower plenum areas was removed using diamond-wire cutting technology. The upper beltline concrete sidewall blocks were approximately eight feet wide, 30 inches thick and 42 feet long. The lower plenum concrete blocks were approximately eight feet wide, 27 inches thick and 26 feet in length .

Steps were required to radioactively decontaminate the entire PCRV cavity (75 feet high by 31 feet in diameter), the reactor building and the support buildings to meet the final acceptance criteria. Decontamination was also required on plant piping and the balance of plant systems and equipment. Depending on their levels of contamination, these systems either were cleaned and left in place or removed for disposal as low-level radioactive waste. During this complex dismantlement, decontamination and system removal process, 511 shipments containing 71 412 curies of low-level waste and weighing approximately 15 million pounds were made without incident to the low-level radioactive waste burial site. This effort was required to meet the NRC's release criteria of 5 microrem (.05 microsievert) per hour exposure rate above background one meter from previously activated surfaces and components, and less than 5000 disintegrations per minute per 100 cm^2 (0.75 becquerel per cm^2) for previously contaminated surfaces and components. The dose from all sources of residual activation had to be less than 10 millirem per year (100 microsievert per year) based on an occupancy factor of 2080 hours per year.

The final radiological survey process began in late 1994 and continued throughout 1995 and into 1996. The final survey plan document, which the NRC had to approve, took more than six person years to complete. The objective of the survey was to allow for unrestricted release of FSV from the NRC license. This survey consisted of characterization, final survey, investigation, and remediation measurements which accounted for the more than 400 000 physical measurements taken throughout the facility. This effort required more than 900 person months over a period of one and a half years to complete. The final survey areas for the entire FSV site were divided in to ten survey groups. Each area was evaluated to determine its classification as “unaffected”, “suspect affected” or “non-suspect affected”. These classifications determined the survey methodology required for each area. By the end of the survey over 300 areas had been surveyed.

To prove the accuracy of the final survey results, PSCo contracted for an independent verification survey. Also, the NRC conducted its own verification survey. Two specific issues that had to be addressed were “hard to detect” nuclides and background determination. Hard to detect nuclides identified at FSV were tritium and iron 55. Since these two nuclides cannot be easily measured as part of a general survey, site specific release criteria were determined for FSV. These release criteria were lower than the regulatory numbers to include the effects of the “hard to detect” nuclide contribution. The background determination was important as well because the release criteria was 0.05 micro Sv per hour above background. Background measurements both onsite and offsite varied between 0.02 micro Sv and 0.35 micro Sv per hour. Permission was obtained from the NRC to use gamma spectroscopy to directly measure exposure rate from licensed material in selected areas. This massive effort cost approximately $20 million and produced a report that covers over eleven feet of shelf space to document the measurements and results.

Safety was of primary concern throughout the decommissioning programme. During the four year decommissioning period, and despite the fact that personnel spent 340 percent more time in the radiologically controlled areas than originally forecast, the project experienced an overall total radiation exposure of only 380 person REM (3.80 person sievert). This number, approximately 12 per cent under the original radiation exposure estimate, is roughly equivalent to the expected person-REM exposure during one year of operation for a LWR. Also, the decommissioning project maintained a low (including all subcontractors) lost-workday incident rate of 0.70 per 200 000 person-hours as compared to the US construction industry average incident rate of 5.5 [8-4].

8.2.3. Decommissioning of the THTR 300 [8-8]

The decommissioning option chosen for the THTR 300 is “SAFSTOR”. Under this programme, the plant is intended for dismantlement after ~30 years of safe enclosure. The establishment of safe enclosure was started 1995 after applying for and the granting of attachments to the core unloading license. The main steps undertaken and finished by a general contractor in 1995 included:

- Enclosing the PCRV by cutting and sealing all ~2000 penetrations
- Sealing all primary circuit system components
- Establishing of an additional enclosure for the sealed components by using the existing vented containment as a type of air flow guidance envelope
- Release of the water-steam-cycle including turbine and generator and the four emergency diesel generators from the restrictions of the Atomic Energy Act
- Preparation work for the establishment of a new ventilation system tailored to the requirement for the safe enclosure operation.

In April 1996, the first part of the next license (safe enclosure establishment and pre-operational tests) was granted and concerned mainly with the erection of a new ventilation and exhaust air measuring system. This work was completed in September, 1996.

The second part of the license was granted in July 1996, and contains the main steps for the establishment of safe enclosure including:

- Dismantlement of the liquid waste storage and evaporation system, decontamination shop, etc.
- Adapt the power supply
- Dismantle contaminated equipment outside safe enclosure that does not fulfill the later requirements concerning contamination limits
- Adapt the building drainage system
- Decommission all other systems that are not needed for operation of safe enclosure
- Install new control equipment appropriate for the task of establishing safe enclosure
- Release all buildings of the site (except the three buildings of the safe enclosed plant: reactor hall, reactor operating and auxiliary building) from the restrictions of the Atomic Energy Act.

A significant issue of this phase was the conversion of the major part (~80%) of the controlled area inside the safe enclosed plant into an “operational supervised” area with a dose level less than 2 μSv. This will allow entrance for maintenance purposes without health physics monitoring. This area is the area outside the "envelope of safe enclosure" but inside the safe enclosed plant.

8.2.4. Decommissioning of the AVR

The scope and schedule of safestore decommissioning of the AVR plant is characterized by three supplements to the original license of March 1994. These planned supplements will bring about the clearance of containment from all auxiliary systems and will eventually leave the sealed outer reactor vessel as the only radioactivity containing area. The three supplements include:

- The first supplement comprises the dismantlement of the fuel handling system, coolant circulators and interspace convection pipe,
- The second supplement to safestore decommissioning addresses the dismantlement of the helium purification system and the condensation coolers,
- The third supplement considers the removal of the shutdown rods including their drive mechanisms.

The long term strategy for the decommissioning of the AVR is being developed. The "continued dismantling" process is under consideration and would have as the ultimate goal the restoration of plant site to green field. The term "continued dismantling" was chosen to indicate the direct transition from the present safestore decommissioning and includes a procedure that can be interrupted after each of a number of dismantling steps. It also may be terminated and transferred into a new safestore mode should any obstacle arise in either the financial or organizational terms, to the continuation of the project. According to the schedule, the transition to continued dismantling, if envisaged, would take place at the beginning of 2002. That date would be the start of the first step of continued dismantling and would address remotely controlled in-situ dismantling of the two reactor vessels and their internals (see Fig. 8.15). The second step, beginning in 2007, would comprise the decontamination of the reactor building (including the hot workshop) and the dismantling of the containment vessel. The third step, starting in late 2009, would concern the free release of the buildings, the release of the site from atomic law, and the demolition of the buildings including the field restoration of the site by late 2011 [8-10].

Pre-engineering of the continued dismantling process resulted in two variants for consideration. Variant 1 would include dismantlement of the reactor with an extension of the containment. Variant 2 would result in dismantlement of the reactor without structural alterations of the containment (the in-situ concept). The conditions required by AVR in this dismantling process are that two barriers be maintained (containment and preservation or replacement of the outer reactor vessel. Also, the dismantling work in the inner reactor vessel is to be performed under an inert atmosphere.

The engineering performed so far has resulted in the conceptional procedure whereby the dismantling of the reactor vessels are executed inside the unaltered containment in a disassembling area, which is created by the enlargement of the outer reactor vessel. To achieve this, a disassembling area is built over the top of the outer reactor vessel, supported by a rack and tightening on the cylindrical wall of the vessel in such a way that in the course of dismantling, the disassembling area serves as the first barrier and the containment as the second barrier.

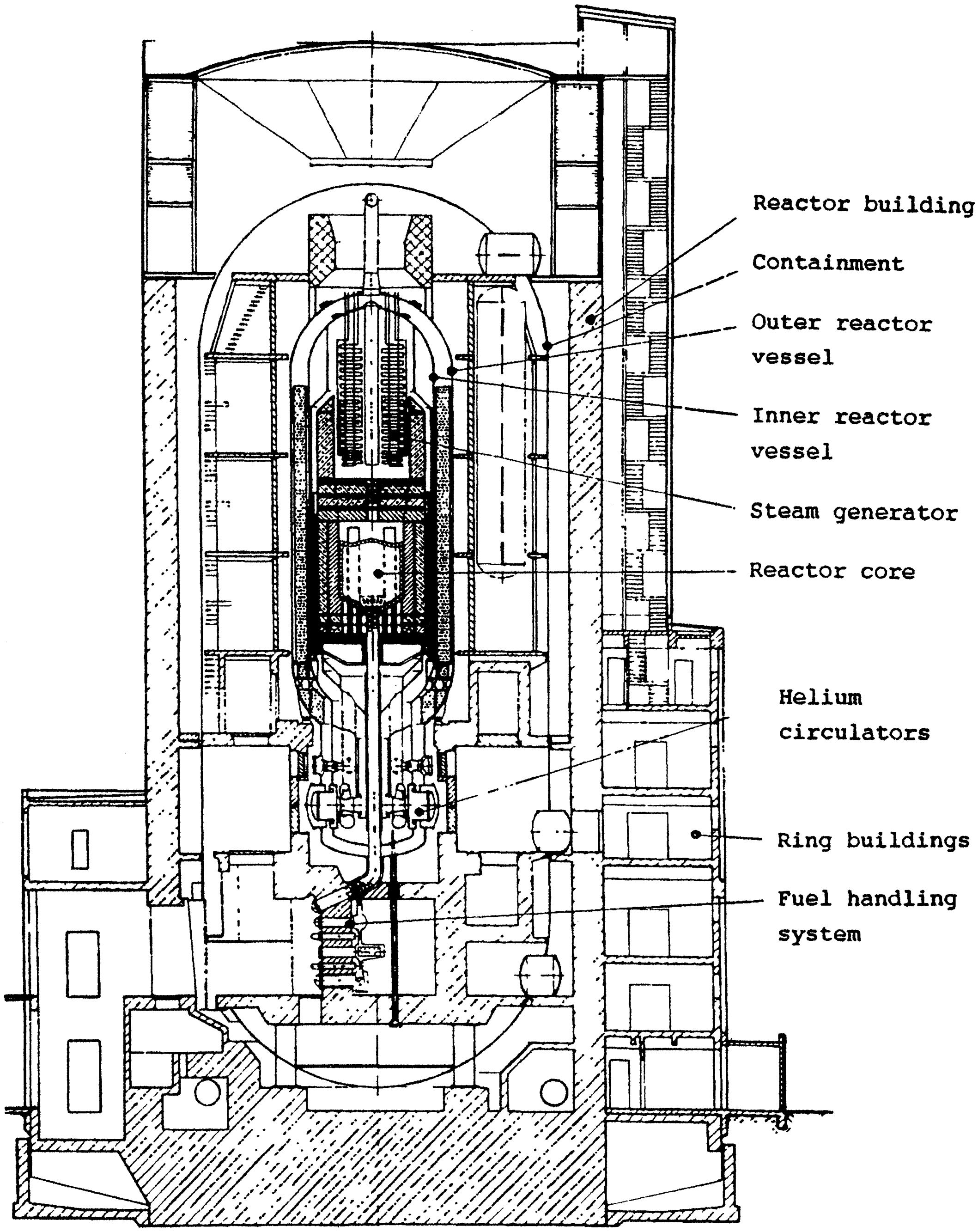

FIG. 8.15. AVR reactor building.

In accordance with the dismantling progress, the disassembling area is performed in a step by step descending manner. The facilities of the remote control dismantling are descending in the same way so that the operating conditions (approach position and handling area) for the remote controlled facilities remain essentially the same.

The important dismantling steps include:

(1) Dismantling of the remaining, but now unnecessary, facilities in the containment and the platforms above the level 17.1 m
(2) Installation of the support rack inside the containment

(3) Installation of the disassembling area with the facilities for remote controlled dismantling, connecting the disassembling area to the outer reactor vessel, and commissioning of auxiliary systems (e.g. vent systems)
(4) Remote controlled dismantling of the steam generator
(5) Remote controlled dismantling of the reactor vessels with internals including the outer vessel
(6) Dismantling of the biological shield 1
(7) Dismantling of the facilities for remote controlled dismantling and the disassembling area.

Upon completion of the measures to dismantle the reactor vessel, the containment will be completely cleared of objects and fully cleaned inside. A provisional platform with a ring-shaped rack will be installed in the containment at the elevation of the manipulator (+5.0 m). Decontamination of the containment shell will be performed from this rack, so that the steel can be conventionally recycled.

All remaining components and facilities which are contaminated or likely to be contaminated (e.g. dowel plates, braces, end plates) and can not be decontaminated together with the building structure, will be dismantled and disposed of. The dismantling of auxiliary and ancillary units may take place simultaneously with the decontamination and radiation measuring of the building structures. This includes facilities and systems such as liquid waste, ventillation, and electrical to the level that they are required to support on-going decommissioning activities.

Principally, the decontamination of buildings is to progress from the higher contaminated locations to the lower contaminated areas, from top to bottom, and from the difficult locations to the areas with room access openings. Decontaminated and surveyed rooms will be sealed in order to avoid recontamination.

After disposal of the decontamination devices and contaminated equipment, and the plant is free of artificial radioactive nuclides generated during the former plant operation, the plant will be assessed by survey measurement, documented and confirmed by the authorities. Based on this documentation, the AVR plant will then be released from the obligations under atomic law [8-9].

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Chapter 9

CHALLENGES TO COMMERCIALIZATION OF THE GAS TURBINE HTGR PLANT

The gas turbine HTGR represents an advanced nuclear power concept that has not been demonstrated with an actual plant. The GT-MHR and PBMR are both in the development phase with detailed design yet to be completed. Successful design, construction and demonstration of these plants is expected to provide the electric utility industry with a highly efficient, relatively low cost, safe and environmentally clean method of producing electricity. However, achieving successful deployment of this plant type will require careful development of the systems and components comprising the design, many of which are either new to the nuclear power industry, involve recent technological advancements, and/or consist of component applications operating in environments and configurations never before demonstrated.

The design organizations readily acknowledge that these plants will require extensive development in order to achieve final commercialization. This is particularly evident with the power conversion system components, where physical size, orientation and operating environment challenge the existing experience base. In this regard, the developing organizations of the PBMR and GT-MHR conceptual plants have subjected their designs to multiple independent international technical and economic reviews resulting in the strong, nearly unanimous, consensus that these designs are viable with no technological “show-stoppers”.

The following challenges are based on the editor’s judgment primarily with focus on the PBMR and GT-MHR closed cycle gas turbine plants because of their advanced stage of development. Due to the basic design of this concept, these challenges all tie directly to the primary coolant system and can be categorized as either “reactor plant” or “power conversion system”.

9.1. REACTOR PLANT

9.1.1. PBMR reactor design

The intent of the PBMR designer is to use the German HTR-MODUL (by Siemens/Interatom) and HTR-100 (by ABB/HRB) concepts coupled with the experiences gained from the operation of the pebble bed AVR and THTR-300 plants as the development basis of the PBMR. However, the final design of the PBMR may include significant differences from these two German reactors that will require code validation(s) and additional development and testing prior to licensing. An example of the difference in core design includes the use of non-fuelled graphite pebbles in the center of the core resulting in an annular configuration. This use of unfueled graphite pebbles in the center of the core will essentially cause the flux profile to expand outward thereby allowing for flux flattening at high power and to provide for better core control capability. This will represent a significant change in the basic core physics characteristics of the PBMR.

Also, acceptance by the South African nuclear regulatory agency, the National Nuclear Regulator (NNR), of the reactor design and its specific HTGR related safety attributes without the need for added requirements and their attendant increase in plant costs represents a risk to commercialization of the PBMR.

The significance to achieving deployment due to this challenge is the reactor performance for both normal and accident operating conditions and associated core physics characteristics for the PBMR will have to be well understood prior to receipt of the final operating license. Any significant deviation from the German design(s) will require modeling and code validation prior to licensing. Although much of the German background can be incorporated into the PBMR, any change in core design represents an additional R&D requirement with an attendant risk for plant licensing. However, if core performance is found to significantly infringe on the safe operational envelope of reactor performance, the consequences could be substantial to future commercialization.

ESKOM has interfaced with the NNR in the development of the PBMR for the past three years. The NNR is currently evaluating the design and supporting documentation relative to their regulatory standards. At issue for the license to construct the first unit is whether the CNS will: a.) Accept and incorporate into the South African nuclear regulatory base the results of the German licensing authority relative to the pebble bed design, or b.) Require further validation which could include additional research, computer modeling, and testing which will most probably be the case for any deviation from the basic German reactor design, and c.) Accept ESKOM's intent to secure the final operating license by "licensing by test" of the first unit.

Acceptance of the PBMR design will require the NNR to determine that the design safety attributes specific to the HTGR meet (or exceed) the overall safety criteria for nuclear power in South Africa. This will include such attributes as very low core power density, use of the TRISO coated fuel particle, an inert single phase (gas) primary coolant and a reactor structure, principally of graphite, with slow heat up characteristics under accident conditions. However, in balance is whether these attributes will provide the equivalent safety dictated by existing regulations such as the need for an overall pressure containment, and allow for a reduced emergency planning zone and a significantly reduced number of safety class systems and components. If ESKOM is unable to provide the supporting justification(s) to the satisfaction of the NNR, the significance could be anywhere from minor to great depending on what additional requirements might be imposed on the PBMR design. This includes the risk of the PBMR becoming cost prohibitive.

9.1.2. GT-MHR reactor design

The GT-MHR includes significant changes in core and internal component design from the existing experience base of reactors utilizing prismatic fuel elements. These changes, coupled with the need for acceptance by the Russian licensing authorities (GAN) of the specific safety attributes inherent in the HTGR without added requirements being imposed on the design, represent significant challenges to the sponsors. Examples of changes anticipated for the GT-MHR core include:

- The GT-MHR is an annular core with the central region consisting of unfueled graphite elements. This is believed to represent first-of-a-kind core geometry.
- Basic operating characteristics of the GT-MHR reactor such as power level, core temperatures, response to accident conditions, etc. represent core physics characteristics which are different from the established experience base.
- Active core height of the GT-MHR will be ten fuel elements, whereas past experience was six elements high (at Ft. St. Vrain).
- The refueling scheme includes shuffling of fuel elements to different core locations for burn-up purposes.

— The control rod drive mechanism and the reserve shutdown actuation mechanism are a new design without a proven experience base.

Acceptance of the GT-MHR design will require GAN to determine that the design safety attributes specific to the HTGR meet (or exceed) the overall safety criteria for nuclear plants in the Russian Federation. This will include evaluation of such attributes as the very low core power density, use of the TRISO coated fuel particle, an inert single phase (gas) primary coolant and a reactor structure, principally of graphite, with slow heat up characteristics under accident conditions. However, in balance is whether these attributes will provide the equivalent safety of existing regulations and allow for regulatory changes such as a reduced emergency planning zone and a reduced number of safety class systems and components. If the sponsors are unable to provide the supporting justification(s) to the satisfaction of the licensing authorities, the significance could be anywhere from minor to high depending on what additional requirements would be imposed on the GT-MHR design. This includes the risk that the GT-MHR could become cost prohibitive. It should be noted that the present design calls for an external, pressure tight containment for the plutonium burning plant. Whether this will also be the case for the commercial GT-MHR has, as yet, not been decided.

Regarding changes, the core and associated component design deviations from the existing experience base will have to be proven analytically prior to acceptance for initial licensing. The GT-MHR core configuration represents physics issues that can only be completely validated when full power is achieved (and then when being tested under accident conditions). If significant issues become evident when modeling the reactor, further research, including testing and validation will be necessary prior to licensing. Similarly, if significant deviation occurs from predicted core response during initial plant testing, the end result could be a substantial issue in attaining competitive deployment.

Technical failures in core reactivity control mechanism design would require additional re-design prior to licensing which would place an economic burden on deployment of the plant. Similarly, shuffling of elements during refueling, if not properly accounted for in the design, would extend the plant outage causing an unanticipated economic burden.

9.1.3. Primary coolant environment and fuel development

The radiological makeup of the PBMR and GT-MHR primary coolant will influence the materials utilized on major components such as the vessel(s) and turbo-compressor blading, and also be a dictating force in the development of tools for component maintenance to assure personnel exposures ALARA.

The GT-MHR and PBMR represents the first application of its kind for many of the primary system components. Past gas cooled reactor plant experience with a helium primary system environment represents a limited base of knowledge. Other system parameters such as helium temperature and chemical contamination, in conjunction with the radiological environment, may compound the challenge placed on the primary system components.

With regard to the fuel, an extensive TRISO particle operational experience base has been established principally in the reactors of FSV, AVR and theTHTR-300. This, coupled with numerous irradiation tests of UO_2 TRISO capsules in the USA, Germany, the Russian Federation and Japan, have provided consistently high performance results for application to the GT-MHR and PBMR. Note, an exception to this high performance was with the HRB-21 capsule in the USA, which exhibited high failure of the silicon carbide (SiC) coating due to unanticipated stresses placed on the coating by the addition of other, (now considered) unnecessary coatings.

This base of experience has given the PBMR and GT-MHR designers confidence that the coated fuel in their respective plants will perform to the required standard of quality and that the fuel is not viewed as a significant challenge. However, the operation of the PBMR and GT-MHR will include high fuel burn-up at high temperatures for extended periods of time. It is a known condition that the silver isotope, Ag-110m, is released from the TRISO particle at high temperatures. The amount and level of influence this isotope has on the metallurgical and maintenance considerations for the PCS components over the long period of plant operation has not yet been fully determined [9-1]. Also, if operation is allowed to exceed a fuel temperature of 1250°C. over an extended period of time, SiC coating thickness deterioration will occur due to palladium attack [9-2]. Also, although the commercial GT-MHR is to utilize low enriched uranium as the fissionable fuel, the initial deployment of this plant is for the burning of weapons plutonium in the Russian Federation. As such, an extensive plutonium fuel particle development program is required for the first (non-commercial) GT-MHR plants.

Failure to understand the isotopic makeup, method of deposition on equipment and associated radiological dose levels for the PBMR and GT-MHR primary system(s) could result in increased failure of equipment, additional plant downtime and larger then anticipated personnel exposures. As an example, the turbo-machines are expected to operate up to seven years without the need for repair/replacement. Seriously misjudging the radiological effects on the machine components could pose a significant problem both for plant availability and risk of substantial equipment failure.

Not knowing the extent of Ag-110m release as well as continuous high temperature operation and possible neutron streaming may significantly increase the maintenance considerations on the PCS components and cause deterioration in the life of the first stage turbine blades. Ag-110m is expected to deposit on the cooler surfaces of the PCS components thereby increasing the need for tooling/shielding when maintaining this equipment. However, the significance for competitive deployment due to this deposition of silver is not expected to be a substantial limitation to the commercial viability of the PBMR or GT-MHR.

Also, these reactors have not, as yet, been validated to assure the requirement that core locations do not exists where fuel is subjected to long term operation in excess of 1250°C. Due to the established experience base and previously validated HTGR codes and models, this is not expected to be a significant challenge to commercialization.

9.2. POWER CONVERSION SYSTEM

9.2.1. Turbine, compressor and generator development

The commercial producers of large turbines, compressors and generators that have provided input to the PBMR and GT-MHR designers are in general agreement that the technological development required for these machines does not represent substantial risk of failure. However, these will be the first application for the utilization of magnetic bearings on machines of this size in a vertical configuration, with high temperature, ultra-dry radiologically contaminated helium as the working fluid. Developmental issues that challenge the designers include:

— Understanding and optimizing the rotor-dynamic characteristics of the machine(s)
— Determination of materials which will operate in the above mentioned helium environment for the extended lifetime of 6 years between overhauls and 60 000 hours without component failure
— Development of magnetic bearings and associated catcher bearings for the intended rotor size, weight, orientation and operating environment[8]
— Application of seals for reliable performance in the required helium environment and in conditions where substantial thermal movement exists on the sealing surfaces
— Generator insulation and stator support designs to prevent migration of the stator windings as a result of insulation deformation coupled with electromagnetically induced vibration in a vertical position
— Stator insulation and exciter diode design to prevent "pop corning" in the event the generator pressure vessel sustains a rapid depressurization following sustained full pressure operation
— Accessibility for maintenance and to assure safe and efficient operational performance.

Economical development of the rotating equipment will be a significant factor in achieving competitive commercialization of the PBMR and GT-MHR plants. Although these machines share many design similarities, the GT-MHR unit incorporates a single rotating shaft including an exciter, electrical generator, turbine, high pressure compressor and low pressure compressor; whereas the PBMR utilizes three rotating machines including a high pressure turbine-compressor, a low pressure turbine-compressor, and a turbine-generator.

A key to the HTGR closed cycle gas turbine plant, with its attendant high efficiency (through the use of the Brayton Cycle) and projected low capital and operating costs, rests with being able to develop this rotating machinery. If this is unattainable, either because of unsolvable technical issues, unanticipated substantial safety and regulatory concerns and/or unexpected developmental or operational costs which result in lack of competitiveness, the future deployment of this advanced nuclear power plant will be at risk and probably will not occur.

9.2.2. Magnetic bearing development

Incorporation of magnetic bearings and their associated catcher bearings on the rotating machines of the GT-MHR and PBMR will represent a significant departure from the established experience base. The areas of departure include: a.) bearing size; the axial (thrust) and radial bearings of these machines are the largest application ever undertaken for magnetic bearings; b.) operational environment of these bearings; particularly with regard to helium with elevated temperature and pressure, ultra dry, but chemically and radiologically contaminated conditions, has not been previously experienced; c.) orientation; all PBMR and GT-MHR rotating machines will have a vertical shaft orientation, which, to my knowledge, represents a first-of-a-kind condition for machines of this size, weight and speed.

Failure, due to a significant (uncorrectable) technical flaw, of the magnetic bearings and/or associated catcher bearings to meet their intended function would result in the GT-MHR and PBMR not attaining commercial status. There is no realistic bearing alternate (except, possibly, a hybrid arrangement for the generator thrust bearing which could

[8] Due to the individual difficulty of challenge and high significance, the areas of magnetic/catcher bearing development and seal design are included in this report as stand-alone technical challenges.

incorporate a conventional bearing system). Within the primary coolant environment, oil or water lubricated bearings are not seen as an acceptable alternate.

A situation could arise where the bearings fail due to a technical flaw that is correctable. This may not represent a fatal situation, depending on the cost associated with redesign, replacement and other changes in the PCS such as possible vessel and/or heat exchanger reconfiguration mandated by the modified bearing arrangement.

9.2.3. Helium seal development

Helium seals will be utilized to prevent leakage and bypass flow between components of the PCS. These seals will be required to operate in an ultra-dry (non-lubricating), high temperature, high pressure, radiologically contaminated helium environment, with the added requirement to be able to grow and move with temperature changes of the PCS components.

Long term operation of these seals and the thermal expansion between PCS components and associated supporting structures represent design requirements and configurations for which no experience base exists.

Failure of the seals to remain leak tight throughout the duty cycles expected for the PCS could result in a situation where the GT-MHR or PBMR are not able to attain commercial status (most severe situation), or result in a situation as minimal as a minor reduction in overall cycle efficiency due to helium bypass leakage slightly in excess of design.

Helium is a difficult gas to contain. Bypass leakage of helium has a direct influence on overall cycle efficiency, and, therefore, reflects on the ability of the plant to achieve competitive deployment.

9.2.4. Recuperator, precooler and intercooler

The recuperator required for both the GT-MHR and the commercial PBMR will represent the largest application of its type ever built. The GT-MHR utilizes the prime surface variant of the plate-fin design. The PBMR will include a standard plate-fin type recuperator. Although the overall recuperator will consist of individual, repetitive modules, its design, size and environmental operating requirements have yet to be proven in an existing application. Similarly, the precooler and intercoolers, although of a generally standard design will require development and qualification testing to assure their successful performance prior to coupling to a nuclear heat source.

Examples of areas of development and testing for these heat exchangers include: a) assuring that the 95% effectiveness criteria for the recuperator is attained, and; b) the ability to test, locate and repair (minor) leakage paths in all heat exchangers without the requirement for removal of the unit.

The inability to attain the required recuperator effectiveness will directly affect overall system efficiency resulting in loss of generation and attendant economic penalties.

Throughout the life of the recuperator, precooler and intercooler heat exchangers there will be the necessity to test the unit(s) for component degradation and to locate and repair minor leaks without the need to remove or replace the unit. Without this capability, the task of replacing a heat exchanger is quite significant, requiring entrance into the PCS vessel and subsequent removal of major PCS components in order to access the faulty unit.

9.2.5. Overall PCS performance

The GT-MHR and PBMR represent the first application(s) of a closed cycle gas turbine PCS coupled to a nuclear power source. Further, the primary system conditions for these plants will achieve higher operating temperatures then past HTGRs. No past experience base exists which can be applied to fully assure the technical and economic performance expected by these plants.

Anticipating and understanding the performance characteristics to assure that the components making up the primary coolant system will collectively operate as an integrated, functionally sound system will be the most significant overall factor in achieving commercial deployment.

Many other challenges described in this report focus on individual components in the PBMR and GT-MHR. This challenge represents the need to provide assurance that the collective integration of all of the PCS components function as a single operating entity in the manner required by the designers, license authorities and vendors. The magnitude of this challenge is considered as high.

Final validation will be required of the first unit in a stringent testing program [9-1].

REFERENCES TO CHAPTER 9

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Chapter 10

IAEA HTGR ACTIVITIES IN SUPPORT OF TECHNOLOGY DEVELOPMENT

The principle focus of the IAEA's programme related to the HTGR is to assist Member States in understanding, developing and ultimately deploying this advanced nuclear power reactor. The objectives of this programme are twofold:

(1) To promote exchange of information on national programmes including new developments and experience, to review and discuss key problems in the development of HTGRs and to facilitate international co-operation in their research and development;
(2) To support the deployment of HTGRs by informing IAEA Member States about their actual technical status and by acting as an information broker between reactor suppliers and possible future customers from Member States.

The IAEA's gas cooled reactor programme is guided by the International Working Group on Gas Cooled Reactors (IWGGCR) which was founded in 1978. Presently, fourteen Member States (Austria, China, France, Germany, Italy, Indonesia, Japan, Netherlands, Poland, the Russian Federation, South Africa, Switzerland, UK and the USA) are members, together with representatives from the CEC and the OECD. The IWGGCR is the only worldwide forum that coordinates discussion of GCR development programmes, their actual status, trends and progress [10-2].

Many of the activities described within this section have been documented by specific, detailed IAEA publications. In response to a recommendation at the 15th meeting of the IWGGCR, the IAEA is in the process of placing these documents on the Internet.

The address for these documents is: http://www.iaea.org/inis/aws/htgr/index.html

10.1. CO-ORDINATED RESEARCH PROGRAMMES (CRPs)

10.1.1. Validation of safety related physics calculations for low-enriched GCRs

This CRP addresses core physics issues for advanced gas cooled reactor designs. This CRP was initiated in 1990 primarily to fill gaps in validation data for physics methods used for core design of advanced HTGRs fuelled with low enriched uranium. Countries participating in this CRP included China, France, Japan, Netherlands, Switzerland, Germany, the Russian Federation and the USA

Although this CRP has now been completed, the main activities were carried out by a team of researchers within an international project at the PROTEUS critical experiment facility at the Paul Scherrer Institute, Villigen, Switzerland. Fuel for the experiments was provided by the KFA Research Center, Juelich, Germany, and initial criticality was achieved on July 7, 1992. Experiments have been conducted for graphite moderated LEU systems over a range of experimental parameters, such as carbon-to-uranium ratio, core height-to-diameter ratio, and simulated moisture ingress concentration, which have been determined by the participating countries as validation data needs. The Paul Scherrer Institute has been very supportive in incorporating experiments as defined by the participating countries to provide results focused on their validation data needs. Key measurements performed at PROTEUS which provided validation data relevant to current advanced HTGR designs include measurements of shutdown rod worth in both the core and side reflector, effects of moisture on reactivity and shutdown rod worth, critical loadings, neutron flux distribution and reaction rate ratios.

Also, data from the uranium fueled criticals at the Japanese VHTRC critical experiment facility on the temperature coefficient (to 2000°C) of low enrichment uranium fuel have been provided by JAERI and analyzed by CRP participants. The results show that calculations of the temperature coefficient are generally accurate to within about 20 per cent.

10.1.2. Validation of predictive methods for fuel and fission product behaviour in GCRS

The experience base for GCR fuel behaviour under accident conditions was reviewed at an IAEA Specialists Meeting in 1990 and the CRP on Validation of Predictive Methods for Fuel and Fission Product Behaviour in GCRs was initiated in 1993. Countries that participated in this CRP included China, France, Japan, Germany, the Russian Federation, and the USA Within this CRP, chief scientific investigators reviewed and documented the status of the experimental database and verified and validated the predictive methods for GCR fuel performance and fission product behaviour. The technical areas addressed included fuel performance during normal and accident oxidizing and non-oxidizing operating conditions, transport of gaseous and metallic fission products during normal and accident conditions, performance of advanced fuels, and GCR fuel design and fabrication programmes. IAEA-TECDOC-978 documents the activities and results of this CRP.

10.1.3. Heat transport and afterheat removal for GCRs under accident conditions

The CRP on Heat Transport and Afterheat Removal for GCRs under Accident Conditions also began in 1993. Countries that participated in the CRP included China, France, Japan, Germany, the Russian Federation, and the USA The objective of this CRP was to establish sufficient experimental data at realistic conditions and validated analytical tools to confirm the predicted safe thermal response of advanced gas cooled reactors during accidents. The scope included experimental and analytical investigations of heat transport by natural convection, conduction and thermal radiation within the core and reactor vessel, and afterheat removal from the reactor. Code-to-code, and code-to-experiment benchmarks were performed for verification and validation of the analytical methods. Assessments of sensitivities of predicted performance of heat transport systems to uncertainties in key parameters were also investigated. Countries participated in these benchmark and experimental activities according to their specific interests and needs. Benchmark activities and cooperation in experiments within the CRP include analyses of heat up accidents associated with the HTTR, VGM, HTR-10 and SANA-1 facilities.

10.1.4. Design and evaluation of heat utilization systems for the high temperature engineering test reactor

The ultimate potential offered by HTGRs derives from their unique ability to provide heat at high-temperatures (e.g. in the range from about 550° to 1000°C) for endothermic chemical processes and, at 850°C and above, for highly efficient generation of electricity with gas turbine technology. Heat from HTGRs can be used for many high temperature industrial processes including; production of hydrogen and/or synthesis gas and methanol by steam-methane reforming, production of hydrogen by high temperature electrolysis of steam and by thermochemical splitting of water, hydrogasification of coal, and for processes which require lower temperatures, such as petroleum refining, seawater desalination, district heating, and generation of steam for heavy oil recovery and tar sand mining. If the heat demand is not in the immediate vicinity of the reactor, a chemical heat pipe can be utilized as a high temperature heat transporter.

Many IAEA Member States are concerned about global environmental problems that result from burning fossil fuels. The application of nuclear process heat can make a significant contribution to resolve these problems. To foster international cooperation in HTGR applications, the IAEA's Division of Nuclear Power and the Division of Physics and Chemistry established this CRP on Design and Evaluation of Heat Utilization Systems for the HTTR in 1994. JAERI is providing the HTTR in support of this CRP. This 30MW thermal facility has the capability of achieving a helium core outlet temperature of 950°C. Countries participating include China, Germany, Indonesia, Israel, Japan, the Russian Federation and the USA The processes being assessed are selected by CRP participants according to their own national interests depending on the status of technology, economic potential, environmental considerations, and other factors. First priority candidate systems being evaluated include steam and carbon dioxide reforming of methane for production of hydrogen and methanol. In addition, testing of advanced intermediate heat exchangers is being examined.

The CRP participants are collaborating by exchanging existing technical information on the technology of the heat utilization systems, by developing design concepts and by performing evaluations of the candidate systems for potential demonstration with the HTTR. This CRP is currently nearing completion with publication of the proceedings in a TECDOC in late 2000.

10.1.5. Evaluation of HTGR performance

In a 1996 recommendation, the IWGGCR asked that the IAEA initiate a new CRP on Evaluation of HTGR Performance. This recommendation was supported by offers from JAERI and from INET to utilize the HTTR and the HTR-10 test reactors, respectively, in evaluation of HTGR performance. This CRP was initiated in 1997 and is expected to continue through 2002 with participation by China, Germany, Indonesia, Japan, Netherlands, the Russian Federation, South Africa and the USA The primary objective of this CRP is for selected Member States to evaluate HTGR analytical and experimental performance models and codes in conjunction with the start-up, steady state and transient operating conditions of the HTTR and the HTR-10. The overall scope of evaluation of HTGR performance includes the areas of core physics, reactor safety characteristics, fission product release and transportation behaviour, thermal hydraulics and reactor dynamics, control and instrumentation and high temperature component performance. Initial benchmark problems for this CRP include HTTR and HTR-10 core physics issues.

10.2. RECENT INFORMATION EXCHANGE MEETINGS

The activities carried out by the IAEA within the frame of the IWGGCR include technical information exchange meetings. Small specialists meetings are convened to review progress on selected technology areas in which there is a mutual interest. For more general participation, larger technical committee meetings TCMs), symposia or workshops are held. Brief reviews of recent meeting sponsored by the IAEA in support of information exchange on the HTGR are provided.

10.2.1. TCM on GCR response under accidental air or water ingress

The TCM on Response of Fuel, Fuel Elements and Gas Cooled Reactor Cores under Accidental Air or Water Ingress Conditions was hosted by the INET, Tsinghua University, in Beijing, China. Some key conclusions from this meeting are summarized in the following.

The response of gas cooled reactors to postulated air and water ingress accidents is highly design dependent and also dependent upon the cause and sequence of events involved. Water ingress may be caused by tube ruptures inside the steam generator due to higher pressure in the secondary loop. The core can only be affected if steam or water is transported from the steam generator to the reactor. Air ingress is possible only after a depressurization accident has already taken place and has to be looked at as an accident with a very low probability.

Considerable experimental data exists regarding behaviour of GCRs under air ingress conditions. These experiments have shown that self sustained reaction of reactor graphite with air does not occur below about 650°C, and above this temperature there is a window of air flow rates: low flows supply insufficient oxidizing gas and fail to remove the reaction products, whereas convective cooling at high flows will overcome the chemical heating. Nuclear grade graphite is much more difficult to burn than coal, coke or charcoal because it has a higher thermal conductivity making it easier to dissipate the heat and because it does not contain impurities which catalyze the oxidation process.

Two serious accidents have occurred which have involved graphite combustion: Windscale (October 1957) and Chernobyl (April 1986). It is important to clearly understand these accident sequences, and the significant differences in the design of these reactors, compared to gas cooled reactors, which use graphite as moderator and either helium or CO_2 as coolant. Windscale was an air cooled, graphite moderated reactor fueled with uranium metal clad in aluminum. The accident was most likely triggered by a rapid rate of increase in nuclear heating (that was being carried out for a controlled release of the Wigner energy) which caused failure of the aluminum cladding. This exposed the uranium metal, which is extremely reactive, to the air coolant, and resulted in a uranium fire, which caused the graphite fire. Water was finally used to cool down the reactor after other efforts failed. Chernobyl was a water cooled, graphite moderated reactor. The rapid surge in nuclear power generation at Chernobyl resulted from a series of safety violations and core neutronic instabilities. Eventually, liquid nitrogen was used to cool the burning debris. It must be emphasized that gas cooled reactors neither use air as coolant (as in Windscale) nor have core neutronic instabilities such as those of the Chernobyl reactor.

Safety examinations of German modular HTR design concepts are addressing even very hypothetical accidents such as the complete rupture of the coaxial hot gas duct. A large scale experiment, called NACOK, is constructed at the FZJ Research Center, Juelich, Germany, to measure the natural convection of ingressing air and to provide data for validating theoretical models.

As a part of the safety review of the HTTR, extensive investigations have been carried out by JAERI of that reactor's response to air ingress accidents including rupture of the primary coaxial hot gas duct and the accident involving the rupture of a stand pipe attached to the top head closure of the reactor pressure vessel. Experimental and analytical investigations have shown that graphite structures would maintain their structural integrity because of the limited amount of oxygen within the volume of the containment that is available to oxidize graphite. Further, there is no possibility of detonation of the produced gases in the containment. Experimental test results showed that there is a large safety margin in the design of the core support posts.

JAERI has examined the response of the HTTR to a design basis accident involving rupture of a pipe in the pressurized water cooler. The ingress of water is sensed by the plant protection system instrumentation resulting in reactor scram and isolation of the pressurized water cooler. Analyses show that the amount of ingressed water is insufficient to result in opening of the primary system safety valves, and the auxiliary cooling system rapidly reduces the core temperatures thereby limiting the oxidation of the graphite structures to acceptable levels. Similar investigations have been conducted by INET for design basis accidents of the HTR-10 reactor assuming the rupture of one or two steam generator pipes.

The neutronic effects of moisture ingress on core reactivity and on control rod worth were examined in Switzerland at the PROTEUS facility. Neutronic effects of water are simulated by inserting polyethylene (CH_2) rods into the core as this material has essentially the same hydrogen density as water. The effect of increasing amounts of "water" is first to increase the core reactivity to a maximum due to under moderation of the neutrons under normal conditions, followed by a reactivity decrease as neutron absorption by hydrogen becomes the dominating factor. Further, water addition into the core has the effect of reducing the worth of the shutdown rods. In the experiments to date, these effects have been well predicted, reflecting perhaps the mature state of reactor physics analysis methods.

To ensure the ultimate goal of a catastrophe-free nuclear energy technology, additional analyses of extreme hypothetical accident scenarios should be performed and, in parallel, methods for enhancing the passive corrosion protection of the graphite fuel elements and structures could be used. Experimental activities in Germany, China, the Russian Federation and Japan have shown that ceramic coatings can considerably increase the corrosion resistance of graphite. At the Technical University in Aachen and the FZJ Research Center Jülich, Germany, a successful coating method has been developed which is a combination of silicon infiltration and slip casting methods to provide a SiC coating on the graphite. Corrosion tests have been conducted simulating accident conditions (massive water and air ingress) at temperatures to 1200°C. Future efforts are required to examine the behaviour of the ceramic coatings especially with neutron irradiation. Activities at INET have involved forming SiC coatings on graphite structures by exposing them to melted silicon. Oxidation experiments have shown very large reduction in oxidation rate compared to uncoated graphite. Other activities at INET have shown that addition of superfine SiC powder to the fuel element matrix graphite greatly reduces graphite oxidation because SiO_2 is formed by SiC-oxygen reaction thereby partly covering and isolating the graphite micropores from further corrosion. Demonstration of the high resistance to oxidation by air or water of SiC coating on graphite surfaces including successful tests on irradiated structures could result in advantages from a public acceptance point of view as well as a technical point of view for the future design of HTGRs.

The close examination of experience presented to the Technical Committee led to the conclusion that plant safety is not compromised for design basis accidents. Continued efforts to validate the predictive methods against experimental data are worthwhile. Protective coatings for fuel and graphite components which provide high corrosion resistance should continue to be developed and tested as these potentially could provide assurance of safety even for very extreme and hypothetical water or air ingress accident conditions [10-1].

10.2.2 TCM on development status of modular HTGRs and their future role

The IAEA TCM on Development Status of Modular HTGRs and their Future Role was hosted by the Netherlands Energy Research Foundation (ECN, now NRG) in Petten on the occasion of the workshop on the Role of Modular High Temperature Reactors in the Netherlands, in November 1994. The meeting included a review of national GCR programmes and experience from operation of GCR's, status of advanced HTGR designs and predicted safety and economic performance, and future prospects for advanced HTRs and the role of national and international organizations in their development. The proceedings of this TCM were published by the Netherlands Energy Research Foundation.

10.2.3. Specialists meeting (SPM) on graphite moderator life cycle behaviour

Graphite has played an important role as a moderator and major structural component of nuclear reactors since the beginning of atomic energy programmes throughout the world. Currently there are many graphite moderated reactors in operation which will continue to produce power for many years to come. Also, there are graphite moderated reactors currently under construction and others in the design stage.

The meeting on Graphite Moderator Lifecycle Behaviour, held at the University of Bath, United Kingdom, was the fourth in the series of meetings on graphite technology convened within the frame of the IWGGCR. Previous SPMs included the subjects of the mechanical behaviour of graphite for HTGRs, graphite component structural design, and the status of graphite development for GCRs. A prominent theme throughout this SPM was the overall concern that the number of experts in nuclear graphite technology has diminished substantially in the past few years. Also, significant changes have taken place worldwide which are bringing about improved relations between countries, and this meeting represented an opportune forum for the current graphite experts to meet and to discuss the nearly forty years of operating experience of graphite moderated plants.

The purpose of the meeting was to exchange information on the status of graphite development, on operation and safety procedures for existing and future graphite moderated reactors, to review experience on the influence of neutron irradiation and oxidizing conditions on key graphite properties and to exchange information useful for decommissioning activities. The programme included topics from the conception of the reactor design through the safe operation and monitoring of the core to the removal and safe disposal of the graphite cores at the end of life.

A number of actions were initiated as the result of this SPM. Due to the strong interest and need expressed from the experts, a plan for the archiving of irradiated nuclear graphite data for the common use of all Member States was developed and subsequently presented to the IWGGCR. This recommendation was the result of a growing need to share in the scientific progress and operational experience on the influence of neutron irradiation of graphite; particularly now with nearly forty years of reactor operational history and growing concerns that the number of experts in this field is diminishing significantly. Follow-up actions of the IWG included the initiation of an inventory of data available from each Member State and development of a data indexing system to obtain consistency of information between countries.

Subsequently, the HSE of the UK, because of the safety implications associated with the structural and other property changes which occur with irradiation of graphite, initiated a follow-on study concerning this archive. In the course of this investigation it became obvious that this graphite data was also valuable for the fusion programme in their design considerations of the properties of carbonaceous plasma-facing reactor materials. Creating a database on irradiated nuclear graphite was also an on-going subject for review by the IWGGCR. In late 1966, the UK Health and Safety Executive (HSE) notified the IAEA of the intent to establish and fund this database for the archiving of data related to the UK GCR programme, and to also explore the possibility of the database being expanded to include other Member States with the final intent to eventually internationalize and place it within the IAEA's nuclear data system. HSE then selected Bath University to undertake archiving of the UK reactor data. Due to the advantages in sharing research and operational data on nuclear graphite for future HTGR programmes, representatives from China, Germany, Japan, the Netherlands, the Russian Federation, South Africa, UK and the USA indicated interest in this project. With this interest and pledges of limited funding support, the IAEA's Nuclear Data and Nuclear Power Technology Development Sections (NPTDS) jointly hosted a Consultancy in Vienna in early 1998 to define the scope and issues related to the establishment and maintenance of the database on an international basis. This database is being established within the Nuclear Data Section.

Another activity undertaken by the IAEA as the result of this SPM is to support the publication of the publication on Irradiation Damage in Graphite due to Fast Neutrons in Fission and Fusion Systems. This publication, which is currently being finalized, represents a comprehensive reference on the changes in dimension and physical properties that graphite components of nuclear reactors undergo when irradiated by high neutron doses.

10.2.4. TCM/workshop on design and development of GCRs with closed cycle gas turbines

The TCM and Workshop on Design and Development of Gas-cooled Reactors with Closed Cycle Gas Turbines was convened within the frame of the IWGGCR, and was held at the INET, Tsinghua University, in Beijing, China. The purpose of the meeting was to provide the opportunity to review the status of design and technology development activities for HTGRs with closed cycle gas turbines, and especially to identify development pathways which may take advantage of the opportunity for international cooperation on common technology elements.

This meeting was very timely as recent advances in turbomachinery, magnetic bearings and heat exchanger technology provide the potential for a quantum improvement in nuclear power generation economics by use of the HTGR with a closed cycle gas turbine. This plant offers the ability to achieve highly efficient generation of electrical power (~48% net) and a high degree of safety based on the inherent features of the HTGR. Areas of investigation at the TCM and Workshop included materials development, component fabrication, qualification of the coated fuel particles and fission product behaviour in the power conversion system, and status and projected activities of national HTGR development and testing programmes including gas turbine related systems and components [10-4]. Enhanced international cooperation among Member States was emphasised to facilitate development of the HTGR/closed cycle gas turbine plant. Technical areas in which this international cooperation was addressed included system fabrication technology and qualification of the coated fuel particles, materials development and qualification, and development and testing of turbo-machines, magnetic bearings and heat exchangers.

10.2.5. Advisory group meeting (AGM) on non-electric applications of nuclear energy

Over half of the world's primary energy consumption is used for the production of hot water, steam and heat for various industrial applications. This requirement is met almost exclusively through the use of fossil fuels. The emission of carbon dioxide and other gases resulting from the burning of fossil fuels poses serious challenges to our climate. Nuclear energy which, to date, has been used almost exclusively in the production of electricity, has the capability of being a significant, clean alternative to fossil fuels for the production of hot water, steam and industrial process heat applications. It was with this focus that the IAEA convened the AGM on Non-electric Applications of Nuclear Energy, which was hosted by the National Atomic Energy Agency of Indonesia in Jakarta.

The AGM brought together a group of international experts to review and assess the present status and recent progress made in the development of systems and processes for non-electric applications of nuclear heat. The technical and economic potential of these systems and processes along with their related environmental and safety issues and requirements were explored and areas were identified for additional research and development necessary before they can be commercialized.

Although unanimous agreement did not exist for international focus on a specific single heat utilization process, a majority consensus felt that the future high temperature application of nuclear energy should be directed to the development of hydrogen production techniques including steam and carbon dioxide reforming of methane. The need to move away from the burning of fossil fuels and to the use of hydrogen is felt by many to be a worldwide requirement for the future. The lower temperature application of desalination was also selected. Other heat utilization processes such as heavy oil recovery, district heating and coal gasification and liquefaction should also receive international consideration.

The participants expressed the need for greater cohesion between national programmes. The prominent issues selected to be addressed in a collective international forum included evaluation of economic and technical feasibility of the total plant and the determination and resolution of the risks and associated safety in combining the different chemical processes with a nuclear heat source [10-5].

10.2.6. TCM on HTGR technology development

The successful introduction of an advanced nuclear power plant programme depends on many key elements. It must be economically competitive with alternative sources of energy, its technical development must assure operational dependability, the support of society requires that it be safe and environmentally acceptable, and it must meet the regulatory standards developed for its use and application. These factors interrelate with each other, and the ability to satisfy the established goals and criteria of all of these requirements is mandatory if a country or a specific industry is to proceed with a new, advanced nuclear power system. It was with this focus on commercializing the HTGR that the IWGGCR recommended this TCM on HTGR Technology Development [10-6]. This meeting brought together key representatives and experts on the HTGR from the national organizations and industries of ten countries and the European Commission. The state electric utility of South Africa, ESKOM, hosted this TCM in Johannesburg. This TCM provided the opportunity to review the status of HTGR design and development activities, and especially to identify international cooperation that could be utilized to bring about the commercialization of the HTGR.

Over the past few years, many Member States have instituted a re-examination of their nuclear power policies and programmes. Significant worldwide activities are occurring in the advancement of HTGR development with regard to the utilization of this reactor to achieve high efficiency in the generation of electricity and in process heat applications. The state electric utility of South Africa, ESKOM, has performed a technical/economic evaluation of a helium cooled pebble bed modular reactor directly coupled to a gas turbine power conversion system for consideration as capacity additions to their electric system (see Chapter 3 for details on the PBMR). Also, work is proceeding on HTGR test reactors in China and Japan which will provide the capability to qualify components and heat utilization processes to temperatures of 950°C. Technological advances in component design and processes such as heat exchangers, turbomachinery, reformers and magnetic bearings coupled with the international capability to fabricate, procure and test these components has provided an excellent opportunity for achieving economic commercialization of the HTGR. Significant areas for commercializing the HTGR were reviewed in this TCM. These included initial plant capital costs and the current state of plant system and component development. It was acknowledged that the on-going economic evaluations for both the PBMR and the GT-MHR programmes are estimating plant capital costs approaching US$ 1000/installed KW(e). The reasons provided for this low cost include a very simplified modular plant design, licensing requirements commensurate with the safety attributes of the HTGR, high plant process efficiency, and competitive procurement of components on an international basis. Also acknowledged for the PBMR is that the current international technology base allows for the design and construction of a full scale prototype plant without the requirement for additional extended research and development.

10.2.7. TCM on gas cooled reactor decommissioning, fuel storage and waste disposal

The purpose of this TCM was to provide an opportunity to review the status of GCR decommissioning and associated spent fuel storage and component waste disposal programmes and related issues including facilities sharing common technological aspects such as other types of reactors which have graphite moderators; and especially to identify pathways which may take advantage of the opportunity for international cooperation on developments addressing these activities. The meeting was hosted by FZJ, Juelich, Germany, and focused on those aspects of decommissioning which are unique and distinctive to GCR plants rather than on generic nuclear power plant decommissioning processes and storage and disposal experiences which are already well understood.

The majority of the presentations addressed decommissioning activities associated with the HTGR and Magnox plants. Some of these plants are shut down and undergoing selected equipment dismantlement with the chosen decommissioning option of safe store for times ranging from 20 to 135 years. Specific to the HTGR, there are three plants in varying stages of decommissioning; the German AVR, the THTR-300 and the FSV station in the USA Defueling of the AVR is now partially complete, with approximately 23% of the (pebble bed) fuel removed from the core. A strategy has been developed for dismantling of the plant which began with the turbine hall and other outside equipment and will progress to include dismantling of the reactor vessel, and, possibly, dismantlement of the containment with the goal of final restoration to "green field" status in 2011. The decommissioning project plan includes the ability to place the plant into the state of safe enclosure (SAFSTOR) following the completion of each dismantling step. Relative to the German THTR-300, defuelling was completed in 1995. This operation required approximately one year to accomplish and included weekly spent fuel shipments to the Ahaus fuel interim storage facility. Establishment of the THTR-300 to SAFSTOR was then initiated for an intended time frame of 30 years.

This will be followed by completion of the remaining dismantlement activities. The FSV plant was shutdown in 1989 after ~10 years of commercial operation. The initial activity was to construct a modular dry vault storage facility and then de-fuel the reactor to this facility. After assessing the different decommissioning options available to nuclear power plant license holders in the USA, the plant owner, Public Service Company of Colorado, chose to proceed with early dismantlement. This decommissioning method was selected in order to eliminate long term financial risks and mitigate extended environmental impacts to the company, its customers and neighboring communities. The major effort in early dismantlement was the removal of the primary system internals and subsequent cutting and sectioning of the prestressed concrete reactor vessel using diamond impregnated wire cutting equipment. Internal vessel component removal was accomplished through the use of underwater dismantlement techniques. Flooding of the reactor vessel provided an excellent radiological exposure shield for the workers who often performed their tasks underwater in diving equipment. On 5 August 1997, the US Nuclear Regulatory Commission determined that the plant site is available for "unrestricted use" and subsequently terminated the plants' reactor license.

The preferred decommissioning strategy in the UK for the Magnox stations is "safestore". This strategy allows for a stepwise approach beginning with defuelling, then a period (~35 years) of "care and maintenance" and ending with deferral of final dismantlement of the reactors for an additional ~100 years to obtain the benefits of radioactive decay. Specific site and equipment decommissioning activities would take place at discrete intervals during the Safestore period. There are three twin Magnox power stations which have now been defuelled and their fuel removed from the plant sites. Initial efforts have been directed to the decommissioning pre-planning processes of defining the organizational needs and requirements, establishment of regulatory processes and the development of procedures to allow for the actual work to be implemented in a safe, efficient and economical manner. Included in this pre-planning is the necessity to show that significant degradation of the physical, mechanical and chemical properties of the graphite moderator will not occur over the Safestore period. Among the areas under examination are the rate of chemical oxidation, formation of explosive dusts, consequences of accidential exposure to moisture, the potential for gas-phase and particulate release including the biological degradation of the graphite [10-7].

Areas of common interest and discussion by participants at the TCM were predominantly centered on the treatment and disposal of the graphite components that constitute the major volume of the core and the requirements related to the handling and final disposition of the spent fuel.

10.2.8. TCM on HTGR applications and future prospects

Within the nuclear power industry, the HTGR has the unique capability of being able to produce core outlet temperatures close to 1000°C. Use of the HTGR for the production of electricity and for high temperature process heat applications has long been recognized as an excellent energy source for the future. This advanced nuclear power plant, with its safety, environmental, economic and technical attributes, has the capability of achieving a net electrical generation efficiency close to 50% through the incorporation of the Brayton cycle in a closed cycle gas turbine configuration. It is also under consideration as an energy source in selected high temperature process heat applications such as the production of hydrogen and the reforming of methane to produce methanol.

It was with the focus of reviewing the status of HTGR development activities and to explore pathways for international cooperation in their advancement that the IWGGCR recommended that the IAEA sponsor this TCM. This meeting was subsequently convened in Petten, Netherlands, and hosted by the National Research Foundation, ECN.

The TCM consisted of presentations on national and international activities associated with high temperature applications and future prospects of the HTGR with specific subject areas centering on design and construction status of HTGRs, status of research and development programmes on HTGRs and related technologies, system and component development activities in support of high temperature nuclear heat utilization and safety, licensing, regulatory and quality assurance aspects of related nuclear applications [10-8]. The proceedings of this TCM were published by ECN as ECN-R-98-004.

10.2.9. TCM on safety related design and economic aspects of HTGRs

The HTGR provides the capability for high temperature energy to generate electricity and for industrial process heat applications such as the production of hydrogen, with a high degree of safety and with capital installation and operation and maintenance costs which are economically competitive with other major energy sources such as coal. This TCM was hosted by the INET of Tsinghua University in Beijing, and included participants from national organizations and industries of ten countries. The TCM provided the forum for participants of Member States to discuss and share the status of individual programmes associated with research, development and commercialization of the HTGR, and to explore opportunities for international cooperation in realizing the potential of this advanced nuclear plant.

Development activities associated with the HTR-10 and HTTR test reactors and the HTGR gas turbine designs by ESKOM in South Africa, ECN of the Netherlands, JAERI of Japan and the consortium of designers from the Russian Federation, the USA, France and Japan for the GT-MHR were addressed. Major focus of the TCM also included research activities of these countries and Germany, Indonesia and the UK in support of the development of the HTGR for future high temperature applications.

10.3. OTHER IAEA ACTIVITIES IN SUPPORT OF HTGR TECHNOLOGY DEVELOPMENT [10-3]

In addition to the CRP and meeting activities discussed previously, the IAEA has supported the development of HTGR technology through production of additional documents, establishment of a database for irradiated graphite and a pending new CRP as summarized below.

10.3.1. Hydrogen as an Energy Carrier and its Production by Nuclear Power (IAEA-TECDOC-1043)

This report was developed under contract to the IAEA, and documents past activities as well as those currently in progress by Member States in the development of hydrogen as an energy carrier and its corresponding production through the use of nuclear power. It provides an introduction to nuclear technology as a means of producing hydrogen or other upgraded fuels and to the energy carrier hydrogen and its main fields of applications. Emphasis is placed on high temperature reactor technology which can achieve the simultaneous generation of electricity and the production of high temperature process heat.

10.3.2. Irradiation damage in graphite due to fast neutrons in fission and fusion systems

This report was developed by with the joint support of the IAEA and the UK HSE to document accumulated information and to provide an understanding of the results from research on the subject of radiation damage in graphite. Topical areas addressed include fundamentals of radiation damage in graphite due to energetic neutrons; the structure and manufacture of nuclear graphite; dimensional changes in graphite and the thermal expansion coefficient; stored energy and the thermo-physical properties of graphite; mechanical properties and irradiation creep of graphite; the electronic properties of irradiated graphite; pyrocarbon in high temperature nuclear reactors; and radiolytic oxidation in graphite.

10.3.3. International database on irradiated nuclear graphite properties

In conjunction with support from Japan, South Africa and the UK, the IAEA is establishing an international database on irradiated nuclear graphite properties. The objective of the database is to preserve existing knowledge on the physical and thermal mechanical properties of irradiated nuclear graphites, and to provide a validated data source for all participating Member States with interest in graphite moderated reactors or development of HTGRs, and to support continued improvement of graphite technology for applications. The database is currently under development and includes a large quantity of irradiated graphite properties data, with further development of the database software and input of additional data in progress. Development of a site on the Internet for the database, with direct access to unrestricted data, is also currently in progress.

10.3.4. Pending CRP on conservation and application of HTGR technology

During the past several meetings, the IWGGCR has been concerned with preservation of HTGR technology developed in earlier progranunes (e.g. in Germany, the UK and the USA), and its application in support of future programmes (e.g. in China, Japan, and South Africa). Formation of a new CRP is in process to address the recommendations of the IWGGCR and related consultancy meetings regarding these concerns. The objective the CRP will be to identify research needs and exchange information on advances in technology for a limited number of topical areas of primary interest to HTR development, and to establish, within these topical areas, a centralised coordination function for the conservation of HTGR know how and for international collaboration, utilizing electronic information exchange, data acquisition and archiving methods. The topical areas identified include high temperature control rod development, research and irradiation testing of graphite for operation to 1000°C, R&D on very high fuel burnup, including plutonium, qualification of pressure vessel steels to 500°C, R&D and component testing of high efficiency recuperator designs, and materials development for turbine blades up to 900°C for long life creep. The CRP is expected to begin in 2000 and be completed in 2005.

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Chapter 11

CONCLUSIONS

Development of the modular HTGR is expected to achieve the goals of being a safe, efficient, environmentally acceptable and economic high temperature energy source for the generation of electricity and for industrial process heat applications such as the production of hydrogen. All HTGRs incorporate graphite moderated, helium cooled cores with ceramic coated fuel particles capable of handling temperatures to 1600° C. The most current HTGR designs possess the capability of continuous operation at average core outlet temperatures between 900 and 950°C.

Among the advances currently under investigation by Member States of the IAEA is the closed cycle gas turbine HTGR concept which (through utilization of the Brayton Cycle) exhibits the capability of achieving net plant efficiencies in the range of 47%. In the area of high temperature process heat, the HTGR is being considered as the energy source for industrial applications including steam and CO_2 reforming of methane for the production of hydrogen as a fuel and/or its subsequent synthesis to other fuels such as methanol. Utilizing the HTGR as the energy source eliminates the need to burn fossil fuels in order to gain the heat required for these industrial processes to occur.

The high level of safety for the modular HTGR can be ensured through core specific structural features and neutron/physical properties as well as technical solutions applied in the design of the plant including the following:

(1) Use of helium as the primary coolant: The helium coolant is chemically inert and does not influence the neutron balance in the reactor due to essentially "zero" neutron absorption and scattering cross sections. This excludes the possibility of uncontrollable increase in reactor power for coolant density variation or loss of coolant in emergency situations. The single phase state of the helium coolant (no liquid to gas transformation) excludes a potential for abrupt changes in heat removal conditions in the core. Low induced activity of helium also causes a low level of radiological consequence with leakage during normal operation and in primary circuit loss of integrity events.

(2) Design for a negative temperature coefficient of reactivity: A core temperature increase caused by an unbalance between heat generation and removal from the reactor automatically leads to a reactor power decrease. This property ensures the self limitation of a fuel temperature rise in the event of an emergency such as loss of heat removal from the reactor or an emergency condition where the reactor shutdown mechanism(s) fail to actuate.

(3) Use of graphite as a core structural material that is capable of withstanding high temperatures (graphite sublimation temperature is ~3000°C): The core graphite structures retain strength at temperatures which significantly exceed the level which is possible under emergency conditions. This property provides assurance of core configuration stability under emergency conditions.

(4) Core and reactor features such as incorporation of a very low power density and corresponding high heat capacity: This, in combination with the steel vessel and reactor system geometry, ensure cool-down of the shutdown reactor under emergency conditions by passive removal of heat from the reactor vessel using heat emission, conductivity and convection, thus maintaining the fuel and core temperature within allowable (for safe operation) limits in all accidents including loss of coolant.

(5) Use of fuel in the form of small particles with multi-layer (TRISO) coatings made of PyC and SiC, which are capable of effectively retaining fission products at high temperatures and fuel burn-up.

The assessment of the capability of the US modular HTGR to control accidental radioactivity releases shows that the doses are a small fraction of the U.S.10CFR100 requirements even for the bounding analyses which consider only the systems, structures and components that require neither operator action nor other than battery power. In fact, the exposures are so low that the protective action guidelines would require no evacuation or sheltering plans for the public as specified in the utility/user requirements. The evaluation confirms that accident dose criteria can be met with a containment system that places primary emphasis on fission product retention within the fuel barriers [11-1].

The emphasis for utilization of the modular HTGR to produce electricity has exhibited a significant change in the past decade. Prior to the 1990s, the HTGR incorporated a steam cycle balance of plant. The focus is now on the modular HTGR coupled to a gas turbine power conversion system for the generation of electricity. This promising alternative for the utilization of nuclear energy to produce electricity is expected to result in achieving a substantial increase in net plant efficiency with a corresponding reduction of capital and O&M costs through plant simplification when compared to NPPs with a steam cycle. This plant is now achievable due to recent technological advancements in the design of magnetic bearings, turbo-machines and compact plate-fin recuperators.

A consortium consisting of organizations belonging to the Russian Federation's MINATOM and companies from the USA (General Atomics), France (FRAMATOME) and Japan (Fuji Electric) are combining their experience and knowledge for the development of the GT-MHR. Conceptual design has been completed on this 600 MW•t/293 MW(e) plant which is currently under development for the destruction of weapons plutonium, but with the longer term goal of commercial deployment utilizing uranium as the fuel. The plant features a prismatic block annular modular HTGR core as the energy source to supply a closed cycle gas turbine power conversion system. The next stage of development is in preliminary design of the plant. Most of the design work on the GT-MHR is being performed within the nuclear organizations of the Russian Federation with financial and management/technical support from all members of the consortium. A recent significant development in the advancement of the GT-MHR is the authorization of limited financial support by the US government on a matching resource basis with MINATOM. Although the GT-MHR is initially to utilize a weapons plutonium fuel cycle which has the capability of achieving a net consumption of Pu-239 of 90-95%, the versatility and flexibility of this core will allow for the application of a wide range of diverse fuel cycles. Fuel derived from uranium, thorium and a variety of plutonium grades is under consideration for long term applications in the GT-MHR [11-3].

In 1997, ESKOM, the state utility of South Africa, completed a technical and economic evaluation of the PBMR as a power generation source for future additions on their electrical grid. ESKOM's determination was that the PBMR represents the nuclear option of choice, as well as a viable and attractive investment opportunity for commercialization. The conceptual design of this plant is now largely complete and the basic design process is underway. The plant features a pebble bed modular HTGR of 265 MW•t coupled to a power conversion system utilizing the closed cycle gas turbine for an electric output of 117 MW(e) per module.

The projected advantages of the PBMR as a power source for the electrical system of South Africa include the capability for distributed generation, particularly along the coast, short construction time, small unit size with excellent load following capability, low environmental impact and economic projections of being competitive with power generation using coal. Commercially, the PBMR is seen to be a viable and attractive investment opportunity in that there is good technical support capability within South African industry, a non-prescriptive nuclear licensing environment, a cost structure for power generation that imposes a strong cost cap, and the backing of a very large utility with credibility and a good public image.

Both the PBMR and the GT-MHR are anticipating plant capital and operating costs significantly below those being experienced by nuclear power plants recently placed in operation or currently under construction. Upon commercialization, the PBMR is projecting a current capital cost in South Africa of less than US$1000 per installed kW with a base load operating cost of <1.7cents/kW•h based on 6% discount and a 40 year plant life. The GT-MHR projects a capital cost of US$928 million (<US$900 per kW installed) with a generation cost of ~1.3cents/kW•h for a four module n^{th} of a kind plant. The economic evaluations of these two plants were derived independently from each other and result from the simplicity of plant design, modularized construction and the high plant efficiency achievable with the closed cycle gas turbine.

The modular HTGR gas turbine plant relies on state-of-the-art technology, particularly with regard to the components within the power conversion system. Both the PBMR and GT-MHR will utilize magnetic bearings on their respective turbo-machines. Compact recuperators will also be key to achieving high cycle efficiency. Although the technology exists on these components, incorporation into the GT-MHR and PBMR will result in the largest application of magnetic bearings and recuperators ever undertaken. Also, coupling these components to an advanced nuclear heat source compounds the challenges facing the designers. However, the designers of both plant concepts have subjected their designs to numerous independent technical evaluations. The general conclusion from these reviewers is that the plant designs contain “no technical show-stoppers” which would result in the plant failing to meet the objectives of the designers.

A necessity in the development process for assuring the capabilities and safety of the modular HTGR is the ability to test and evaluate actual system and component performance and safety. Interest in the HTGR as an advanced nuclear power source for co-generation applications of electricity production and high temperature heat for industrial processes has resulted in the construction of the HTTR by JAERI and the HTR-10 by INET. These nuclear test facilities have the capability of achieving core outlet temperatures to 950°C and 900°C, respectively, and will be utilized to support research and development activities including validation of HTGR safety and general performance characteristics, electricity generation and validation of high temperature process heat applications [11-3].

Construction of the HTTR at JAERI’s research facilities in Oarai, Japan, began in 1991 with first criticality achieved on 10 November 1998. This plant includes a 30 MW•t graphite-moderated, helium cooled reactor comprised of hexagonal fuel elements with TRISO coated fuel particles in compact form. The HTTR will be utilized to establish and improve on HTGR related technology, for the performance of innovative research, as a test facility for fuel and materials irradiation, and to demonstrate process heat applications such as hydrogen production. Relative to commercialization of the closed cycle gas turbine HTGR plant, the HTTR will be a valuable facility in demonstrating the safety features of the HTGR, performance and qualification of materials and components and in fuel and fission product behavior.

The HTR-10 test module is a major project in the Chinese National High Technology Program and is currently under construction at the INET research site northwest of Beijing. Construction of the plant is anticipated to be completed in 2000, with initial criticality scheduled shortly thereafter. The HTR-10 is a pebble bed 10 MW•t HTGR. The objectives to be achieved with this plant include the evaluation of the design, operation and safety aspects of the HTGR, to test and validate its co-generation capabilities including the gas turbine for generation of electricity, to support the development of nuclear process heat applications, and as an experimental and irradiation facility for HTGR related component and system evaluation [11-3].

The HTTR and HTR-10 test reactors are being provided by JAERI and INET, respectively, in support of national gas cooled reactor programmes. Chief scientific investigators from participating Member States are evaluating the safety and performance characteristics of these two test reactors within the IAEA Co-ordinated Research Programme (CRP) on Evaluation of HTGR Performance. The HTTR was also provided to participating Member States for the CRP on Design and Evaluation of Heat Utilization Systems for the HTTR.

The Member States of the IAEA's IWGGCR provide mutual support and close collaboration in the R&D associated with the HTGR as an advanced energy source of the future. This interrelationship is very evident in the design of the GT-MHR and PBMR. The GT-MHR includes organizations and industry from the Russian Federation, the USA, France and Japan. The PBMR is being fostered by ESKOM of South Africa and includes technical support from Germany, where the original pebble bed design was developed. Other technical support of the PBMR includes NRG of the Netherlands, AEA Technology of the UK, OKBM and RRC Kurchatov of the Russian Federation.

In addition to the international collaboration between Member States on the GT-MHR, PBMR, HTTR and HTR-10, significant individual activities in HTGR R&D are taking place in the world.

In the Netherlands, NRG (ECN Nuclear Research) is developing a conceptual design of a 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 MW•t pebble bed HTGR to produce 14 MW of electricity and 17 tonnes of 10 bar, 220°C steam per hour [11-4]. NRG is developing the ACACIA plant because of their belief that the HTGR, with its favorable characteristics of safety and simplicity, can become a competitive heat source for co-generation application. Current activities at ECN regarding this plant and the HTGR include research in the areas of core physics (including core neutronics, thermal hydraulics, PBMR shielding and dynamics of the core and PCU, and HTR plutonium burning), waste behavior during long term storage, thermal hydraulic safety analysis on the PBMR and HTGR co-generation energy conversion [11-7].

In China, with the HTR-10 presenting a new generation of nuclear power plant, extensive development efforts by INET are being pursued in the areas of HTGR safety and in R&D of the plant's components and systems. INET is also involved in the development of future applications of the HTGR including the desalination of seawater. In this regard, a technical/economic analysis has been developed to assess diverse desalting systems. Initial comparative analyses has shown the steam turbine-MED as the favorable choice for the dual production of potable water and electricity [11-3].

In the USA, the Massachusetts Institute of Technology and the Idaho National Engineering and Environmental Laboratory are developing a modular HTGR plant with the reference reactor design from the ESKOM PBMR, but with a significantly different balance of plant. Associated with the development activities related to the GT-MHR by General Atomics, the US Department of Energy has instituted a government programme to support this plant for the burning in the Russian Federation of weapons plutonium. Also, the US Electric Power Research Institute has initiated a HTGR R&D programme to enhance its technical capability to support member utilities in the development and potential deployment of this advanced nuclear plant as a competitive electricity generator.

The National Atomic Energy Agency of Indonesia established a HTGR programme in 1993 to conduct studies of the HTR technology and its application. The programme currently supports five general areas of HTGR development including reactor technology and optimization of electricity and steam co-generation, safety and environmental, coal liquefaction and desalination, instrumentation and control and the HTR fuel cycle.

Within Japan, the HTTR is the predominant focus for the R&D associated with the HTGR. However, numerous plant design activities are also being investigated. These plants are primarily under development by JAERI within the framework of the Japanese HTGR-GT feasibility study program, and include gas turbine cycle units with reactors of 400 MW•t, 300 MW•t, and two variations with a power level of 600 MW•t. The Japanese organization, RAHP, composed of members from private industry, the utilities and academia, are monitoring and supporting the HTGR and its commercial feasibility.

German technical resources are currently focusing on support of ESKOM with the design of the PBMR as well as the decommissioning and safe closure of the AVR and THTR 300 plants. Complementary to the commercial projects on evolutionary LWRs, additional generic R&D is being performed in Germany with the aim of identifying the fundamentals of innovative nuclear reactor technologies as being requested by the German Atomic Act of 1994, even for beyond design basis accidents. The proven safety characteristics and principles of HTGRs form a sound start-up basis for these considerations aiming to self-acting (passive) nuclear, thermal, chemical and mechanical stability of the primary circuit and an independence of the fission product barriers [11-5].

In France, CEA and Framatome have been most active on the HTGR. They have initiated an analytical research program to determine the best utilization of a GT-MHR for fissile material including plutonium, uranium-235, with or without thorium, in order to maximize fuel and core efficiency and minimize the stock of actinides and long life fission products. Coupled to these studies, the fuel management of the core is also being investigated starting with the traditional 18 month fuel cycle length to the feasibility of a 3 year fuel cycle length. Framatome, as a partner in the development of the GT-MHR, is contributing in technical areas that are directly linked to its experience and work in PWR design and construction. These include heavy component design, fabrication and installation, site erection and seismic conditions satisfaction and cost estimation.

Within the UK, the recent addition of BNFL as a Joint Venture partner in the holding company which governs the overall PBMR project will provide a major level of support in commercializing this gas turbine modular HTGR plant. Another significant area of support for the HTGR is in nuclear graphite performance with plant life and associated structural changes due to the effects of irradiation and temperature. The UK HSE, because of the safety implications associated with the structural and other property changes which occur with irradiation of graphite, has funded a programme for the archiving of UK irradiated nuclear graphite data. This has subsequently been expanded to the IAEA for the common use of all participating Member States. Major international support and development activities on nuclear graphite are focused at AEA Technologies plc.

Although the principle focus of the Russian Federation's current HTGR programme is with the development of the GT-MHR plant, extensive design capability and testing facilities are available for international R&D on the HTGR. Development of the HTGR was initiated in the Russian Federation approximately 30 years ago and have included designs for the VGR-50, VG-400, VGM and VGW reactor plants. These activities are providing a wealth of scientific resources and facilities now being utilized primarily for the GT-MHR, and also for R&D associated with the PBMR.

The principle HTGR development in South Africa is the PBMR. Significant utilization of international expertise and R&D facilities is a key aspect of this HTGR developmental project. The nuclear facilities at the South African AEC include a test stand for evaluation of small turbo-machines with magnetic bearings, the SAFARI material test reactor, hot cell facilities and a fuel fabrication facility which previously provided fuel for the PWRs at Koeberg. This facility is being made available for the transfer of fuel development technology for fabrication of coated fuel particles and spherical fuel elements for the PBMR.

Until recently, the Paul Scherrer Institute in Switzerland, performed significant work in the fields of high temperature materials and reactor theory [11-6]. Through 1996, the Institute provided the use of the PROTEUS facility for a series of critical experiments on low enriched fuelled small MHTGR type cores within the IAEA CRP on Validation of Safety Related Physics Calculations for Low Enriched Gas Cooled Reactors.

The sustainability and advancement of nuclear energy is being supported by the European nuclear industry through its "Michelangelo Initiative" and the "Safety-related Innovative Nuclear Reactor Technology Elements-R&D" funded through the European Union. With regard to the HTGR, the western European countries of the IWGGCR (France, Germany, Italy, the Netherlands and the UK) have formed a partnership (European Concerted Action on Innovative HTR) for the coordinated development of the HTGR which will make maximum use of available technology.

Advising the IAEA on international gas cooled reactor developmental activities is the primary function of the IWGGCR. As witnessed within this TECDOC, the developmental activities for the modular HTGR has been the result of very substantial collaborative efforts between Member States. The members comprising the IWGGCR have actively utilized this international forum to the advantage of sharing technical personnel and facilities for the common goal of developing the modular HTGR into an energy source for the future.

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