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Safety related design and economic aspects of HTGRs

Proceedings of a Technical Committee meeting held in Beijing, China, 24 November 1998

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FOREWORD

The high temperature gas cooled reactor (HTGR) is a promising energy source to help meet the needs of society in the twenty-first century. This advanced nuclear power reactor has the capability to provide high temperature energy for the generation of 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. It was with the objective of reviewing the status of international development activities associated with the safety related design and economic aspects of the HTGR that the International Working Group on Gas Cooled Reactors recommended that the IAEA convene this Technical Committee Meeting (TCM) on Safety Related Design and Economic Aspects of HTGRs, which was held from 2-4 November 1998 in Beijing, People's Republic of China. It was hosted by the Institute of Nuclear Energy Technology of Tsinghua University, 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 their individual programmes associated with research, development and commercialization of the HTGR, and especially to identify pathways which can provide the opportunity for international cooperation in realizing the potential of the HTGR.

The status information presented in the papers is as of the time of drafting. Some of this information has been superceded by material in a recently completed publication on Current Status and Future Development of Modular High Temperature Gas Cooled Reactor Technology, IAEA-TECDOC-1198 (2000). Thus, the first three papers which were presented on status of the HTTR, PBMR and HTR-10 projects have not been included in this report.

The IAEA officer responsible for this publication was J. Kendall of the Division of Nuclear Power.

EDITORIAL NOTE

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SUMMARY

The Technical Committee Meeting (TCM) on "Safety Related Design and Economic Aspects of High Temperature Gas Cooled Reactors (HTGRs)" was held from 2-4 November 1998 at Tsinghua University, Beijing, People's Republic of China. The meeting was hosted by the Institute of Nuclear Energy Technology (INET), and was convened by the IAEA on the recommendation of its International Working Group on Gas Cooled Reactors (IWGGCR). Approximately fifty participants and observers from ten countries (France, Germany, Indonesia, Japan, Netherlands, People's Republic of China, Russian Federation, South Africa, United Kingdom and the United States of America) attended the TCM. Twenty-four papers were presented with time set aside for questions and comments following each presentation. Tours of the INET research facilities and the High Temperature Reactor (HTR-10) construction site followed the meeting.

The purpose of the TCM was to provide the opportunity to review the status of design and development activities associated with safety related and economic aspects of HTGRs, and to identify pathways which may provide the opportunity for international cooperation in addressing these issues. The HTGR, as a nuclear heat source for the safe, economic and efficient production of electricity and high temperature industrial processes has, within the past few years, become a significantly increasing influence in the future of nuclear power. Nuclear test facilities with the capability of achieving core outlet temperatures to 950°C are presently under construction in the People's Republic of China and Japan. These plants will be utilized to support HTGR research and development activities, including electricity generation via the gas turbine and validation of high temperature process heat applications. Also, major development programmes focusing on the generation of electricity through the direct cycle gas turbine are in progress by ESKOM, the state electric utility of South Africa, and by a consortium of organizations from the Russian Federation, USA, France and Japan. Other national programmes focusing on research and development of the HTGR are underway including the Netherlands, where an evaluation is being completed on a heat and power co-generation plant utilizing a small direct cycle HTR; in Germany, where the primary focus is centered on basic issues of reactor safety and innovative reactor technology; in Indonesia with the evaluation of process heat applications such as coal liquefaction, hydrogen production and high temperature reforming of methane; and in the USA with the recent re-introduction of national support for the HTGR specifically directed to the burning of weapons plutonium.

The status information presented in several of the papers is as of the time of drafting. Thus other later material should be referenced for more current status information. One source of current information is the IAEA Gas Cooled Reactor Project web site at http://www.iaea.org/inis/aws/htgr/index.html, which also provides links to modular HTGR research and power reactor web sites. However, the technical information provided in the papers, which constitutes the majority of the information presented, remains valid.

The TCM was opened by Professor Z. Wu, Director of INET, and Mr. J. Luo on behalf of the IAEA. Professor Wu indicated that The People's Republic of China now has a population of approximately one and one quarter billion people. The electrical requirements necessary to support the future societal development needs for the people of this vast country are extensive.

Currently, approximately three-quarters of the energy supply in China is from coal. This has placed a burden on the environment and a high demand on the transportation infrastructure. The need for extensive nuclear power development within China is significant. Mr. Luo indicated the timeliness and importance of the TCM based on current and anticipated HTGR activities of Member States comprising the IWGGCR. He indicated that the meeting was convened as an activity of the IAEA in the application of its mission to foster the exchange of scientific and technical information regarding the peaceful use of atomic energy throughout the world. Specifically to the HTGR, he also reviewed the proposed activities within the Agency's gas cooled reactor project scheduled for 1999.

The HTR-10 Test Module is a major project in the Chinese National High Technology Programme and is currently under construction at the INET research site northwest of Beijing. Construction of the reactor building was completed in October 1997 and initial criticality is scheduled for late 1999. The objectives to be achieved with the HTR-10 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. The design of the HTR-10 is significantly influenced by the passive safety features of the HTGR. The safety philosophy for this plant deviated from the traditional approach for nuclear power plants which relies on redundant and diverse active components and systems including their power supplies. This test reactor will serve to demonstrate passive safety features and will be instrumental in support of future licensing and regulatory developments for the HTGR. As the HTR-10 is a new generation of nuclear power plant, there has been extensive development efforts by INET in its components and systems. Typical of this is evaluation of the safety characteristics of the core pressure vessel which was evaluated for the case of a large rupture accident and included studies on the effects of pressure loss and associated transient stresses. INET is also involved in the development of future applications of the HTGR including the desalination of seawater through the energy source of the HTGR. In this regard, a techno-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.

Construction of the High Temperature Engineering Test Reactor (HTTR) at the Japan Atomic Energy Research Institute's (JAERI) facilities in Oarai, Japan, is now complete. Initial fuel loading of this 40 MWt test reactor continues with first criticality achieved on 11 November 1998, with an annular core fuel configuration. Annular core physics tests will include control rod reactivity worth, scram reactivity, neutron flux distribution, excess reactivity and reactor noise analysis. Subsequent fuelling of the full core will then proceed followed by zero power physics tests including those similar to the annular core and also including tests to determine maximum reactivity insertion rate, excess reactivity, shutdown margin, nuclear power correlation and the temperature coefficient of reactivity. Rise-to-power tests will be conducted to confirm plant design, validate respective codes, demonstrate typical HTGR safety features and confirm the integrity of high temperature components.

A major safety component of the HTGR is the coated fuel particle. To initially fuel the HTTR, Japan's Nuclear Fuel Industries Ltd. established the fabrication technology and successfully produced approximately 900 kgU of coated fuel particles. Testing of this fuel resulted in the quality values of $8x10^{-5}$ for the SiC defective fraction and $2x10^{-6}$ for the bare uranium fraction. The safety evaluation of the HTTR required confirmation that there is no chance of core damage and that the barrier design against fission product release is appropriate. One requirement to assure that the maximum temperature of the reactor pressure boundary does not exceed permissible values was to demonstrate that the helium circulator will brake within ten seconds following a pressurized water cooler pipe rupture and reactor scram. Upon demonstration, the DC electrical braking system designed for the HTTR circulators allowed the pressurized water cooler temperature piping to reach 368 degrees °C, well within the accident limit of 650 °C.

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. JAERI currently has under design a hydrogen production system which will utilize steam reforming of natural gas with HTTR nuclear heat of 10 MWt and a core outlet temperature of 905°C. This project is beginning with an out-of-pile test facility to simulate key components on a scale of one to thirty.

The safety attributes, high thermal efficiency, flexibility of high temperature heat applications such as electricity generation and industrial processes, and low capital and operating costs are the key attributes in seeking commercialization of the HTGR. Two specific programmes are currently under design for the generation of electricity through the use of an HTGR coupled to a closed cycle gas turbine power conversion system. These include the Gas Turbine-Modular Helium Reactor (GT-MHR) and the Pebble Bed Modular Reactor (PBMR). A consortium consisting of organizations belonging to the Russian Ministry on Atomic Energy (MINATOM) and companies from the USA (General Atomics), France (FRAMATOME) and Japan (Fuji Electric) are combining their experience and knowledge in the HTGR and the design and fabrication of components such as recuperators and turbo-compressors for development of the GT-MHR. Conceptual design has been completed on this 600 MWt/293 MWe plant which is currently under development for the destruction of weapons plutonium, but with the longer term goal of commercial deployment. The next stage of development is in preliminary design of the plant which will begin early 1999. 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 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 burnup approaching 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.

ESKOM, the state utility of South Africa, has completed a detailed technical and economic evaluation of the PBMR as a power generation source for future additions on their electrical grid. ESKOM's determination is 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 it features a pebble bed HTGR of 265 MWt coupled to a power conversion system utilizing the closed cycle gas turbine for an electric output of 117 MWe per module. The projected advantages of the PBMR as a power source for the South African electrical system include the capability for distributed generation, particularly along the coast, short construction time, small unit size with excellent load following characteristics, 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 adequate technology within South African industry with a limited anti-nuclear movement and 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 of less than US$1,000 per installed kW with an operating cost of < 1.5cents/kWh 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 1.76 cents/kWh for a uranium fuelled four module commercial 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 unique safety characteristics of the HTGR, with its slow response to accident conditions and ability to rely on passive rather then active safety systems, is a significant area of continued development by virtually all Member States of the IWGGCR. A major reason for this includes the capability to significantly reduce plant complexity and corresponding cost without a reduction in safety to the public or environment. In this regard, Mitsubishi Heavy Industries is investigating possible changes in the safety and code class to allow expansion in the use of non-nuclear grade equipment in the plant. JAERI also continues to evaluate the safety characteristics of advanced HTGRs including the development of a code to model reactor dynamics and heat removal from the surface of the reactor pressure vessel for the 400 MWt severe accident free HTR (SFHTR). It has been analytically determined that no reactor melt will occur, even under the worst case design basis accident. Future plans are to perform the safety demonstration test in the HTTR as validation of this code.

The South African nuclear regulatory authority is currently in the first stage of safety review of the PBMR. The licensing approach envisioned for this plant would include a design basis which respects prevailing international norms and practices and a quantitive risk assessment in accordance with the fundamental safety standards of the South African Council for Nuclear Safety (CNS). As this is a new advanced nuclear power plant, it will be necessary to establish general design criteria and design rules in order to assure that the PBMR complies with CNS's currently established risk criteria, and that, as a minimum, the same degree of protection is afforded to the public, operator and environment as that required for the current generation of nuclear plants, including societal trends with regard to levels of safety.

Development activities incorporating the gas turbine are also underway in the Netherlands and Japan. The Nuclear Research and Consultancy Group is developing a conceptual design of an HTGR for combined generation of heat and power for industry. The Advanced

Atomic Co-generator for Industrial Applications (ACACIA) plant incorporates a 40 MWt pebble bed HTGR for the production of 14 MW of electricity and 17 tons of steam (10 bars, 220 degrees) per hour. Transient analysis development work in support of this plant is continuing and includes the calculational coupling between the (Panthermix) high temperature reactor code and the (Relap5) thermal hydraulics code for the energy conversion system. Evaluation of this coupling has resulted in a more realistic simulation of the entire plant system.

JAERI continues investigation of new designs for incorporation of the closed cycle gas turbine with an HTGR. The preliminary design of a 600 MWt plant utilizing the monolithic fuel compact, control rod sheaths of carbon/carbon composite material and development of a plate-fin recuperator model were carried out in 1997. Currently, design work on the smaller, 300 MWt direct cycle plant is in process. This plant includes an annular core pebble bed reactor and the incorporation of thermal insulation internal to the reactor pressure vessel to achieve a plant efficiency of approximately 50%. Investigation into improvements in advanced HTGRs are also being carried out within Japanese industry and educational organizations. Fuji Electric Co. and Tokai University have completed experiments to determine the fundamental operating characteristics for use of a heat pipe decay heat removal system. It was determined that by adopting the use of a variable conductance heat pipe, it is possible to have a fully passive decay heat removal design which would minimize heat loss during normal operation and achieve lower reactor vessel temperatures during accidents.

The National Atomic Energy Agency of Indonesia is in the process of developing nuclear design codes for fuel management and safety analysis of HTGRs. These codes include the simulation of flow of pebble fuel for the once-through-then-out HTGR core as well as multi-pass pebble bed fuelling schemes. Two codes have also been developed to investigate safety and accidents in a pebble-bed HTGR. These include simulation of core reactivity accidents and plant depressurization. Simulation of the severest accident, i.e., depressurization resulted in a fuel temperature of 1600°C in the upper core after approximately 10 hours, and approximately 1800°C at 23.5 hours, assuming no corrective actions are taken.

Graphite is a key component in all HTGR cores, both as a structural member and as a moderator. The design life of these reactors is governed by the ageing of the graphite core primarily because of fast neutron damage. Due to the polycrystalline nature of nuclear graphites, the life of a graphite component in a nuclear reactor can be related to the radiation-induced damage to the individual crystals. Nuclear graphites typically shrink with age and then a "turnaround" occurs resulting in swelling. The safety implications of these changes are very important and AEA Technology and the University of Manchester have developed a two-phase finite element model of a single graphite crystal to investigate this behaviour.

Analysis of international R&D trends and the development of associated strategies to achieve commercialization of the HTGR are of primary focus of the Research Association of HTGR Plant (RAHP). This organization represents the private and industrial sector of Japan for HTGR development and is currently investigating the fuel cycle aspects and the international programmes of small modular HTGRs. RAHP has shown considerable support for the Agency's GCR programme, particularly the R&D related activities and is currently helping in the development of international HTGR data telecommunications.

Coordination of international R&D activities, including the dissemination and archiving of information and data related to existing and past HTGR development work has been of primary importance to the IWGGCR. The sustainability and advancement of nuclear energy has been supported by the European nuclear industry through its "Michelangelo Initiative" and the "Safety-related Innovative Nuclear Reactor Technology Elements-R&D" funded by the European Commission. The western European countries of the IWGGCR 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. Advancements in electronic communication such as use of the internet and world wide web are also being utilized, with the HTGR now having its own "Homepage", as noted earlier. Collectively, these activities have helped form the basis for a pending new Coordinated Research Programme (CRP) on Conservation and Application of HTGR Technology, which is focused on HTGR related R&D and the conservation and application of HTGR technology.

This TCM provided a forum for participants from research organizations and industry of IWGGCR Member States to share the results of their individual programmes in the advancement of the HTGR as a future nuclear energy source for both the generation of electricity and as a high temperature process heat source for industrial applications such as the production of hydrogen. The major focus of this TCM was on the safety related design and economic aspects of the HTGR, and included research developments and a review of the major technological considerations necessary for the commercialization of the HTGR.

PRESENT STATUS OF THE HIGH TEMPERATURE ENGINEERING TEST REACTOR (HTTR)

(Session 1)

RECOMMENDATIONS FOR THE DEVELOPMENT OF HIGH TEMPERATURE REACTORS (HTRs)

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Abstract

The Concerted Action "Assessment of safety and innovative technology for an HTR power generating plant (INNOHTR)" is funded by the European Commission within its 4th RTD Framework Programme. It is a joint project of four European nuclear industry organisations and five European research centres. This paper presents the preliminary conclusions of the INNOHTR Concerted Action. It addresses existing experience in designing, building, and operating HTRs in the areas of fuel, reactor core, and plant systems. It also addresses the need for additional research and development to support forthcoming industrial prototypes as well as for long term advanced system development. Specific areas of development are recommended along with the need for international cooperation.

1 – Introduction

The Concerted Action "Assessment of safety and innovative technology for an HTR power generating plant (INNOHTR)" is funded by the European Commission within its 4th RTD Framework Programme. It is a joint project with nine partners, four of then coming from the European nuclear industry and five from European research centers, two of the latter being institutes of the Joint Research Center of the European Commission.

Its objectives are:
- to report on the state of the art of high temperature gas-cooled reactor (HTGR) technology;
- to assess the technology available for HTGR development, coming both from past HTGR activities and from other industrial areas (e.g., gas turbine technology, high temperature processes);
- to advise the European Commission about the R&D needs for the development of modern HTGRs.

The action began in January 1998, and will end in June 1999. In the meantime, the 5th Framework Programme of the European Commission will start in the beginning of 1999, and, as it appears that the Commission could support HTGR development as soon as possible, the INNOHTR partners are organising a coordinated response to the first call for proposals under the 5th Framework Programme, which will be issued early next year. This would mean that a European R&D programme on HTGR could start within one year from now.

This paper presents the preliminary conclusions of the INNOHTR Concerted Action.

2 – Achievements from Previous Experience in Designing, Building, and Operating HTRs

From the 1960s to the 1980s, several experimental HTGRs and prototype power plants were operated for a total of nearly 50 reactor-years (about 7000 f.p.d.). They proved the feasibility of HTGR technology for electricity generation with a steam cycle power conversion system. Among the different elements inherited from this experience, the main significant achievements concern the fuel and the core.

2.1. Fuel

The major achievement of previous experience with HTGR technology is probably the development of a very reliable coated particle technology, up to the industrial scale.

Different materials have been used for fuel fabrication (U, Th, and, on a small experimental scale, Pu) in different chemical forms (oxides, carbides, carboxides), and very low defect fractions have been achieved in particle production (down to 10^{-5}).

For U-Th fuel, a maximum burnup of 160 GWd/tHM (18 % FIMA) was reached in the German experimental AVR reactor, with fast flux up to 8 x 10^{21} n/cm^2 (E>0.18 MeV). During one-third of the operating time of this reactor (21 years), the temperature of helium was maintained above 900°C (up to 950°C), and therefore the maximum fuel temperature reached more than 1300°C. For plutonium, experience is much more limited, nevertheless a few plutonium particles have reached a burnup of 750 GWd/tHM (nearly 80 % FIMA) in the experimental Dragon reactor without any problem.

Load following has been widely tested in the German prototype THTR plant for load changes of 8 %/min over a power range of 25 %.

The major advantage of coated particle fuel is its leaktightness to fission products, both under normal and accident conditions. In normal operation, practically all the releases in the primary circuit are due to a small fraction of fissile material, which can be found outside the coated particles. The collective doses in the prototype plants, which were operated in United States and Germany, give evidence of good coated particle fuel performance, in terms of fission products release, far better than the records of LWRs.

Under accident conditions, there is no additional release, at least up to a fuel temperature of 1600°C. This is illustrated in Figure 1. And, as can be seen from this figure, above this temperature there is no "cliff effect." This was demonstrated in the first HTGR that ever worked, Dragon, in an 80-day heating experiment, during which the temperature of the fuel was kept at 1800°C. Even though the coated particle technology was, at that time, less sophisticated than now, only 1 % of the fission products were released.

2.2. Core

The HTGR core offers the advantage of accepting many different fuel materials and fuel cycles without any major design changes. This results from the decoupling between the parameters of the geometry of the coolant channels and the parameters of reactor physics optimisation (concentration and distribution of heavy nuclei, moderation ratio). An illustration of this ability of HTGRs to accept many different types of fuel can be seen in Table 1, giving the different types of fuel that have been irradiated in the AVR.

Pebble bed reactors offer one additional advantage in fuel management flexibility: the possibility of changing gradually the type of fuel on line, without stopping the reactor. This was experimented in the AVR, where HEU fuel was changed for LEU fuel during reactor operation.

But the most attractive characteristics of the HTGR core are its inherent safety features:

- a strongly negative temperature coefficient for any irradiation condition of the fuel,
- high thermal inertia,
- a large margin between the normal fuel operating temperature (below 1300°C) and the leaktightness limit of the coated particles.

With such characteristics, proper design of the reactor (limited power, possibly annular geometry of the core and use of passive heat transfer mechanisms under accident conditions - conduction, natural convection, radiative heat transfer) can result in keeping the

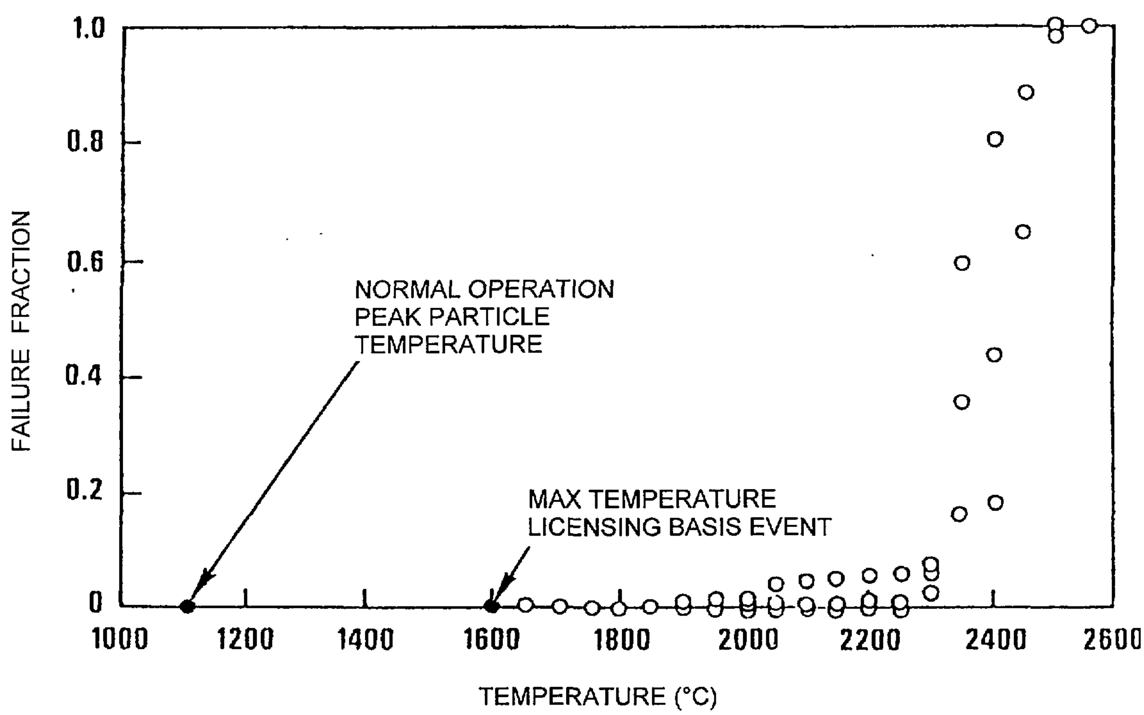

Figure 1 : failure rate of irradiated particle fuel

Table 1

Fuel elements (FE) inserted into AVR

FE type	Fuel	Particle coating	Fuel per FE (g)			Number of FE used in AVR
			U^{235}	U_{tot}	Th	
First core FE	(U,Th)C2	HTI-Biso	1.00	1.08	5	30155
Wall-paper FE	(U,Th)C2	HTI-Biso	1.00	1.07	5	7504
Carbide	(U,Th)C2	HTI-Biso	1.00	1.08	5	50840
Oxide	(U,Th)O2	HTI-Biso	1.00	1.08	5	72418
	(U,Th)O2	LTI-Triso	1.00	1.08	5	6083
(THTR-FE)	(U,Th)O2	HTI-Biso	0.96	1.03	10.2	34415
Separate feed and breed particles	UO2, ThO2	LTI-Biso	1.00	1.08	10	1440
		LTI-Biso				
	UO2, ThO2	LTI-Triso	1.00	1.08	10	1610
		LTI-Biso				
	UC2, ThO2	LTI-Triso	1.00	1.08	5	6067
		LTI-Biso				
	UC2, ThO2	LTI-Triso	1.00	1.08	5	5860
		LTI-Biso				
	UCO, ThO2	LTI-Triso	1.00	1.08	5	5363
		LTI-Triso				
Low enriched Oxide	UO2	LTI-Biso	1.40	20	-	2400
	UO2	LTI-Triso		1.00	10	24611
	UO2	LTI-Triso		1.00	6	29090

temperature of the fuel under its leaktightness limit during the most severe accident scenarios. This is the basic principle of the design of modern modular HTGR reactors. Experimental demonstration of the possibility of relying entirely on the intrinsic safety features of the reactor during an accident was provided by the AVR safety tests. In Figure 2, for example, temperatures in different reactor structures are measured during a test of a loss of coolant accident without action of any engineered safeguard system. Of course, the temperature of the fuel could not be measured continuously, even if the maximum temperature of some experimental pebbles could be checked with fusible wires. Nevertheless the highest temperature which could be recorded continuously was in a structure located near the center of the core. All these experimental data are compatible with a maximum fuel temperature significantly below 1600°C. The slowness of the transient is also to be noticed: more than 10 hours are necessary to reach the maximum temperature.

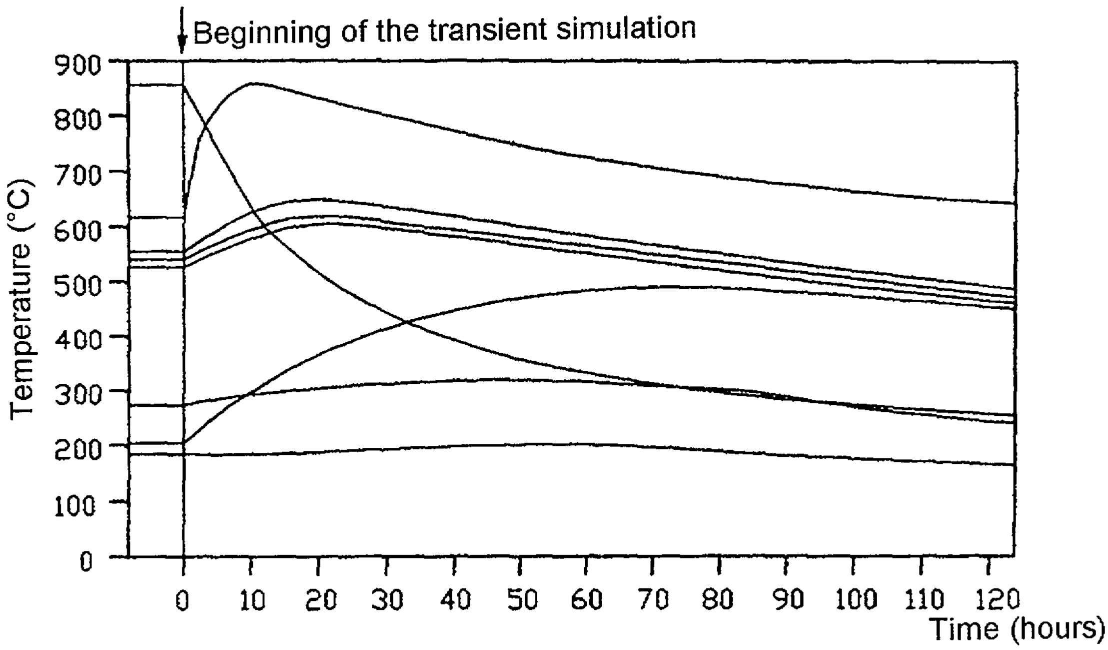

Figure 2 : experimental simulation of a LOCA in AVR

2.3. Systems

Achievements in operating the systems of HTGR reactors are more mixed, which is not surprising for experimental and prototypical facilities. For example, one of the permanent difficulties was with moisture ingress (at Fort Saint Vrain). The AVR even encountered a major water ingress (without any significant consequence).

These difficulties with the systems explain the mixed availability records of the HTGR facilities:

Peach Bottom	88 % for 7.5 years
AVR	67 % for 21 years
THTR	37 % for 5 years
Fort Saint Vrain	29 % for 6.5 years.

But none of the problems that had to be faced involved a basic threat to the viability of HTGRs. Moreover, many of these problems will not be encountered in modern HTGRs, which will use different technologies (direct cycle gas turbine, magnetic bearings, etc.).

3 – Why are There New R&D Needs for the Industrial Development of HTGRs?

3.1. Needs for the forthcoming HTGR industrial prototype projects

Even though the design of the next generation or HTGR prototypes (PBMR, GT-MHR, INCOGEN/ACACIA) will be mostly based on existing technologies, the feasibility and reliability of which have been widely proved, they have design options that are quite different from those of past reactors. Some of the new technologies still need further development, some need to be scaled up, and others just need to be validated.

The two main new features of modern HTGRs, compared to all the HTGRs that were built and operated in the past, are the use of a direct helium cycle for energy conversion and modular design.

Moreover, even if existing technologies are used, this does not mean that they are still available. Development of HTGR technology has been stopped for many years, and the danger of losing past knowledge is very great. The first and most urgent task for a new R&D programme on HTGR technology is to organise the conservation of past know-how, by opening up the archives and making them available through a modern database and by consulting with the experts who were involved in past projects on their experience in designing, building, and operating HTGRs. The global HTR Network (GTHRN), which is now under development, will be a very useful tool for performing this task.

The power conversion system (PCS), with a direct helium cycle, is not the result of a simple connection of a standard natural gas turbine to a nuclear reactor. It has specific features, which require many important adaptations and modifications from standard gas turbine technology:

- Contrary to the case of natural gas, the PCS of a HTGR works in a closed cycle, completely isolated from the atmosphere.
- The PCS is part of the reactor's primary system, which results in some radioactive contamination of its components and in the requirement for a very compact design, in order to be able to enclose it completely in a pressure vessel of a reasonable size.
- The reactor contains a large amount of graphite, so to avoid safety and availability problems, the use of water or organic fluids (oil) is to be prohibited in the primary system.

Due to these very specific features, some new problems have to be solved and some new challenges have to be taken up in designing the PCS and its components:

- There will be heavier loads on some components than in an open-cycle system, particularly for certain transients.
- The performance of a closed-cycle PCS will be more sensitive to deviations from the nominal characteristics of its components (which can be due to manufacturing or ageing).
- It is a challenge, due to its compactness and its enclosure in a pressure vessel, to design the PCS in such a way that its maintenance remains feasible and not too time-consuming. One part of the solution for the PCS maintenance can be found in designing components in such a way that they satisfy very high reliability requirements (e.g., operation of the turbine-generator during 10 years without maintenance).
- There is no industrial experience with designing and operating a generator in a confined and pressurised helium environment.

- To avoid the use of oil, the turbine-generator will make extensive use of magnetic bearings, coupled with catcher bearings. In some projects, with vertical shafts for the rotating machines, the load to be borne by the magnetic bearings is much higher than what is presently achieved in industry (in the GTMHR, for example, six times higher). Catcher bearings, which will not be operated very often, but, when operated, have to withstand very high loads without damage, are also a development challenge.
- The intricate circulation of helium with different temperatures and pressures in the PCS vessel requires splitting the volume inside the vessel into several compartments, separated by sliding seals (sliding in order to accommodate expansion due to varying temperatures in these compartments). The largest dimensions of these sliding seals and their most severe operating conditions (e.g., in the GTMHR project, maximum diameter 3.2 m, pressure difference 4.5 MPa, and temperature 510°C) exceed present industrial experience.
- To avoid additional complexity and cost in the PCS, it has been decided not to use internal cooling of the turbine blades, which is used for the most efficient natural gas turbines. The temperature of hot helium (850°C – 900°C) is at the limit of what can be accepted without internal cooling for the usual blade materials. The mechanical behaviour of the turbine and the choice of materials for its blades must be very carefully examined.
- To rule out some of the above-mentioned problems, a helium-leaktight rotating seal could be developed. With such a seal, the generator could be outside the pressure vessel, which would have several advantages. The generator could be of standard design, so its maintenance would be easier, the weight of the entire turbine-generator rotor could be supported outside the vessel by a standard mechanical bearing and therefore the magnetic and catcher bearings would be used only to control the lateral position of the shaft, which would make their design much easier, and finally the water circuit for generator cooling would not penetrate into the vessel. Such a rotating seal technology exists for explosive gas used in industry, but the scale is smaller and there is no experience with helium.

If all the PCS design problems, some of which are mentioned above, are considered, it appears that the design of such a system is rather innovative and is the major challenge for the development of future HTGR projects. Even if most of the technologies to be used already exist, they will have to satisfy new requirements and be scaled up to dimensions and operating conditions in the range of which there is not yet any industrial experience. Therefore, the components and the whole system will have to be extensively proven in large scale tests before being operated in a reactor.

As for the modular design, it has been extensively studied and the challenge is less difficult. Nevertheless, as the unique intrinsic safety features of modular HTGRs are the major advantages of these reactors for achieving public acceptance, it is important that these advantages can be demonstrated beyond dispute. As already mentioned, integral experiments with the behaviour of the reactor during accidents have already been performed in the AVR. But due to the special configuration of this reactor, with the steam generator right on the top of the reactor vessel, the major contributor to the heat release in accident situations was natural convection and not radiative heat transfer, like in modern HTGR configurations. Therefore, even if the effects of the strong negative temperature coefficient and of the large inertia of the reactor are perfectly illustrated by these experiments, the special AVR configuration puts a restriction on their significance for showing the intrinsic safety of modern HTGRs. That is why it is very important to follow and analyse very carefully the safety tests that will be performed in the experimental reactors HTR-10 (China) and HTTR (Japan), the design of which is very similar to that of modern HTGRs.

On the other hand, to be able to validate the calculation of the maximum core temperature during an accident, large-scale nonnuclear experiments of heat transfer at high temperatures without forced convection have been performed in the past for a pebble bed core. Even if similar experiments are perhaps less necessary for a block-type core, where heat transfer, mostly conductive, can be easily calculated due to the simple core geometry, the main uncertainty for this type of core is the heat conductivity value of irradiated graphite: for the graphite proposed by producers today, which is not the same as in the past, data are very few or non-existent for high fast fluence, and more data will have to be obtained.

To be able to profit from the intrinsic safety of a modular HTGR, so as to simplify its safety systems and therefore keep this type of reactor competitive, the licensing approach used for LWRs has to be amended: this is the conclusion of an exercise conducted in the Netherlands in the framework of the INCOGEN study (see Table 2). It would be beneficial for all the promoters of HTGRs in the world if they do not try to prepare such an adaptation of the rules in separate discussions with each safety authority, project by project, but if there is, as soon as possible, an international discussion, if possible under the auspices of the IAEA, on such an adaptation of the safety approach. If such an approach is initiated early enough and is internationally recognised, there is a chance that any modern HTGR design can, in the future, be accepted by the safety authorities of many different countries without major changes, which could have a very significant positive effect on the competitiveness of these reactors.

Even if there are no other major R&D needs for the coming HTGR industrial prototypes, there are a few needs to validate existing technologies for use under HTGR conditions, which must not be forgotten. This is the case, for example, for:

- The core calculation method in the case of an annular core, chosen in some projects in order to increase the power without exceeding the fuel temperature limit of 1600°C under accident conditions. The existence of a central reflectorl of graphite in the core results in a very heterogeneous neutron spectrum, thermal in this central reflector and epithermal in the fuel annulus region, with a very steep transition at the interface of these two regions (see Figure 3). Such conditions are not usually met in other types of reactors. It must be verified that the usual tools of reactor physics and the usual calculation procedures can be applied without particular problems to HTGR annular cores.

Table 2

Applicability of the IAEA Code and Guides for the design of nuclear power plants to the HTR

Topic	Applicable	Main issues	Remarks
Code of Design	amended	- barrier concept/ containment - shut down systems - second control room - PIES[1)] - water elements	
Classification	amended	- barrier concept - reactor coolant	this guide is written for LWR but can be used if it is transformed (He-cooling) to the HTR
Fire protection	yes		
Protection system	yes		
Internally generated missiles	yes		
External man-induced events	yes		
Ultimate heat sink	amended	- decay heat removal	this guide may be used for the INCOGEN plant with the atmosphere as the ultimate heat sink and passive heat conduction and radiation for decay heat removal
Emergency power	yes		
Instrumentation and Control	yes		
Radiation protection	amended	- source terms - steam line - ^{14}N - ^{14}C	with some amendments applicable to INCOGEN
Fuel handling and storage	amended	- no water pools - fire prevention	fire prevention and protection has to be addressed in a different way than for LWR
General design safety principles	amended	- PIES - ATWS	for INCOGEN a specific set of PIES has to be defined
Containment	amended	- containment concept - H_2-control	with specific amendments applicable to INCOGEN
Reactor coolant	amended	- pressure boundary	this guide is written for LWR (and a little bit for AGR) the requirements on the pressure boundary has to be replaced by specific requirements on the integrity of the fuel elements and the HTR reactor vessel
Reactor core	amended	- thermal hydraulics - cladding effects	the specific safety functions and performance of the fuel elements of INCOGEN have to be included
Seismic events	yes		
Note: 1) PIES = Postulated Initiating Events			

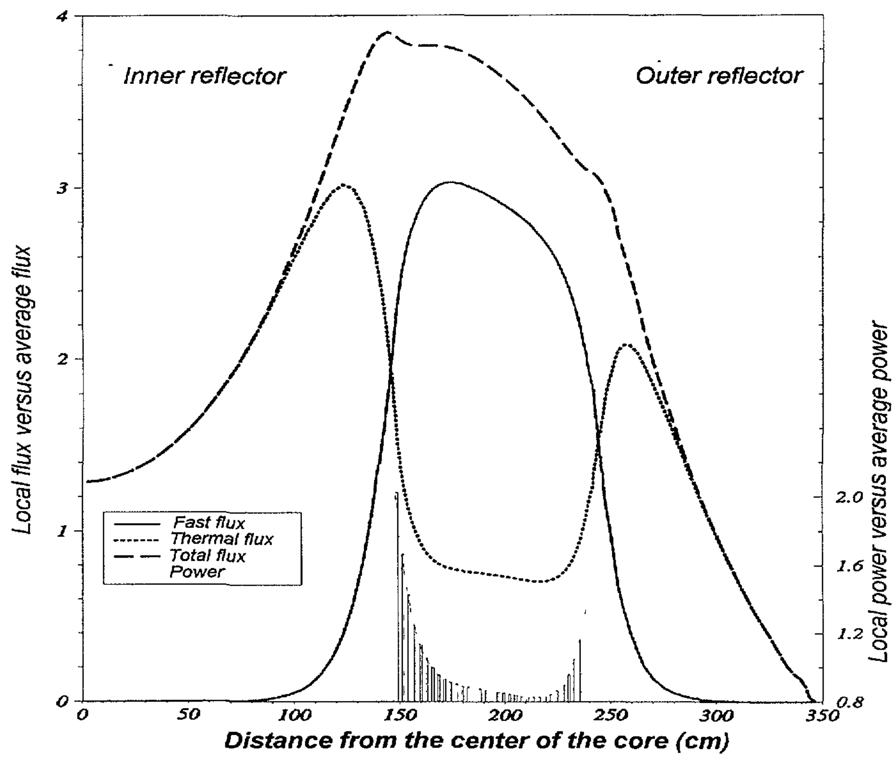

Figure 3 : transverse distribution of neutron flux in a plutonium annular core (Pu from 45 GWd/t PWR - UO_2 fuel)

- The reactor vessel material: If the reactor design is such that the temperature of the reactor vessel cannot be kept low enough, the reactor vessel must be made of 9 Cr steel. This steel is well known in industry, but it has not been used before in nuclear reactors. Therefore, the use of this material with thick welds for nuclear applications has to be qualified.

3.2. Needs for long term development of HTGR technology

In the coming years, the development, construction, and operation of HTGR prototypes can again start the nuclear energy industry in motion towards innovation, which has been stopped for a long period, and can prove that the modern HTGR is one of the safe and competitive solutions for the future of nuclear energy. But in the longer term, HTGR development should be pursued, because nuclear energy will be faced with renewed competitiveness challenges and increased social concerns, which will require still more innovative solutions.

As for the competitiveness challenge, it will not be met merely by ensuring a favourable comparison of HTGRs with other types of reactors. This challenge concerns the entire nuclear industry, which has to compete with other sources of energy, most particularly with natural gas. HTGRs will have a future only if they make a favourable contribution to this competition. And in that prospect, the needs of the market do not concern only the per-kWh cost. There is also a need for energy sources with:

- low capital cost
- a short construction period
- a medium power level (200 to 600 MWe), and
- the possibility of competitive use of the released heat.

The HTGR seems to be a good candidate to satisfy the last three criteria. As for the capital cost, the present status is rather uncertain, because of the very preliminary stage of the industrial projects. Nevertheless, even if there is a large uncertainty, the evaluations made by different organisations give the same order of magnitude for the investment cost for generating one kW with an HTGR and with a LWR. This is not enough, however: with the progress of natural gas turbine technology and with the current cheap price of gas, the advantage of nuclear electricity production is vanishing. Natural gas turbine technology is still progressing, therefore nuclear energy in general and HTGRs in particular will also have to progress.

On the other hand, people are more and more concerned about the impact of nuclear energy. Safety requirements are more and more stringent: no releases under any accident situations and no evacuation. The solution of the waste issue is also considered to be vital for the long-term sustainability of nuclear energy. People do not only expect that the wastes, which have to be stored anyway, can be stored safely, but that the quantity of such wastes be minimised. Finally, there is considerable concern with the plutonium issue. The nuclear industrial complex must find solutions to stop the accumulation of civil and military plutonium and burn it as much as possible.

Even if the present achievements of HTGR technology can already meet some of the competitiveness and public acceptability requirements, it is very encouraging for the future role of HTGRs to note that there is still a large potential of development of this technology in order to meet more and more stringent requirements. We can give a few examples here.

The high operating temperature, which characterises HTGRs, with the corresponding high energy conversion efficiency, contributes both to the competitiveness of this type of reactor and to the minimisation of wastes. There are nevertheless several innovations that can result in increasing either the operating temperature, the power density, or the power of the reactor. These innovations, used separately or in combination, can further improve competitiveness, the minimisation of wastes, or both:

- Advanced particle coatings (zirconium carbide) can increase the temperature limit for leaktightness of the particles under accident conditions from 1600°C to 1800°C.
- High-temperature materials can be used to increase the operating temperature of different components of the HTGR primary system.

- The prestressed cast iron vessel concept can be a solution to alleviate an important constraint on HTGR design: the limit on the vessel diameter due to the limitations in the workshop facilities and in the transportable size. This limit directly impacts the maximum core size and the residual power that can be passively removed during an accident.

The principle of the prestressed cast iron vessel is illustrated in Figure 4. The vessel is made of cast iron blocks assembled on the site and put under compressive stress by hoop and axial tendons, the leaktightness of the system being ensured by an inner liner. The fact that the vessel walls are put under compressive stress has another important consequence that is very advantageous for safety improvement: the vessel is burst-proof. Some feasibility studies and tests have been performed in Germany on this concept. A vessel with a diameter of 4 m and a height of 4.8 m was built and has successfully experienced a pressure test up to 122 bar (in water).

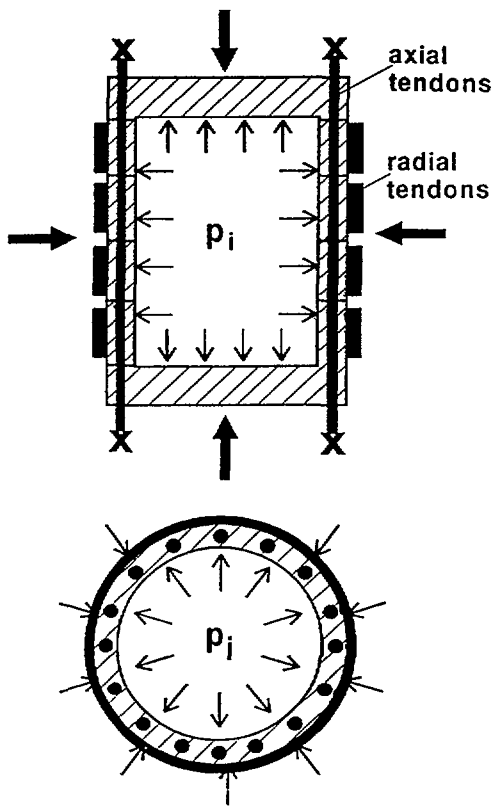

Figure 4 : principle of the prestressed cast iron vessel

Another innovation, important for improvement of the reactor's availability and safety, is the development of a silicone carbide coating, which can protect the fuel elements (pebbles or blocks) and internal graphite structures from corrosion in case of air or water ingress. Irradiated pebbles protected with such a coating stayed during 200 hours in air at 1400°C without suffering any corrosion, while standard pebbles completely disappeared through corrosion in the same conditions.

For the waste issue, apart from the high efficiency of the reactor, the fuel and the fuel cycle can play a major role. The capability of particle fuel for high burnup (160 GWd/tHM), widely proved, has an effect on the mass of irradiated fuel in inverse proportion to the achieved burnup. As mentioned above, very high burnup (750 GWd/tHM) has been reached during experimental irradiations of plutonium and highly enriched uranium. But the feasibility of such very high burnup on a large scale is still to be proved; it must be verified that under such irradiation conditions there is no increase in the proportion of leaking particles.

Another solution for the waste issue is the minimisation of long-term activity by using a fuel that reduces the actinide production. This is the case with thorium cycles and, as noted before, HTGRs are very well suited to accept different fuel types, most particularly thorium fuel: the thorium cycles are optimised for the HTGR epithermal neutron spectrum. Different cycle solutions can be studied, possibly in synergy with other types of reactors.

Reprocessing particle fuel is rather difficult, and, at least for very high burnup fuel, probably not economically attractive. When 80 % of the initial heavy metal inventory has disappeared through fission, there is not much fissile material left to recover. Therefore particle fuel, or at least part of it, is intended for direct disposal. Moreover, particle fuel is particularly adapted to direct disposal. According to the available experimental data, the coating keeps its excellent leaktightness performance under long-term storage conditions, but this has to be confirmed by complementary experiments and also has to be verified for advanced coatings.

As for the potential of consuming plutonium, preliminary analyses show higher performance than for PWRs or FBRs (see Table 3). Moreover, due to the high burnup the inventory of heavy nuclei left after irradiation is smaller and the degradation of the isotopic composition of plutonium (decrease of the fissile plutonium proportion) is higher than with other types of reactors: for example, with plutonium recovered from 45 GWd/t UO_2 PWR fuel, in which the proportion of fissile isotope is 65%, a MOX recycling in a

Table 3

Plutonium burning potential (kg/TWhe)

PWR (900 Mwe)	FBR (CAPRA)	HTR (GT MHR)
65	75	100

PWR with a discharge burnup of 45 GWd/t would leave after irradiation an inventory of plutonium which still represents 73% of the initial mass of Pu, with a proportion of 55% of fissile isotopes, while a HTGR recycling would leave only 28% of the initial mass of Pu, with a proportion of 36% of fissile isotopes. This is one more incentive for burning plutonium in a once-through HTGR cycle: at the end of such an open cycle, the economic value of recovering plutonium is very low, because of this low fissile inventory, and on the other hand, because of the very low quality of the remaining plutonium, storing it involves little proliferation risks.

4 – Conclusion

The industrial feasibility of the high temperature fuel and core has been widely proven, but the intrinsic safety features of HTGRs still need some integral experimental demonstrations. The technologies for the direct cycle power conversion system exist, but most of them are to be used beyond the limits of their present industrial applications. The integration of the whole power conversion system is also far from present industrial experience (closed cycle, severe transients, nuclear environment, compactness, very stringent availability requirements, etc.).

In the long term, HTGRs in particular and nuclear energy in general will be faced with more and more demanding requirements for competitiveness and public acceptability. The HTGR still has a large potential for future development, to face up to these new challenges. Therefore, right from the start, these long-term developments must not be neglected, because they will probably play an important role in the future development of nuclear energy.

For all these developments, broad international cooperation is needed:

- Available experimental facilities (reactors for irradiation, integral safety demonstration, and reactor physics measurements; loops and test benches for tests of component performances) are not numerous. They have to be carefully identified and to be supported and shared on the basis of international research programmes.
- The safeguarding of available knowledge and experience from past HTGR programmes could be organised worldwide through the "Global HTR network" (GHTRN).
- Licensing standards adapted to modular HTGRs could be developed through international collaboration. Such an approach, if started when the industrial projects are still in an embryonic state, could lead to more uniform standards in different countries, which could help HTGRs to better penetrate the world market.

The IAEA could play an important role in promoting such broad international cooperation.

GT-MHR AS ECONOMICAL HIGHLY EFFICIENT INHERENTLY SAFE MODULAR GAS COOLED REACTOR FOR ELECTRIC POWER GENERATION

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Abstract

The Gas Turbine Modular Helium Reactor (GT-MHR) project is a joint effort of organisations in the Russian Federation co-ordinated by the the Ministry on Atomic Energy (MINATOM), General Atomics in the United States, Framatome in France, and Fuji Electric in Japan. This paper describes the structure, objectives and history of the GT-MHR project, as well as the plant design, including plant layout, components and fuel. It summarises the results of evaluations of the design safety characteristics and economics, for both a plutonium fueled single module and a uranium fueled four module plant. The plans and anticipated structure for an international project to construct the plant are also summarised.

INTRODUCTION

The GT-MHR project is a joint elaboration of organizations belonging to Russian Ministry on Atomic Energy (MINATOM), and companies of General Atomics, Framatome and Fuji Electric that have experience and knowledge in areas of HTGR technology and fabrication of large-size components. An interest to this project is called due to its economic effectiveness, inherent safety features, production of minimum amount of long-lived radioactive wastes, minimum heat environment impact and possibility to be applied in different configurations for both electricity and high potential heat generation.

International cooperation in development of components and systems and possibility to vary total power of module-composed plant and choose options of generated energy use (electricity, technology heat for chemistry, metallurgy, oil refining, synthetic and hydrogen fuel production, etc.) should decrease technical and economic risk of elaboration. Replacement of fossil burn to nuclear energy source should make favourable ecological effect for environment and towns.

All that might be applied both to the reactor itself with its inherent safety properties ensuring prevention of impermissible releases of radiotoxic substances and to the fuel cycle. Application of ceramic materials in HTGR fuel elements ensures minimum relative generation of long-lived radioactive wastes. These materials are resistant against a corrosion under final deployment conditions. Therefore, safety of radwaste storing is ensured for the very long period of time independently of final deployment cask integrity.

A perfection in such complicated engineering system as GT-MHR is, might be achieved only as a result of a large volume of work on stages of designing, researches and engineering development

1. GT-MHR PROJECT

The GT-MHR project development was started in 1995 accordingly to the Agreement on designing and development of the GT-MHR plant and its further construction in Russia, signed by MINATOM and General Atomics. In 1996 the Framatome company joined the Agreement, and in 1997 the Fuji Electric company joined too. In 1997 the GT-MHR Conceptual Design had been completed.

Most engineering decisions accepted in the GT-MHR are based on the ones mastered in construction and exploitation of Peach-Bottom and Fort St. Vrain reactors, and on thirty-year Russian experience of HTGR designing (the VG-400 reactor plant of loop configuration and the modular pebble bed core reactors VGR-50 and VGM).

The GT-MHR program aims a creation of the prototype for commercial nuclear power plants on the base of the modular helium-cooled reactor and the direct cycle gas turbine, that are competitive to existing energy sources, including LWRs and fossil fuel plants, in criteria of safe, reliability, manoeuverability and economy.

The GT-MHR high economic indices ensuring its competitiveness are caused by the following four main factors accepted in the concept of the project:

- modular arrangement;
- inherent safety;
- direct closed gas turbine cycle with ensuring the efficiency about 48%;
- use of the advanced modern technologies such as the high efficient gas turbine, electromagnetic bearings operating without friction, plate and finned tube high effiective compact heat exchangers.

You can see the GT-MHR major parameters in the **Table 1.**

The GT-MHR reactor is designed taking into account use of the plutonium fuel based on the WGPu oxide. The degree of Pu burnup is accepted sufficiently high (up to 90% of initial loaded Pu-239) to eliminate both technical and economic expediency of its conversion and repeated use in the reactor and make impossible its military employment.

The design of the prototype nuclear power plant (NPP) with one GT-MHR Pu loaded modular reactor is destined to be arranged on the Siberian Chemical Combine (SCC) site at the town of Seversk.

Commercial NPPs are supposed as U loaded four-module (4x600 MWt) ones.

The GT-MHR reactor module layout is shown in the **Figure 1.**

The prototype GT-MHR NPP has one reactor module placed within the underground containment. The components of the reactor module primary circuit are arranged within the pressure vessel system consisting of the reactor vessel and the Power Conversion System vessel connected by the cross vessel.

The reactor vessel is surrounded by the surface cooler of the Reactor Cavity Cooling System. This passive activity system ensures cooling of the cavity concrete and heat removal out of the reactor vessel under natural circulation both in normal and emergency operating modes.

Looking on the **Figure 2,** you can see the layout of the GT-MHR reactor module components.

TABLE 1

GT-MHR MAJOR TECHNICAL CHARACTERISTICS

• Thermal power, MW	600
• Electricity generation efficiency (net), %	up to 47.6
• Helium temperature (core inlet / outlet), °C	490 / 850
• Core configuration	annular core consisting of prismatic graphite blocks
• Fuel	Plutonium oxide
• Fuel type	coated particles
• Initial Plutonium load, kg	750
• Pu-239 destruction level, %	~ 90
• Annual Plutonium consumption, kg per year	250
• Quantity of WGPu processed for reactor lifetime (60 years), t	15

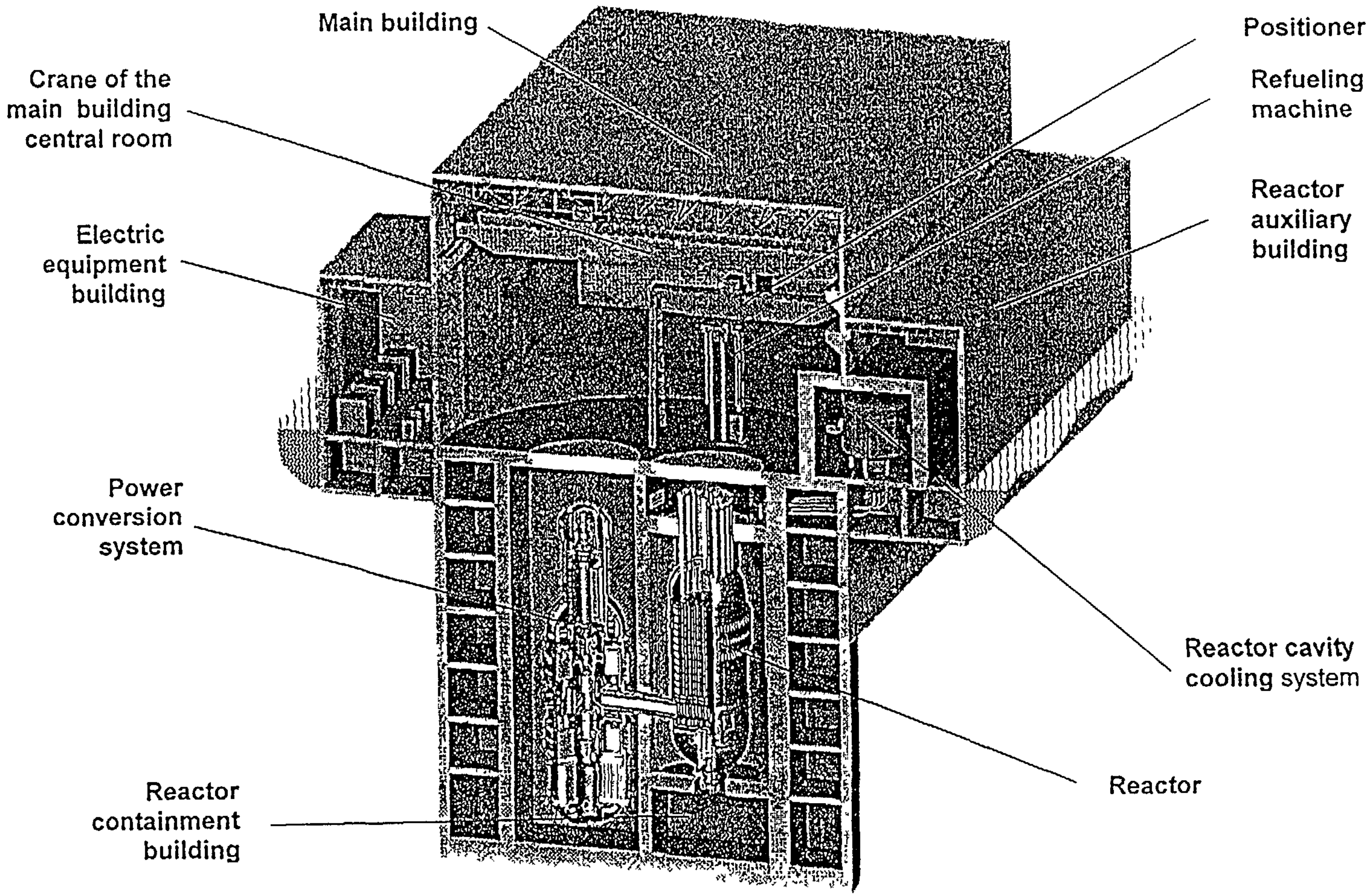


FIG. 1. GT-MHR reactor module arrangement in reactor building.

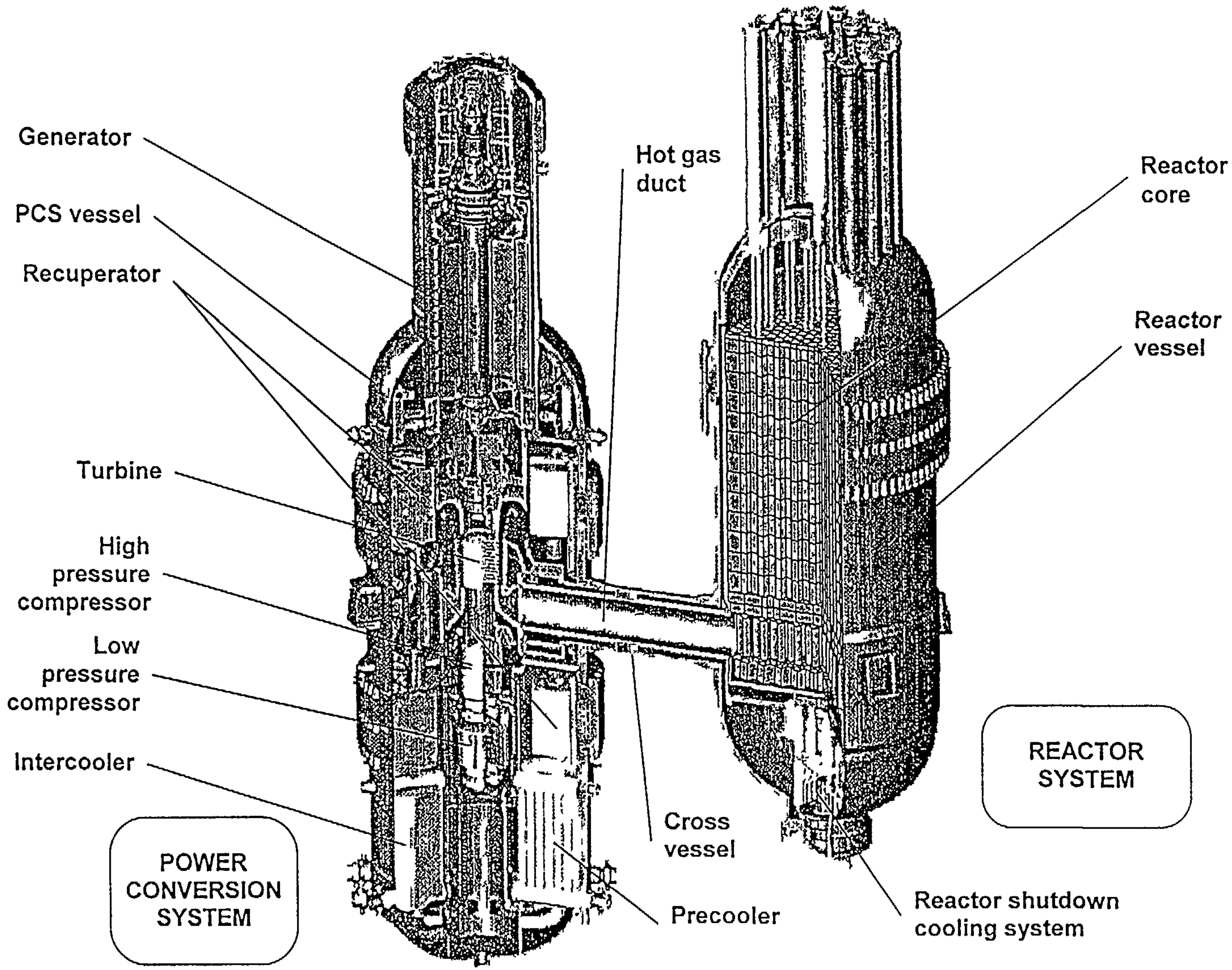


FIG. 2. GT-MHR reactor module layout.

The reactor has the annular core assembled of prismatic graphite blocks. The blocks have channels where fuel elements are placed. In the bottom part of the reactor vessel the Reactor Shutdown Cooling System is placed. This system ensures heat removal out of the shutdown reactor core by active means, including modes of core reloading, PCS maintenance and also accidents within the design basis.

The Power Conversion System (PCS) is placed within the pressure vessel, connected with the reactor vessel by the cross vessel, and consists of the one-shaft vertical arranged turbomachine, recuperator, precooler, intercooler and internal gas ducts and supports. The turbomachine consists of gas turbine, electric generator, LP and HP compressors.

The primary circuit helium coolant leaves the reactor core and through the hot gas duct inside of the cross vessel comes to the turbine inlet. The turbine rotates directly the low pressure and high pressure compressors. From the turbine outlet the coolant passes through the recuperator aiming to return the maximum possible amount of heat to the cycle. Then it comes into the precooler for heat removal to sink.

Cooled in the precooler helium comes consecutively to the LP compressor, intercooler and HP compressor. Compressed coolant is heated in the recuperator, and then through the annular gap between the hot gas duct and cross vessel wall moves into the reactor vessel. Through channels of reactor internals it is delivered to the reactor core and, passing through the core from top to bottom, closes the circuit.

The GT-MHR fuel design shown in the **Figure 3** is as fuel micro-particles covered by multi-layer coating of pyrolitic carbon and silicon carbide. The particles are dispersed within a cylindrical graphite matrix (compact) of 12.5 mm diameter and 50 mm length. Compacts are inserted into fuel channels of prismatic graphite fuel blocks.

The fuel micro-particles are plutonium oxide-made spherical kernels of 0.2 mm diameter with coatings. These coatings prevent solid and gaseous fission products release in the process of the reactor exploitation.

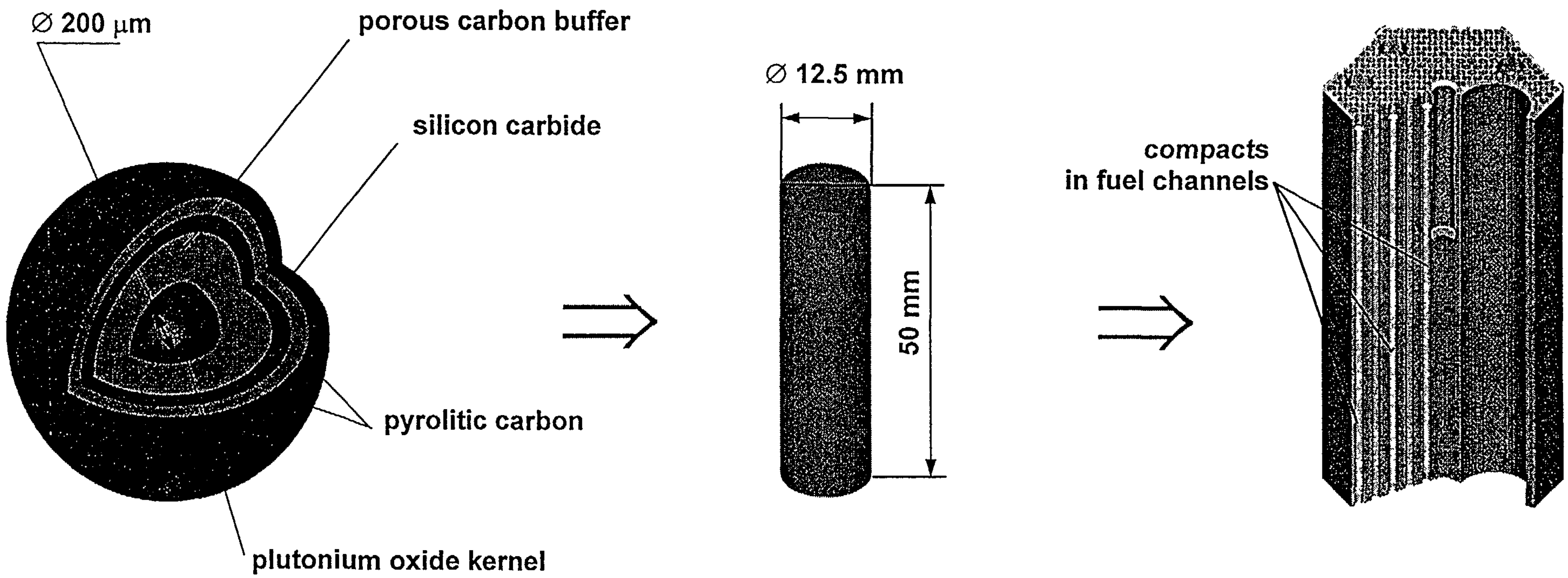
Average content of plutonium is 0.24 grams per a compact.

2. MODULAR ARRANGEMENT

All NPP projects being developed in last some years are based on the modular arrangement concept independently of the reactor type. The modular arrangement gives nice prospects for improvement of the NPP economic indices, and the GT-MHR design has maximally soaked up all merits of this concept.

The accepted configuration of the reactor island (underground location of reactors and the surface common services building), compact layout of the reactor plant main equipment, including the turbine generator, inside of the high pressure vessel unit, and also the vertical arrangement of the turbine shaft make possible to decrease building volumes and reduce construction time. Time and expenditures for designing of serial plants become minimal. The possibility arises to use some systems and buildings for servicing of several reactor modules. Thus, in the design of the GT-MHR commercial four-module plant one reloading system, two interim spent fuel storage facilities and one services building operate four reactor modules. Unification of components and serial production enable wide use of the international cooperation resources in manufacturing, reduction of manufacturing time and cost by simplification of the license procedure and the learning effect.

GT-MHR FUEL ASSEMBLY COMPONENTS



FUEL PARTICLE

FUEL COMPACT

FUEL ASSEMBLY

FIG. 3.

3. SAFETY

In the GT-MHR design the well-known HTGRs inherent safety characteristics, such as the inert coolant, the core made of graphite materials, the fuel as micro-particles with multi-layer protective coating are supplemented with especial technical decisions directed to ensuring of passive principle based safety. The release of impermissible large quantity of radionuclides out of the fuel is prevented significantly by ceramic coatings of fuel micro-particles that are planned for both normal and probable emergency operational modes.

The integrity of the coating of fuel micro-particles, as the safety barrier, is reached in emergency by elimination of excess of the maximum permissible fuel temperature of 1600 °C (the safe exploitation limit), that is ensured by the following factors:

- low core power density;
- high heat capacity of the graphite-made core and reactor internals;
- large maximum permissible fuel temperature margin (up to 200 °C);
- negative reactivity coefficient;
- core design decisions ensuring heat removal through the reactor vessel wall;
- existence of the effective passive system of emergency heat removal, that is the reactor cavity cooling system;
- limitation of air and water/steam potential impact on fuel microparticles in accidents.

These properties make the significant positively influence on the plant economic effectiveness. So it enables, on the one hand, to refuse some emergency operation systems, such as the emergency core cooling system or the emergency shutdown cooling system that exist at NPPs with LWRs, and, on the other hand, to combine the functions of normal and emergency operation in the same system. For instance, the normal operating cavity concrete cooling system is simultaneously the system of emergency heat removal.

Passive safety properties, design characteristics and the fuel design ensuring prevention of impermissible releases of radioactive substances, allows to refuse the containment construction for the U fuel variant, that amounts up to 7% of the module construction capital cost.

4. SINGLE-MODULE PROTOTYPE PLANT ECONOMICS

The GT-MHR plant technical and economic characteristics applied in estimation of the electricity production cost are shown in the **Table 2.**

The following assumptions were accepted for the estimation:

- *the prototype single-module NPP is placed on the SCC site;*
- design and engineering development costs on the GT-MHR equipment and fuel are included into the prototype single-module NPP cost;

TABLE 2

GT-MHR ECONOMIC CHARACTERISTICS

	Pu Fuel Single Module Prototype Plant	U Fuel Four Module Commercial Plant
• Power, MW(th) / MW(e)	600 / 293	2400 / 1172
• Annual electricity generation (K = 0.8), 10^6 kWh	2051	8204
• Annual electricity sale, 10^6 kWh	1999	7996
• Operational and Maintenance Staff, persons	230	493
• Plant Design and Engineering Development, $M	180	19
• Plant Capital Cost, $M	273	928
• Fuel Researchs and Engineering Development, $M	70	-
• Fuel Fabrication Capital Cost, $M	45	42
• TOTAL INVESTMENT, $M	568	1031
• POWER GENERATION PRIME COST, mills per kWh	22	17.6

- the fuel cost is accepted equal to its primary cost (without benefit and VAT) because the fuel production and the NPP are the subdivisions in the same enterprise SCC;
- the cost of SCC services for NPP (security, transport, etc.) not accounted in the operating and maintenance (O&M) cost are accounted in the clause "Site owner's expenses";

The composition of the annual exploitation cost is as follows:

- Fuel cost;
- Operating and maintenance cost;
- Capital cost;
- Site owner's (SCC) expenses;
- Decommissioning fund fees;
- Other funds fees quoted from the generation cost.

The prime cost of one kWh electricity generation estimated as 2.2 cent for the prototype Pu-loaded single-module NPP, and 1.76 cent for the commercial U-loaded four-module NPP.

The sale price of the electricity generated by existing energy plants on the SCC site is about 2 cents per kWh (data of July 1998). Thus, taking into account the fact that the prime cost was estimated under a conservative approach, we may say that even the single-module prototype plant repays investments in 60 years.

In the **Table 3** you can see estimated variants of the GT-MHR NPP waste heat utilization for heat supply. In this estimation the following assumptions were accepted:

- the existing heat networks at the site will be employed;
- the costs of modification of the existing heat networks and reactor module design are negligible in comparison with the NPP capital cost (might be included in the Contingency account)

In this conditions we may say about profitability of the GT-MHR prototype single-module NPP, that is a rather rare case in power engineering.

5. INTERNATIONAL COOPERATION

Until the recent time, owing to both objective and subjective reasons, HTGRs were out of the mainstream of the atomic power engineering development. The properties of these reactors had been demonstrated on the plants of AVR, DRAGON, Peach Bottom of the first generation, and then THTR-300 and FSV. But that time was not yet the HTGR time. These projects were too exotic for that period to be used word-wide commercially, and after Chernobyl accident that had influenced extremely negatively on the politicians' and the public relations to atomic energy at whole, these projects were stopped.

In nineties, in conditions of word-wide sudden quality progress in technologies, the idea of HTGRs attracted an attention to itself, and its main direction in development becomes the optimization of designs of modular low and medium power reactors, concentrating on the inherent safety, optimization of their power, maintainability, positive economic effect of serial fabrication, use of the most modern advanced

TABLE 3

VARIANTS OF GT-MHR SINGLE MODULE NPP WASTE HEAT UTILIZATION

	Nominal mode	Hot water supply	Heat supply
• Thermal power, MW	600	600	600
• Net electric power, MW	285	276	226
• Network circuit power, MW	-	314	364
• Annual heat sale volume, 10^3 Gcal	-	1884	2184
Sale price, cents per kWh / dollars per Gcal	2 / 0	2 / 6	2 / 6
• Annual average profit, $ M per year	0	+ 11,3	+ 13,1
• Investment repaying period, years	60	50	43

technologies, such as high speed turbines, electromagnetic bearings, compact high effective recuperators, corrosion-protected fuel elements.

Many technical problems yet need to be solved in the HTGR development process. So, the technologies of manufacturing of pressure vessels and high effective heat exchangers have to be improved and mastered, the regulatory base must be developed for the HTGRs taking into account their inherent safety, etc.

The principal way for solution of so complicated problems must be an international cooperation giving conditions for consolidation of financial and engineering resources, efforts of various countries designers, developers, manufacturers of NPP components and IAEA, exchange of existing advanced technologies, optimal employment of existing industrial capacities.

The GT-MHR project has a status as the international one from the very beginning. Companies from four countries took part in its development on the Conceptual Design stage. Designing and development were fulfilled by Russian organizations, companies of USA, France and Japan granted their technologies existing in this area. At present, investigations are fulfilled on possibility and expediency of the GT-MHR components manufacturing on the Framatome company plants.

The next stage in the GT-MHR project will be the start of the Preliminary Design development in 1999 and continuation of WGPu-made fuel development. This work is expected to be financed by USA and Russia on a parity basis.

As the next step in the process of the GT-MHR creation, formation of the international technology company MHRCo is now on the stage of completion. This company functions will be collection, storage and capitalization of technologies and know-how in the HTGR area and sale of licenses for their employment.

In the further development of the project it is planned to form the company that will be occupied in construction of plants with GT-MHR-type reactors.

RELEVANT SAFETY ISSUES IN DESIGNING THE HTR-10 REACTOR

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Abstract

The HTR-10 is a 10 MWth pebble bed high temperature gas cooled reactor being constructed as a research facility at the Institute of Nuclear Energy Technology. This paper discusses design issues of the HTR-10 which are related to safety. It addresses the safety criteria used in the development and assessment of the design, the safety important systems, and the safety classification of components. It also summarises the results of safety analysis, including the approach used for the radioactive source term, as well as the approach to containment design.

1. Introduction

The paper discusses some design issues of the HTR-10 test reactor which are related to safety. The design of the HTR-10 reactor represents the design features of the modular HTGR design which is mostly characterized by inherent safety features. The safety design philosophy deviates very much from the traditional approach which relies on highly reliable redundant and diversified active components and systems as well as their power supply. How much credit is given to the new safety design approach which is based on inherent safety philosophy affects the plant economy balance very much. The HTR-10 test reactor should serve as a test facility to demonstrate the inherent safety features of the modular reactor design and to help win the credit from regulatory bodies, utilities and the public.

2. Safety Design Criteria

In the current nuclear culture, the issue of design criteria is almost always the first encountered issue when one starts to design a nuclear reactor, especially when the safety design is concerned, but up to now there exist in China not enough nuclear standards, codes and guides specifically compiled for high temperature gas cooled reactors which are directly applicable to the HTR-10 reactor. Of course there are general standards which specify the limits on off-site doses and radioactive release rate into the atmosphere as well as doses for the operating staff, and other general standards for the design of nuclear power plants, but these standards serve only as the top-level standards to follow. For the concrete design of the HTR-10 reactor and its systems, specific design criteria are needed. Before the basic design of the HTR-10 reactor, the design engineers had firstly compiled the **Design Criteria for the 10MW High Temperature Gas-cooled Test Reactor** and had it reviewed by the licensing authority. This document contains chapters for the specific systems of the test reactor. It is based on making reference to international and national nuclear standards or regulations, and to a large extent on the experience of the concept design work. This document mainly deals with the requirement on system designs. Therefore, it outlines the configuration and design of those nuclear safety important systems for the HTR-10. As far as the design, manufacture as well as installation of safety class components, specific codes have to be chosen to follow, mostly the ASME code.

In the current situation when the worldwide gas-cooled reactor community seeks much cooperation, it is probably a worthwhile area to study on the design criteria for modular high temperature gas-cooled reactors by internationally cooperative efforts.

3. Safety Important Systems

Safety important systems are those which perform such safety functions as reactor shutdown, decay heat removal and radioactivity release limitation. For the HTR-10 reactor, safety important systems are primarily the following:

- Reactor protection system and its related instrumentations and power supplies
- Reactor shutdown systems (the control rod system and the small-absorber-ball system)
- Decay heat removal system
- Primary coolant pressure boundary and its pressure relief system

Proper definition of the safety functions and proper configuration of the safety important systems, which take proper credit for the inherent safety features of the modular reactor design, are a key issue in terms of maintaining enough nuclear safety and cost effectiveness.

4. Safety Classification

Safety classification with related Quality Assurance classification and seismic categorization of components is another important issue in maintaining safety and cost effectiveness. High quality, nuclear classified components are of course always good, but it has to be made clear whether the nuclear quality requirement is necessary from the viewpoint of safety or of operational reliability. For the HTR-10 test reactor, safety classifications of components departures from the way as it is done for water cooled power reactors in China. For example, primary pressure boundary is defined to the first isolation valve. Steam generator tubes are classified as Safety Class II component. Diesel generators are not required to be as highly qualified as those used for large water cooled power reactors, since no systems or components with large power demand would require an immediate start of the diesel engines at a plant black-out accident. So, some credit is given to the inherent safety features of the test reactor. It is worthy to study carefully the issue of component safety classification of the modular HTGR design.

5. Accident analysis

As usual, design basis accidents (DBA) are classified into several categories for the HTR-10. These categories are:

- Increase of heat removal capacity in primary circuit
- Decrease of heat removal capacity in primary circuit
- Decrease of primary flow rate
- Abnormality of reactivity and power distribution
- Rupture of primary pressure boundary tubes
- Anticipated transients without scram (ATWS)

The reactor is designed against these accidents. The analysis of these accidents is done with conservatism. The analysis results show excellent safe response of the reactor to accidental events. Within the framework of DBA, no accident would lead to relevant release of fission products from the fuel elements.

The issue of severe accidents has to be addressed. A number of postulated accidents are selected to be analyzed. These postulated events mainly include:

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

The definition, choice and analysis of severe accidents are one important aspect during licensing. These accidents are chosen as severe accidents of HTR-10 based on conversations with the licensing people and on the reference to practices in Germany and USA. It is important to address the severe accident issue in order to win credit for the inherent safety, but reasonableness and justification in the choice and analysis of severe accidents need to be handled carefully. For the HTR-10 reactor, as long as the reactor protection system works, most of the above accidents lead to no damages to the fuel elements. The rupture of the connecting vessel and the hot gas duct is probably the most severe one which does lead to some damages to the fuel elements but impermitted release of radioactivity into the environment is not expected. One has to block, within a relatively long time scale, say a few days, the rupture area in order to prevent air from continuously entering into the reactor core.

6. Source term

A mechanistic approach is adopted for determining the radioactive source term. Severe core damages are not arbitrarily postulated at siting evaluation, as it is done for large water cooled power reactors. The release of radioactivity is calculated specifically for those individual demanding accidents which 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 safety features, but it is directly related to the quality of the fuel elements and to the knowledge of fission product release behavior during normal operation and accidental conditions. The HTR-10 reactor shall serve as a test bed for fuel elements and as a facility to study the fission product behavior.

7. Containment design

For the HTR-10 test reactor, no pressure-containing and leak-tight containment is designed. The concrete compartments, which house the reactor and the steam generator as well as other parts of the primary pressure boundary and which are preferably regarded as confinement, together with the accident ventilation system, serve as the last barrier to the radioactivity release into the environment (see Figure 1). During normal operation, the confinement is ventilated to be kept sub-atmospheric. When 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 low radioactivity content. Afterwards the confinement is ventilated again, gases in it will be filtered before they reach the environment.

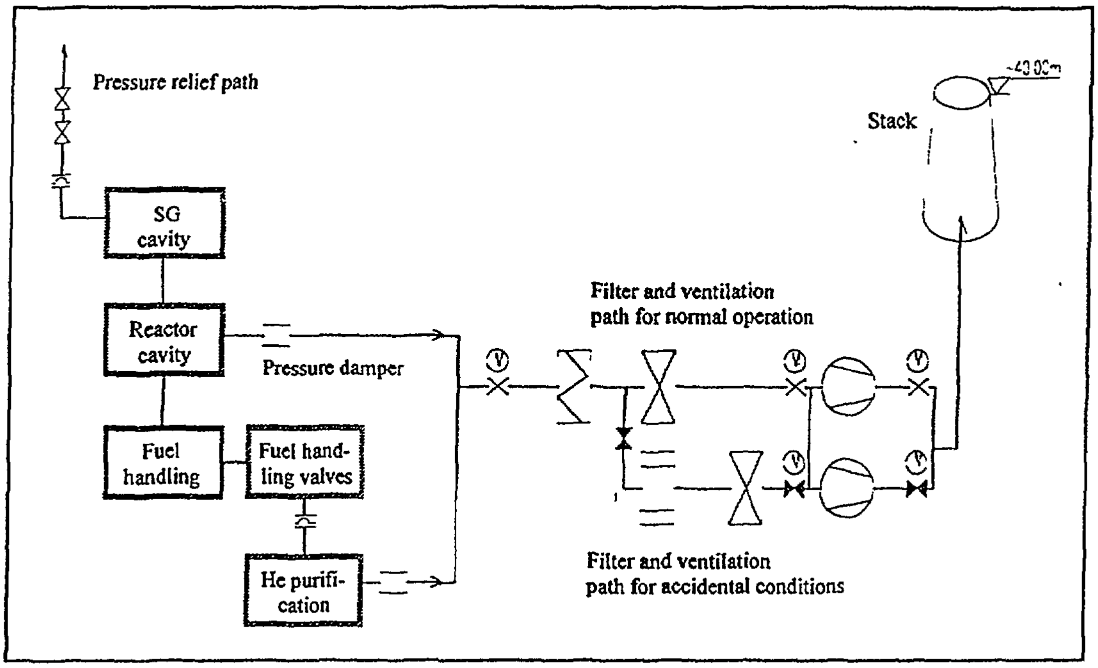

Figure 1 Schematic Diagram of the HTR-10 Confinement Design

8. Summary

Safety is the underlying feature of the modular HTGR design. The HTR-10 test reactor is a test module for demonstrating the safety features of the modular HTGR design. The reference value of the HTR-10 reactor safety design to modular HTGR of commercial size is a worthwhile issue to be evaluated in the future along with the operational experience of the test reactor.

PRESENT ACTIVITY OF THE FEASIBILITY STUDY OF HTGR-GT SYSTEM

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Abstract

In JAERI a feasibility study of the High Temperature Gas-cooled Reactor - Gas Turbine (HTGR-GT) system has been carried out since January, 1997 as an assigned work by the Science and Technology Agency. The study aims at obtaining a promising concept of HTGR-GT system that yields a high thermal efficiency and at the same time is economically competitive. Designs of a few candidate systems will be undertaken and their power generation costs will be evaluated. In parallel with design works, some experimental works such as the fabrication of a plate-fin type heat exchanger core and material tests will be carried out. The study will be continued till 2000 fiscal year.

In 1997 fiscal year, a preliminary design of a direct cycle plant of 600MWt was developed. A reactor inlet gas temperature of 460℃, a reactor outlet gas temperature of 850℃ and a helium gas pressure of 6MPa were selected. Some advanced technologies were adopted such as a monolithic fuel compact and a control rod sheath made of carbon/carbon composite material. They were very effective to enhance the heat transfer of fuel and to reduce the core bypass flow. As a result, a power density of 6MW/m^3 and the maximum burnup of 10^{5}MWD/ton were achieved. A single-shaft horizontal turbomachine of 3600rpm was selected to ease the mechanical design of the rotor supported by magnetic bearings. The turbine, two compressors, a generator and six units of intercooler were placed in a turbine vessel. Plate-fin type recuperator and precooler are installed in a vertical heat exchanger vessel. By this design, a net thermal efficiency of 45.7% is expected to be achieved. To develop a high performance plate-fin recuperator, a core model of W200mm×L200mm×H200mm with small fin size of 1.15mm height was fabricated and as a result of tests, leak tightness, component strength and bonding appearance were found to be satisfactory.

In 1998 fiscal year, a design of a direct cycle plant of 300MWt is undertaken. The new concept of an annular pebble bed core and a reactor pressure vessel provided with inner insulator were employed. By this challenging concept, a very high thermal efficiency could be expected.

1. Introduction

In JAERI a feasibility study of the High Temperature Gas-cooled Reactor - Gas Turbine (HTGR-GT) system has been carried out since January 1997 as an assigned work by the Science and Technology Agency. The study aims at establishing a promising concept of HTGR-GT system that yields a high thermal efficiency at competitive cost. Designs of a few kinds of candidate system concepts will be undertaken and their power generation costs will be evaluated. In parallel with design works, some experimental works such as the fabrication of a plate-fin type heat exchanger core and material tests will be carried out. The study will be continued till 2000 fiscal year. Figure 1 shows the scope and time schedule of the feasibility study.

In 1997 fiscal year, that is, from April 1997 to March 1998, a design of a 600MWt direct cycle plant was carried out in cooperation with nuclear industries such as Fuji Electric Co. Ltd., Hitachi Ltd., Mitsubishi Heavy Industries Ltd. and Toshiba Corporation. In addition, the fabrication and testing of a heat transfer core model of recuperator with small offset-fins was carried out. In 1998 fiscal year, a design of a direct cycle plant with a smaller capacity of 300MWt is being developed.

In the following, the results of the design of the 600MWt direct cycle plant and the fabrication and testing of the recuperator-core model are described. In addition, an initial concept of the design of the 300MWt direct cycle plant is introduced.

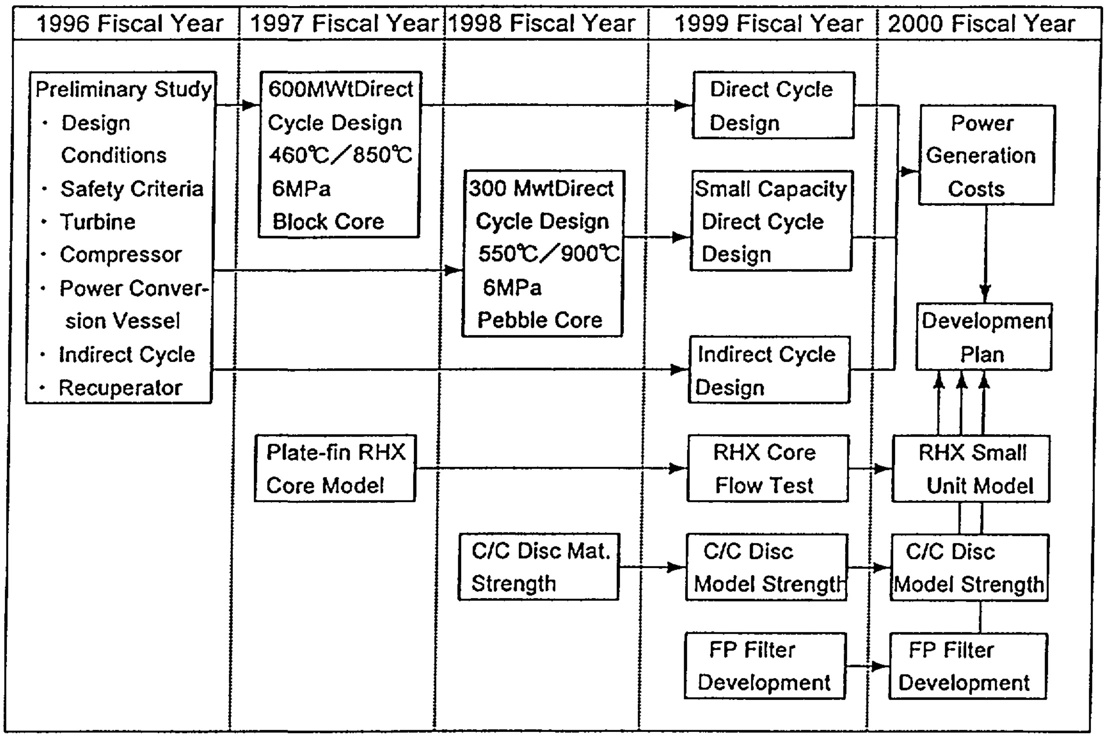

Fig. 1 Scope and time schedule of the feasibility study.

2. Design of the 600MWt Direct Cycle Plant

(1) Selection of Main Design Parameters

The thermal power output of 600MWt, a reactor outlet gas temperature of 850℃, a reactor inlet gas temperature of 460℃ and a helium gas pressure of 6MPa were selected based on the following considerations.

The effects of the reactor inlet and outlet gas temperatures ($T_{R,IN}$ and $T_{R,OUT}$) on the thermal cycle efficiency are shown in Fig. 2 for the typical values of turbine and compressor adiabatic efficiencies, recuperator effectiveness and system pressure drop ratio of 5% (summation of $\Delta p/p$). Naturally, the higher $T_{R,OUT}$, the higher the cycle efficiency. It should be noted that there exists an optimum $T_{R,IN}$ for a particular $T_{R,OUT}$, and it becomes around 500℃ for the $T_{R,OUT}$ of 850℃. The optimal value of $T_{R,IN}$ becomes higher gradually, as $T_{R,OUT}$ increases. In a usual design, the inner surface of the reactor pressure vessel (RPV) is in contact with the inlet gas flow, the temperature of which coincides with the operating temperature of RPV. As the strength of the RPV steel, 9Cr-1Mo-V decreases rapidly around 500℃, the value of $T_{R,IN}$ affects strongly the RPV structural design as well as the permissible helium gas pressure. Figure 3 shows the effects of both parameters on the weight of RPV. The horizontal dotted line in Fig. 3 shows the maximum capacity of existing RPV-factory crane. From this figure, it becomes clear that both a $T_{R,IN}$ of less than 500℃ and a gas pressure of less than 6MPa are required.

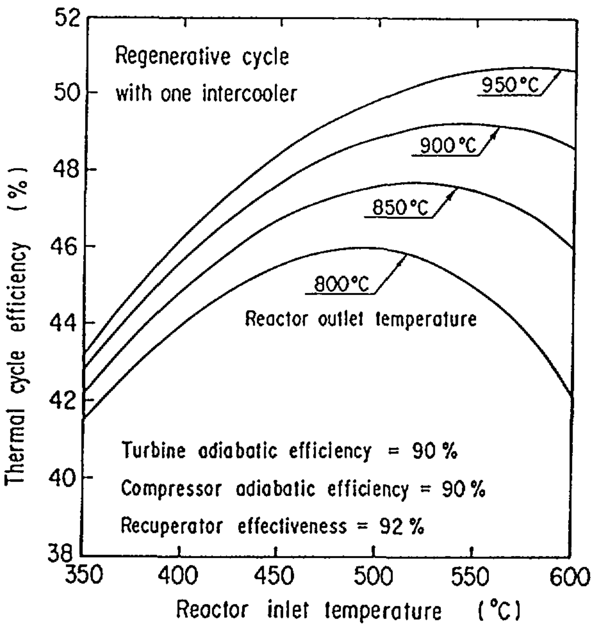

Fig. 2 Effects of reactor inlet /outlet gas temperatures on thermal cycle efficiency.

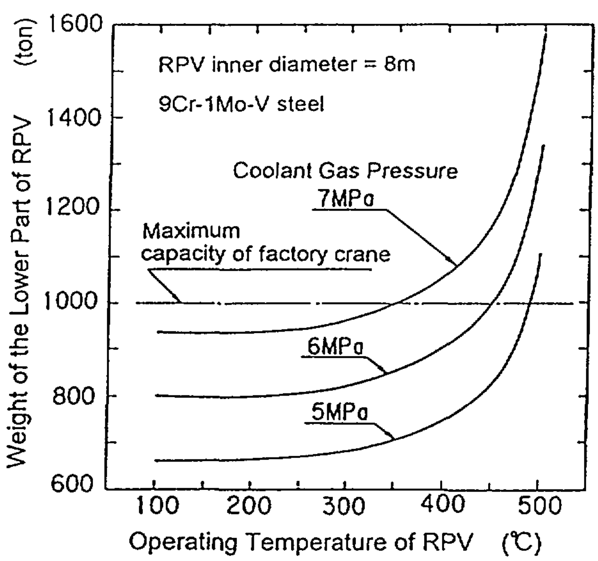

Fig. 3 The relationship of RPV weight and gas temperature and pressure.

The thermal power output is limited by both the maximum core size and the maximum fuel temperature. Simplified calculations were conducted to examine the dependence of the maximum fuel temperature on the $T_{R,IN}$, $T_{R,OUT}$ and a linear heat rate. The result is shown in Fig. 4, which projects the values of achievable linear heat rate to be 206, 140 and 83 W/cm, respectively for the cases of 500 / 850℃, 500℃ / 950℃ and 350℃ / 950℃, as indicated by the values of intersection between the curves of reactor inlet/outlet temperature and the horizontal line of 1400℃ fuel temperature. This fuel temperature is the allowable limit to restrict all the failure modes of coated fuel particles such as a thermal resolution, a palladium attack, an internal pressure burst and an amoeba effect. Based on these linear heat rates, the relative values of achievable reactor power can be

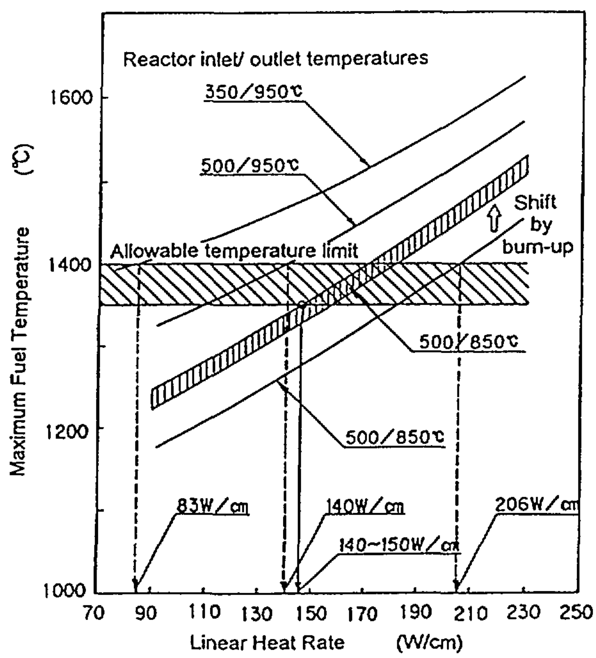

Fig. 4 Relationship between core linear heat rate and maximum fuel temperature.

predicted with respect to the various temperature conditions as follows.

50℃ increase of $T_{R,OUT}$ → multiplying the reactor power by $0.824 = (140/206)^{1/2}$

50℃ decrease of $T_{R,IN}$ → multiplying the reactor power by $0.840 = (83/140)^{1/3}$

In case of 500/850℃, the actual maximum fuel temperature is additionally affected by the burn-up and therefore the curve must be shifted up as indicated in the figure. As a result, the achievable linear heat rate is reduced to 140 to 150W/cm, corresponding to the power density of 5.8 to 6.2MW/m³ because the arrangement and the dimension of fuel pins in the horizontal cross section is already fixed. In the case of this power density, the necessary core diameter and height become 7.6m and 6.7m, respectively, for generation of 600MWt power output. The diameter of RPV becomes 8.2m by adding the dimensions of the annular flow passage within the RPV. Though this value is slightly exceeding the structural limit for the RPV, the reactor power of 600MWt was believed to be achievable. Therefore, the parameter set of 600MWt reactor power and 500℃/ 850℃ core inlet/ outlet gas temperature was selected as a starting design point..

Both reactor thermal power and gas pressure have effects on the volumetric flow rate and eventually on both the performance of turbomachine and the system pressure drop. To clarify these relations, parametric aerodynamic design study of turbine and compressor was done for a reactor power range of 450 to 600MW and gas pressure of 5 to 7MPa. The study resulted in the following relationships to be applicable for projection of adiabatic efficiencies for helium turbine(η_t) and compressor(η_c):

$$\eta_t = \left\{\frac{350}{\Delta T}\cdot\frac{7}{p}\cdot\frac{Q}{600}\right\}^{0.045} \times 90\%$$

$$\eta_c = \left\{\frac{350}{\Delta T}\cdot\frac{7}{p}\cdot\frac{Q}{600}\right\}^{0.025} \times 90\%$$

where, ΔT, Q and p denote a temperature rise(℃) from the reactor inlet to outlet, the reactor thermal power(MW) and the gas pressure(MPa), respectively. As seen, the aerodynamic performance of the turbine and compressor is a strong function of the volume flow rate.

Starting from the parameter set of 600MWt, 850℃/500℃ and 7MPa, which was adopted in the pioneering design[1] established by GA, cycle thermal efficiencies and relative weights of RPV were evaluated. Keeping the pressure constant, the effects of core outlet/inlet temperatures were first examined as shown in Fig. 5. The trade-off among the thermal efficiency, the reactor power and a relative figure (W/R) of RPV weight normalized by that of GA's design indicates an optimum combination of

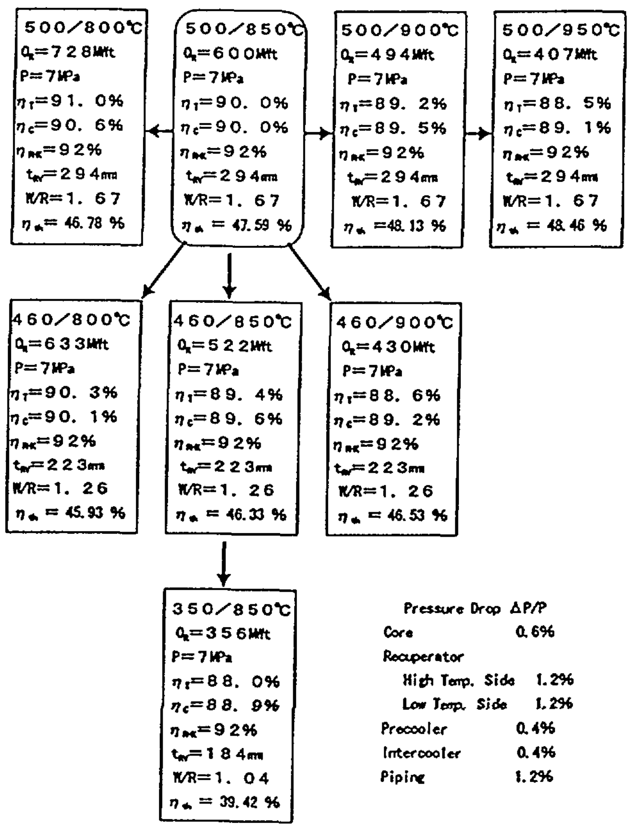

Fig. 5 Effects of reactor inlet and outlet gas temperatures on the cycle efficiency and the RPV weight.

460/850℃ were inlet/outlet gas temperatures. Even though a moderate $T_{R,IN}$ of 460℃ was selected, the value of W/R equals to 1.26 which exceeds considerably the allowable limit of 1.0. Then, the gas pressure must be reduced, as shown in Fig. 6, to 6MPa, in order to obtain a W/R values of 1.07, under which the vessel was considered to be fabricable.

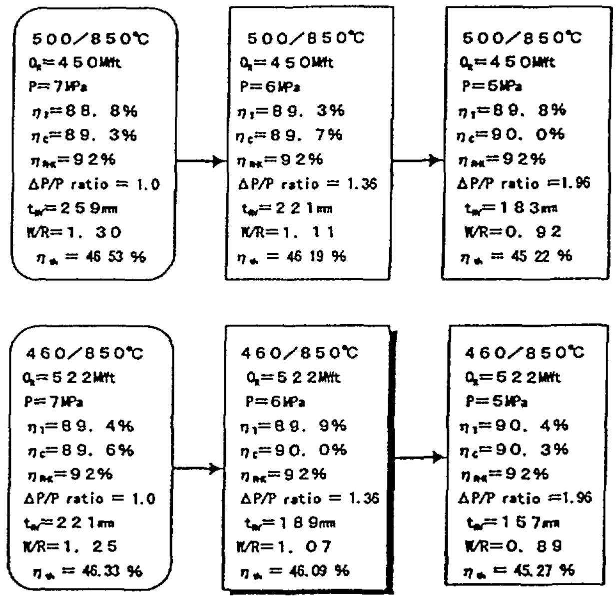

Fig. 6 Effects of reactor gas pressure on the RPV weight and the cycle efficiency.

(2) Reactor Design

The following design specifications were assumed.

- Reactor power=600MWt
- $T_{R,IN}$=460℃
- $T_{R,OUT}$=850℃
- Reactor outlet gas pressure = 6MPa
- Monolithic type fuel element
- Maximum burnup = 10^5 MWD/ton

The following dimensions of fuel block were determined based on the data for the pin-in-block fuel.

- Inner/outer diameter of fuel compact = φ9mm/φ25mm
- Cooling channel inner diameter = φ37mm
- Fuel triangular pitch = 44mm

For the monolithic fuel compact, a coating layer of graphite of 1mm was provided on the surface of fuel compact instead of the sleeve of 3mm thickness. The saved thickness of 2mm was allocated to the cooling channel. Then, the cooling channel became 2mm larger in diameter than that of the pin-in-block fuel element for the HTTR[2]. Figure 7 shows the schematic drawing of this fuel element.

A core configuration was determined to realize the linear heat rate of 140W/cm and the power density of 6MW/m^3 from parametric studies of dimensions and numbers for the fuel block. The optimum values of the core design parameters were established, as shown below, in considerations of a desirable core height to radius ratio, minimized pressure drop through the core, a sufficient core shutdown margin and the maximum effective core flow rate:

- Fuel-block size : distance between parallel walls 390mm
- Number of fuel channels per block 57

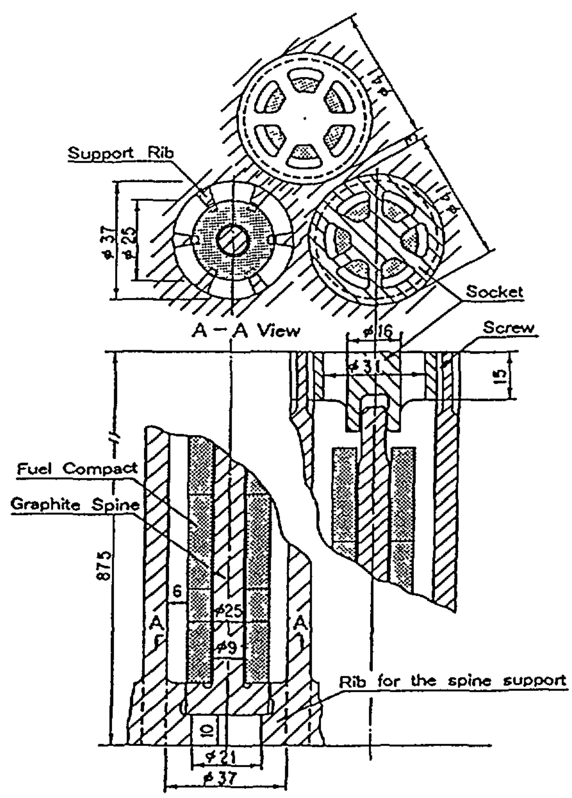

Fig. 7 Schematic drawing of monolithic fuel element.

・Number of fuel columns 108

・Number of fuel-column layers 4

・Number of inner graphite reflector columns 43

The total height of the core and the inner diameter of RPV became 7m (one fuel block height =875mm) and 7.7m, respectively. The cross-section of the reactor is shown in Fig. 8. The control rods were divided into 24 columns, of which 6, 12 and 12 columns were located at the inner reflector boundary, the annular fuel region and the outer replaceable reflector layer, respectively.

To achieve as a flat radial power distribution as possible, the fuel enrichment was varied to 6, 8, 10, 12, 14 and 16%. To achieve an axially uniform maximum fuel temperature, a four-batch fuel exchange method was employed, horizontal layers of burned fuel blocks are removed from the core bottom while an equal two number of layers of fresh fuel blocks are added at the core top every 300days. As there are 8 layers, one complete fuel cycle takes 1200days. By this means, the fresh fuel blocks generating the highest power density are always located at the top at the lowest gas temperature and on the other hand, the burned fuels at the bottom at the highest gas temperature. Therefore, the uniform axial maximum fuel temperature can be established as shown in Fig. 9. The maximum fuel temperatures are maintained just below the allowable limit of 1400℃.

The time dependent temperature behavior was calculated for the depressurization accident. Both the decay heat and after heat in the core are assumed to be transferred from the core to RPV by the heat transfer mechanisms of heat conduction and radiation only, and then from RPV to a reactor cavity cooling system by natural convection and radiation only. The temperature history showing the highest temperature for each of the key core components

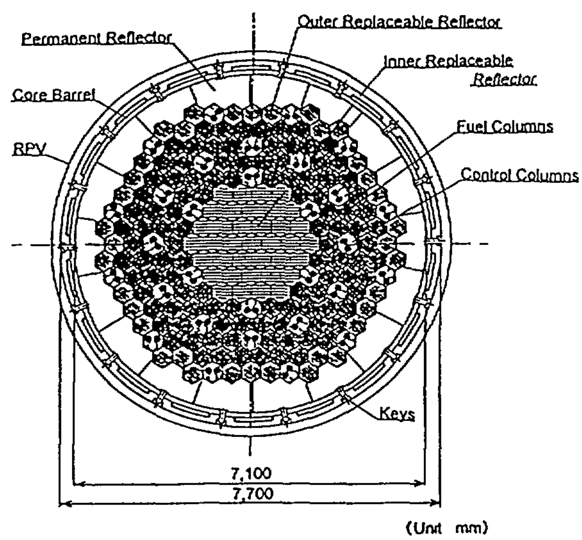

Fig. 8 Horizontal cross section of reactor pressure vessel.

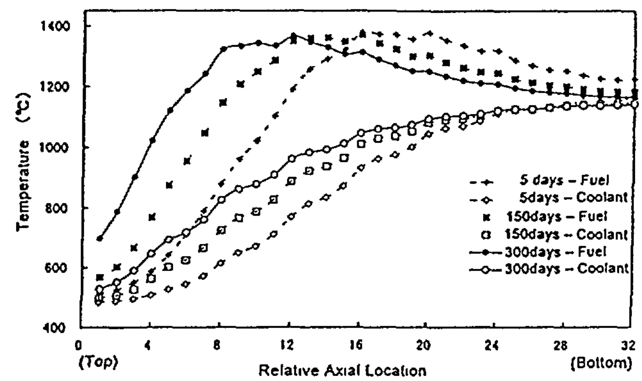

Fig. 9 Axial distribution of the maximum fuel temperature.

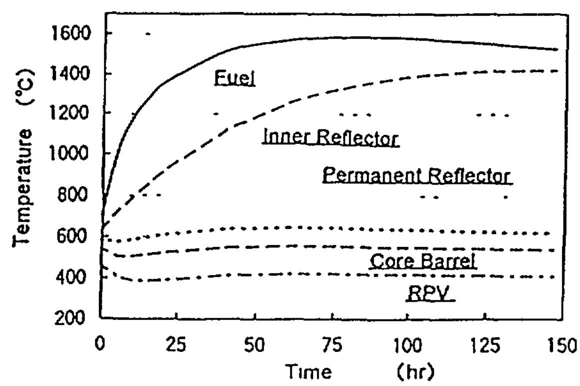

Fig. 10 Time history of temperature at the depressurization accident.

is given in Fig. 10. The maximum fuel temperature remains under the allowable limit of 1600℃ and the passive cooling ability was confirmed.

(3) Gas Turbine Design

As the rotational speed was specified to be 3,600rpm, the other major parameter to be determined is the number of stages. Aerodynamic designs of turbines with 5, 8 and 11 stages have been carried out by varying both loading and flow coefficients. Under the strength limitations for the blades and discs, the condition achieving the maximum adiabatic efficiency was obtained. At this condition, the adiabatic efficiency became 90.78, 93.48 and 93.84% for the 5, 8 and 11 stages, respectively. The rotor lengths of bladed section were 1.20, 1.92 and 2.64m for the above stages, respectively. The increase in the number of stages from 5 to 8 was very effective to increase the adiabatic efficiency. On the other hand, the further increase to 11 stages improves the efficiency marginally but increases the rotor length significantly. The 8 stages was thus considered to be optimum.

As for the compressors, the aerodynamic design calculations were conducted by increasing stages one by one. As the number of stages increases, the diffusion factor, which measures pressure loss per stage, decreases, leading to improved efficiency. However, the amount of reduction in diffusion factor is gradually saturated as the number of stages became large and the total stage loss increases corresponding to the increase of stages. Therefore, an optimum number exists to achieve the maximum efficiency. They were 27 and 29 for the low pressure compressor and high pressure compressor, respectively. Nevertheless, a smaller number of stages, that is, 16 and 17, were selected to shorten the axial length for each compressor, respectively.

The capacities of turbine, low pressure compressor and high pressure compressor are 558MW, 135MW and 135MW, respectively. With a generator efficiency of 98.5% the electric power output becomes 284MWe. At this capacity with a speed of 3,600rpm, the generator becomes necessarily large, that is, ϕ 1,200mm×4,900mm core length. The total length of generator including bearings and exciter became 13,311mm.

The total length of the gas turbine including the generator became more than 20m. Then, it was

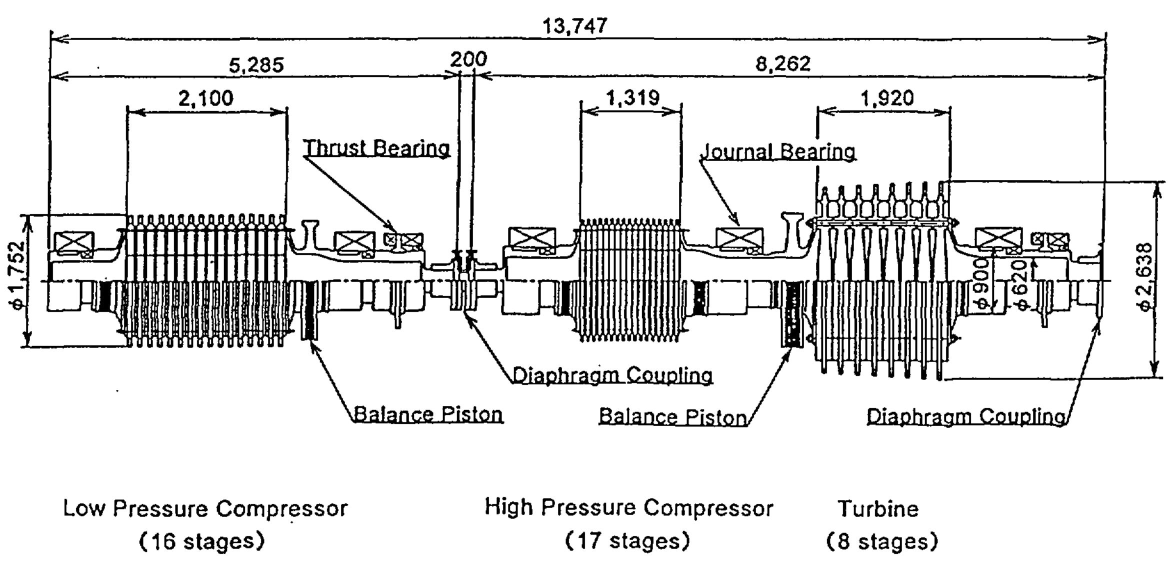

Fig. 11 Helium turbine rotor (284MWe, 296.4kg/s, 3,600rpm).

considered desirable if the rotor could be divided into smaller sections to raise the natural frequencies. Two diaphragm couplings were provided for this purpose as shown in Fig. 11.

(4) Layout Design

In the direct cycle, it is designed to install turbomachinery and heat exchangers in a single vessel called a power conversion vessel(PCV), because the piping between primary components could mostly be eliminated and as a result the containment space and helium-gas leakage could be minimized. The critical issue, however, is the fabricability of such a PCV, where different thermal expansions must be absorbed, pressure seals among components assured at large temperature and pressure differences, and the easy replaceability of contaminated rotating machinery satisfied. To date, the turbomachinery is installed in one vertical PCV in the design of General Atomics[1)] or in two vertical pressure vessels in the design of ESKOM[3)]. These vertical arrangements were chosen to allow easy replaceability by cranes and to save space.

However, the following modification was considered desirable:

- Horizontal arrangement for the turbine rotor,
- Two power conversion vessels, that is, a turbine vessel and a heat exchanger vessel, which are shown in Figs. 12 and 13, respectively.

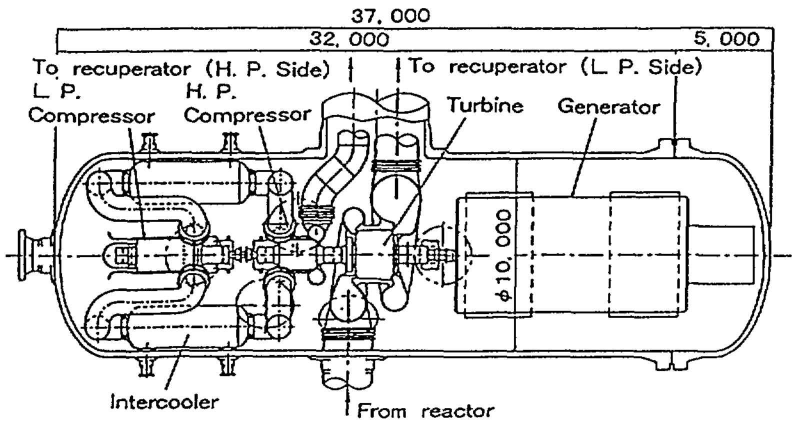

Fig. 12 Vertical arrangement of the turbine vessel.

The horizontal turbine arrangement was preferred because firstly the rotor could be fixed firmly to the vessel wall, secondly the division of the rotor by the diaphragm couplings became possible, and thirdly the large thrust bearing could be eliminated.

The turbomachinery and generator are mounted on the rail provided in the turbine vessel. Since the recuperator and the precooler were transferred to the other vessel, the sufficient space surrounding the turbomachinery is made available for lateral access to the rotor. In case any replacement is necessary, remotely controlled devises are inserted into the vessel through four hatches at the side wall, to remove the bolts and clamps of flange connections. The turbine rotor is then pulled out axially on the rail with its casing, after which, the machine is cleaned, serviced and replaced. If the cleaning is not

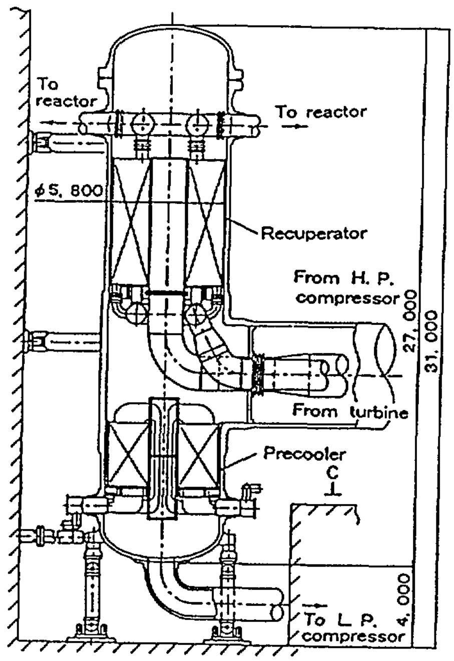

Fig. 13 Vertical arrangement of the heat exchanger vessel.

sufficient for the personal access, another new machine would be installed and the old one would be stored for several years to allow radiation decay.

Figures 14 and 15 show vertical and horizontal views of the total plant, respectively. The reactor vessel is fixed horizontally while the coaxial ducts, the turbine and heat exchanger vessels are permitted to move relative to the reactor by thermal expansion. Sliding mechanisms are employed for support of the turbine vessel to absorb the relatively small movement. The same type of flexible crank supports as used in the light water reactors are employed for support of the heat exchanger vessel.

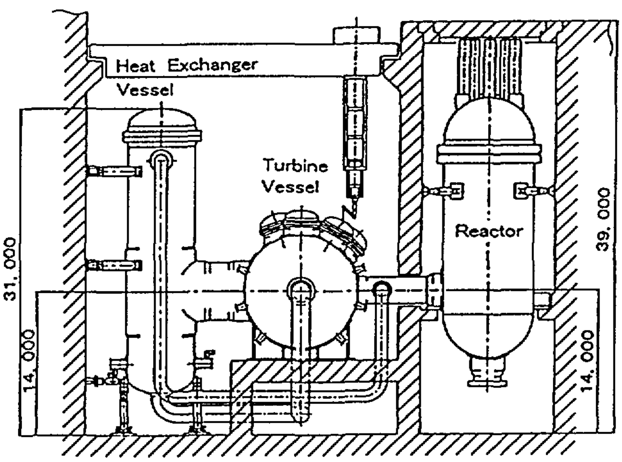

Fig. 14 Elevated view of the plant arrangement.

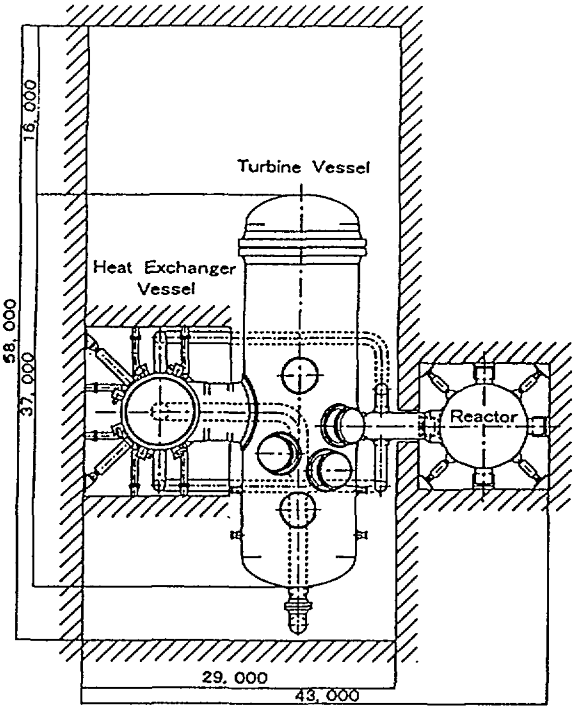

Fig. 15 Horizontal view of the plant arrangement.

(5) Thermal Efficiency

For the above layout, the pressure loss as well as the mass and heat balance were calculated as shown in Fig. 16 and the cycle thermal efficiency was estimated. The recuperator effectiveness was assumed to be 94%, which was considered achievable from the experience of the model fabrication and flow analyses which are described at the next section. The bypass flow of 9.1kg/s from the high compressor outlet to the turbine is needed to isolate the turbine discs from the exposure to the high temperature working helium gas. The efficiency of the generator is 98.5%. The cycle thermal efficiency was calculated to be 47%. Taking into account of the in-plant electric power consumption(7.5MWe) and the parasitic heat loss from RPV(5MWt), the net thermal efficiency of the plant became 45.7%.

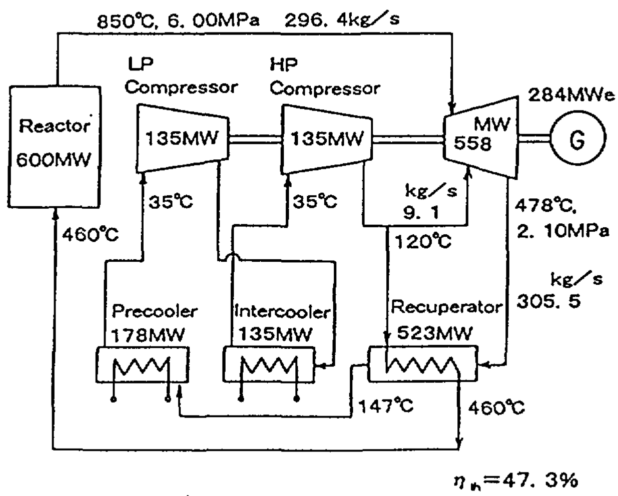

Fig. 16 Mass and heat balance of the 600MWt direct cycle plant.

3. Fabrication of Recuperator-Core Model

The recuperator effectiveness among others affects the cycle thermal efficiency strongly. The maximum values of current industrial plate-fin heat exchangers are around 89%. To obtain the cycle thermal efficiency of nearly 50%, much higher effectiveness up to 95% is required. Therefore, the development of such a high performance recuperator was undertaken.

There are two factors critical to the goal of achieving the 95% recuperator effectiveness. They are the basic heat transfer characteristic and the uniformity of the flow distribution in the heat exchanging core. The former is determined by heat transfer surface design while the latter is affected by the configurations of inlet and outlet headers. According to the calculation conducted in 1996FY, a core surface structure made from small and dense offset-fins of ~ 1mm height could achieve the performance of 95% recuperator effectiveness. A heat transfer core sectional model of 200mm×200mm×200mm shown in Fig. 17 was fabricated. The fin size is 1.15 mm height and 1.4 mm pitch and the material is 304SS. After the fabrication, leak proof tests and pressure proof tests were conducted. At the completion of the tests, the test body was cut into 9 pieces to examine the integrity of bonding structures. All the results were satisfactoy and the leakage rate and the defect ratio were confirmed to be below the allowable limits. Therefore, the fabricability of such a plate-small offset-fin core was experimentally verified. Figure 18 shows the photo of the model and a magnified plate-fin structure.

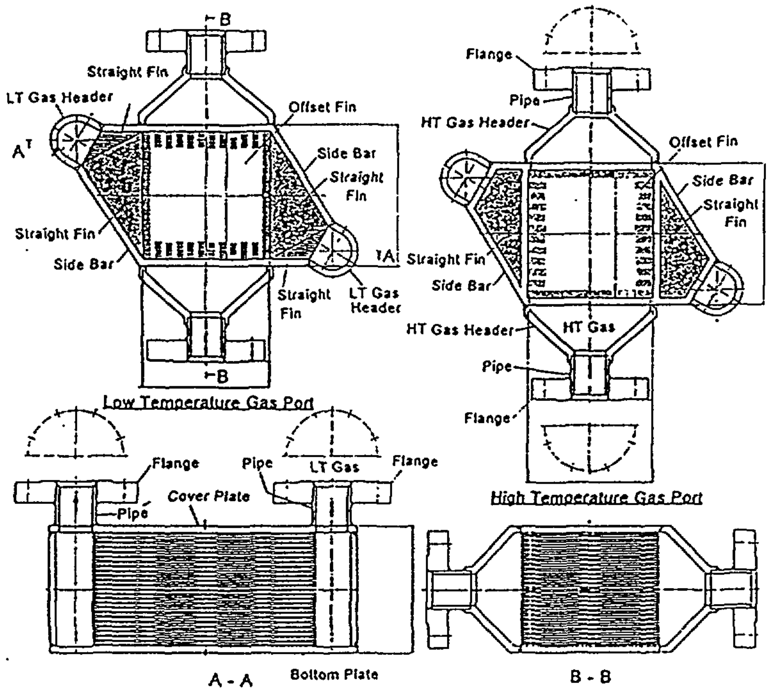

Fig. 17 Heat transfer sectional model of recuperator.

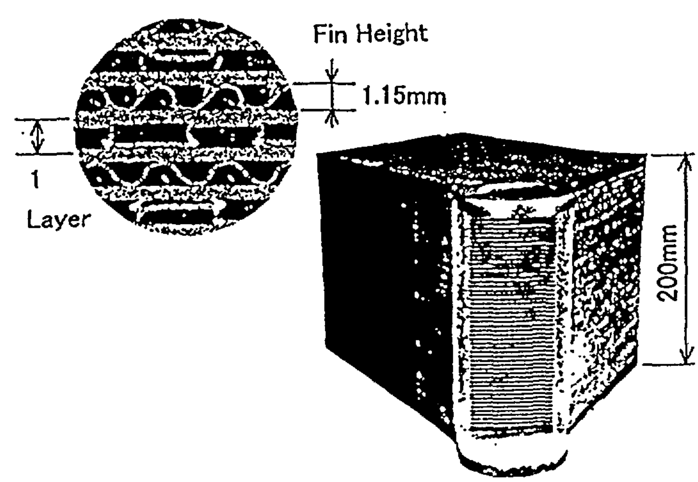

Fig. 18 Photo of the model and a magnified plate-fin structure.

Regarding the flow distribution, analytical research has been carried out for the flow channels at the low temperature gas side of the recuperator for the 600MWt plant. It was revealed that the mass velocity varies 95 to 103% for the average of 100%. To estimate the recuperator effectiveness, the flow distributions of the opposite gas sides must be known. Then, a completely reversed flow distribution was assumed as the worst case. The result of such evaluation showed a recuperator effectiveness of 93.3%. Therefore, the maximum reduction in recuperator effectiveness due to any non-uniform flow distribution was clarified to be 1.7%. In an actual component, the flow distribution could be significantly improved.

To establish the technology for the recuperator of 95% temperature effectiveness, experimental verifications of both the heat transfer characteristics of small size offset-fin and the fabricability of a larger scale model are required.

4. Concept of 300MWt Direct Cycle Plant

In the case of smaller size plant compared with 600MWt plant, it is generally believed that a pebble bed core is more suitable than the block type core. Then, the pebble bed core was selected with two kinds of new challenging concepts under consideration. They are an annular core and a reactor vessel provided with internal thermal insulation.

Regarding the annular core, two designs have been reported so far[4), 5)]. The former has a typical pebble bed core. In this case, a bypass flow will be induced through the central graphite ball region. In the latter, the core is divided to three regions by the radial partition plates and the strength of the

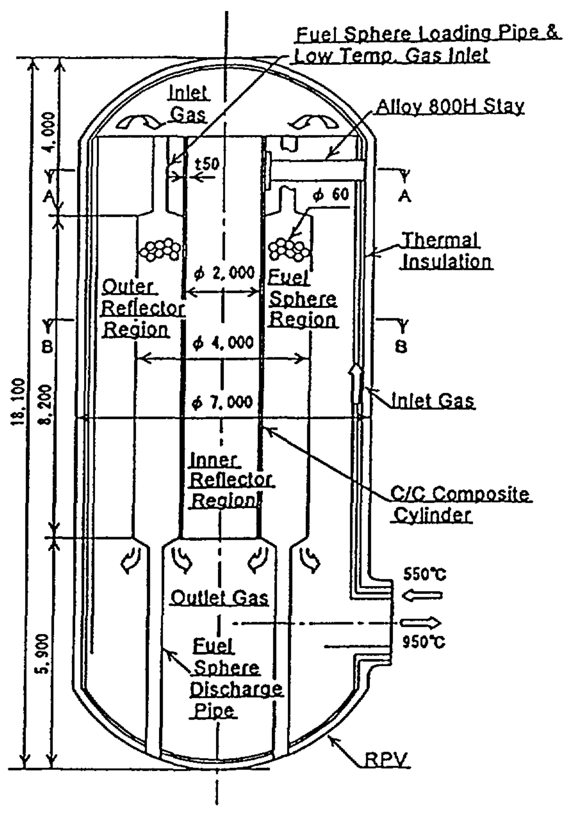

Fig. 19 Annular pebble core with a C/C composite cylinder packed by graphite blocks.

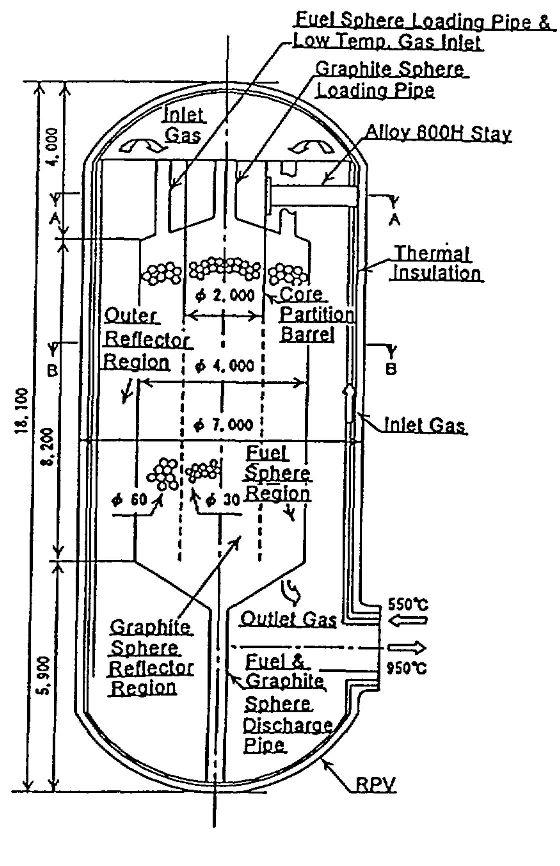

Fig. 20 Annular pebble core with a central reflector region consists of smaller graphite spheres.

partition plates is of concern. We are now considering the following two new concepts shown in Figs. 19 and 20.

The first has a solid central cylinder consisting of an outer cylindrical barrel made of carbon/carbon composite material and the inner stacked circular graphite blocks. The cylindrical C/C composite barrel is supported at the bottom by a mating structure and at the top by radial beams made of alloy 800H.

In the second concept, a cylindrical partition plate is provided only at the top entrance section, where the gas temperature is low enough to use metallic material for the plate. Therefore the strength of partition plate is no longer of concern. In order to reduce the bypass flow through the central region, the size of graphite balls is reduced to half of that of the fuel balls. As the flow area in the central region is a third of the annular area and the pressure drop in the central region is four times larger than that at the annular region, the estimated bypass flow is only less than 10%. In addition, the effect of bypass flow could be further mitigated by means of mixing a small amount of fuel particles to the graphite balls to raise the temperature of the bypass flow.

The second important feature of our concepts exists in the RPV structural design which employs internal thermal insulation. Keeping the $T_{R,IN}$ high enough, for example 550 ℃ is necessary for optimum turbine cycle design. With the thermal insulation on the inner surface of RPV, the temperature of RPV can be limited to less than 350℃, which makes the employment of MnMo steel (A-533) possible. This structure is realized owing to the excellent cooling characteristic at the depressurization accident by both the annular core and the relatively low thermal power density of pebble bed core.

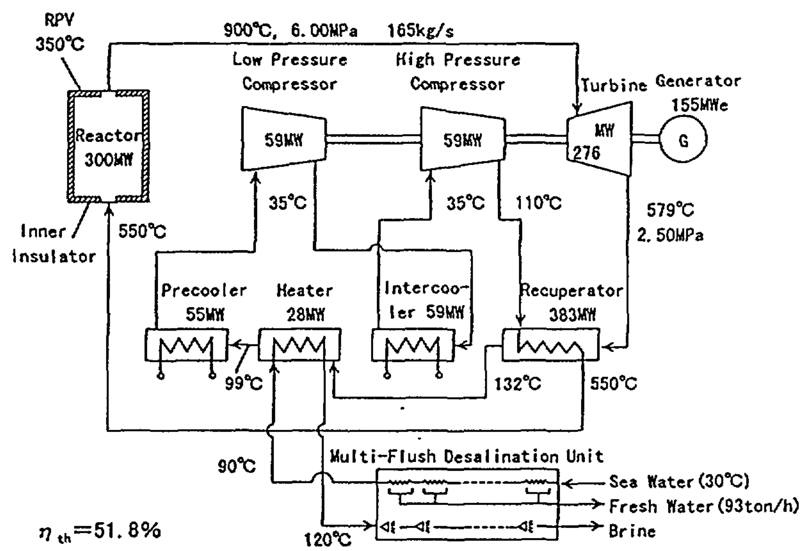

Fig. 21 Flow diagram for the 300MWt direct cycle plant.

By this concept, a very high thermal efficiency is expected as shown in Fig. 21. The design is now ongoing and will be finished in March 1999.

5. Summary

A preliminary design of 600MWt direct cycle plant and the fabrication of a plate-fin core model for the recuperater were carried out in 1997 fiscal year as a feasibility study of HTGR-GT.

The main specifications of the preliminary design include a reactor thermal output of 600MWt, a reactor inlet gas temperature of 460℃, a reactor outlet gas temperature of 850℃ and a helium gas pressure of 6MPa. The 600MWt was selected as the maximum achievable value within the limit of the maximum allowable fuel temperature. The values of 460℃ and 6MPa were selected to ease the structural design of the reactor pressure vessel. Some advanced technologies were adopted such as a monolithic fuel compact and a control rod sheath made of carbon/carbon composite material. These were very effective to enhance the heat transfer of fuel and to reduce the core bypass flow. As a result, a power density of $6MW/m^3$ and the maximum burnup of 10^5MWD/ton were achieved. A single shaft horizontal turbine rotor of 3600rpm was selected. The horizontal arrangement was preferred to ease the mechanical design of the rotor. The turbine, two compressors, a generator and six intercoolers were placed in the turbine vessel. The turbomachinary may be decoupled from the piping by a device manipulated remotely and can be removed in axial direction by rail. The turbomachinery can be replaced under the clean condition by means of FP filtering or by several-year storage for radiation decay. Recuperators and precoolers are installed in the vertical heat exchanger vessel. By this design, the net thermal efficiency of 45.7% was expected to be achieved. A core model of W200mm7L200mm7H200mm with plate-offset-fins of 1.15mm height was fabricated. The results of leak tests, pressurized tests and observation of bonded cross sections were satisfactory. The reduction of temperature effectiveness due to the non-uniform flow distribution across the core was calculated to be 1.7% at most.

In 1998 fiscal year, a design of a direct cycle plant based on pebble bed core at smaller capacity with a desalination system is carried out. In this design, unique design concepts, such as an annular core and internal thermal insulation for the RPV, were adopted. By this means, a very high thermal efficiency is expected.

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FUEL CYCLE RELATED PARAMETRIC STUDY CONSIDERING LONG LIVED ACTINIDE PRODUCTION, DECAY HEAT AND FUEL CYCLE PERFORMANCES

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Abstract

One of the very attractive HTGR reactor characteristics is its highly versatile and flexible core that can fulfil a wide range of diverse fuel cycles. Based on a GTMHR-600 MWth reactor, analyses of several fuel cycles were carried out without taking into account common fuel particle performance limits (burnup, fast fluence, temperature). These values are, however, indicated in each case. Fuel derived from uranium, thorium and a wide variety of plutonium grades has been considered. Long-lived actinide production and total residual decay heat were evaluated for the various types of fuel. The results presented in this papers provide a comparison of the potential and limits of each fuel cycle and allow to define specific cycles offering lowest actinide production and residual heat associated with a long life cycle.

1. INTRODUCTION

Among the well-known identified HTGR characteristics often mentioned, one can remind essentially the inherently safe behaviour of the concept vis-à-vis of the Lost Of Forced Circulation (LOFC) major accident, the highly flexible reactor core allowing many different fuel applications with generally a low long-lived radiotoxic waste production whatever the fuel type concerned. In collaboration with Framatome, a study has been initiated in CEA, to compare the potential of a wide range of diverse fuel cycles from the neutronic point of view (life cycle and generated amount of long-lived actinides) without taking into account the common limits of the fuel particles performances usually respected.

Indeed, being optimistic enough concerning the future fuel performances and technology, it is interesting to perform investigations on long-life fuel cycles without, in a first step, limiting them by the fuel technological recommendations (fluence, %FIMA and burnup). That does not exclude, however, to notice the values achieved in each case. Due to the neutron moderation and reactor cooling functions which are totally independent, HTGR cores allow to accommodate different neutronic parameters, i.e. the fuel/graphite ratio, neutron spectrum, power density, etc... Important parameters such as fissile/fertile particle fraction, particle volume fraction in the graphite matrix, type of fuel (enrichment, plutonium quality and content...), burnable poison,... can be adjusted in order to tentatively approach the optimum cycle life that can be achieved for several types of fuel. For each one and based on the GTMHR-600 MWth concept, equilibrium cycle lengths are given and generated amounts of long-lived actinides are estimated by varying these parameters.

Moreover, it is useful to consider the impact of each possible fuel cycle on the inherent safe behaviour of the concept. The presence in fuel of plutonium, thorium or minor actinides (in case of incineration) does not always induce lower or similar residual decay heat level than in the case of low enriched uranium. Therefore, for each promising fuel cycles, the residual decay heat has been evaluated. Only calculations related to the evolution of the decay heat after a reactor shutdown are performed in this study. The results are compared to the existing ones (benchmark calculation or detailed conceptual studies).

This study is not a complete and exhaustive study of the HTGR fuel cycles, more detailed core neutronics analyses on the capacity of operating with the same high level of safety would be necessary. Preliminary results of the present paper could allow to address in the near future a much more complete reactor core study including the reactivity control aspects, the temperature coefficient, neutronic core stability...

2. CATEGORIES OF FUEL EXAMINED

In the present study, UCO/U_{nat} and UCO/Th have been considered as a first step to validate the calculation process employed. Indeed, UCO/U_{nat} with a uranium enrichment of 20 %, is the basic cycle of the commercial version of the GTMHR-600 MWth power plant proposed by General Atomics. On the other hand, UCO/Th which has been considered for a long time as the most promising fuel of the MHTGR-350 MWth, provides also a possible comparison for establishing a calculation method.

For both fuel types, cycle length in the range of 420 and 462 Equivalent Full Power Days (EFPD) for respectively the UCO/U_{nat} and UCO/Th are commonly announced. In this case, the equilibrium burnups taken into account by General Atomics were 110 and 92 GWd/t.

Moreover, previous studies have shown that due to the high burnups achievable with the coated particles, the fissile isotope composition of the spent HTGR fuel is degraded much beyond the spent fuel standard. Calculations have been then performed with different categories of plutonium exclusively in oxide form inside the particle kernel.

First, plutonium coming from the spent light water reactor fuel (LWR) has been envisaged. Partially burnt in France in the same reactor (LWR) as MOX fuel, it generates a second generation of plutonium that has also been taken into account in the present study. Plutonium containing typically 36 % to 54 % of ^{239}Pu (50 to 66 % in fissile Pu) has allowed to cover the various isotopic vector changes of the plutonium that could be found today or in a near future (planned increases fuel burnups in LWR will lead to lower fissile plutonium content in the spent LWR fuel).

Second, plutonium coming from the dismantling of nuclear warheads has also been studied. Since the end of December 1995, GA, MINATOM, Framatome, and Fuji Electric have been working together on the design of a plutonium-consuming GTMHR with the final goal of replacing the existing weapons-grade plutonium production reactors at Tomsk-7, or other regions of the Russian Federation [1]. In this framework, CEA in collaboration with Framatome is investigating the possibilities of burning highly enriched plutonium in an HTGR. Finally, all the plutonium gathered in the Table I below are classified by their spent fuel origin (enrichment, burnups and fuel management).

Table I. Plutonium isotopic compositions

Cycle name	*Plutonium origin*	Pu^{238}	Pu^{239}	Pu^{240}	Pu^{241}	Pu^{242}	Am^{241}	Pu_f/Pu_t
PU1	PWR 3,7 % (U5/U) 42 GWj/t - 1/4	2,7	54,5	22,9	11,7	7	1,0	66,2 %
PU2	EPR 5,0 % (U5/U) 60 GWj/t	4,49	50,31	23,56	12,39	8	1,25	62,7 %
PU3	PWR 4,5 % (U5/U) 55 GWj/t - 1/6	4	50,4	23	12,3	9,1	1,20	62,7 %
PU4	First generation MOX-PWR 45 GWj/t - 1/4	3,5	47,4	27,1	11,1	9,8	1,10	58,5 %
MOX1	Second generation MOX-PWR 33 GWj/t - 1/3	3,3	41,70	28,70	14,50	10,7	1,10	56,2 %
MOX2	Second generation MOX-EPR 60 GWj/t	5,62	36,75	28,76	13,40	14,13	1,35	50,15 %
PUW	Weapons	0,01	94,0	5,4	0,6	0,01	0,01	94,6 %

3. COMPUTER CODES AND METHODOLOGY

3.1 - Core neutronic calculations

For the following calculations, the French reactor physics code system SAPHYR is used. Usually applied to industrial PWR calculations and research & development purposes, SAPHYR gathered several CEA codes like APOLLO2 (transport) based on data base given by THEMIS/NJOY, CRONOS2 (diffusion-transport), FLICA4 (3D thermal hydraulics)..., which are interconnected. The first calculations presently performed on the GTMHR indicate that SAPHYR is adapted for the assessment of the HTGR performances and characteristics.

The core neutronics analysis is essentially based on a specific calculation process employing the APOLLO2 transport code [2], which is a portable/modular code for multigroup transport calculations. The formalism used to solve the Boltzmann transport equation is either the Integral-Equation (collision probability 1D and 2D) or Integral-Differential-Equation (discrete ordinates and nodal methods in 2D). The main functions of the code are: the self shielding computation, the leakage calculations, the direct and adjoin multigroup flux solver (source and eigenvalue problem, criticality search), the fuel depletion calculation and homogenization by an equivalence procedure.

Containing over 400 isotopes issued mainly from JEF 2.2, the standard 99-groups (47 thermal) APOLIB library has been used in the present study. The calculations have been performed in fundamental mode (critical buckling), considering a linear anisotropic collision hypothesis for the calculation of graphite diffusion coefficient.

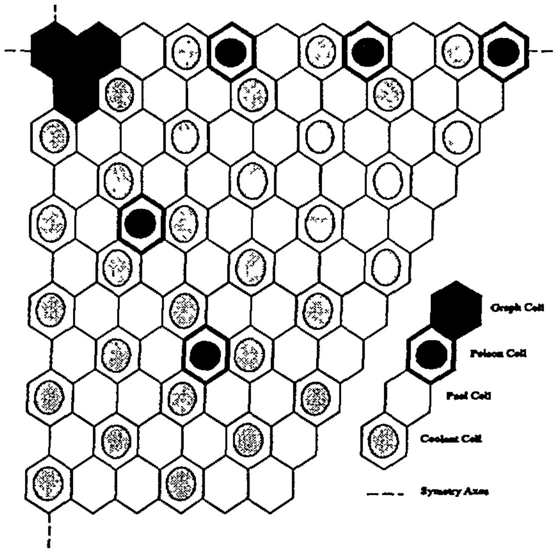

Figure 1 : Multicell geometry - ¼ element

The geometry representative of the standard fuel element is a hexahedral multicell geometry (Fig. 1). 216 channels are available in the element for compact columns of fuel and burnable poison. Some calculations have been performed without fixed burnable poison, such as, for example, in the case of low fissile isotope contained in the plutonium or in the case of high fuel loaded in the core. The poison compacts are therefore substituted with fuel compact, taking care that the plutonium mass loaded in the element is still the same.

Likewise, the double heterogeneity of the geometry characterised by the spherical fuel particles on the one hand and the hexagonal multicell on the other hand are taken into account by a two step internal flux resolution in the code.

The standard depletion chain consists of 20 heavy nuclei and 77 fission products. Nine of the principal heavy nuclides are self shielded during the fuel evolution. The depletion calculations are made independently for each of the media and the isotopic concentrations evolution can be given for fertile, poison and fissile material. Erbium and bore depletion are also taken into account by simplified chains :

$$^{162}_{68}Er \quad ^{164}_{68}Er \quad ^{166}_{68}Er \xrightarrow{n,\gamma} {}^{167}_{68}Er \xrightarrow{n,\gamma} {}^{168}_{68}Er \qquad\qquad ^{10}_{5}B \xrightarrow{n,\alpha} {}^{7}_{3}Li$$

In order to estimate the multiplication coefficient (k_{eff}) of the reactor core, the relation below has been used to determine the k_{eff} from the infinite multiplication coefficient of the fuel element in evolution:

$$k_{eff}^{3D}(J) = \frac{k_{\infty}(J)}{1 + M^2(J)B_g^2} \qquad (1)$$

where :

- ❒ M^2 is the migration area in cm²,
- ❒ B_g^2 is the geometric buckling,
- ❒ k_{∞} is the ratio Production/Absorption in fundamental mode (fuel element calculations),
- ❒ J the burnup.

The equivalent geometric buckling mentioned above is assumed to be constant in the estimation of the evolution of the neutron volumetric leakage. Much more representative of the characteristic core geometry than the global neutron leakage, it takes into account the reflector efficiency function of the neutron spectrum. This parameter is determined for each type of fuel, at the core's beginning of cycle, by using the following formulation and an homogeneous annular core calculation:

$$B_g^2 = \frac{1}{M^2(BOC)}\left(\frac{k_\infty(BOC)}{k_{eff}^{3D}(BOC)} - 1\right) \qquad (2)$$

For each type of fuel cycle, two calculations have been made so as to estimate the radial and axial leakage. The geometry used for the radial calculation is a cylindrical one. The core structure consists of four layers, including internal reflector (graphite), active zone (homogenised fuel element), outer reflector (graphite) and core barrel (steel). The axial leakage is estimated using a 1D plane geometry with a similar description for internal structure.

The formalism used to solve the Boltzmann equation in this configuration is the Integral-Differential Equation, using the discrete ordinates methods (P_1-S_8). The calculation performed on the fuel element in fundamental mode allows to generate a 99-group macroscopic cross section library for each burnup step. These macroscopic cross sections are used for the 99-groups transport calculation described previously (Fig. 2). Finally, the multiplication coefficient of the 3D-annular core can be estimated using the formulation:

$$k_{eff}^{3D}(BOC) = \frac{k_{eff}^{2D}(BOC)}{1+F_z} \qquad (3)$$

where F_z are the axial leakage and $k_{eff}^{2D}(BOC)$ the effective multiplication coefficient calculated on the annular geometry.

The average burnup of the core is estimated at the beginning of the cycle and depends of course on the type of the fuel management. However, the annular core geometry, with a low active core thickness, leads to an important gap between the spectrum in fundamental mode and the average spectrum in the annular core. Considering a core fuelled uniformly with the *PU1* plutonium at a burnup of 250 GWd/t, Figure 2 displays an example of the differences obtained between the average spectrum in the fuel element and in the annular core region.

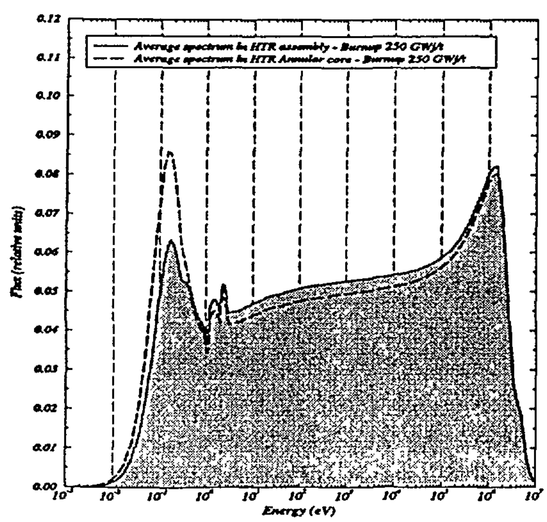

Figure 2 : Neutron spectrum comparison between the fuel element and the core

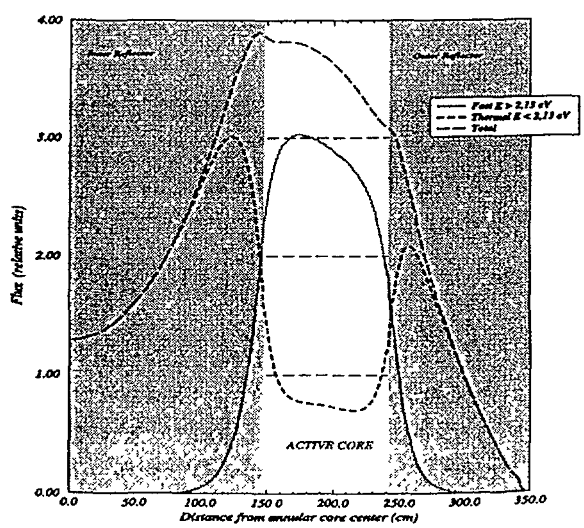

Figure 3 : Core radial fluxes in a 2D annular geometry

Figure 3 exhibits the radial variation of fast and thermal flux in the annular core region. These effects, which are important in the regions close to the reflector, lead to an increase of 3D leakage during evolution. Taking into account the error induced by the calculation, we can estimate a global uncertainty on the fuel cycle length to 20 EFPD.

3.2 - Decay heat calculations

The annular core geometry of the GT-MHR was selected to maximise the power density and still permit passive core heat removal while maintaining reasonable fuel temperatures during accident conditions. Only calculations related to the decay heat removal after the reactor shutdown allow to say if the amount of energy that can be extracted passively during a LOFC accident will not lead to excessive fuel particles temperatures. Moreover, recent benchmark calculations [5] or well defined existing design like the GT-MHR of General Atomic in its commercial version offer a possible comparison of the residual heat evolution curves.

In the code system SAPHYR, the DARWIN/PEPIN2 code [3] calculates, by an analytical method, the radioactive decay equations of 762 fission products and 88 heavy nuclides using a specific CEA library based on JEF2. It allows to assess the concentrations and activities of each nuclide so as to calculate the decay heat and the activity of a reactor, depending of its operating time. The historical irradiation is reconstructed from the multigroup self-shielded cross sections and fluxes (σ_g, ϕ_g) directly issued from the evolution transport calculation for the nuclides present in APOLLO2. The cross sections of the others isotopes are evaluated from the library by using the ϕ_g fluxes.

Only the β and γ decay heat of the fission products on the one hand and α, β and γ of the heavy nuclei on the other hand have been considered. No fission power decreasing kinetic has been considered here. Depending on the temperature coefficient, then of the type of fuel, the major contribution to the fission power decrease is the total reactivity inserted in the core during the reactor shutdown. This fission power evolution correspond to a very low fraction of energy deposited into the core during a LOFC accident (0.1 % of the total energy accumulated during 240 h). It may be omitted in a first approximation.

4. RESULTS AND DISCUSSIONS

4.1 - Fuel element analyses

4.1.1 - Reactivity and neutron spectrum change in evolution

Some of the characteristics of the fundamental mode calculations are presented hereafter. Figure 4 shows the evolution of the multiplication coefficient for three types of plutonium: one weapons grade plutonium (*PUW*) and two reactor grade plutonium with the lower and the higher fissile isotopic weight (*PU1* and *MOX2*). The strong slope observed for the *PUW* at the beginning of life is due to the ^{239}Pu consumption and ^{240}Pu build-up. On the opposite, the erbium's consumption (^{167}Er isotope), combined with the ^{241}Pu production induces a low and steady decrease between 250 and 450 GWj/t. These phenomena do not exist for other plutonium because of the very low mass of erbium charged in the fuel element (there is a ratio of 8 between the erbium charged with *PUW* or *PU1*).

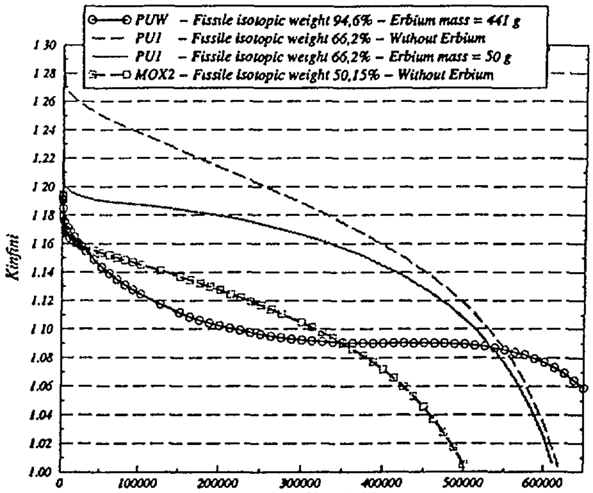

Figure 4 : Infinite multiplication coefficient in evolution for divers plutonium

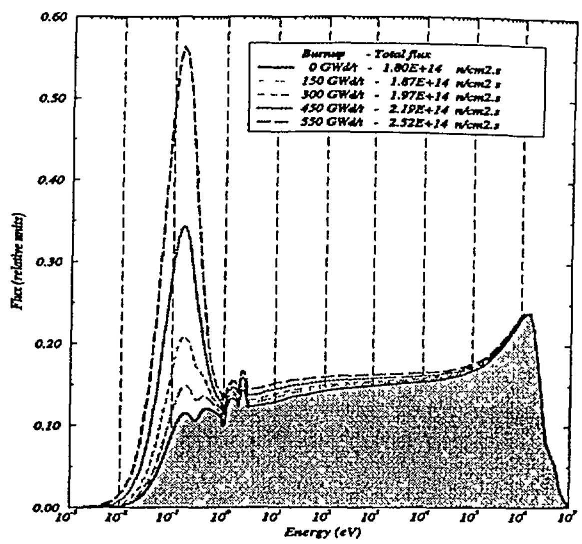

Figure 5 : Evolution of the neutron spectrum in the fuel element - Plutonium *PU1* -

The evolution of the average neutron spectrum in the fuel element is described in Figure 5. The loss of both erbium and ^{240}Pu during the fuel depletion induces an increase of the thermal flux (there is a factor greater than 3 on the thermal flux between the beginning and the end of life).

4.1.2 - The burnable poison impact

It should be stressed that compared to a similar fuel evolution without erbium (*Pu1* in Fig. 4), the fuel with poison presents an initial negative reactivity of around 9000 pcm which strongly decreases towards 600 GWd/t, corresponding to the loss of 90 % of ^{167}Er. The cycle lengths are however comparable. This important feature of the burnable poison equivalent to that observed with ^{10}B allows to adjust the initial reactivity of the different fuel cycle without changing the cycle lengths.

The optimum load of fixed burnable poison has not been estimated in the present study and therefore the different fuel cycles analysed do not exhibit the same initial reactivity. Nevertheless, we should not forget that the presence of the erbium in fuel element may play an important part in the core behaviour with regard to the negative reactivity feedback. Unlike the Doppler coefficient which is always negative, the graphite temperature reactivity coefficient can be positive in presence of plutonium, especially at the end of life.

4.1.3 - Power distribution in the fuel element

Figures 6 and 7 present the power distributions (in percent) for the standard fuel element at the beginning and at the end of life for two types of fuel loaded in the core (respectively 701 and 1200 kg). The average compact power is done taking into account both fuel and burnable poison compact.

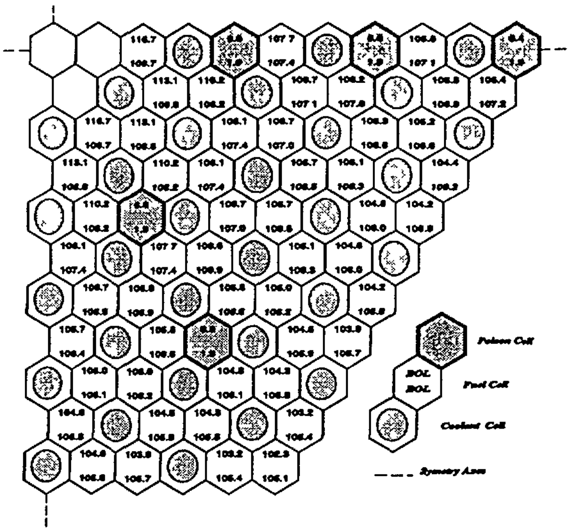

Figure 6 : Fuel element power distribution - Core with 701 kg plutonium *PU1* (with Er)

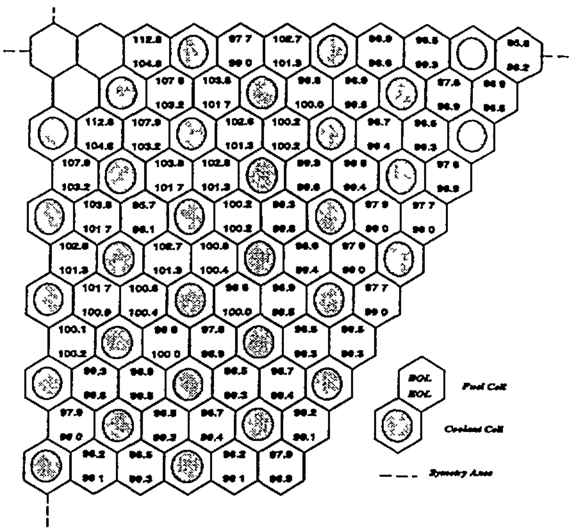

Figure 7 : Fuel element power distribution - Core with 1200 kg plutonium *PU1* (without Er)

One should note that an important power is emitted by the fuel compact located at the centre of the element. Similar to the one observed at the interface core-reflector, this phenomenon is amplified by the presence of the plutonium (a peak power of only 6 to 7 % is obtained with the uranium fuel).

4.2 - Fuel cycles study

Table II below gathers the various parameters characterising the uranium and weapon grade fuel cycles. All the isotopic balances are given for a cooling time of 5 years after the fuel discharge. Uranium cycles employ low enriched uranium (20 %) in fissile particles and natural uranium or thorium as fertile fuel while in the others cycles only one type of particles containing plutonium in oxide form is used.

For the different fuel types feeding the reactor, the discharged burnup was determined in order to achieve a reactivity margin of 2000 pcm at the end of cycle (k_{eff}=1.02) embracing the possible uncertainties. The values of k_{eff} at the beginning of life are not indicated in the tables. They depend essentially of the presence of the ^{240}Pu for the plutonium cycles and must be adjusted by addition of burnable poison without really shortening the cycle length.

The cycle length results obtained with the uranium fuels (U/U_{nat} and U/Th) are in good agreement with those generally encountered in this type of reactor. For both fuels, the production of plutonium and minor actinides is two times smaller than the one achieved in a standard uranium fuel cycle of a pressurised water reactor (PWR). However, as far as the U/Th cycle is concerned, the minor actinides build-up is comparable to the one estimated for a PWR loaded with a similar fuel [4].

In general, the generation of transuranium nuclides is lower in presence of thorium due on the one hand to its position in the heavy nuclei chain and on the other hand to the small capture to fission ratio of ^{233}U produce from the thorium. This tendency is highlighted in Table II between both uranium cycles. That is the reason why, associated with the large epithermal neutron spectrum of HTGR, $^{233}U/Th$ was for a long time considered as the reference cycle of the concept.

The *PUW* cycle presents a cycle length in agreement with the previous studies [1] and a very high fuel burn-up compare to the uranium cycles. Its strong initial reactivity imposes nevertheless the use of burnable poison and demands a multi-batch fuel management scheme. The small ^{240}Pu content in the *PUW* leads to a relatively low build-up of minor actinides. Likewise, a very good utilisation of the heavy metal load in the reactor is observed with some Fission rates greater than 70 % per Initial Fissile Atom (FIFA) and then around 70 % per Initial Metallic Atom (FIMA).

Table II. Uranium and weapon-grade plutonium fuel cycles characteristics

Fuel cycle		UCO/Unat	UCO/Th	PUW
Fuel core loading (kg)		3496/1028	3400/2380	701
Fuel element loading (g)		3543/1042	3450/2416	727,51
compact fuel volume fraction (%)		*20.8/2.8*	*20.8/7.3*	*14.0*
Type of burnable poison		natural B_4C	natural B_4C	natural Er_2O_3
Burnable poison loading		3 g / std elem	3 g / std elem	441 g / std elem
compact poison volume fraction (%)		*0.87*	*0.87*	*22.56*
Fuel power density (W/g)		132.6	99.6	855.9
Total neutron flux (BOC/EOC in n/cm²/s)		1,86 / 2,20 10^{14}	1,83 / 2,04 10^{14}	2,023 / 2,163 10^{14}
Std element peak power (BOC/EOC)		1,075 / 1,0161	1,071 / 1,0161	1,146 / 1,131
Equilibrium cycle length EFPD		**490**	**460**	**260**
Fraction of core refueled per cycle		1/2	1/2	1/3
Average discharged fuel burnup (GWd/t)		130	97.5	668
% FIFA		*87.5*	*81.9*	*73.0*
% FIMA		*13.7*	*9.6*	*69.1*
Particles average fluence (E_n > 0.18 MeV)		1.8 n/kb	1.45 n/kb	3.2 n/kb
Uranium balance per cycle :		**- 368 kg (-16.3%)** **- 109 kg/TWhe**	**- 268 kg (-15.7%)** **- 85 kg/TWhe**	+37 g + 21 g/TWhe
Plutonium balance per cycle :		+ 51.5 kg + 15.3 kg/TWhe	+ 38.6 kg + 12.2 kg/TWhe	**- 169.3 kg(-72.5%)** **- 95.5 kg/TWhe**
Minor actinides balance per cycle :	Np	+ 4.3 kg + 1.2 kg/TWhe	+ 3.4 kg + 1.1 kg/TWhe	+ 3 g + 1.7 g/TWhe
	Am	+ 3.5 kg + 1.1 kg/TWhe	+ 2.4 kg + 0.8 kg/TWhe	+ 7.3 kg + 4.1 kg/TWhe
	Cm	+ 0.3 kg + 89 g/TWhe	+ 150 g + 47 g/TWhe	+ 0.5 kg + 0.3 kg/TWhe
	Th		- 50.4 kg - 17.2 kg/TWhe	
	Pa231		+ 9.2 g + 3 g/TWhe	
	Total : Np+Am+Cm	**+ 8.1 kg** **+ 2.3 kg/TWhe**	**+ 6.1 kg** **+ 1.9 kg/TWhe**	**+ 7.8 kg** **+ 4.4 kg/TWhe**
Np+Am+Cm in % of the consume matter		***2.2 %***	***2.3 %***	***4.6 %***

Table III gathers the results obtained for the plutonium fuels. What should be established at the very outset is that ^{240}Pu appears especially as a fertile material. The higher conversion process existing inside this type of core implies that a great part of the initial plutonium loaded in the reactor core is burnt (between 55 and 72 %). The conversion level of each fuel is underscored by the comparison between the %FIFA and the residual plutonium enrichment. Up to 98 % FIFA can be reached and the fraction of the heavy metal used ranges from 44 to 70 % FIMA.

Table III. Plutonium fuel cycles characteristics (core power density of 6.6 MW/m^3)

Fuel cycles	PuW	Pu1	Pu1	Pu2	Pu3	Pu4	Mox1	Mox2
Fuel core loading (kg)	701	701	1200	1200	1200	1200	1200	1200
Fuel element loading (g)	727,51	727.51	1245.38	1245.38	1245.38	1245.38	1245.38	1245.38
compact fuel volume fraction (%)	*14.0*	*14.0*	*22.5*	*22.5*	*22.5*	*22.5*	*22.5*	*22.5*
Type of burnable poison	natural Er_2O_3	natural Er_2O_3	no erbium	no erbium	no erbium	no erbium	no erbium	no erbium
Burnable poison loading (sdt elem)	441 g	50 g						
compact poison volume fraction (%)	*22.56*	*2.56*						
Fuel power density (W/g)	855.9	855.9	500.0	500.0	500.0	500.0	500.0	500.0
Neutron flux (BOC/EOC in 10^{14} n/cm²/s)	2,023 / 2,163	1,83 / 2,04	1,774 / 1,881	1,793 / 1,908	1,796 / 1,914	1,826 / 1,927	1,827 / 1,923	1,867 / 1,936
Std element peak power (BOC/EOC)	1,146 / 1,131	1,154 / 1,131	1,112 / 1,088	1,110 / 1,086	1,110 / 1,088	1,109 / 1,084	1,107 / 1,085	1,105 / 1,087
Equilibrium cycle length EFPD	260	223	406	384	377	370	356	287
Fraction of core refueled per cycle	1/3	1/3	1/3	1/3	1/3	1/3	1/3	1/3
Average discharged fuel burnup (GWd/t)	668	572.8	609.0	576.0	566.5	555	535.0	431.5
% FIFA	*73.0*	*89.2*	*95.3*	*95.3*	*93.6*	*98.2*	*98.5*	*89.0*
% FIMA	*69.1*	*59.0*	*63.1*	*59.7*	*58.7*	*57.4*	*55.3*	*44.6*
Particles average fluence# (E_n > 0.18 MeV)	3.2 n/kb	2.72 n/kb	4.94 n/kb	4.67 n/kb	4.60 n/kb	4.51 n/kb	4.34 n/kb	3.51 n/kb
Plutonium balance per cycle :	- 169.3 kg	- 151.7 kg	- 283.3 kg	- 271.3 kg	- 269.1 kg	- 266.1 kg	- 259.9 kg	- 218.9 kg
	-72.5%	-65.7%	-71.7%	-68.7%	-68.1%	- 67.2 %	-65.7%	-55.5%
(kg/TWhe)	- 95.5	- 99.8	- 102.4	- 103.7	- 104.5	- 105.5	- 106.9	- 111.7
Residual enrichment (Pu_f/Pu_{tot} at the EOL)	54.7 %	31.2 %	35.9 %	35.7 %	36 %	35.6 %	34.9 %	36.6 %
Minor actinides balance per cycle : Am	+ 7.3 kg	+ 10.6 kg	+ 21.2 kg	+ 22.5 kg	+ 23.8 kg	+ 25.4 kg	+ 27.3 kg	+ 30.4 kg
(kg/TWhe)	+ 4.1	+ 7.1	+ 7.7	+ 8.6	+ 9.2	+ 10.0	+ 11.2	+ 15.5
Cm	+ 0.5 kg	+ 3.1 kg	+ 9.8 kg	+ 9.9 kg	+ 10.4 kg	+ 10.9 kg	+ 11.2 kg	+ 10.0 kg
(kg/TWhe)	+ 0.3	+ 2	+ 3.5	+ 3.8	+ 4.1	+ 4.3	+ 4.6	+ 5.0
Total M.A. : Np+Am+Cm	+ 7.8 kg	+ 13.8 kg	+ 31.0 kg	+ 32.3 kg	+ 34.2 kg	+ 36.3 kg	+ 38.5 kg	+ 40.4 kg
(kg/TWhe)	+ 4.4	+ 9.1	+ 11.2	+ 12.3	+ 13.3	+ 14.4	+ 15.8	+ 20.6
M.A. in % of the initial metal burnt	*4.6 %*	*9.1 %*	*10.9 %*	*11.9 %*	*12.7 %*	*13.6%*	*14.8 %*	*18.4 %*

the maximum fluence can be estimated by taking into account a core peak power in the order of 2.4 multiply by the element peak power mentioned above.

Besides, the reactivity swings presented in Figure 4 are significantly smaller for the reactor grade plutonium than those observed for the weapon grade plutonium. In fact, the greater the fertile isotopes content, the smaller the criticality swing, because of the high conversion process.

Therefore, the reactor grade plutonium might not necessitate additional poison if its initial mass was adjusted. This is illustrated by the *PU1* cycle envisaged with two different masses of fuel fed in the core. A very flat curve of the k_{eff} evolution is observed for the highest load case and permit much more longer cycle lengths. A once through management scheme could even be envisaged if adequate means were used to minimise the radial peak power at the core-reflector interface (fuel managed on a three-batch basis allows to soften annular-radial power distribution by placing irradiated fuel near the reflector).

It is noteworthy that second generation plutonium can also be used as fuel in the GT-MHR but leads, unfortunately, to a higher minor actinide level of production due to the important ^{242}Pu initial concentration. Finally, the HTGR seems to be a good candidate to use all type of plutonium and then could reduce the plutonium stockpiles.

4.3 - Residual decay heat results

Figure 8 and 9 describes for various types of fuel the evolution over 100 days of the residual power of a the GTMHR-600 MWth after a reactor shutdown arisen at the end of cycle.

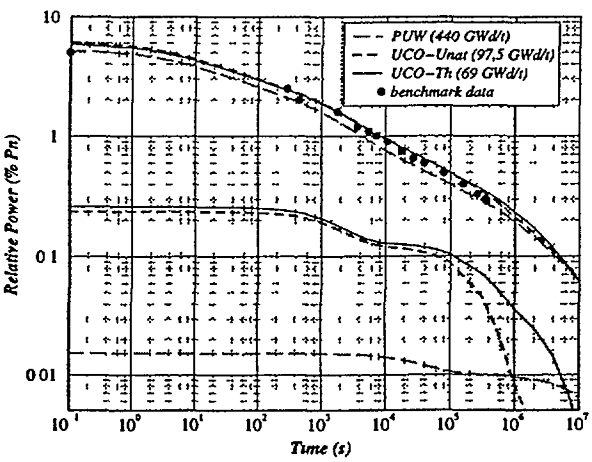

Figure 8 : Decay heat evolution (heavy nuclei and total) for the *PUW* and Uranium fuels.

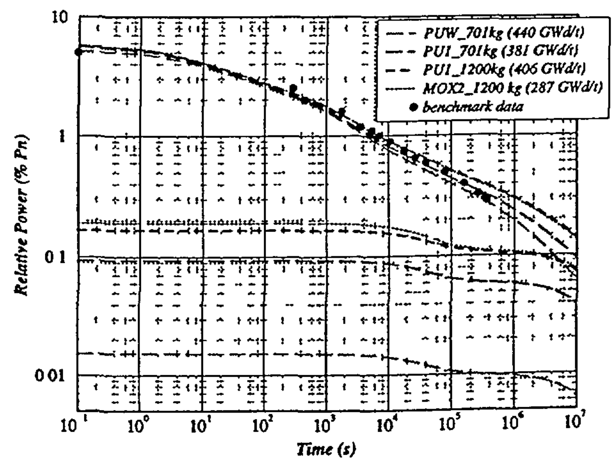

Figure 9 : Decay heat evolution (heavy nuclei and total) for different plutonium fuels.

The residual power obtained in the cases of the uranium fuels are very similar (Fig. 8). For the first 100 hours, the high heavy nuclei heat source resulting essentially from beta emission of ^{239}Np and ^{239}U on the one hand and, on the other hand, the high fission product afterheat (total energy generated during one cycle greater in the case of the uranium fuel) leads to a total residual power greater (10-20%) than for the *PUW* fuel.

On the Figures 8 and 9 are also indicated the power values used for the last calculational benchmark aiming at estimating the fuel and vessel maximum temperatures achieved during a thermal transient following the LOFC major accident [5]. These values appear slightly conservative compared to the *PUW* rundown curves observed on Figure 8 and obtained for identical cycle length and fuel type. One of the essential characteristics of the *PUW* fuel is the low heavy nuclei residual heat due to the small initial amount of ^{240}Pu which therefore did not produce a significant amount of actinides. In spite of the high operating power density of the *PUW* fuel, the total residual power remains lower (10 to 20 %) than the one of the uranium fuel with a cycle length almost two times longer.

When other plutonium isotopic compositions are taken into account (Fig. 9), the higher amount of minor actinides created, generates a higher heat source of radiation decay of the heavy nuclei. In particular, a factor 2 between the *PU1* and the *MOX2* on the quantity of ^{242}Pu (yielding the strong alpha emitter ^{242}Cm) and ^{238}Pu, leads to a greater actinides residual power level for the MOX2.

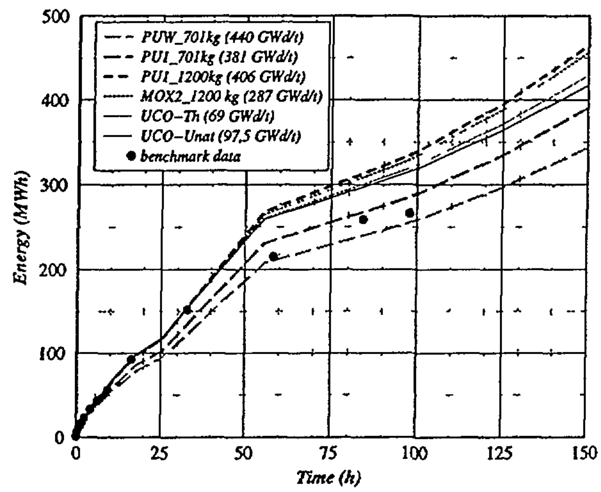

Figure 10 : Energy stored in the core after the reactor shutdown

Figure 10 opposite shows detailed evolution of the stored energy during a typical length of time of the LOFC accident. The maximum difference achieved between all the examined fuels is on the order of 30 % (100 h). This values should of course have a significant impact on the maximum temperature reached by the fuel in the course of the accident. Independently of the fuel performances, only a sensitivity study of the GT-MHR to the residual power, taking into account the radial power profile in the annular core, will permit to define the acceptable limits to recycle highly degraded plutonium fuel. Finally, it should be stressed that an increase of the fuel loading (*PU1* case) induces directly a growth of the energy deposited in the core and limits therefore the possibility to adjust the initial mass of fuel.

5. CONCLUDING REMARKS

This paper is a part of a much more general and exhaustive study which is aimed at assessing the potential application of several fuels in the HTGR providing long cycle lengths. Starting from fuel cycles previously studied, a calculation method dedicated to a parametric analysis, has been set up to evaluate the evolution of the reactor core reactivity during the fuel depletion. The core modelling takes into account particular features of the HTGR such as the double geometric heterogeneity and neutron leakage evolution in the annular core.

The first results obtained with a wide spectrum of plutonium isotopic compositions prove HTGR potentials to use the plutonium as fuel without generating large amounts of minor actinide and respecting the same high level of safety. If the very high burnups reached by fuel particles were confirmed, the GT-MHR would be a good candidate to burn economically all type of plutonium. Providing energy, this option could therefore reduce the plutonium stockpiles.

Long cycles are possible if burnups as high as 700 GWd/t and fluences in the order of 12 n/kb (a factor 2 with the common requirements) sustained by the fuel particles are technologically feasible. Nevertheless, more detailed core neutronic analysis are necessary to assess the reactivity control aspects, the temperature coefficients, neutronic core stability... These additional analyses should also permit to define the appropriate fuel management and to answer the power distributions related issues especially important in the case of the plutonium use. Only such an effort will allow to conclude on the feasibility to have in an HTGR long plutonium cycles greater than three years.

ACKNOWLEDGEMENT

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ACTIVITIES OF RAHP AND PROBLEMS TO BE CLEARED TOWARDS COMMERIALIZATION

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Abstract

Research Association of HTGR Plant (RAHP) is the sole research association on HTGR Plants in industrial sector of Japan, since its establishment in 1985.

Activities in these years were to analyze world R&D trends, to set-up fundamental development strategies, and to incorporate the study results into actions towards commercialization of HTGR.

RAHP is now mainly investigating fuel cycle aspects and development programs of small modular reactors, such as US-Russian GT-MHR and South African PBMR.

Conclusions obtained through the activities so far are;

(a) From view points of effective use of energies and reduction of environmental impacts in global scale, development of nuclear power is essential, and that of HTGR is desirable in particular, because of its very highly inherent safety and feasibility of high temperature heat utilizations. The role of HTGR is inter-complementary with those of LWR and FBR.

(b) Future subjects and problems to be cleared towards commercializing HTGR are; (1) safety, technical and economical demonstrations, including inherent safety, fuel integrity as key of the safety, gas turbine cycle, high temperature system performances, etc., (2) acceptance of safety design concept, and preparation for its evaluation and licensing.

(c) Construction of demonstration plant(s) as a total system, through effective international cooperation, such as internationally common or sharing development on basic machinery systems between GT-MHR and PBMR Programs, becomes most significant.

1. BRIEFING OF RAHP

RAHP is the sole research association in private or industrial sector of Japan, with respect to HTGR plant. It was established in 1985, and has been chaired by Shigehiro An, Professor (Professor Emeritus, at present) of University of Tokyo.

Theme of the research has consistently been "Plant Concept of HTGR and its Commercial Feasibility".

Office is, at present, set up in The Institute of Applied Energy (IAE), Tokyo, which is one of the affiliated associations of Ministry of International Trade and Industry (MITI).

Membership is increasing with increased interest in recent HTGR related trends, which are observed in South African Program, US-Russian Program in Russia, or preparation of setting up of Global HTGR R&D Network (GHTRN) in IAEA, for example. RAHP is, at present, composed of 5 specialists from academia, and representatives from member companies, which are all 11 electric power companies, and 7 fabricators in nuclear industry. And recently representatives of New Energy and Industrial Technology Development Organization (NEDO), as well as Japan Atomic Energy Research Institute (JAERI), which has been constructing High Temperature Engineering Test Reactor (HTTR) and conducting fundamental R&D of HTGR as a national research institute, are being invited as observers.

Main activities are categorized as follows;

(a) Periodical research meetings for investigation on global trends, including special invited lectures as well as exchange of mutual views,
(b) Working group meetings for study on specific problems,
(c) Activities, related to public and industrial acceptance in the country and abroad, including presentations at domestic or international symposiums,
(d) Publishing documents, such as specific reports, executive summaries or annual reports.

Main activities in these years and problems considered necessary to be cleared are presented below.

2. MAIN ACTIVITIES OF RAHP IN THE PAST SEVERAL YEARS

2.1 Fiscal years 1993 and 94

In 1993, RAHP changed the style of study, from previous specific topics oriented approach to strategic approach towards the object theme of the association, so as to analyze the present status and to have a common future outlook among the participating members, based on information obtained through the activities.

That is, RAHP set up "PA Working Group", and performed a series of sorting studies, standing really on its starting point, on "What is HTGR?", "Why to develop HTGR now?", "What are the problem areas?", "What concepts and Which size?", "Until When, Where and How to commercialize?" and so on[1].

2.2 Fiscal year 1995

Investigated present status and future subjects on graphite technology and seismic design, as well as evaluation methods on HTGR plant economy.

As a part of the investigation of international trends of R&D and as one of the PA activities, participated in IAEA Advisory Group Meeting on Sea-water Desalination Using Nuclear Energy, July 24-26, 1995, Vienna, as well as in IAEA Technical Committee Meeting (TCM) and Workshop on Gas Cooled Reactors with Closed Cycle Gas Turbine, Oct.30-Nov.2, 1995, Beijing.

2.3 Fiscal year 1996

Reviewed and studied present status of High Temperature Engineering Test Reactor (HTTR) of JAERI, Gas Turbine Modular Helium Reactor (GT-MHR) plant technology under conceptual design by GA (USA) and MINATOM (Russia), and methodology for long term forecasting of energy supply and demand, etc..

Participated in IAEA-TCM and GHTRN Meeting, Nov.11-19, 1996, Johannesburg[2],[3], and visited "planned construction site" of the first module HTGR "Pebble Bed Modular Reactor (PBMR)"[4], at Koeberg near Cape Town, which site was internationally announced for the first time at the TCM in Johannesburg.

As a part of activities related to public and industrial acceptance, visited Japan Atomic Industrial Forum (JAIF), and reported on the present status and new trends of R&D on HTGR in the world, including the above mentioned South African and US-Russian Programs.

2.4 Fiscal year 1997

Investigated international trends of R&D on HTGR and relative positioning of HTGR in light of Next Generation Reactors, through attending International Conference on Nuclear Engineering (ICONE-5), May 25-29, 1997, Nice, and HTGR TCM, Nov.10-12, 1997, Petten[5],[6].

Continued investigating on fundamental scenario for promotion of international commercialization, and on targets of economy, etc..

Started study on nuclear fuel cycle aspects of HTGR.

Along the above line, attended at International Nuclear Fuel Cycle Symposium, June 3-5, 1997, Vienna, and also newly set up "Fuel Cycle Working Group" in RAHP.

At Petten, proposed financial contribution to GHTRN activities.

2.5 Fiscal year 1998 (as of Sep. 1998)

Investigating present status of international R&D on HTGR and Next Generation Reactors, through attending Pacific Basin Nuclear Conference (PBNC-98), May 3-7, 1998, Banff, and ICONE-6, May 10-14, 1998, San Diego.

Through such activities, RAHP is clearly recognizing that small modular HTGRs, such as PBMR and GT-MHR, are becoming the candidates for most desirable and promising countermeasures for safe, economical, and environmentally friendly energy sources, in case of adoption in global scale.

Through Fuel Cycle Working Group activities, is developing several scenarios of HTGR fuel cycle, ranging from Non-reprocessing (Once-through) to Reprocessing cases, in combination with fuel types (Uranium Oxide, Plutonium Oxide, and others) and fuel burn-ups.

Made substantial efforts to try to contribute to GHTRN activities as in previous year.

3. PROBLEMS NECESSARY TO BE CLEARED TOWARDS COMMERCIALIZATION AND INTERNATIONAL COOPERATION

Through above RAHP activities, problems considered necessary to be cleared towards commercialization, and international cooperation considered necessary, especially towards design and construction of the first module(s), are as follows;

3.1 Safety, Technical, Institutional and National Policy Problems

(a) Recognition and acceptance of inherent safety design concept by authorities
- National authorities
    - US, Russia, South Africa, Japan, China
    - Other interested countries
- International organizations
    - IAEA, OECD/NEA, etc.

(b) Preparation of evaluation and licensing on inherent safety
- Establishment of data base on inherent safety
  - Fuel integrity, as a key of the safety
    - integrity during fabrication and irradiation of coated fuel particle, especially at high burn-ups
  - System safety of Containment Vessel- free design
    - countermeasures for large scale unexpected fuel failure
  - Technical standards and licensing criteria for the safety

(c) Technical development and demonstration of machineries and systems at high temperatures
- Reactor pressure vessel
  - material integrity at high temperatures (in case of GT-MHR)
- Closed He gas turbine system
- Magnetic bearing of large scale
- Recuperator
- Countermeasures for problems caused by thermal expansion among machineries within Power Conversion Unit
- Graphite material
  - specifications and quality control (for impurities)
- FP plate-out
  - performances
  - countermeasures
- Maintenability of machineries
  - replaceability, repairability

(d) Recognition and acceptance of Non-recycling policy
- Expansion of fuel cycle options to include Once-through cycle (especially in case of Japan)

3.2 Economic Problems

- Economic impact of quality instability of materials, equipments or sub-systems, to be expected in connection with international competitive bid of the plant (in case of PBMR)
- Demonstration of plant operation and maintenance

3.3 International Understanding and Cooperation

- International grasp of comments (needs, demands, hopes, etc.) from developed and developing countries
  - understanding on internationally common desire for safer and cheaper nuclear power

- understanding on internationally common hope for evacuation free nuclear power plant
- Avoidance of international undue commercial competitions or meaningless duplication of development works (especially during initial stages of development)
- International cooperation and effective share in role, such as internationally common or sharing development of basic machinery systems between GT-MHR and PBMR Programs
- Construction of first module(s), as test/ demonstration/ first commercial plant(s), on international cooperation bases, through information exchanges, joint works, for example.
- Importance of the role of IAEA in establishment of international coordination system

4. CONCLUSIONS OBTAINED THROUGH RAHP ACTIVITIES

4.1 Nuclear Power and Desirability of HTGR

From view points of effective use of energies and reduction of environmental impacts, in global scale, development of nuclear power is essential, and that of HTGR is desirable in particular.

HTGR is of very highly inherent safety and of feasibility of heat energy utilization at high temperatures such as nearly 1000 deg.C, as well as electric power use. And the role of HTGR is in inter-complementary relationship, rather than conflicting with those of LWR-FBR, which has long been adopted as a basic line of nuclear energy development, in Japan at least.

4.2 Problems to be cleared

Problems or future subjects to be cleared on HTGR are considered to be;

(a) Technical demonstration of inherent safety and high temperature technologies

(b) Authoritative, industrial and public acceptance of inherent safety

(c) Economic prospects towards its commercialization

(d) R&D through international cooperation and effective share in roles, and

(e) Successful construction and operation of demonstration plant(s), which turns out to be the first module(s) for commercialization

4.3 Planned activities of RAHP in future

(a) RAHP is planning to continue survey and analysis of global trends of HTGR development, refinement of national and international development scenario, identification of remaining subjects to be cleared, in technical and economic fields, investigation on fuel cycle features, and activities for acceptance of HTGR and its inherent safety.

(b) RAHP is also planning to support Test Reactor Programs such as HTTR and HTR-10, global activities like GHTRN, and Programs for HTGR realization, such as PBMR and GT-MHR.

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(Session 2)

SAFETY EVALUATION DURING A DEPRESSURIZATION ACCIDENT OF THE SFHTR

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Abstract

High temperature gas-cooled reactor (HTGR) has many good characteristics that inherent safety is very high, high temperature heat of near 1000℃ is available, high heat efficiency and high fuel burn-up are expected, etc. To obtain a public acceptance for the HTGR having many good characteristics mentioned above, it is a crucial point to show and demonstrate the outstanding inherent safety, for example, the safety during a severe accident. The reactor transient during the depressurization accident without reactor scram (DAWS) has been analyzed for the severe accident free HTR (SFHTR) with 400MWt. The DAWS has a possibility to become the severe accident in the HTGR. To simulate the reactor dynamics and heat removal from the surface of reactor pressure vessel during the DAWS, the one point-approximation reactor dynamics code with single channel model representing the core was combined with the thermal analysis code with whole reactor model. The combined calculation code is able to calculate the reactivity balance due to xenon build up and reactor cool down during such accident. From the analytical result, it was found that no reactor meltdown occurs even in the worst scenario. The calculation code developed to simulate the DAWS is applicable to the evaluation of beyond design basis accidents for the SFHTR. Also this code is applicable to the simulation of the safety demonstration test planned in the High Temperature engineering Test Reactor under conducting a reactor physics test at present. This paper describes the reactor transient during the DAWS and the newly developed analytical code.

1. INTRODUCTION

It would be required to confirm the safety during the severe accident for the future advanced reactor from the viewpoint of getting public acceptance easily. High Temperature Gas-cooled Reactor (HTGR) has a high potential to the safety during the severe accident because of the highly inherent safety feature. It is necessary to demonstrate the safety during the severe accident. The analytical model to simulate the severe accident is an indispensable tool to design advanced HTGRs in the future. In the HTGR, the depressurization accident without reactor scram (DAWS) has a possibility to become the severe accident. The severe accident is selected as an event that the probability of occurrence is smaller than accident and reactor core damage might be expected.

The analytical code to simulate the reactor transient during the DAWS was newly developed. In order to simulate such transient, the single channel model to calculate the neutron kinetics with one point approximation was combined to the whole core model to calculate the thermal dynamics in the reactor. The single channel model calculates the reactor power and it is applied as a heat generation in the whole core model. The whole core model calculates the temperature distribution and a heat flow estimated from it is applied as a thermal boundary condition in the single channel model. The single channel model calculates the reactivity balance due to a change of poison (xenon) density and core temperature. The analytical model combining the single channel with the whole reactor core is very useful to evaluate the consequence of DAWS.

The analytical simulation during the DAWS was performed with newly developed code for the severe accident free HTR (SFHTR). Japan Atomic Energy Research Institute (JAERI) started to design the severe accident free HTR (SFHTR) with 450MWt [1]. A modular HTGR with a medium thermal output would become a best power source for the island and big city because of that appropriate power size and inherent safety. The SFHTR has many inherent safety features that are the high fuel performance, negative temperature coefficient, low power density and large heat capacity of the core, inert and single-phase characteristics of helium coolant, etc. The modular SFHTR is one of candidates of future HTGR. From the analytical result, it was found that no reactor meltdown occurs even in the worst scenario.

The developed calculation code is applicable to the evaluation of beyond design basis accidents for the SFHTR. Also this code is applicable to the simulation of the safety demonstration test planned in the High Temperature engineering Test Reactor (HTTR) under conducting a reactor physics test at present. The HTTR has a highly inherent safety feature and the first high temperature gas-cooled reactor in Japan [2]. The construction of HTTR has been completed and the performance test to confirm the function of each component is almost completed. The fuel loading is under going and the first critical approach will be held soon. JAERI has a plan to perform the safety demonstration test using the HTTR. The partial trip test of helium gas circulator and the control rod withdrawal test have been licensed in the HTTR. The purpose of the safety demonstration test is to confirm the safety margin to the reactor transient is large even if the severe accident would be assumed in the HTTR, that is in the HTGR.

2. OUTLINE OF SFHTR [1]

Table 2.1 shows major specification of SFHTR with 450MWt. Fig.2.1 shows the conceptual view of SFHTR. The reactor core consists of 108 columns and 5 pin-in-block typed fuels are piled up vertically in each column. Some columns except fuel column are located in the center of reactor core.

Table 2.1 Major Specification of modular SFHTR

Thermal output	450 MW
Core	Annular arrangement
Height	6.19 m
Outer flat length	4.9 m
Inner flat length	2.9 m
RPV outer diameter	8.4 m
Outlet/inlet coolant temperature	950/350 °C
Coolant(helium) press.	6.0 Mpa
Fuel block	Prismatic (pin-in) block
Power density	Average 6 W/cc
Fuel coating material	ZrC
Burn-up	100 GWd/t
Allowable fuel temperature	1800 °C
Fuel shuffling	Axial shuffling every year
Safety system	Complete passive cooling

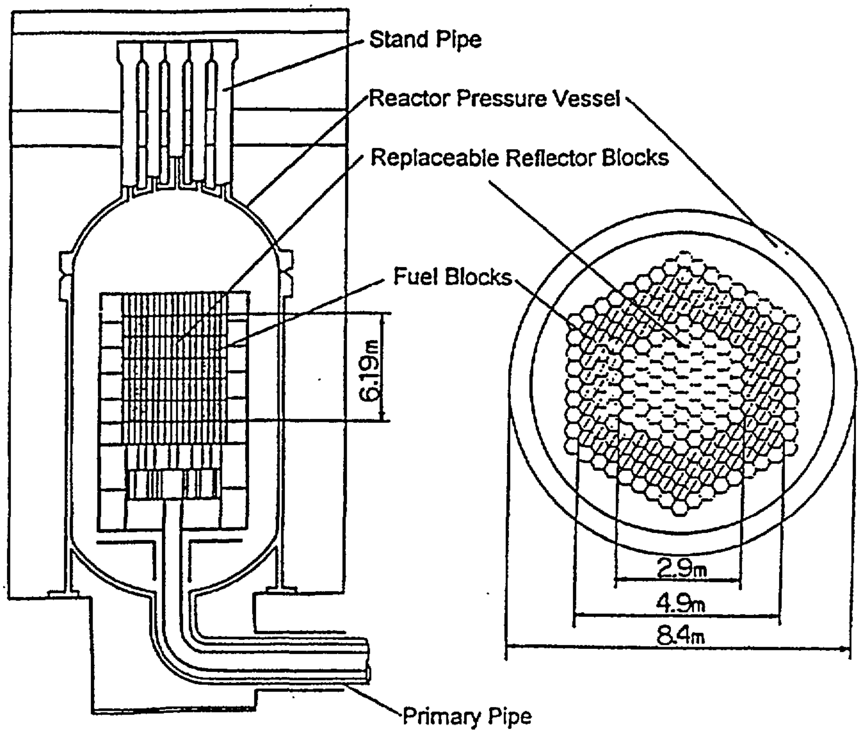

Fig.2.1 Conceptual view of SFHTR

Those columns consist of graphite block without any fuel and coolant hole and are surrounded by the fuel columns. The heat capacity of those columns is functioning to prevent the fuel temperature from rising up during depressurization accident. The average power density of reactor core is 6 W/cc. The maximum fuel temperature in the SFHTR is limited under 1500℃ during the normal operation. The fuel can keep its integrity up to 1800℃ during accident condition because zirconium carbide is applied as a fuel coating material. The SFHTR can get more than 100GWd/t of burnup thanks to high fuel performance and axial shuffling every year. A confinement to prevent fission product from release to the environment during accident is applied and filled with inert gas.

3. SIMULATION OF REACTOR TRANSIENT DURING DAWS

3.1 Accident scenario

Figure 3.1 shows the event sequence of the DAWS. In the case that such accident is initiated due to the rupture of the primary concentric hot gas duct in the rated power operation, the reactor power goes down by the negative temperature feedback effect just after the accident. Then, the reactor becomes deeply under critical due to xenon build up. The primary coolant is released outside through a break and forced cooling is lost after the accident. In this case, conduction and radiation are dominant to remove the residual heat from the reactor core. The residual heat is finally removed by radiation from the surface of RPV to the passive vessel cooling system. After the several tens hours, the reactor becomes recritical as a result of reactivity balance between temperature and xenon. The reactor power rises up again but goes down because of negative temperature feedback effect. The reactor power oscillates damply with repeating critical and subcritical. Finally the reactor power converges at the balancing point of xenon and temperature reactivity effects.

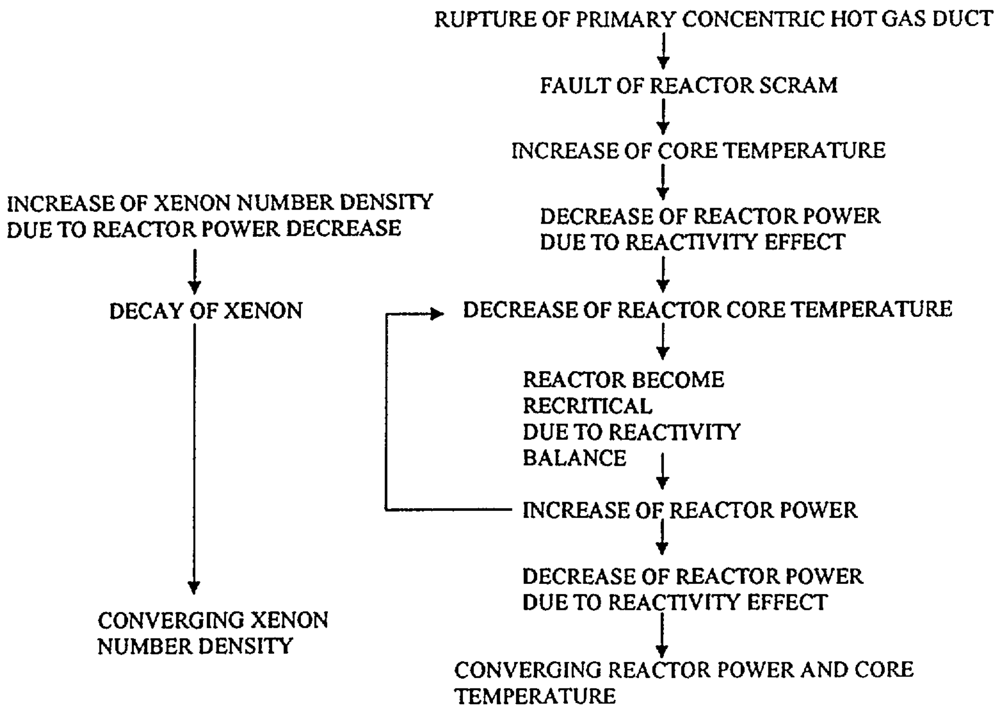

Fig.3.1 Event sequence of DAWS

3.2 Analytical model

(1) Basic equation

The reactor kinetics is analyzed by the point kinetics equation shown as follows:

$$\frac{dn}{dt}=\frac{\rho-\beta_{eff}}{l}n+\sum\lambda_i C_i$$

$$\frac{dC_i}{dt}=\frac{\beta_i}{l}-\lambda_i C_i$$

Where,	n	:	Reactor thermal power (Thermal power is assumed to be proportional to the number of neutrons)
	ρ	:	Reactivity
	β_i	:	Fraction of delayed neutron $\beta_{eff}=\Sigma\beta_i$
	l	:	Neutron lifetime
	λ_i	:	Decay constant of precursor
	C_i	:	Number density of precursor

The number density of xenon is calculated by the equation shown as follows :

$$\frac{dI}{dt}=\gamma_{Te}\Sigma_f\phi-\lambda_I I$$

$$\frac{dX}{dt}=\gamma_{Xe}\Sigma_f\phi+\lambda_I I-X\sigma_{aXe}\phi-\lambda_{Xe}X$$

Where,			
	I	:	Number density of iodine
	X	:	Number density of xenon
	t	:	Time
	γ_{Te}	:	Fission yield of tellurium
	γ_{Xe}	:	Fission yield of xenon
	Σ_f	:	Fission cross section
	ϕ	:	Neutron flux
	σ_{aXe}	:	Absorption cross section of xenon
	λ_I	:	Decay constant of iodine
	λ_{Xe}	:	Decay constant of xenon

Reactivity is calculated as follows :

$$\rho = \rho_{temp} + \rho_{Xe} + \rho_{dstb}$$

$$\rho_{Xe} = \frac{\sigma_{aXe} X}{\Sigma_{atotal}}$$

$$\rho_{temp} = \alpha \Delta T$$

Where,			
	ρ_{temp}	:	Reactivity due to change of temperature
	ρ_{Xe}	:	Reactivity due to change of xenon number density
	ρ_{dstb}	:	Reactivity due to change of control rod position
	α	:	Temperature reactivity coefficients of fuel and moderator
	ΔT	:	Temperature changes of fuel and moderator
	Σ_{atotal}	:	Total absorption cross section

The thermal dynamics is analyzed by two-dimensional thermal conduction equation shown as follows :

$$\rho C \frac{\partial T}{\partial t} = k_R \left(\frac{\partial^2 T}{\partial R^2} + \frac{1}{R}\frac{\partial T}{\partial R}\right) + k_Z \frac{\partial^2 T}{\partial Z^2} + q$$

Where,			
	ρ	:	Density of structure
	C	:	Heat capacity of structure
	T	:	Temperature of structure
	k_R	:	Radial heat conductivity
	k_Z	:	Axial heat conductivity
	q	:	Heat density

The fuel and moderator temperatures to evaluate the reactivity feedback effect are calculated by averaging each mesh temperature with a weight of mesh volume.

(2) Combined model

In order to simulate such reactor transient, the single channel model to calculate neutron kinetics by one point approximation was combined with the whole core model to calculate the heat flow dynamics inside the RPV. A schematic diagram of cmbined model is shown in Fig.3.2. The heat transfer in the reactor core calculated in the whole core model is given as a thermal boundary condition for the single channel model. On the other hand, the thermal power of the reactor calculated in the single channel model is given as a heat generation in the reactor core for the whole core model. The reactivity balance due to a change of xenon density and core (fuel and moderator) temperature is calculated in the single channel model. The heat removal form the surface of RPV is calculated in the whole core model. The single channel model and the whole core model are applied to the "BLOOST-J2" [3] code and the "TAC-NC" [4] code, respectively. Both codes were used to evaluate the consequence of the accident in the HTTR and have been validated sufficiently.

The single channel model is an asymmetric simulation of fuel compact, graphite sleeve, coolant channel and graphite block. The fuel compact is modified as the homogeneous solid to smear the coated fuel particle and graphite matrix. The fuel block is modified as the equivalent annular slab having the same thermal capacity of it. The equivalent heat transfer rate from the whole core model is applied as outer boundary condition of graphite block.

The whole core model is an asymmetric simulation of the reactor core, internal structure, RPV and VCS surrounding it. The reactor core is modified as the homogeneous solid to smear the fuel compact, graphite sleeve and graphite block. The heat sink for the simulation of temperature transient during the DAWS is modified as the constant temperature boundary outside the RPV.

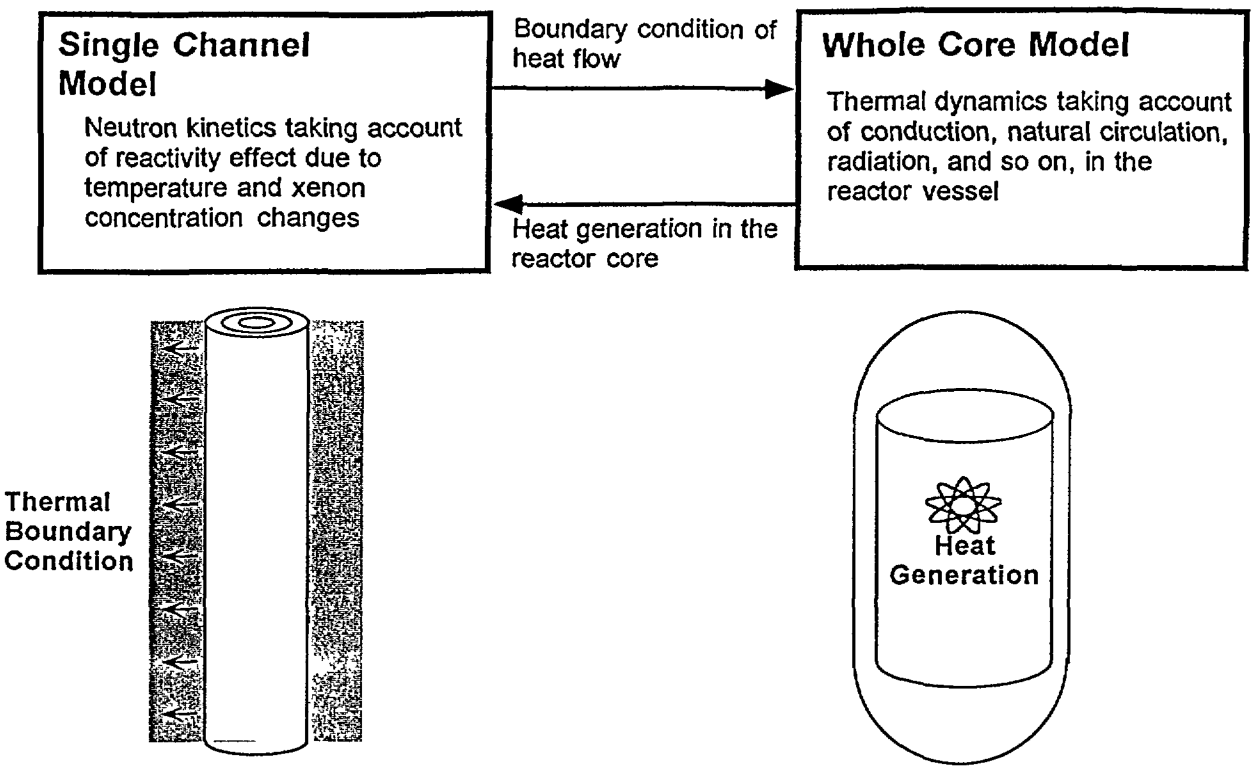

Fig.3.2 Parameter exchanged between single channel model and whole core model

3.3 Analytical result of DAWS

Figures 3.3 and 3.4 show the transient of reactor power and fuel temperature. The reactor power decreases rapidly due to negative temperature reactivity effect just after the accident. The fuel temperature decreases to same temperature of graphite due to heat transfer from fuel compact to graphite block. Xenon density increases due to the reactor power decrease at first and turns to decrease due to its decay.

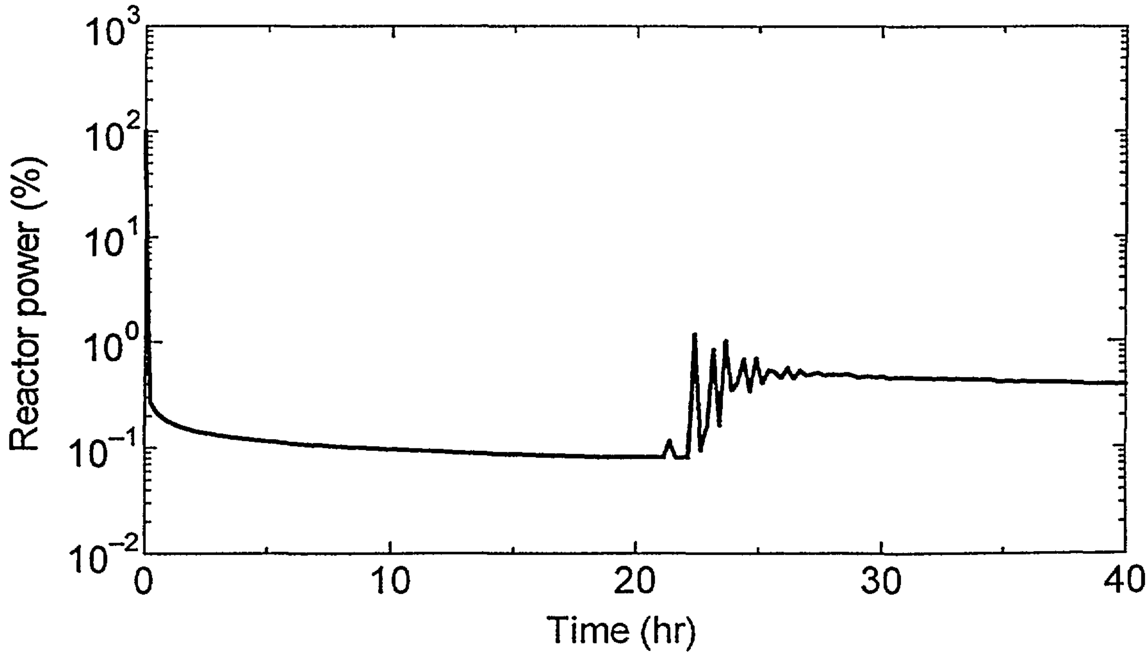

Fig.3.3 Reactor power transient during DAWS in SFHTR

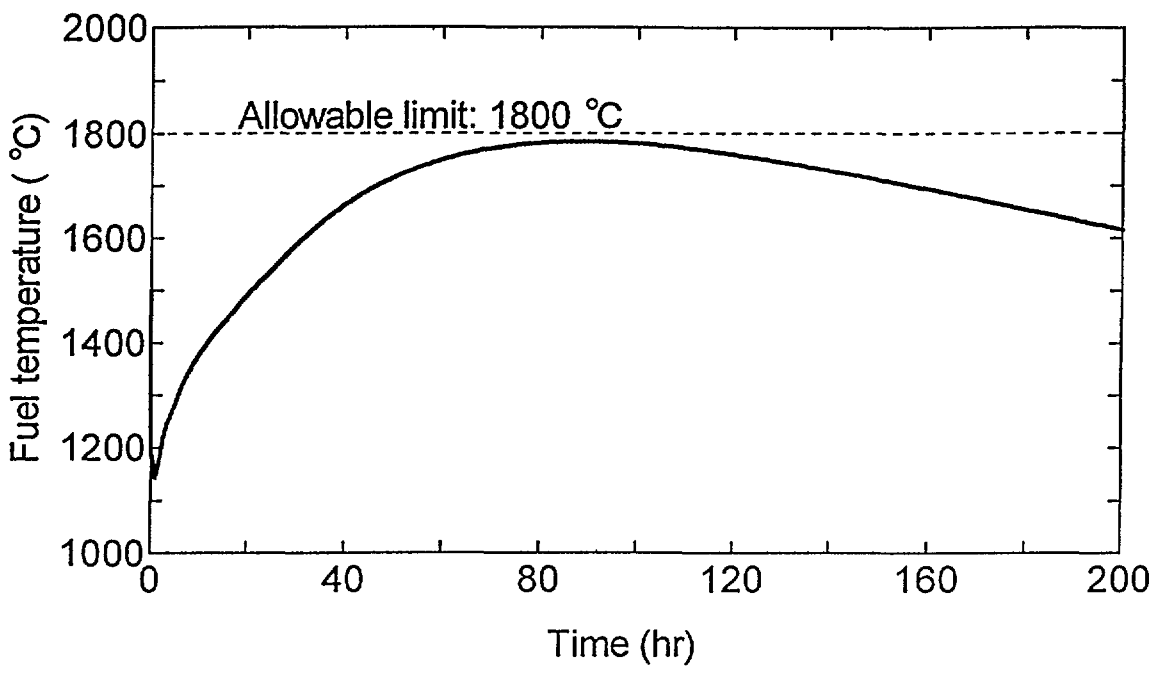

Fig.3.4 Fuel temperature transient during DAWS in SFHTR

The reactor becomes recritical at about 22 hours after the occurrence of the accident. The reactor power increases up to about 3.8MW at this time and converges finally to about 1.0MW after damped oscillation. In this accident, the fuel integrity is ensured because the fuel temperature rise is limited up to about 1785°C.

4. CONCLUSION

The analytical model to simulate the reactor transient during the DAWS was developed. The analysis of the DAWS having the possibility to become the severe accident in the SFHTR was performed. From the analytical result, it was found that no reactor meltdown occurs even in the worst scenario because the maximum fuel temperature increase up to about 1785°C and does not exceed the limit temperature of 1800°C.

The newly developed code is applicable to the evaluation of beyond design basis accidents including the DAWS. Also this code is applicable to the simulation of the safety demonstration test planned in the HTTR. JAERI has a plan to perform the safety demonstration test after getting the sufficient experiment data and operation experience in the HTTR. The partial trip test of helium gas circulator and the control rod withdrawal test have been licensed in the HTTR. The purpose of the safety demonstration test is to confirm the safety margin to the reactor transient is large even if the severe accident would be assumed in the HTTR, that is in the HTGR. The power-up test in the HTTR will start in September 1999 and the rated power operation of 30MW will come in May 2000. The safety demonstration test will be performed in 2001. The developed code will be useful to simulate and evaluate the transient in the HTTR safety demonstration test.

ACKNOWLEDGEMENT

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NUCLEAR DESIGN CODE DEVELOPMENT FOR FUEL MANAGEMENT AND SAFETY ANALYSIS OF HTGR IN INDONESIA

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Abstract

The development works and some results on the nuclear design codes for fuel management and safety analysis of HTGRs conducted in Batan is reported. Batan-MPASS code, an in-core fuel management code, have been developed and verified to simulate the continuously flow of the pebble fuel elements in a pebble-bed type HTGR core for both once-through-then-out (OTTO) as well as multipass fueling schemes. One important feature of the code is that it can search directly the equilibrium core condition without simulating the transient cores. A similar code, Batan-PEU code, has also been built to simulate the peu a peu fueling scheme. These codes are equipped with in-core thermal-hydraulics modules for estimating the pebble fuel temperature. For prismatic/block type HTGR, Batan-FUEL code has been compiled where originally the code was developed for in-core fuel management of a research reactor with ordinary batch refueling scheme. The thermal-hydraulic modules for Batan-FUEL are planned to be developed in the future. The diffusion calculation module within Batan-FUEL code has already successfully used for the benchmark problems of the Japanese HTTR's 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 V.S.O.P and DELIGHT-7 cell calculation codes, developed by KFA and JAERI, respectively. Some application results of the codes for the modular HTR-M 200 MWth design are reported. The accomplishment of these codes are expected to contribute for assessing the techno-economic of small and medium-scale modular HTR design presently being conducted by Batan, especially for providing the fuel cost estimation.

For safety and accident analysis, two codes have been developed to simulate the depressurization and reactivity accidents, respectively, 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 application of the two codes for assessment of the modular HTR-M 200 MWth safety under such accidents is reported.

Introduction

Through bilateral and international co-operation, as well as self development by the National Atomic Energy Agency, the availability of nuclear data, nuclear data processing codes, and reactor physics calculation codes in the Agency is growing higher. This is partly attributed to the increasing needs of the Agency to operate and utilize more efficiently and safely the three research reactors, i.e., two, relatively low power, Triga-type research reactors located in Bandung and Yogyakarta, and one 30 MWth multipurpose reactor (RSG GAS, formerly called MPR-30) located in Serpong, Tangerang.

As a long term objective, the activity is expected to enhance also the power reactor technology transfer and development, in which, the high temperature gas-cooled reactor (HTGR) is considered as one potential candidate for the near future or next generation power reactors. This paper reports briefly the nuclear code development for fuel management and safety analysis of HTGRs conducted in Indonesia, specifically in the Agency, and some transient and accident analysis results for a typical modular pebble-bed HTGR, i.e. the HTR-M 200 MWth [1].

Nuclear Data and Libraries

Concerning the nuclear data and their relevant processing codes, within the Agency's organization, up to the present date, there is no special center for nuclear data, however, an ad-hoc team consisting of several persons for maintaining the existing nuclear data and the relevant processing codes has long been established. The team's members come from the Informatics Development Center, Center for Multipurpose Reactor, and the Center for Assessment of Nuclear Technology.

As shown in Table 1, the main sources of the nuclear data libraries maintained by the team are the Evaluated Nuclear Data File (ENDF/B) and the Japan Evaluated Nuclear Data Library (JENDL). A general nuclear data processing code for these nuclear data libraries available in the Agency is the NJOY91.38 code [2]. The processing code is used for generating additional cross sections fed into the 1-D cell calculation code, the WIMS/D4 [3]. Despite the fact that this cell calculation code is potential for HTGR applications, up to present, we have not yet investigated the applicability of the code for HTGR neutronics analysis.

The (processed) MGCL-137 and MGCL-26 nuclear libraries with their processing code systems, MAIL, REMAIL etc. [4] are being used for cell calculations and control rod studies. However, the libraries are relatively old and their scattering cross sections are limited to P_0 and P_1 scattering cross sections at room temperature. These nuclear libraries together with 1-D transport code, ANISN-Jr [5], were used for cell calculations to produce group constants. Concerning the HTGR applications, this code system have been used for generating the cross sections of the block-type fuel elements and control rod studies. For example, we have utilized them and a generic 3-D diffusion code, Batan-3DIFF [6], for the JAERI HTTR first criticality reactor physics benchmark calculations [7].

The author has also compiled problem dependent cross section sets for high temperature reactor, especially for a typical modular pebble-bed HTGR, the HTR-M 200 MWth, using JAERI's DELIGHT-7 [8] and V.S.O.P [9] cell calculation codes. These libraries, together with self-developed Batan-MPASS code [10], a code for pebble bed HTR fuel management, have been used for preliminary design evaluation of the HTR-M 200 MWth which is being conducted by the Agency's HTR Team and HTR Design Project Team.

Neutronics and In-Core Fuel Management Codes

The reactor physics codes available in Batan are summarized in Table 2. A workstation version of SRAC-95 [11] has been successfully installed and used in Batan for research reactor analyses. The code is found very useful for detailed analyses of research reactors which require full 3-D many group neutron diffusion calculations, or 2-D few group neutron transport calculations. Concerning the HTGR applications, we

Table 1. Original and processed nuclear data (related to HTGR) available in Batan (1998)

NUCLEAR DATA ORIGINAL	NUCLEAR DATA PROCESSED	PROCESSING CODE	SPECIFIC LIBRARY OUTPUT
	MGCL-137 (JAERI)	MAIL, REMAIL etc. (JAERI)	• P_0, P_1 cross section library for transport code (ANISN-Jr)
ENDF/B-IV ENDF/B-VI JENDL-3.2		NJOY93.38 (IAEA)	• Cross sections for WIMS/D4 library or for direct core calculation (Batan-FUEL code)
ENDF/B-VI			• MCNPXS (DLC-189) for MCNP-IV code
ENDF/B-IV ENDF/B-III		DELIGHT-7 *) (JAERI)	• Operational parameters dependent cross section libraries for Batan-MPASS, -PEU, -FUEL codes (limited to HTR-200 Modul)
ENDF/B-IV		V.S.O.P *) (KFA-JULICH)	

*) Not available in Batan

Table 2. HTGR analysis codes available in Batan (1998)

ORIGIN	CODE NAME	CELL CALC.	CORE CALC.	FUEL MANAGEMENT	APPLICATIONS
JAERI (Japan)	SRAC-95	Yes	Yes	Yes	• Benchmark calculations
ANL (USA)	CITATION	Yes	Yes		• Code verification
RSICC (USA)	MCNP-4B		Yes		• Criticality calculations
Batan (Indonesia)	Batan-FUEL		Yes	Yes	• Batch refueling, block-type HTGR neutronic analysis and in-core fuel burnup simulation and fuel management
	Batan-MPASS		yes	Yes	• Continuous refueling, pebble bed HTGR neutronic & thermal hydraulic analysis and fuel management (OTTO and Multipass refueling schema)
	Batan-PEU		yes	Yes	• Continuous refueling, pebble bed HTGR neutronic & thermal hydraulic analysis and fuel management (Peu a Peu refueling scheme)

have also used the SRAC-95 code system for the above mentioned JAERI HTTR first criticality reactor physics benchmark calculations [7]. The recently available MCNP-4B Monte Carlo code [12] will be used for more advance applications where accuracy is of the most importance.

Batan-FUEL in-core fuel management code [13] originally developed by the Agency for research reactor in-core fuel management and to support the RSG GAS core conversion program from oxide to silicide fuel [14]. The code was designed to simulate and record the fuel burnup in a reactor core with batch refueling scheme within the framework of multigroup neutron diffusion theory in 2-D XY or RZ reactor geometry. Although not for routine fuel management, Batan-FUEL code has a 3-D diffusion module (identical to Batan-3DIFF module) to treat the axial dependent problems such as when the axial position of the control rods must be evaluated accurately. One important feature of the code is the equilibrium core searching option; using this option the reactor designer can directly obtain the core equilibrium condition without simulating the transition cores [15]. The code has been further developed for the in-core fuel management of the batch refueling block type HTGRs by adding the trigonal mesh option to treat the hexagonal geometry of the HTGR fuel element in a more precise way. Considering the heterogeneity in the axial direction of block-type HTGRs' core the code will be modified in the future to simulate fuel burnup in 3-D reactor geometry.

Batan-MPASS code [10] was developed as a general fuel management code for calculating the core equilibrium condition of pebble-bed high temperature reactors with multipass and Once-Through-Then-Out (OTTO) refueling schemes. In the code, the flow of the pebble fuel elements across the core is simulated continuously. The code adopts an iterative method to obtain directly both the equilibrium core and critical conditions of the reactors without simulating the transition cores. Two-dimensional, RZ reactor geometry as well as 1-D reactor geometry options are available. The code is also equipped with 1-D and 2-D thermal-hydraulic routines to assess the coolant flow pattern and fuel temperature distribution in the core. With these options the designers can obtain comprehensive results from neutronics, fuel management, and thermal aspects of the reactor.

For peu a peu refueling scheme, a similar code, Batan-PEU code [16] was developed. This code is almost identical with Batan-MPASS code except for the fuel burnup calculation. The fuelling operation which increases the core height is simulated in the code by moving the boundary between the core and upper core void regions.

Batan-MPASS and -PEU codes have been used in the assessment and prefeasibility study on the utilization of high temperature gas reactors for future national industrialization programs, particularly in the fields of (1) process heat applications, (2) process steam and electricity co-generation, and (3) electric generation for isolated/remote islands [17].

Safety and Accident Analysis Codes

For safety and accident analysis, two codes have been developed to simulate the depressurization and reactivity accidents, respectively, in a typical pebble-bed type HTGR. These two severest accidents have been of the most interest for the HTGR communities.

For the depressurization accident simulation, firstly, the equilibrium condition of the core (neutronics and thermal-hydraulics) is calculated using either Batan-MPASS or Batan-PEU code to obtain the initial condition. Secondly, the decay heat generated afterward in the core is calculated using the JNDC Fission Product and Yield Data [18] compiled by JAERI, taking into account also the contribution of decay heat of heavy metals. During the depressurization accident, the heat transfer processes in the core are (1) natural convective heat transfer between fuel balls and He coolant, (2) radiative heat transfer between the adjacent fuel balls, (3) heat conduction inside the fuel balls and (4) heat convection in the He coolant. Thirdly, 2-D RZ geometry heat conduction from the core to the reflector regions and finally to the outest boundaries is then calculated. For conservative safety analyses, it is common to apply the adiabatic boundary conditions on the outest boundaries. In the next section, a simulation result of the depressurization accident for the HTR-M 200 MWth with OTTO fueling scheme is given.

For the reactivity accidents where the a rapid and local perturbation of the neutron flux and thermal properties occur, the neutron dynamics is simulated in the code with the improved quasi-static model under 2-D, RZ geometry, four group neutron diffusion theory. During the reactor power transient, the time dependent pebble fuel temperature is solved using the 1-D spherical thermal conduction equation using a semi-implicit time differencing scheme. On the other hand, the helium coolant flow through the core is treated in a quasistatic manner. The mass, momentum, energy balance for the fluid and solid phases are solved for 2-D RZ geometry. A complete discussion on the iterative schemes to solve the in-core thermal hydraulic problem was presented in Ref. (19).

In the above mentioned simulations, the transient of the whole primary cooling system is simulated in a approximate way by assuming that the secondary cooling system is always working in its nominal rate.

In addition to the reactivity accident, some other transient events can be simulated using this code, for e.g. the primary coolant flow change and the inlet coolant temperature change. In the next section, some simulation results of the reactivity accident and other transient events for the HTR-M 200 MWth with OTTO fueling scheme are given.

Simulation Results

The equilibrium condition of the OTTO refueled HTR-M 200 MWth is first calculated to obtain the initial conditions. The design parameters are shown in Table 3 and the reactor geometry used in the calculations is shown in Fig. 2. The calculated power density distribution is shown in Fig. 3. As predicted, a peak located in the upper part of the core is negligible. For this equilibrium core condition, the discharged fuel burn-up reaches the same level as the original multipass HTR-M. The neutronics calculations are accompanied by thermal-hydraulic ones to provide the initial conditions for the following accident simulations.

The simulation results for the severest accident, i.e., the depressurization accident (in which the helium forced circulation is totally lost) are shown in Fig. 4 and 5. They show the axial distribution of the fuel center temperature and decay heat density distributions along the center line of the core, respectively. The fuel center temperature in the upper part of the core reaches 1600 °C at ~ 10 hours and its maximum temperature of ~ 1800 °C at 23.5 hours after the depressurization. The decay heat

Table 3. Main design parameters for HTR-M 200 MWth [1].

Power (MWth)	200
Power density (W/cc)	3
Core diameter (m)	3
Core height (m)	9.6
No. of control rods	6
No. of absorber ball systems	18
Diameter of pressure vessel (m)	5.8
Height of pressure vessel (m)	23.8
He temperature (°C)	250 / 750
He gas flow (kg/s)	76
System pressure (atm)	60
Loading scheme	OTTO[¶]
Fuel cycle	U/Pu
Enrichment (w/o)	8
Heavy metal loading/fuel element (g)	7
Burn-up (GWd/t)	80
Fuel residence time (day)	1020

[¶] In Ref. [1] the fuel loading scheme is multipass.

distribution with a peak in the upper part of the core is the main cause to the result. A grace time of about 10 hours is available for corrective actions. If no corrective action is allowed then some part of the core would exceed the fuel failure temperature of 1600 ~ 1700 °C and fractional release of fission products might occur. If the integrity of the reactor pressure vessel or concrete reactor cell is assumed then a significant release of fission products to the surroundings could be avoided.

The simulation results for partial loss of flow and blower over-speed are shown in Figs. 6 and 7. An excess coolant flow (F=1.1) enhances the core heat removal and introduces positive reactivity. The power level increases and stabilizes at a higher value, so that the positive reactivity is compensated. On the contrary, a partial loss of flow (F=0.75, 0.5 or 0.25) reduces the core heat removal so that the core temperature increases. This gives negative reactivity which brings the power level down.

Mismatches of primary and secondary coolant circuits are simulated by an increase (δT_{in} = +50 or + 100 °C) or decrease (δT_{in} = -50 or - 100 °C) of the helium inlet temperature. The results are shown in Figs. 8 and 9. The power and temperature transients are similar to the previous cases for flow disturbances.

Compares to the depressurization accident, the fuel temperatures for the flow and inlet temperature disturbances are much lower than the fuel failure temperature 1600 ~ 1700 °C.

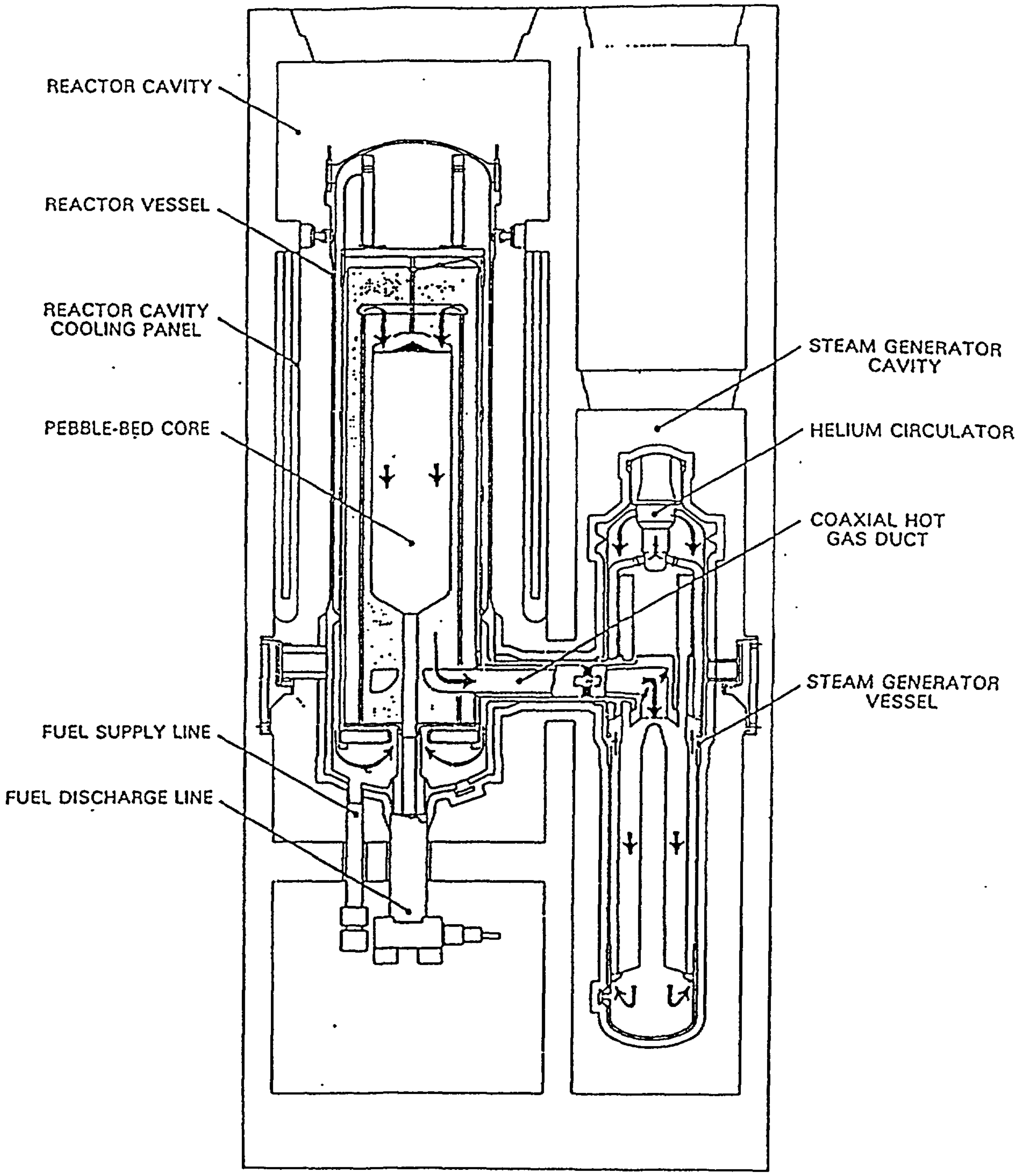

Figure 1. HTR-Modul configuration.

The simulation results shown in Figs. 10 and 11 are for the reactivity accidents initiated from nominal power and start-up conditions, respectively. The external reactivity introduced by withdrawal of all absorber rods in the reflector is ~ 1.5 $ which is equal to the excess reactivity during normal operation of HTR-M. The power transient from nominal power is characterized by a rapid heat accumulation rate in the core so that the negative reactivity feedback promptly appears. On the contrary, the power transient from start-up condition, in which the initial power is just 1 % of the nominal value, shows that the power excursion needs longer time to attain a considerably high level before negative reactivity feedback can be expected to stop the excursion.

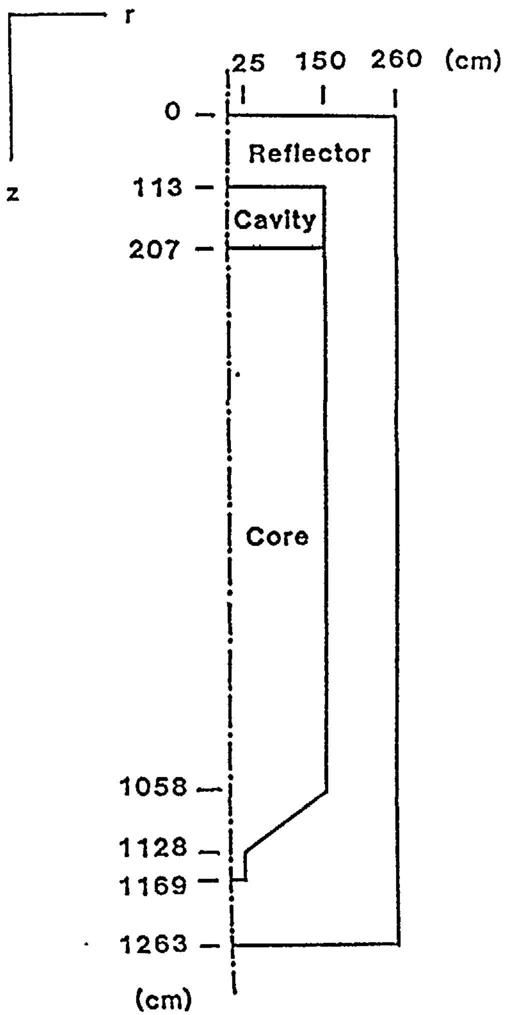

Figure 2. Core and reflector geometry for neutronic and thermal hydraulic calculation models.

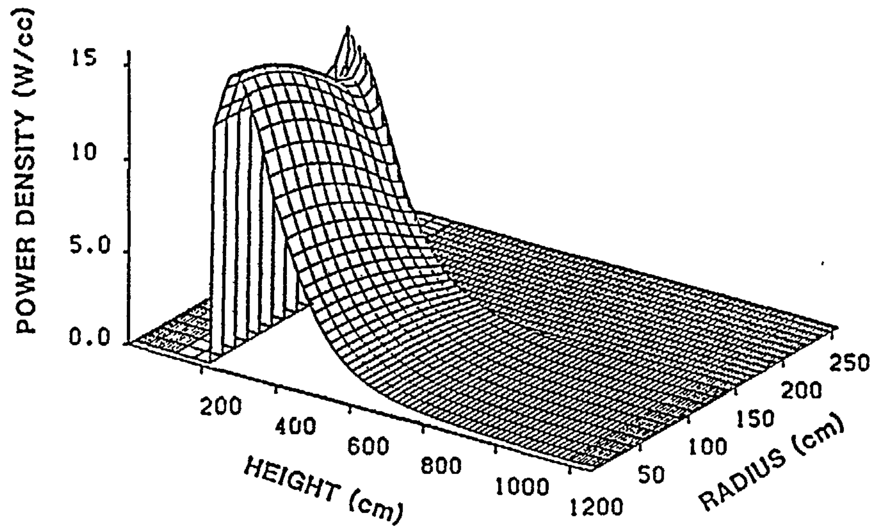

Figure 3. Power density distribution for the core equilibrium condition.

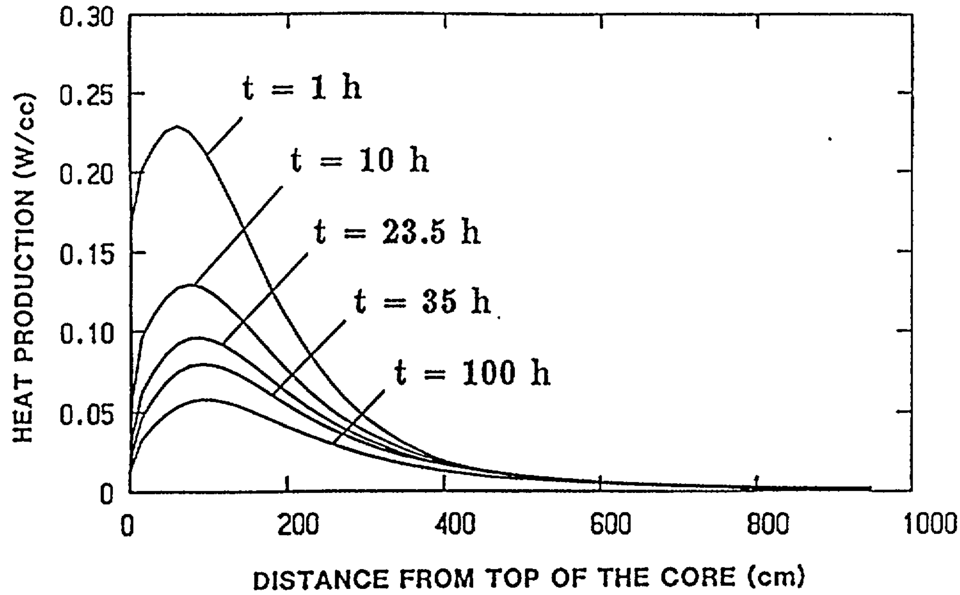

Figure 4. Time-dependent, axial distribution of decay heat production.

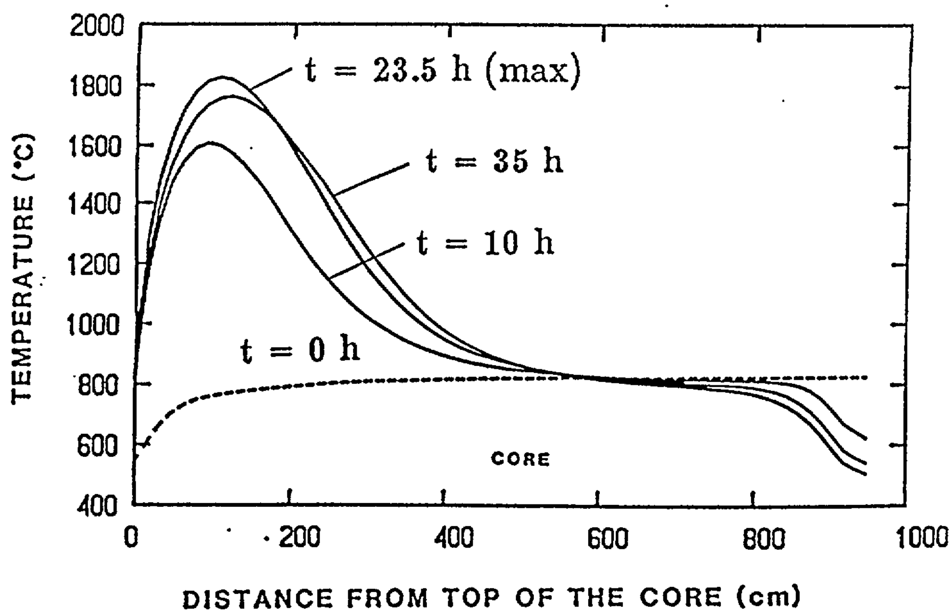

Figure 5. Time-dependent, axial distribution of pebble center temperature for depressurization accident.

In Fig. 12 the axial fuel center temperature and power distributions are shown. As discussed previously, the OTTO fueling scheme HTR-M provides an almost flat initial fuel temperature distribution. During the beginning of power excursion the power density peak in the upper part of the core results in higher fuel center temperatures at those locations. At the end of the power excursions, with the primary coolant system works normally, helium coolant takes heat from the fuel elements in the lower part of the core The final fuel center temperature profile is similar to the initial profile but at a higher level corresponds to the new reactor power level. The new power level and fuel temperature compensate the positive reactivity. In the case of reactivity accident from the nominal power, the fuel center temperature reaches ~ 1200 °C, hence, it is still below the fuel failure temperature.

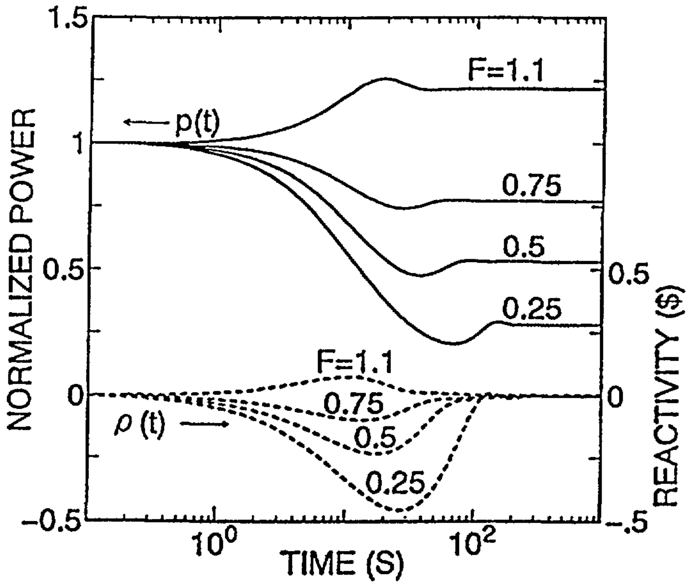

Figure 6. Power and reactivity transients for primary coolant system flow change events.

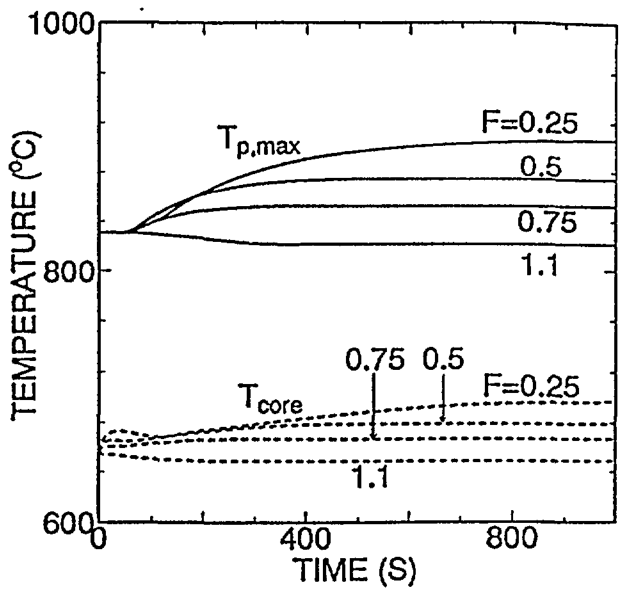

Figure 7. Core averaged and maximum pebble center temperature transients for primary coolant system flow change events.

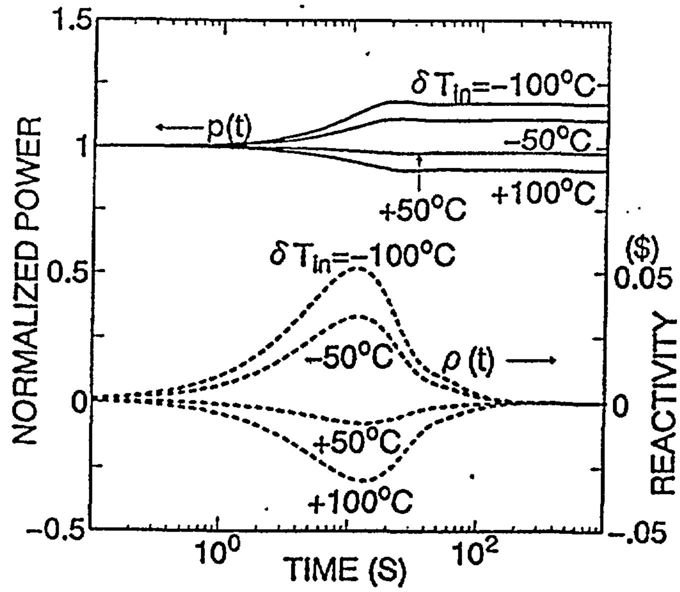

Figure 8. Power and reactivity transients for inlet coolant temperature change events.

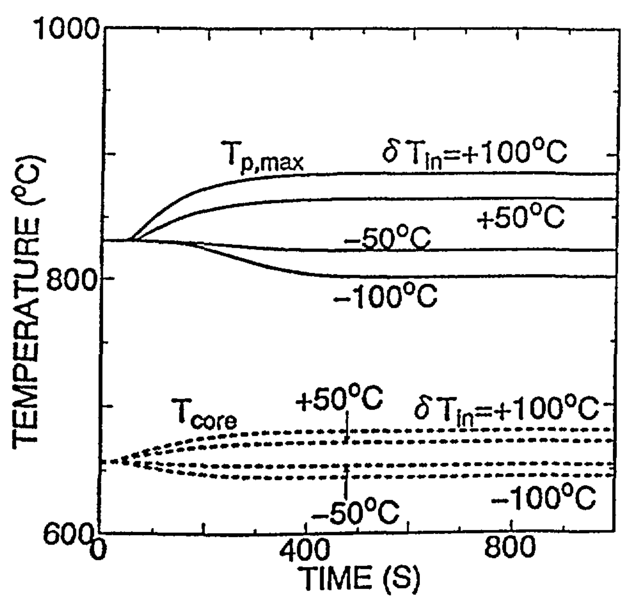

Figure 9. Core averaged and maximum pebble center temperature transients for inlet coolant temperature change events.

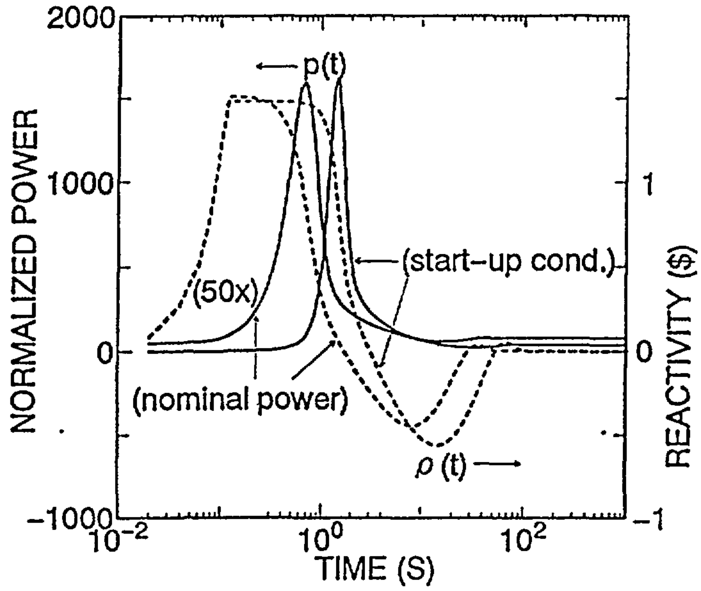

Figure 10. Power and reactivity transients for rods withdrawal accidents from nominal power and start-up condition.

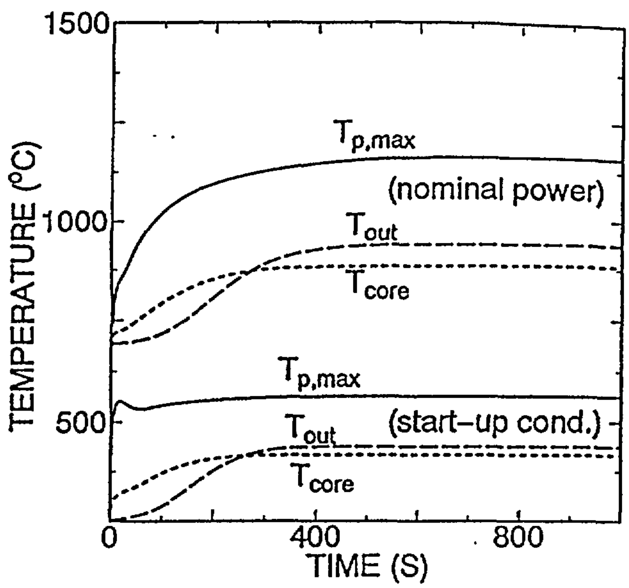

Figure 11. Core averaged, helium outlet and pebble center temperature transients for rods withdrawal accidents from nominal power and start-up condition.

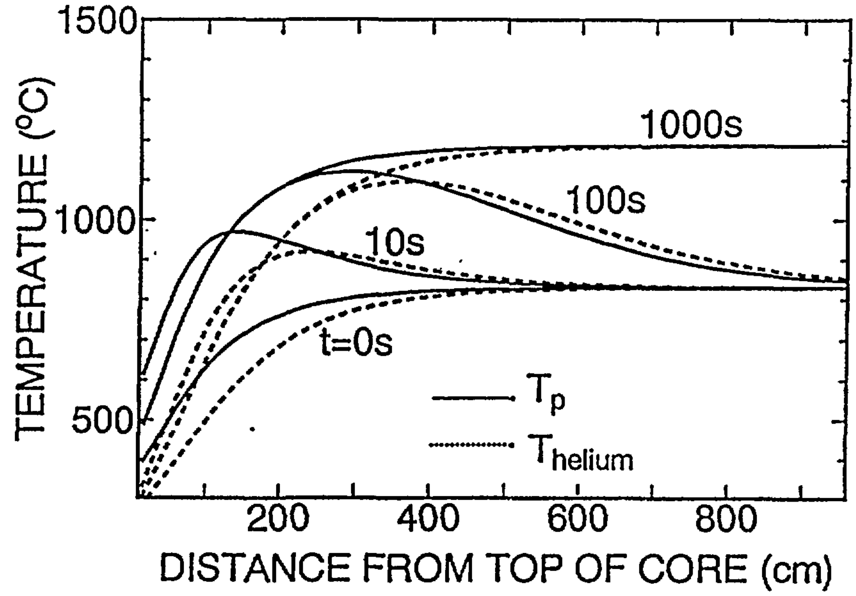

Figure 12. Axial pebble center and helium temperature distributions for rods withdrawal accident from nominal power condition.

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PBMR-SA LICENSING PROJECT ORGANIZATION

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Abstract

The South African nuclear regulatory authority, the Council for Nuclear Safety (CNS) is beginning the safety review of the Pebble Bed Modular Reactor (PBMR) design under development by the South African National Electrical Utility, Eskom. This paper describes the CNS licensing process, including the establishment of basic licensing criteria, general design criteria, and specific design rules, as well the safety assessment to be conducted in accordance with the established structure. It also summarises the CNS PBMR review project activities, including the overall organisational arrangements, licensing basis, safety and risk assessment, general operating rules and plant design engineering, and pre-operational testing.

1. INTRODUCTION

The South African nuclear regulatory authority, the Council for Nuclear Safety (CNS) is currently at the first licensing stage of the safety review of the South African high temperature gas-cooled Pebble Bed Modular Reactor (PBMR).

Eskom, the National Electricity Utility, has officially submitted an application for a staged licensing process requesting that the first stage considers the concept design without any specific site. The CNS will evaluate the acceptability of the safety bases for the proposed PBMR.

The current PWR Koeberg 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 CNS 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 CNS fundamental safety standards.

Based on the outcome of the safety assessment process to demonstrate compliance with the above criteria, conditions of licence 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 is proposed that a similar approach should be adopted in respect of the PBMR.

However, unlike the light water reactor situation, the same level of international consensus has not yet been developed in respect of 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 high temperature gas reactors (HTGR) 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.[1].

The CNS made an estimate of the time and costs which will be incurred performing all the activities required to reach a decision on the feasibility of licensing a Pebble Bed Modular Reactor.

The activities associated with the fuel procurement or manufacture are not discussed in this paper.

2. LICENSING PROCESS

The process proposed for licensing the PBMR-SA is set down below. The chronology does not necessarily represent the sequence within which the various activities will be carried out. Many of the activities will be conducted in parallel and will be scheduled following further discussions with Eskom.

2.1 Establishment of Basic Licensing Criteria

The first step of the process is the establishment, agreement and documentation of the Basic Licensing Criteria to be applied to the PBMR. The following is proposed:

	SAFETY REQUIREMENTS	EVENT FREQUENCY	SAFETY CRITERIA
a	The design shall be such to ensure that under anticipated conditions of normal operation there shall be no radiation hazard to the workforce and members of the public. This must be demonstrated by conservative deterministic analysis.	Normal operational conditions shall be those which may occur with a frequency up to but not exceeding 10^{-2} per annum.	Individual radiation dose limits per annum of 20 mSv to workers and 250 μSv to members of the public shall not be exceeded. +ALARA+Defence in depth criteria
b	Design to be such to prevent and mitigate potential equipment failure or withstand externally or internally originating events which could give rise to plant damage leading to radiation hazards to workers or the public. This must be demonstrated by conservative deterministic analysis.	Events with a frequency in the range 10^{-2} to 10^{-6} per annum shall be considered.	Radiation doses of 500 mSv to workers and 50 mSv to members of the public shall not be exceeded. +ALARA + Defence in depth criteria
c	The design shall be demonstrated to respect the CNS risk criteria. This must be demonstrated by probabilistic risk assessment using best estimate + uncertainty analysis	Consideration shall be given to all possible event sequences.	CNS risk criteria apply. $5X10^{-6}$ Individual risk 10^{-8} Population risk Bias against larger accidents. +ALARA

The above criteria are based on current CNS standards which are from time to time validated/benchmarked against internationally endorsed standards. In comparison with safety criteria for PWR plant defined by ANSI/ANS-51.1 standard [2], the criteria "a" covers Plant Conditions (PC) 1-3 and criteria "b" covers PC 4 and PC 5.

In defining the above criteria it will be necessary to establish interpretation statements pertaining to specific terms used in the above table e.g ALARA , Defence in depth , conservative analysis, best estimate , cut off criteria etc.. This will be undertaken by the following process.

Activity #	Description	Responsibility
1	Agree, establish and document the basis for the fundamental approach outlined above	CNS/Eskom

In addition to the above requirements consideration will be given to identify contingency measures for an Emergency Plan.

2.2 Establishment of General Design Criteria and Specific Design Rules

The next step of the process will be to establish the PBMR General Design Criteria and Design Rules. The following are the main steps of the process :

Activity #	Description	Responsibility
2	Set up and agree an event/accident classification scheme/process	CNS/Eskom
3	Concept design proposal	Eskom
4	Identify and agree list of event/accidents applicable to PBMR and classify according to 2 above	Eskom/CNS
5	Establish General Design Criteria for PBMR according to an agreed process	Eskom/CNS
6	Review and agree General Design Criteria	CNS/Eskom
7	Establish guidelines and interpretation of General Design Criteria leading to establishment of Specific Design Rules	Eskom/CNS

2.3 Detailed Design of Plant

The next main step of the process will be for Eskom to propose a plant design in accordance with the above design criteria and rules

Activity #	Description	Responsibility
8	Proposal of a plant design in accordance with above General Design Criteria and Specific Design Rules	Eskom

2.4 Safety Assessment

In accordance with the general CNS Licensing approach indicated in 1.0 above, it is expected that the proposed installation meets specific requirements. These will be formulated and demonstrated in the Safety Assessment.

Activity #	Description	Responsibility
9	Establish requirements to demonstrate compliance with agreed General Design Criteria and Specific Design Rules	CNS/Eskom
10	Establish requirements to comply with CNS risk criteria	CNS/Eskom

11	Establish code requirements and application methodology – e.g. benchmarking, validation etc.	CNS/**Eskom**
12	Carry out and document Safety Assessment to meet the above requirements in 9, 10,11	Eskom
13	Review and evaluate Safety Assessment	CNS

2.5 Establishment and Performance of Prototype Test Programme

Due to the limited operational experience feedback and data availability from this type of reactors it will be necessary to implement a comprehensive step by step test programme to verify/validate some of the critical safety assessment parameters.

Activity #	Description	Responsibility
14	Establish and agree the scope of the test programme	Eskom/CNS
15	Identify and agree on additional hardware to support the test programme	Eskom/CNS
16	Agree objectives and acceptance criteria for the various activities of the test programme	Eskom/CNS

2.6 Establishment of Testing and Inspection Programmes

This step is standard practice in the Nuclear Industry

Activity #	Description	Responsibility
17	Establish and agree Testing and Inspection programme for the manufacturing phase	Eskom/CNS
18	Establish and agree Testing and Inspection programme for the construction phase	Eskom/CNS

2.7 Establishment of the requirements for General Operating Rules - GORs

The outcome of the Safety Assessment will result in the identification and establishment of GORs requirements, which will have to be complied with upon the plant becoming operational:

Activity #	Description	Responsibility
19	Identify, establish and agree GORs requirements. This will include the standard GORs associated with a NPP e.g. Operating technical specifications In service inspection programme Maintenance programme Radiation protection programme Waste management programme Emergency planning Etc..	Eskom/CNS

2.8 Establishment of Quality Assurance Requirements

These QA requirements will apply at every stage of the Project.

Activity #	Description	Responsibility
20	Propose Licensing guidelines for QA requirements	CNS
21	Establish and agree QA requirements to be applied at every stage of the Project	Eskom/**CNS**

2.9 Documentation

As for the above QA step, documentation requirements will be identified at every step of the process

Activity #	Description	Responsibility
22	Establish and agree the format and content of the Safety Analysis Report - SAR	Eskom/CNS
23	Establish and agree documentation requirements	Eskom/CNS

3. PBMR PROJECT ORGANIZATION

To perform the required licensing activities, a corresponding Department has been created as an integral part of the Power Reactor Group of the CNS. The overall CNS Organigram is presented in Figure 1. The main objective of the Department is to undertake the overall control and management of Licensing activities for the Pebble Bed Modular Reactor.

3.1 Organisational arrangements

The organizational structure of the PBMR department and its integration with other CNS divisions is presented in Figure 2. A number of technical subprojects (Projects A-D) have been identified to reflect the project scope indicated in activities above. Each of these subprojects is lead by a Technical Project Leader (TPL), who will use, as required, different technical specialists from other CNS departments.

In order to establish a credible Licensing process, the CNS identified that, due to the lack of expertise in HTGR technology within the current staffing, specialised scientific and engineering services will have to be obtained from competent outside organisations. Thus contacts were established with various organisations who could possibly provide the CNS with the necessary expertise and experience with gas graphite reactor technology. The organisations were selected on the bases of advice solicited from contacts overseas and the involvment we have had up to date with the IAEA group on gas cooled reactors.

Following discussions with the various organisations the CNS is finalising Agreements for specialised technical support services within the particular disciplines key areas relating to HTGR technology and safety assessments with the companies from the following countries: France, UK, Germany and USA. Each company has been "allocated" a specific key discipline:

* Licensing Basis + Nuclear Engineering,
* Materials including Graphite
* Radiation Safety Engineering and Fuel
* Peer review of important safety issues on a case by case basis.

The main activities of the Licensing process as indicated above in Section 2, are reflected in the following sub-projects breakdown.

CNS Organigram

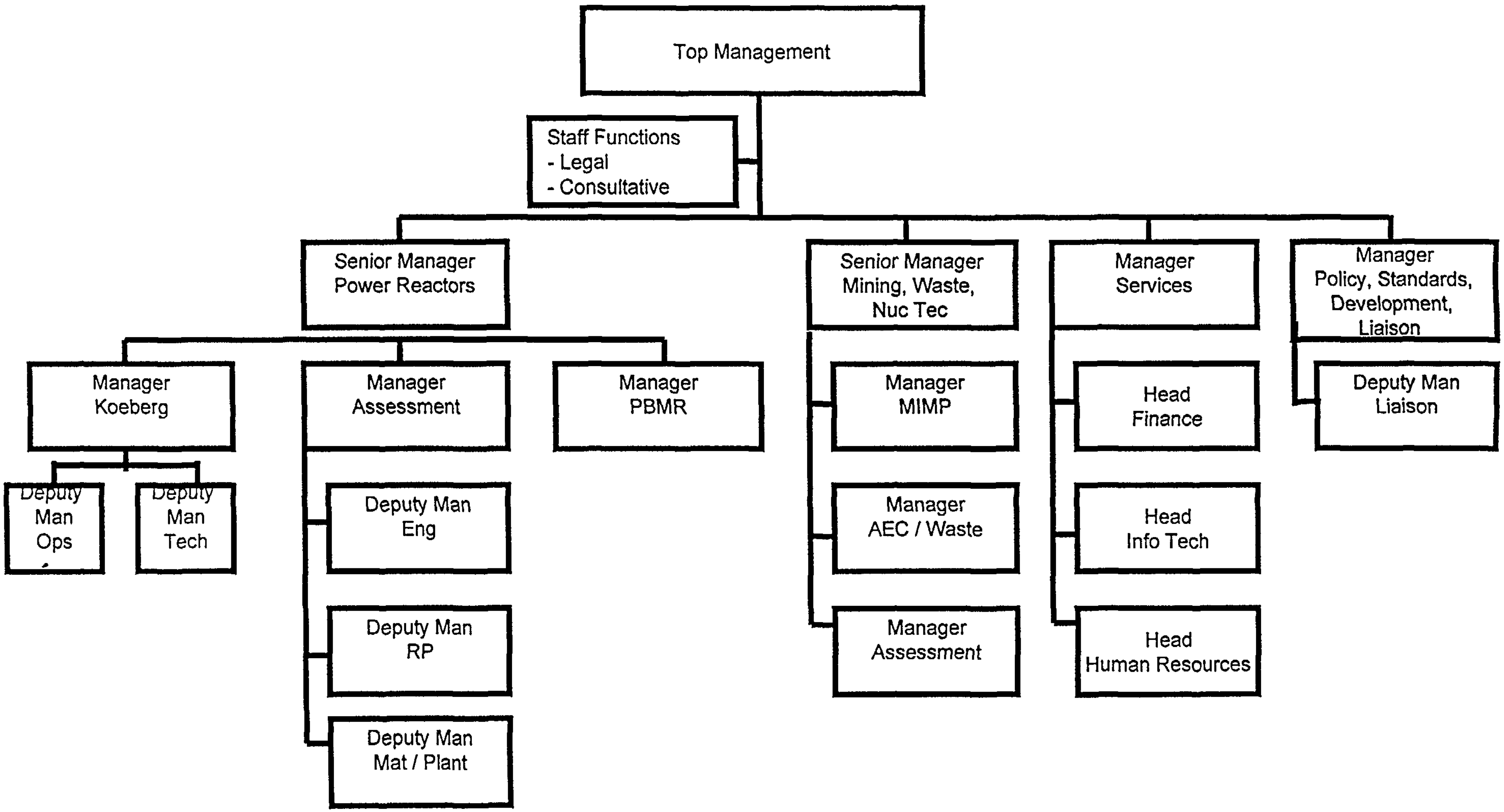

FIGURE 1

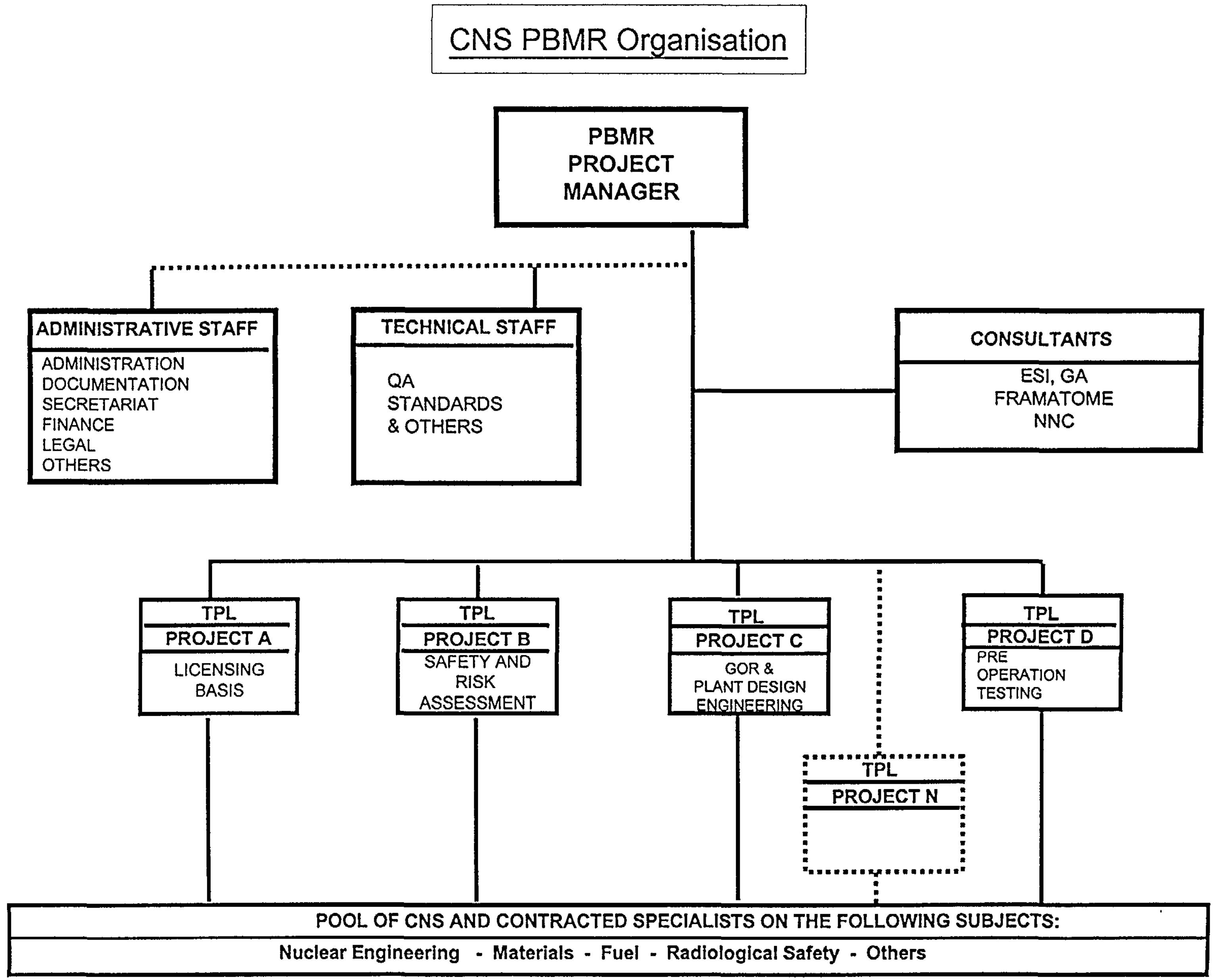
CNS PBMR Organisation
PBMR PROJECT MANAGER
ADMINISTRATIVE STAFF
ADMINISTRATION
DOCUMENTATION
SECRETARIAT
FINANCE
LEGAL
OTHERS
TECHNICAL STAFF
QA
STANDARDS
& OTHERS
CONSULTANTS
ESI, GA
FRAMATOME
NNC
TPL
PROJECT A
LICENSING BASIS
TPL
PROJECT B
SAFETY AND RISK ASSESSMENT
TPL
PROJECT C
GOR & PLANT DESIGN ENGINEERING
TPL
PROJECT N
TPL
PROJECT D
PRE OPERATION TESTING
POOL OF CNS AND CONTRACTED SPECIALISTS ON THE FOLLOWING SUBJECTS:
Nuclear Engineering - Materials - Fuel - Radiological Safety - Others

FIGURE 2

3.2 PROJECT A: Licensing Basis

Objective:

to establish rules which will assure that the PBMR complies with the CNS risk criteria.

Scope

- Basic Licensing criteria
- Quality Management
- PBMR design criteria
- Glossary
- Fuel design limits
- Acceptance criteria
- Licensing basis events
- Codes and standards
- Emergency planning requirements
- Format and content of SAR
- Safety Evaluation Review procedure
- Requirements to evaluation models and computer codes
- Treating of uncertainties
- Licence Draft
- Etc

3.3 Project B: Safety- and Risk Assessment

Objective:

Ensure the development, maintenance and evaluation of a valid safety- and risk assessment of the design and operation of the PBMR.

Scope:

- Safety standards
- PBMR-specific criteria, consistent with the CNS's fundamental safety standards and -criteria
- Thermodynamic and heat transfer analyses related to the nuclear safety of the design and operation and consequent thermodynamic behaviour of reactor core, fuel elements and peripheral systems
- Accident- and upset condition analysis (thermodynamics, determination of success criteria for the aversion of the onset of core damage, accident- and event categorization, etc.)
- Fission product behaviour (Release of fission products from fuel elements, fission product transport, dust behaviour [plate out and resuspension])
- Evaluation of defense-in-depth attributes and safety margins
- Accident management
- Emergency Planning
- Risk assessment (initiating events, component- and system failures [fault trees], accident sequences [event trees], plant damage states, accident progression [source terms], consequences, levels 1, 2 and 3 PRA, data analysis, uncertainty analysis) during both full power and low power/shutdown conditions
- Determine attributes and limitations of importance analyses and qualitative ranking methods that are most appropriate for use in screening analyses and in categorization of structures, systems, and components and human activities according to their contribution to risk and safety

- Computational analyses, computer codes evaluation
- Safety analysis report

3.4 PROJECT C: General Operating Rules & Plant Design Engineering

Identify, establish and agree General Operating Rules requirements. This will include the standard GORs associated with a NPP e.g:

- Operating technical specifications
- Operating procedures
- Accident Management
- In service inspection programme
- Maintenance programme
- Radiation protection programme
- Waste management programme
- Emergency planning

Plant Design Engineering:

- Overall safety concept for the plan
- System design
- Assessment of design and its integrity for structures, components, equipment and systems against criteria developed by Project A
- Materials properties and behaviour under operational and accident conditions
- Component performance and stress analyses.
- Maintainability
- Classification of structures, components, equipment and systems

3.5 PROJECT D: Pre-Operational Testing

The main tasks are:

a) Establishment of Prototype Test Programme
The first reactor is to be considered as a prototype plant for which an extensive step-by-step start-up and commissioning test programme will be required. The expected major testing areas are: fuel design, reactor physics, in-vessel-flow distribution and flow-induced vibration, the reactor vessel, the passive heat removal system, the assumptions and shortcoming of the safety analysis.

The main objectives of this test programme are to:

- resolve safety questions in order to proceed to the next licensing stage;
- decrease and justify uncertainties in the design and safety analysis;
- validate evaluation models and computer codes;
- demonstrate and validate expected inherent safety features of the PBMR-SA design.

A preliminary scope of this activity:

- Establish objectives and acceptance criteria for the various activities of the test programme
- Establish scope of programme
- Identify and agree additional hardware to support the test programme
- Identify the requirements for the commissioning program.

b) Establishment of Testing and Inspection Programmes covering the following scope:

- Identify the requirements for the Testing and Inspection Program, prior and during components manufacture, installation e.g. fuel, reactor vessel, graphite etc...
- Establish and agree programme for manufacturing
- Establish and agree programme for construction
- Pre- & In-service inspection.

Additional subprojects will be organised if need arises.

4. CONCLUSION

This paper represents the licensing project organization and conceptual approach which the CNS intends to follow in evaluation of licencability of the PBMR-SA concept. It is considered that this process will determine whether the concept design can meet the CNS licensing criteria and allow the design to be finalised and for Eskom to move on to the next stages of siting and construction.

The positions and conclusions on all matters discussed in this paper are preliminary and are subject to change.

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TRANSIENT ANALYSIS FOR THE HIGH TEMPERATURE PEBBLE BED REACTOR COUPLED TO THE ENERGY CONVERSION SYSTEM

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Abstract

This paper describes the results of the calculational coupling between a high temperature reactor code and a thermal hydraulic code for the energy conversion system. This coupling has been developed in order to come to a more detailed and realistic simulation of the entire HTR system. Combining the two codes reduces the number of assumptions that have to be made related to the boundary conditions of the two separate codes. The paper describes the models used for the dynamic components of the energy conversion system, and shows the results of the calculation for two operational transients in order to demonstrate the effects of the interaction between reactor core and its energy conversion system.

2 Introduction

Previous studies have been performed on the dynamic behaviour of the Dutch conceptual HTR reactor core [1], [2], but these studies have always considered the reactor core as a stand-alone system. The influence of the energy conversion system (ECS) which converts the produced heat to electric power or industrial steam, was not included in the calculations. A fixed coolant mass flow, a fixed temperature and pressure at the reactor core inlet represented the state of the ECS at which the reactor dynamics were analysed. This approximation holds well for the severe transients investigated, such as Loss Of Forced Cooling (LOFC) and Depressurised Loss Of Forced Cooling (DLOFC). In both cases the flow comes to a halt almost completely, thus effectively cancelling the interaction between the reactor core and the ECS. Only in the first minutes of these transients interaction takes place.
However, when studying normal operational transients the mutual interaction between reactor and ECS is governed by the helium mass flow. This flow rate is influenced by essentially two parameters, viz. the helium inventory and the bypass flow around the core and turbine. The helium inventory is the amount of helium circulating in the system and is adjusted each time load following is required. Bypass flow is used in situations of rapid or near-instantaneous power demand reductions and protects the turbine against excessive overspeed.
As the HTR is a very flexible reactor system and capable of load following, the helium mass flow will vary frequently, and a calculational coupling between reactor core and ECS becomes necessary in order to analyse the plant behaviour in sufficient detail. In the past long calculation times frustrated simultaneous coupled transient calculations, but that is less problematic nowadays so the code for reactor core calculations (Panthermix [3]) and the code for thermal hydraulic calculations for the ECS (Relap5/mod3.2) have been coupled recently [4]. It should be clear from the beginning, that actually two code couplings were required in order to perform total plant calculations. Firstly, the neutronics code Panther [5] and the pebble-bed thermal hydraulic code Thermix-Direkt [6] were coupled for reactor core calculations. This resulted in the HTR-version of Panther, Panthermix. Secondly, this Panthermix code has been coupled to the thermal hydraulic code which describes the ECS, Relap5/mod3.2.
This paper will discuss this second coupling and consider Panthermix as a single code. It presents the analysis of operational transients which clearly influence the helium mass flow through the system (cases of load following, load rejection, generator trip, opening bypass valve). This demonstrates the interaction between the nuclear reactor and its ECS.

3 HTR Reactor Design

The HTR system modelled is the reactor and ECS design of the Dutch INCOGEN study [2]. The reactor cavity measures 2.5 m in diameter and 4.5 m in height. The active core consists of spherical graphite elements (6.0 cm diameter) filled with TRISO coated fuel particles (0.9 mm diameter). The coating is widely acknowledged to be capable of retaining the fission products even at high temperatures (up to 1600°C). The maximum temperature the fuel can reach in the core - even in case of loss of coolant - is expected to be well below this temperature due to a low power density. Helium flows through this pebble bed and heats up from 500°C to 800°C (at 2.3 MPa), thus extracting 40 MW of thermal power. In order to limit the overreactivity in the core, the initial core is filled with pebbles until it just reaches criticality (active core height approximately 1.10 m). New fuel pebbles are added each day in order to remain critical as the depletion proceeds. This is called peu à peu fuelling. After 10 years of operation the core reaches its maximum height and has to be discharged in order to start a new cycle. Due to the minimal overreactivity in the core, and its low power density, the use of control rods is only required during start up and shutdown. The reactor relies on the feed-back mechanism of the negative fuel temperature coefficient to obtain a critical state during operation, or reach a subcritical state after an incident. Shutdown and recriticality behaviour are analysed in [7]. Because of this autonomous reactor shutdown the reactor can be called inherently save.

4 Energy Conversion System (ECS)

The direct Brayton cycle is used as thermodynamic cycle for the ECS. It resembles the Rankine cycle for steam cycles, but a single phase gaseous working fluid (helium) is used instead of a condensing fluid. In this design the Brayton cycle can be termed 'direct', because the helium does not transfer its heat to a secondary steam cycle but powers the turbine directly, resulting in a higher efficiency. The system consists of a single-shaft turbine-compressor with a directly coupled electrical generator. A precooler before the compressor and a recuperator further enhance the overall efficiency (nominal 42%). This HTR system is shown in figure 1.

Unique to a closed cycle power generating ECS is the ability to control the output power on the generator by adjusting the amount of helium circulating in the system. This type of control hardly affects the work point of the cycle, as cycle temperatures and flow velocities do not change at partial power. As a result, even at low power output when only a proportionally small part of the helium inventory circulates, a high efficiency close to nominal can be attained. During these transients helium is stored in or retrieved from large storage vessels [8]. The removal or injection of helium from or to the ECS is a slow process (minutes) and will only be used for long term adjustments. For instantaneous power reduction (load rejection) a bypass valve over the turbine is used.

In the following sections the components of the ECS will be discussed in greater detail.

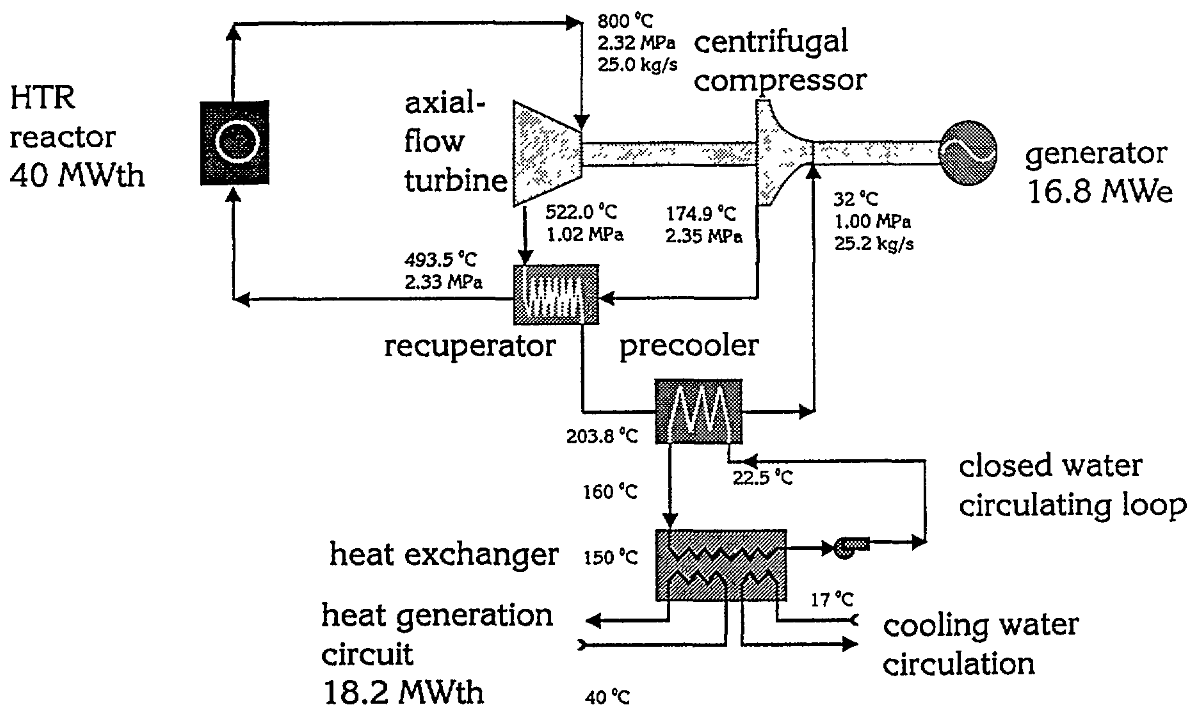

Figure 1: The cycle diagram of the basis configuration.

4.1 Turbine

The Relap5/mod3.2 steam turbine model has been used in order to model the helium turbine. A lumped-parameter model is used wherein a sequence of stages forms the turbine. One single stage consists of a single-row fixed-blade system (nozzle) followed by a single-row rotating-blade system. An efficiency factor based upon simple momentum and energy considerations is used to represent the non-ideal internal processes.
The theoretical design [9] of the turbine contains 7 stages. In our Relap5 model one dummy stage with efficiency zero (used for governing purposes) precedes 4 reaction stages with reaction fraction r=0.5, i.e. the fraction of the stage energy released (enthalpy change) in the moving blade system is half the total enthalpy change over the stage. This is a common design value. The general efficiency formula for a reaction stage turbine reads [10],[11]:

$$\eta = \eta_0 \frac{2v_t}{v^2}(vb - v_t) + [(vb - v_t)^2 + rv^2]^{1/2}, \tag{1}$$

where

- η = turbine efficiency, defined as the actual enthalpy change divided by the isentropic enthalpy change across the stage
- η_0 = maximum turbine efficiency
- b = $(1-r)^{1/2}$
- r = reaction fraction
- v = the helium velocity at nozzle outlet
- v_t = tangential velocity of rotating blades

The maximum efficiency can be found by differentiation to occur when

$$\frac{v_t}{v} = \frac{1}{2(r-1)^{1/2}}. \tag{2}$$

Figure 2 gives the efficiency curve as a function of v_t/v for a turbine stage with r=0.5 and η_0=0.90.
The actual work, W, produced by the fluid on the rotating blades as its momentum is changed, is written as an efficiency factor times the idealised process, the isentropic enthalpy $h_{isentropic}$ change across the stage. From the first law of thermodynamics follows for such a process:

$$W = -\eta \int dh_{isentropic} = -\eta \int \frac{1}{\rho} dP. \tag{3}$$

Performing the integral over the turbine stage for the pressure P at inlet, P_i, and outlet, P_o, gives

$$W = \eta \frac{1}{\rho}(P_e - P_i) \tag{4}$$

where ρ is assumed to be an average constant density for the stage. This assumption is important, and is the reason that a complete turbine cannot modeled with a single stage, as the density differs a factor 2 between turbine in- and outlet. In this calculation with 4 stages, the density is approximately constant within 10% across each stage, which is satisfying.

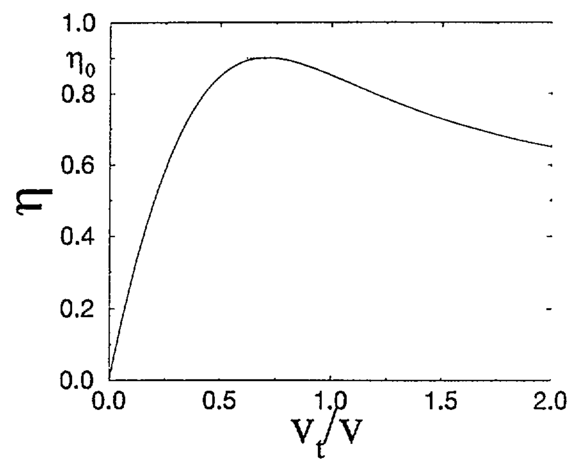

Figure 2: Analytic efficiency curve for a reaction stage turbine with r=0.5 and η_0=0.90.

The remaining fraction $(1 - \eta)$ of the pressure gradient will contribute to changes in the kinetic energy of the helium flow. Note that the inefficiency of the turbine should give some dissipation that is a source of internal energy. The dissipation is assumed to be a small effect and is neglected in the model.

4.2 Radial Helium Compressor

For the initial modeling of the helium compressor the Relap5 water pump model was used. This model has been abandonded now as it assumed constant density during compression. Although being correct for liquid water, it looses meaning for helium gas which is compressed more than a factor 2. Therefore a new routine has been developed which replaces the pump model with a compressor stage model when the fluid quality q equals 1.0, i.e. the fluid is completely in the gas phase. It calculates all variables Relap5 would require from the pump model, including the compressor dissipation. The complicated configuration of a radial compressor precludes a complete first-principle model, so empirical relations are used instead for stage by stage modeling. The current compressor model contains 5 stages. The relations are defined for the work input coefficient λ and dimensionless polytropic head $h_{ndim.}$ both as function of a flow coefficient ϕ. Neglecting the weak dependence on Reynolds-number and tip-speed Mach number gives for one compressor stage [12]:

$$\phi = \frac{\dot{m}}{\rho A u} \tag{5}$$

$$\lambda = \frac{W_c}{u^2 \dot{m}} \tag{6}$$

$$h_{ndim.} = \frac{gh}{u^2} = \frac{g\Delta p/\rho g}{u^2} \tag{7}$$

with W_c the work (J s^{-1}) required to drive the compressor, $\dot{m}$ the mass flow (kg s^{-1}), u the rotor velocity (rad s^{-1}), A the cross sectional area (m^2), g the gravitational acceleration (m s^{-1}), h the head (m) and Δp the pressure drop (Pa). The work point of a compressor stage is subsequently determined from the two compressor maps given in figure 3 and 4. The data for the compressor stage maps have been obtained by scaling the stage characteristics of another compressor due to lack of design- and test data for helium turbines. The dimensionless numbers λ, h_{ndim} and $\dot{m}$ have been normalised to their design values, indicated by the subscript $_{scaled}$. No attempt has been made to describe the stage characteristics during stall and surge operation, as it (presently) is beyond the scope of this study. It could be incorporated in this compressor model, as has been shown [13], but it would also require another turbine model which handles stall and surge correctly.

4.3 Recuperator

The counterflow recuperator has been included in the ECS in order to enhance its efficiency. Instead of rejecting all the heat in the precooler before compression, the heat is used to warm the high pressure helium flow towards the reactor. Both the high and low pressure side of the recuperator have been modeled as a pipe divided in 29 volume elements. Volume element 1 of the one side has a thermal contact with element 29 of the other side, element 2 with 28, etc., in order to model the counterflow. The recuperator is a passive component in these calculations, for instance no effort has been made here to model an attemperation valve in order to protect the recuperator against thermal shock.

4.4 Bypass Valve

The bypass valve connects the core inlet with the turbine outlet, effectively short-circuiting the recuperator high- and low pressure side for part of the helium flow. During steady state operation the bypass valve remains closed, or - if desirable - remains slightly opened in order to adapt easily to small variations in the power demand. If the power demand suddenly drops, which occurs in cases like load rejection or generator trip, the shaft rapidly starts accelerating. Immediately opening the bypass valve prevents overspeed as the helium mass flow through the turbine reduces and subsequently the excess in developed power decreases. Fully opened, the bypass will be able to divert a third of the helium flow, which results in a reduction of power output to around 5% of the nominal conditions.
As a first thought, one could choose to bypass only the turbine instead of both turbine and reactor, but this will upset the work point and the temperatures of the components more than is necessary: in that case the high temperature flow from the reactor is bypassed over the turbine and mixes with the lower temperature turbine output flow. A flow at higher temperature enters the low pressure side of the recuperator. The effect is evident on the high pressure side of

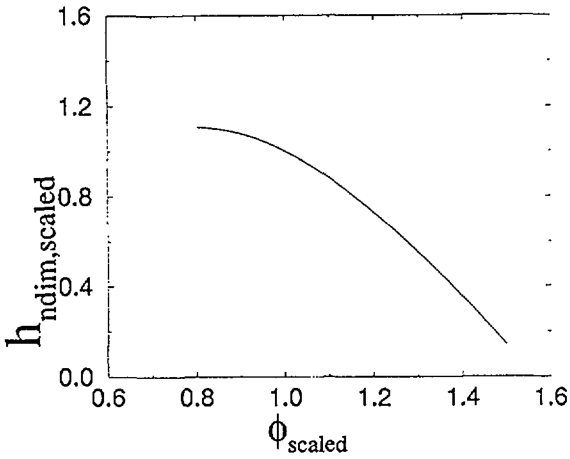

Figure 3: Radial compressor stage map in order to determine the generated head as function of the helium gas flow.

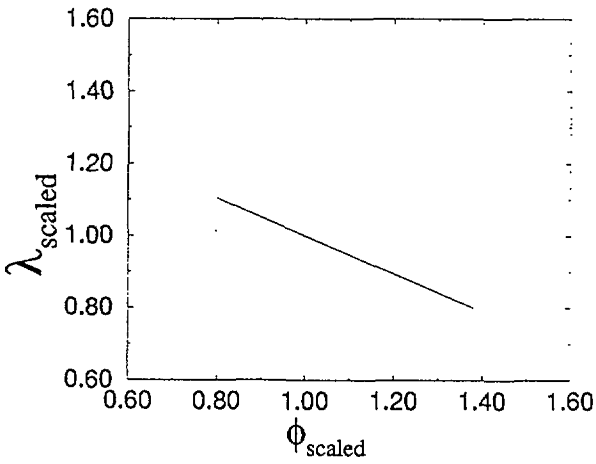

Figure 4: Radial compressor stage map in order to determine the required torque for compression as function of the helium gas flow.

the recuperator; a mass flow at higher than nominal temperature will enter the reactor. The reactor outlet temperature will be higher in return, effectively reinforcing the process. Of course the temperature is bounded; on the reactor side by the feedback of the negative fuel temperature coefficient, on the ECS side by the precooler which functions as a heat sink and keeps the compressor inlet temperature constant at 305 K.
These 'problems' are reduced if both reactor and turbine are bypassed. The difference in temperature between reactor inlet and turbine outlet flow is small (30 K), and mixing the flows gives nearly the same temperature. The overall effect will be that the system recovers faster from the transient and will return sooner to a steady state.

4.5 ECS Control System

The control system has been kept extremely simple, and focuses on two things: 1) to prevent the shaft from overspeed and 2) to produce the demanded power at high efficiency, which means to regulate the helium mass flow. In order to achieve this, three valves are used, viz. the bypass valve and two valves from which one connects the ECS to a helium mass sink and the other to a helium mass source. The bypass valve solely controls the shaft speed: if the shaft rotates too fast, the signal becomes positive and opens the bypass valve proportional to the magnitude of the signal; if the shaft rotates too slow the bypass signal becomes negative and the bypass valve closes or remains closed. The control system has been designed in such a way, that the bypass valve signal also controls the other two valves. A positive bypass signal occurs due to shaft overspeed, and helium can be extracted from the ECS: the valve connecting the sink can be opened. On the other hand, a negative bypass signal indicates a lower than nominal shaft speed, and a larger mass flow is needed to accelerate the shaft back to nominal speed: the valve connecting the mass source can be opened. In order to avoid a system that continually opens and closes valves during steady state operation, the bypass valve remains slightly open and accommodates small variations in the shaft speed.

5 Calculation of Operational Transients

As has been explained in the introduction, the interesting transient behaviour is to be expected during transients with sustained helium mass flow. This means that the LOFC and DLOFC scenarios will not be investigated here; they have been calculated as stand-alone reactor cases in the INCOGEN study [2].
The two transient cases presented here are a load rejection and a load following transient. Both transients involve a strong variation in helium mass flow. This immediately affects the reactor core as more (or less) heat can be transferred to the helium, which in turn gives a change in inlet temperature for the ECS. Because the reactor lacks control rods, reactivity changes will be compensated by temperature changes via the fuel and moderator temperature coefficients. The ECS control system initiates variations in the helium mass flow by injecting helium from the mass source into the ECS, or by opening the bypass valve and removing helium from the system.

5.1 Transient 1: Load Rejection

The load rejection transient resembles the generator trip situation. In a short interval, typically less than a second, the power produced by the turbine cannot be delivered to the electricity grid (in the case of a generator trip because the generator is disconnected electrically, in the case of load rejection because the total plant has been disconnected from the grid). The power surplus causes an immediate acceleration of the shaft rotational speed, and the bypass valve must open in order to protect the turbine, compressor and generator against overspeed. Diverting part of the helium mass flow from the turbine to the compressor creates the situation in which the turbine has to deliver more power (more helium flows through the compressor) from a reduced helium flow (part of the flow has been bypassed). A balance is sought for a situation where the shaft speed is close to its design value (12000 rpm). Operating the ECS with the bypass valve open is inefficient, as the power normally used to drive the generator now is used for extra compression. The solution is to decrease the power supplied by the reactor in the form of heat, which can be achieved by reducing the amount of medium that transports the heat, i.e. helium. Therefore helium will be removed from the system up to the point that the bypass valve is only just opened; the shaft speed has returned to design speed and the high efficiency of the loop has been established once again. In the case of load rejection, only power for house load has to be produced which equals 5% to 10% of nominal power output. Here 10% has been chosen.
The progress in time for the load rejection transient has been chosen as follows: in 1 second the generator load is rejected from nominal to 10%. During 10 minutes the control system removes helium from the ECS until normal efficiency has been restored. At time = 30 min the power demand rises from house load (10%) to nominal load in 1 minute. The control system starts filling the ECS again, but will not be able to do so in 1 minute. After about 10 minutes the system will be refilled and the transient will be followed for another 40 minutes.
In figure 5 the power demand, power output and reactor thermal power are shown as function of time. The power output is in good agreement with the demanded power up to the point where the demand rises from house load to nominal power. Due to a (small) xenon build-up the reactor is not able to reach nominal power at once and has to wait until some xenon has been 'burned away'. This problem is analysed in more detail in the following section about the load following transient. Figure 6 shows the flow through the three valves in the system: the bypass valve which opens immediately if the load is rejected, the valve connected to the helium mass sink by which helium is removed from the system, and the valve connected to the helium mass source which adds helium to the system when the power demand rises. Finally, with figure 7 the mass flow through the turbine can be compared to the mass flow through the compressor during the period that the bypass valve is opened.

5.2 Transient 2: Load Following

The load following transient can occur daily, depending on the power demand behaviour of the customer, and on other electricity plants. The progress in time for the load following transient has been chosen as follows: Initially the reactor and ECS are in a steady state situation at nominal power. At a rate of 10% $\min^{-1}$ the power demand is linearly decreased to 20%, and the ECS follows by removing helium until the bypass valve closes. This procedure has already been discussed for the load rejection transient. The system is kept at this state for 5 hours in order to monitor the xenon build-up which reaches its maximum after approximately 6 hours. In a previous study [4] the xenon transient has been followed for 50 hours for the coupled system. This will not be repeated here. Instead, we look in more detail to what happens in the first 5 hours.
In figure 8 the power demand, power output and reactor thermal power are shown as function of time. Power demand and output are in good agreement, but in order to explain why the thermal reactor power output increases the xenon build-up has been plotted in figure 9 in terms of xenon worth. The core reacts to the increase of xenon reactivity by reducing the power. As a consequence the fuel temperature decreases, causing an increase in reactivity and thus the xenon build-up has been balanced. The lower fuel temperature gives a lower core outlet temperature for the helium, which gives a lower turbine outlet temperature. That means that the flow on the low pressure side of the recuperator is cooler, and as a result the core inlet temperature is cooler. In the Relap5 model the turbine extracts less power from a mass flow at lower temperature and is not able to sustain the nominal rotational speed (one should realise here that the compressor is nearly always operated under design conditions as long as the shaft speed is nominal because the precooler produces a constant compressor inlet temperature of 305 K. This means that the load the turbine experiences only depends on the mass flow (equation (7)) , because the volumetric flow remains fairly constant (equation (6)) during the transient). Adding more mass to the system enables the turbine to extract relatively more energy than the compressor load increases, and the power demand can be met. The increase of the helium mass flow through the compressor is shown in figure 10, the temperature behaviour at core inlet, core outlet (=turbine inlet) and turbine outlet in figure 11.
Whether the reactor plant behaviour is acceptable during such transients or not, is still a point of discussion. One could argue that adding control rods would stabilise the temperatures, however, doing so would increase the complexity of reactor design and control system.

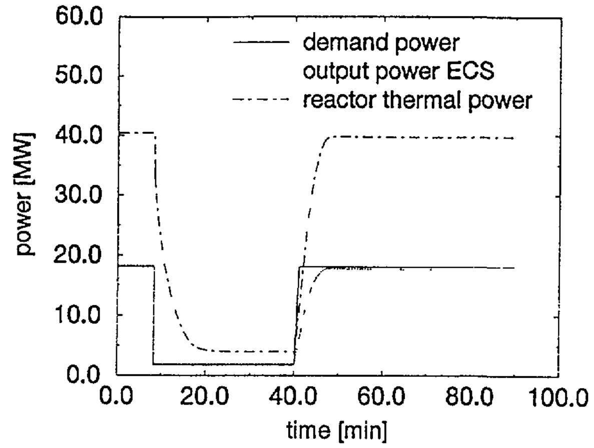

Figure 5: The power demand, power output by ECS and the thermal reactor power as function of transient time.

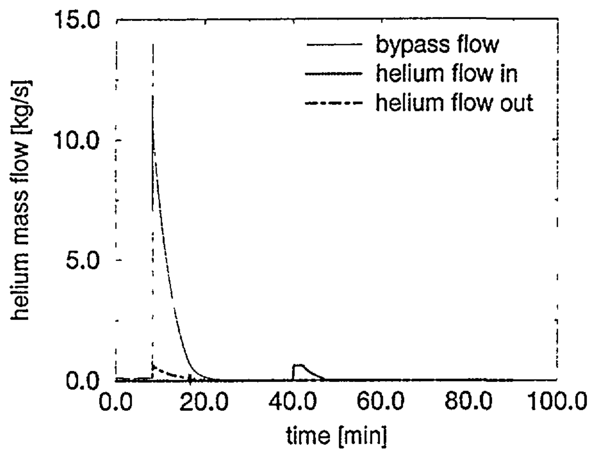

Figure 6: The helium mass flow as function of transient time for the bypass valve, the valve connected to the helium mass sink, and the valve connected to the helium mass source.

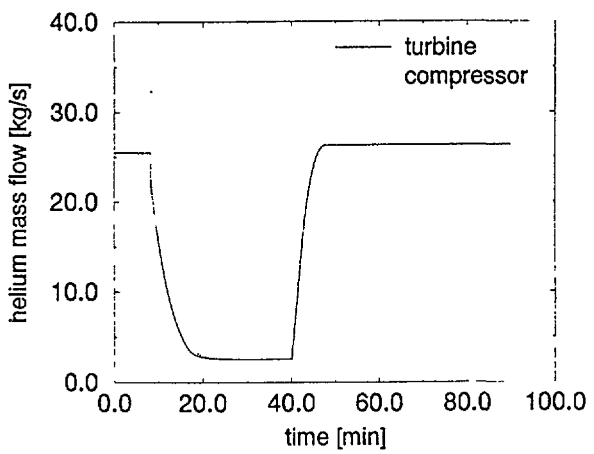

Figure 7: The helium mass flow through the turbine and compressor as function of transient time

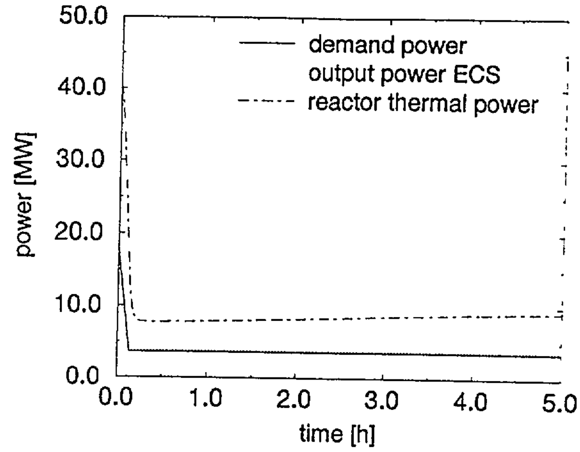

Figure 8: The power demand, power output by ECS and the thermal reactor power as function of transient time.

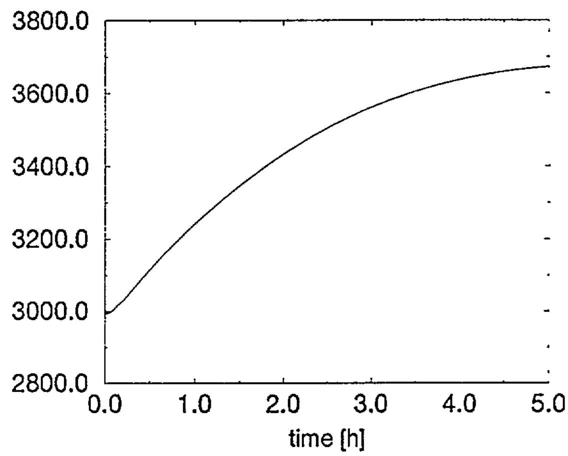

Figure 9: The xenon build-up in terms of xenon worth as function of transient time.

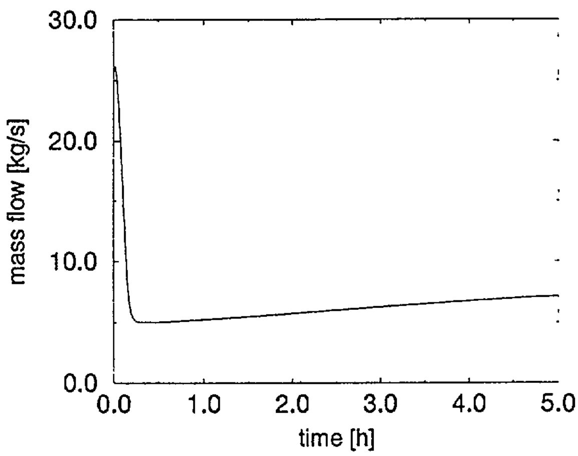

Figure 10: The helium mass flow (kg/s) through the compressor as function of transient time.

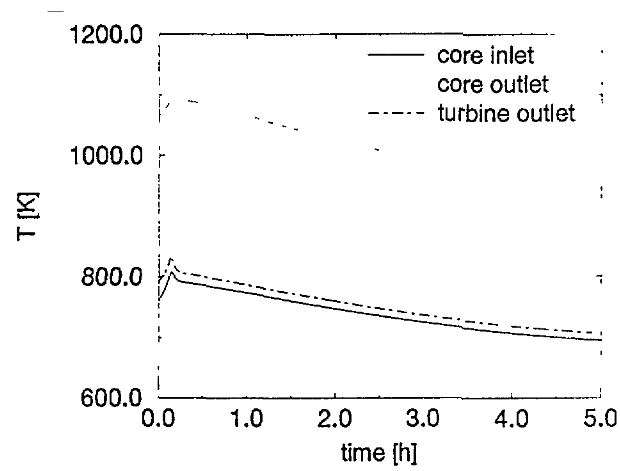

Figure 11: The core inlet, outlet and turbine outlet temperatures as function of transient time.

6 Conclusions + Future Outlook

Linking the Relap5 code with helium as a working fluid to Panthermix through the coupling software Talink opens up the possibility to model the thermal hydraulics and 3-D neutronics of a pebble bed HTR together with its ECS. A more realistic simulation of the entire system can be attained, as the coupling removes the necessity of forcing boundary conditions on the simulation models at the data transfer points. Two transient cases have been discussed which are representative as operational transients for an HTR. The first transient case represents a load rejection case and shows the functioning of the control system, in particular the bypass valve. The second transient is a load following transient where the influence of the xenon build up on the rest of the system becomes apparent. Both transients can only be calculated with sufficient accuracy if the reactor core and ECS are coupled.
Future developments will include the refinement of the control system, and the components-wise comparison of the ECS with other calculations [13]. Also, the question will have to be answered whether such a simple reactor design as described justifies full 3-D core calculations or that a point-kinetic model is sufficient (of course, the point-kinetic parameters will be calculated from the full 3-D model).

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APPLICATION STUDY OF THE HEAT PIPE TO THE PASSIVE DECAY HEAT REMOVAL SYSTEM OF THE MODULAR HTR

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Abstract

To investigate the applicability of the heat pipe to the decay hat removal (DHR) system of the modular HTRs, preliminary study of the Heat Pipe DHR System was performed. The results show that the Heat Pipe DHR System is applicable to the modular HTRs and its heat removal capability is sufficient. Especially, by applying the variable conductance heat pipe, possibility of a fully passive DHR system with lower heat loss during normal operation is suggested.

The experiments to obtain the fundamental characteristics data of the variable conductance heat pipe were carried out. The experimental results show very clear features of self-control characteristics. The experimental results and the experimental analysis results are also shown.

INTRODUCTION

One of the most important features of the modular HTR is a passive decay heat removal (DHR) characteristic. It is performed by the cooling system installed outside the reactor pressure vessel. Residual heat and decay heat generated in the core is transferred to the reactor pressure vessel and then to the decay heat removal system passively by the natural phenomena such as conduction, radiation and convection.
The purpose of our study is to establish the concept of the fully passive and highly reliable decay heat removal system utilizing the separated type of heat pipe. Moreover the aim of our concept is a self-regulation characteristics to minimize the heat loss under the normal operation with maintaining the sufficient heat removal capability during the accidents.

The authors are examining the application of the heat pipe system to the DHR system expecting to have comparatively high heat removal capacity and low pressure vessel temperature without any active components.

This paper provides the outline of the preliminary study of the Heat Pipe DHR system. The experiments to obtain the fundamental characteristics data of the variable conductance heat pipe were carried out and the experimental results with the experimental analysis results are also shown.

PRELIMINARY STUDY OF HEAT PIPE DHR SYSTEM

System Concept

The concept of the Heat-Pipe DHR System is shown in Fig. 1.

The separated type of heat pipe is suitable for DHR system because of its large heat removal capacity. It consists of four parts, i.e., pipe for vapor, pipe for liquid, the evaporator part and the condenser part. The evaporator part is installed on the reactor cavity wall. Decay heat of the core is transferred from the reactor pressure vessel to the evaporator part by radiation and natural convection in the reactor cavity. In the evaporator part, the heat from the reactor pressure vessel is changed into the latent heat of the working fluid and flow to the condenser part outside of the reactor building. The condenser part comprises heat exchanging pipes cooled by the atmospheric air through natural convection strengthened by the stack. Decay heat of the core is transferred to the atmosphere in this condenser part and condensed working fluid returns to the evaporator part by the gravity.

To improve the advantages of the Heat Pipe DHR System, an introduction of Variable Conductance Heat Pipe (VCHP) was investigated. Basic principle of VCHP is adding a quantity of non-condensable gas into the system for controlling the system working temperature at the intended temperature level. By adjusting the amount of the non-condensable gas in the system, the volume of the surge tank connected to the condenser, as shown in Fig.1, or the gas pressure initially enclosed into the system, it can be designed that the heat loss during the normal operation can be reduced with maintaining the maximum heat removal rate under accident.

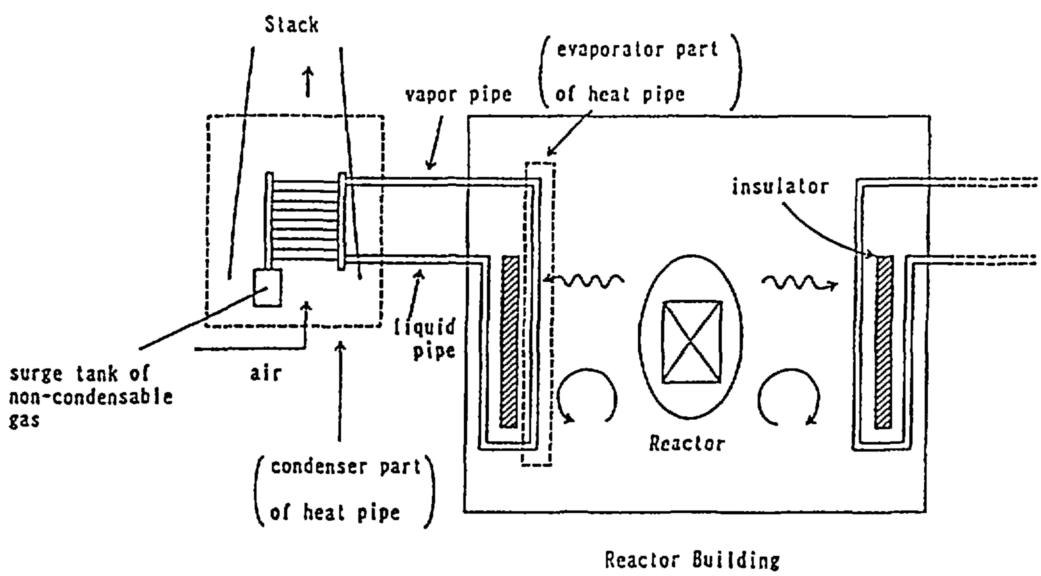

Fig.1. Concept of Variable Conductance Heat Pipe (VCHP) Decay Heat Removal System for Modular HTR

No active components are used in the system and the core decay heat is removed by fully passive measures.

Temperature Distribution Analysis under Depressurization Accident

Temperature distribution analyses of the reactor with heat pipe DHR system under the normal operating condition and the depressurization accident were performed. 350MWth MHTGR is taken as a reference reactor here.

Computer program and analytical model

Computer program used in the analysis is TAC-2D which is a two dimensional heat transfer calculation program using the finite difference method.[4] The reactor and the decay heat removal system outside of the reactor vessel were modeled in a r-z cylindrical configuration as shown in Fig. 2. The structures above the upper reflector and below the lower reflector were neglected and the top of the upper reflector and the bottom of the lower reflector were modeled as adiabatic boundaries for simplicity. This simplification does not have much effect on the estimation of the maximum core temperature and the maximum reactor vessel temperature.

The active core was modeled into one homogeneous material, which has an equivalent thermal conductivity and heat capacity of the fuel block. The decay heat was calculated with Sure's equation.

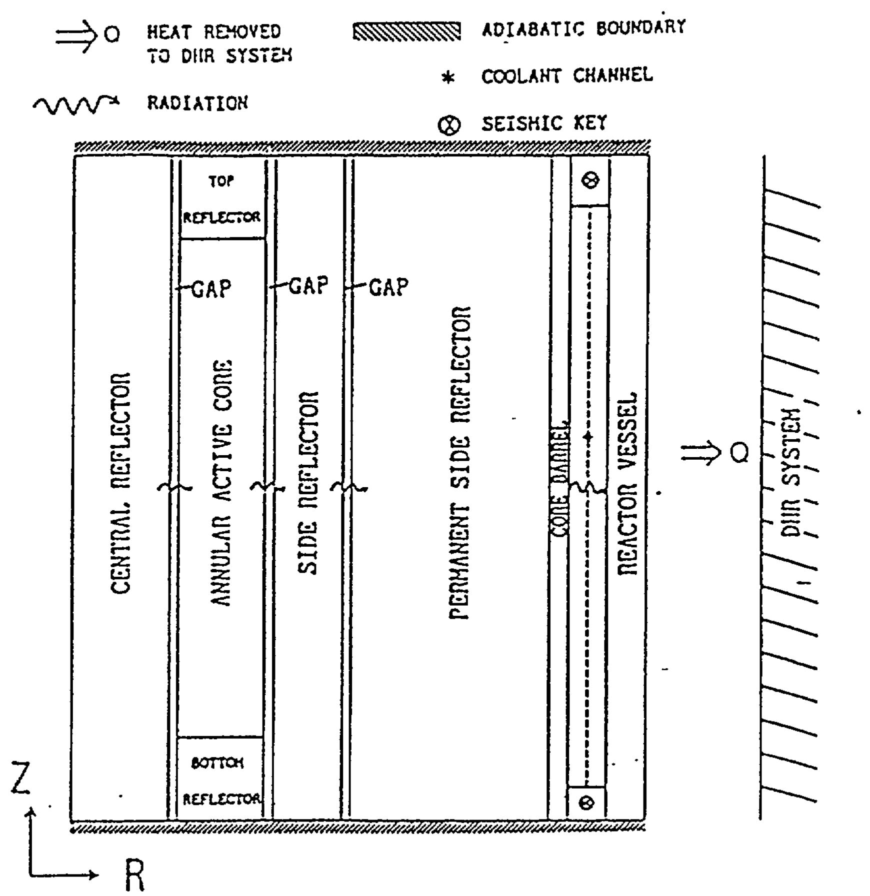

Fig.2. Analytical model for the temperature Distribution calculation

Analytical model for Heat Pipe DHR System

The heat removal characteristics of the Heat Pipe DHR System previously derived as a function of the reactor vessel temperature were modeled as a boundary condition outside the reactor vessel.

Initial condition

The core temperature distribution under the normal operation were previously calculated with an adiabatic single channel model for each radial region of the core. Then the temperature of the reactor vessel and the other reactor internals were calculated with the reactor model shown in Fig. 2 under the boundary condition of the core temperature distribution derived above.

Results of the transient temperature calculation

The calculation results of the transient temperature under the depressurization accident are shown in Fig.3.

After loss of primary system pressure and forced cooling from full power operation, the core temperature begins to increase. The temperature of RPV and DHR system under normal operating condition are 203℃ and 61℃, respectively. The heat loss during the normal operation is 560 kW, which is lower than the case of the ordinary heat pipe system. The maximum core temperature during accident reaches to 1399 ℃ and the maximum reactor vessel temperature is 415℃, which are well below than the safety limits.

The advantage of the proposed system is to be capable of selecting the preferable working temperature with consideration on both the normal condition and accident condition. In this study, the working temperature at 100 ℃, for example, is selected.

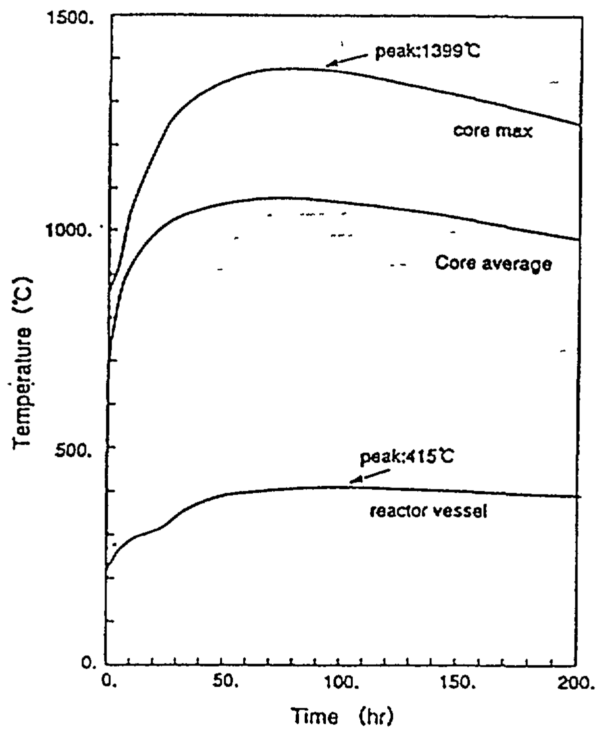

Fig.3. Temperature Behavior under Depressurization Accident

EXPERIMENTS

Experimental Apparatus

To confirm the fundamental characteristics of the VCHP, we have conducted a series of experiments with three experimental apparatuses which are (1) horizontal condenser type with forced air cooling, (2) vertical condenser type with forced air cooling and (3) vertical condenser type with water pool natural cooling. These results are already reported in the previous paper.[5)]

As the cooling condition of the third apparatus are not clear, we have prepared a new apparatus with forced water cooling, which is shown in Fig.4. Nitrogen gas was fed from N_2 gas bomb through a reducing valve at an intended pressure level. The system was also capable of being vacuums by a rotary pump. The heat was supplied by a sheathed heater fixed around the evaporator covered with thermal insulation and was controlled by the voltage regulator.

Experimental Results

Nitrogen gas was used for the non-condensable gas and water or methanol was used as working fluid. The temperature distributions in the condenser tubes were recorded by scanning a sheath thermo-couple through a thin tube enclosed in the condenser tubes. Experiments have been done under the various initial N_2 gas pressure Po and heat input Q .

Fig.5 shows temperature distribution behaviors in the case water is used as working fluid when the heat input is varied from 300 W to 1100W on the condition without non-condensable gas initially enclosed. In this case it is characterized that the system temperature,i.e., vapor temperature inside the tube is comparatively low and the temperature distribution is flat. The temperature level of the system is increased in accordance with the increase of heat input. In this case, the condensation seems to be occurred for the whole region of the condenser tube, and consequently temperature level bocomes relatively low.

If non-condensable gas is enclosed initially, however, the temperature behaviors drastically change. Fig.6 shows the temperature distribution in the condenser tube in response to the increase of the heat input. The heat input was varied from Q=300 W to 1100W. Nitrogen gas was initially enclosed at 0.014 MPa (about 0.14 ata). Temperature in the evaporator side, that is the system temperature, is fairy increased while the temperature in the return tube side slopes down. It seems that the most part of the condensation area is filled with nitrogen gas and the condensation is limited to the area starting the temperature sloping down. When the heat input is increased, condensation area increased by itself, thus the controlling the system temperature rise. This result shows clearly the self-control features.

The steam boundary seems to automatically expand in accordance with the increase of the heat input, resulting in moderate system temperature rise.

The system temperature behaviors in response to the various heat input Q and the initial N_2 gas pressure enclosed in the case of water as working fluid are shown in Fig.7. With no N_2 gas enclosed, the system temperature increases with the increase of heat input because the system has no self-control

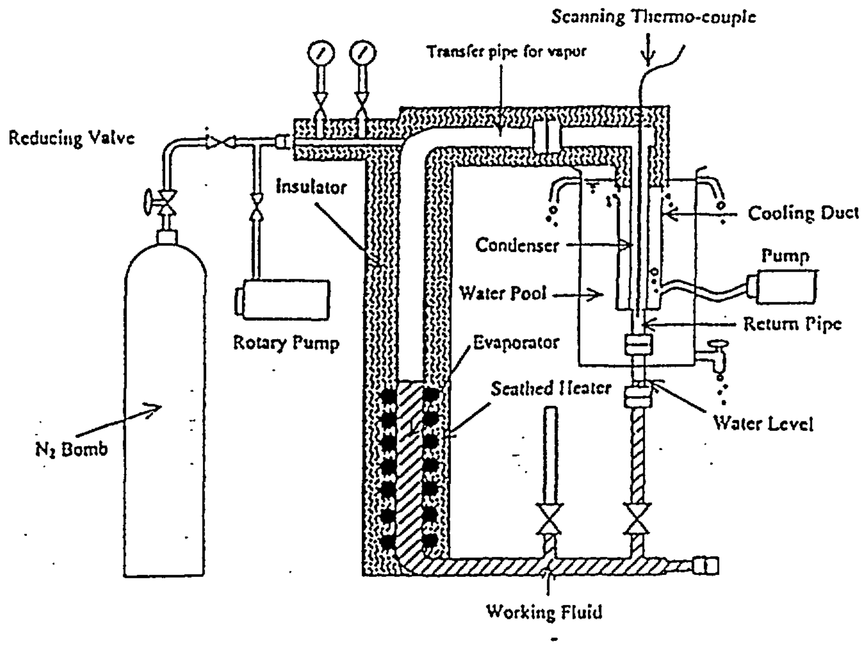

Fig.4. Experimental Apparatus

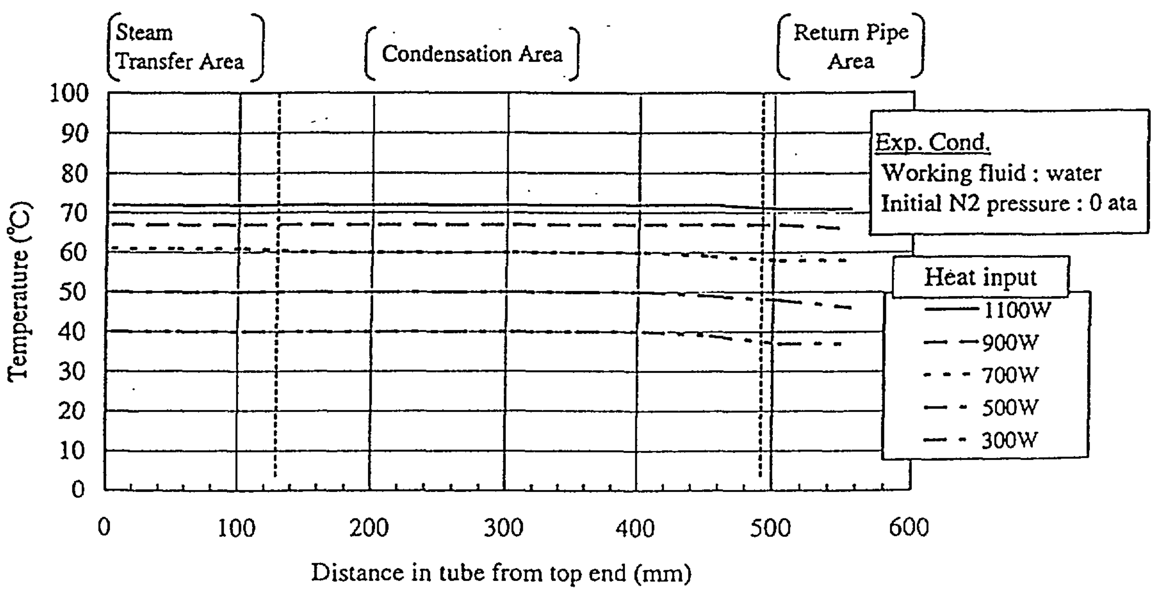

Fig.5 Temperature distribution in the tube for various heat input
in the case of "without N2 gas"

features. If non-condensable gas is enclosed, the system temperature steps up with the nitrogen gas pressure, but becomes fairly flat.

Fig.8 shows results in the case that methanol was used as working fluid. Though the system temperatures are relatively lower than the case of water, it shows similar characteristics on the water as working fluid.

These facts mentioned above means that by selection of gas and by designing the gas pressure initially enclosed the intended temperature can be achieved and the system temperature can be controlled in such a system of the separated type, variable conductance heat pipe systems.

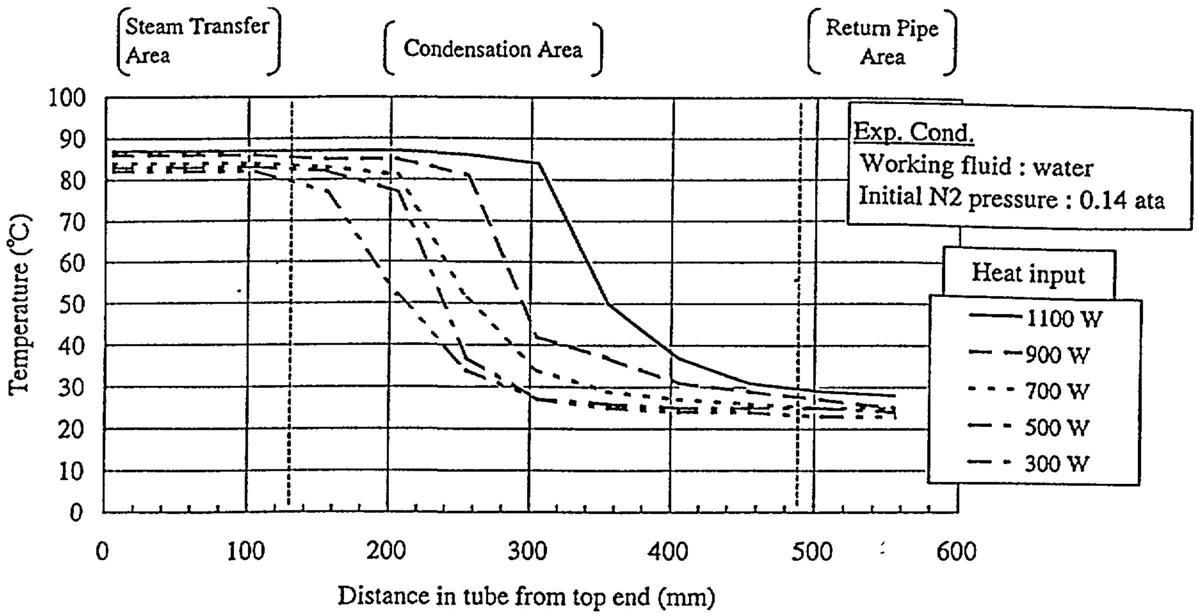

Fig.6 Temperature distribution in the tube for various heat input
in the case of "N2 gas enclosed"

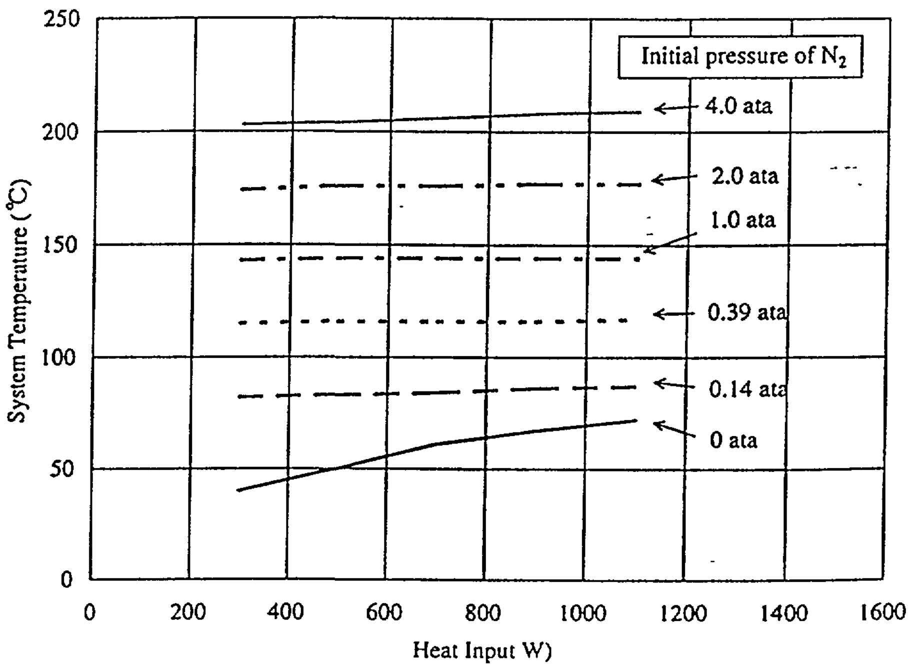

Fig.7 System Temperature Behaviors in Response to the Parametric Changes of Heat Inpu
-Working fluid : Water-

Experimental Analyses

Model of self-control mechanism

The experimental results show very clear features of self-control of the system to the wide range of heat input changes. Though these facts it was confirmed that the characteristics of the self regulation behaviors are true for the heat pipe system of separate type.

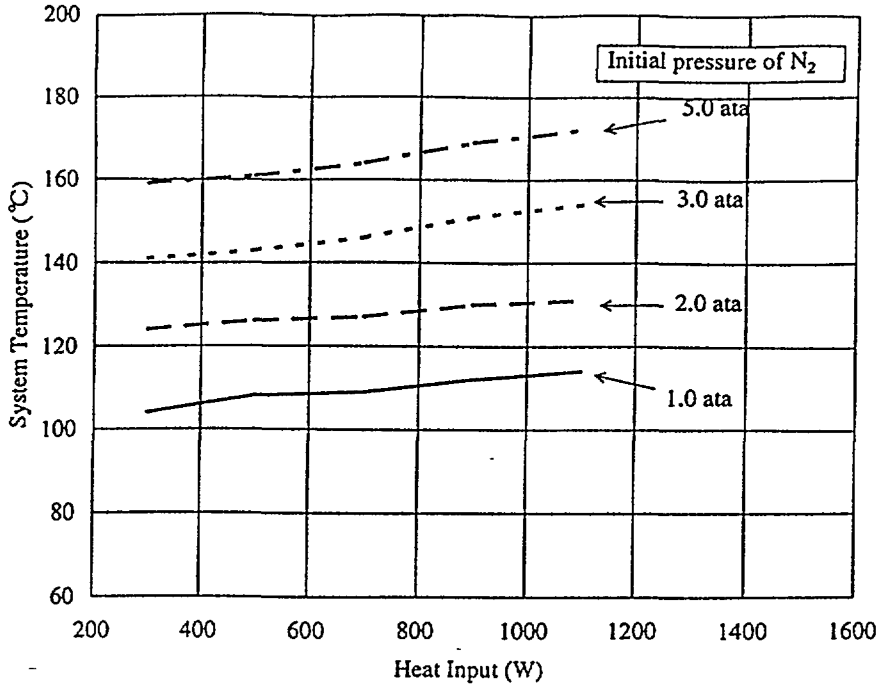

Fig.8 System Temperature Behaviors in Response to the Parametric Changes of Heat Inp
-Working fluid : Methnol-

The self-control mechanism supposed for these heat pipe systems is shown in Fig.9. During operation there may be formed a boundary between the vapor region and non-condensible gas region. Vapor region expands with the increase of heat input and compresses the gas volume to widen the condensing area, suppressing the system temperature rise consequence.

Analysis

The analysis to predict the thermal conditions of the system was done with focusing on the volume of nitrogen gas enclosed in the system, by combining the equation of state about the conditions of the nitrogen gas before and during operation with the relation between the heat input and the heat transfer and the relation between the saturated temperature and the pressure of the working fluid such as water or methanol.

With the nitrogen gas enclosed in the system, the thermal conditions can be expressed by equation of state before and during the system operation as follows;

$$\frac{PoVo}{To} = \frac{(Pv - Pvi)\{Vc(1-x) + Vr\}}{T2'} \qquad (1)$$

In equation (1), Po, Vo, and To is the pressure, volume and temperature of the nitrogen gas before operation , and Vc and Vr are the volume of the condenser and the volume of the rest, i.e., the volume of the gas space below the condenser to the water surface in the return tube. Similarly, Pv and Pvi are the saturated pressure of the steam and the partial vapor pressure in the nitrogen gas respectively, and T2' is the temperature of the gas and x is the volumetric ratio of the condensing part in the condenser tube, all during operation. The saturated pressure Pv (ata) for the steam and methanol used in our analysis is shown in Table 1.

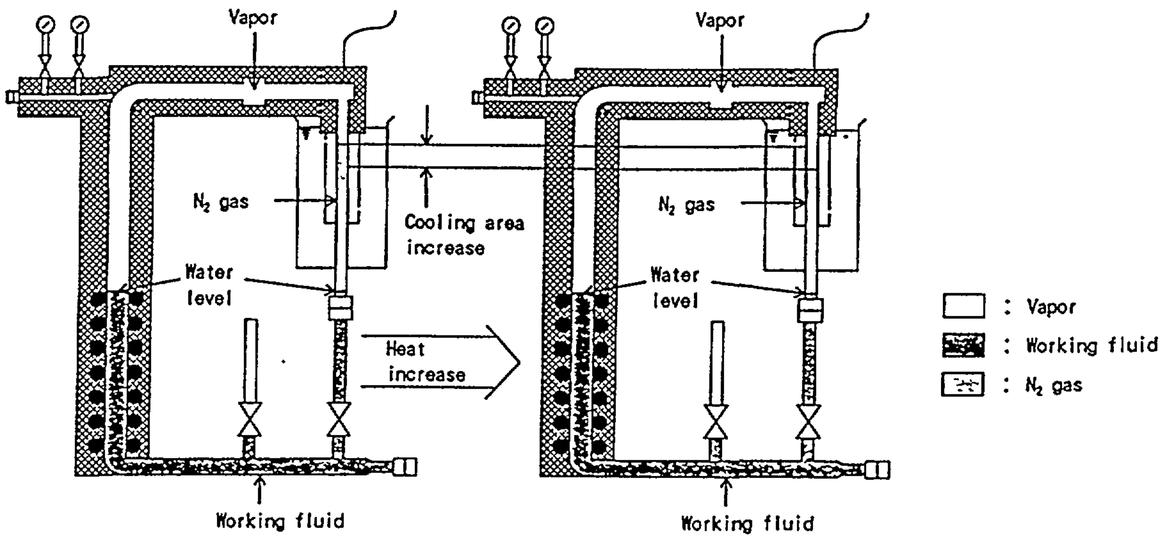

Fig.9 The self-control mechanism supposed in the separated type heat

Table 1. Saturated pressure Pv(ata) for steam and methanol

Working Fluid	Equation of saturated pressure	Note
Steam	$Pv=0.00031(Tv+273)^2 - 0.200(Tv+273)+32$ (Tv:50～100℃) $Pv=0.00129(Tv+273)^2 - 0.200(Tv+273)+32$ (Tv:100～200℃)	
Methanol	$\mathrm{Log}\ Pv=8.07 - \dfrac{1575}{Tv+238.9}$	Ref. 7

Tv; Saturated temperature (℃)

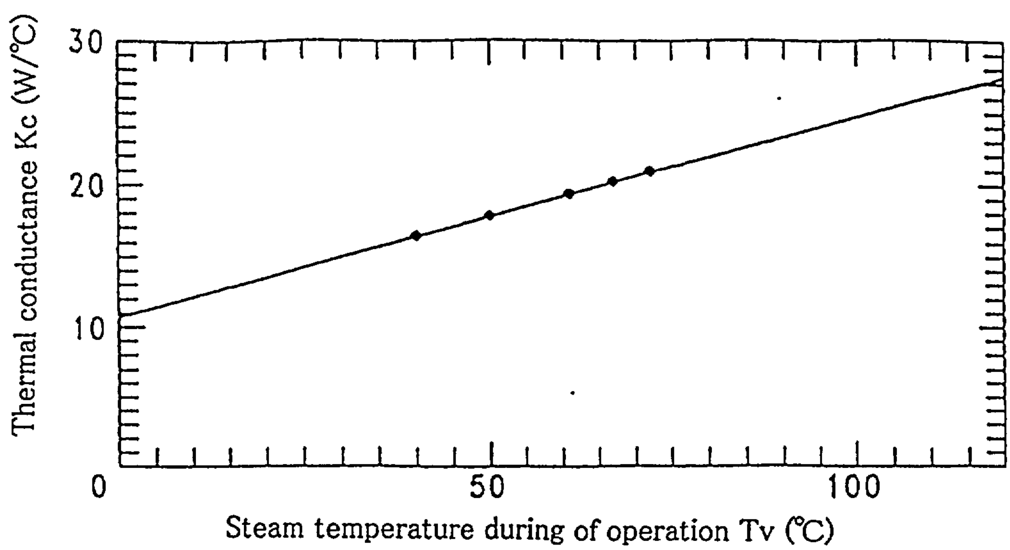

Fig.10 Thermal conductance Kc of the condenser v.s system temperature

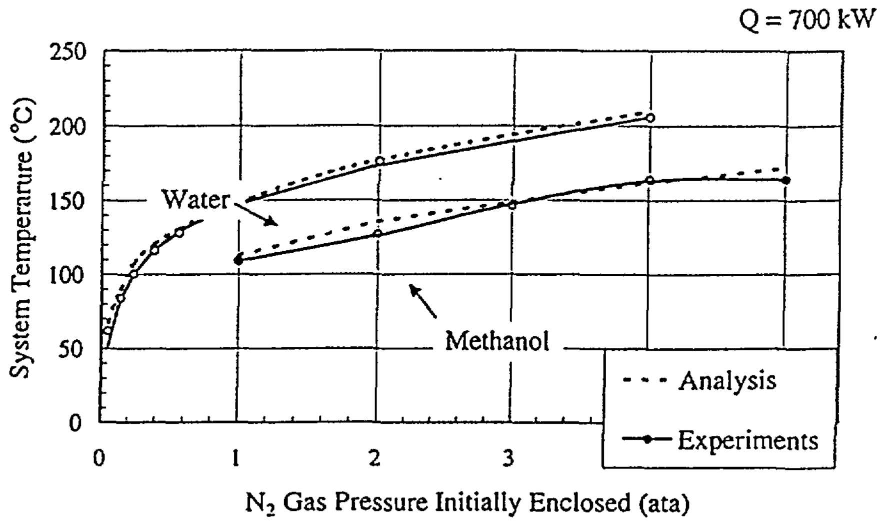

Fig.11 Comparison of the analysis with the experiments

The relation between the heat input Q and the heat discharging at the condensing part of the condenser is expressed as following;

$Q = Kc \cdot x(Tv - Ts)$,

where Kc is the thermal conductance of the condenser and Ts is the temperature of the heat sink. The values of the Kc used were derived from the data obtained from the experiments in the case of both water and methanol. For example, the thermal conductance Kc for water as working fluid is derived from the Fig.10.

Fig.11 shows the comparison of the system temperature to the nitrogen gas pressures enclosed on both the water and methanol as the working fluids calculated at the heat input Q=700W. The analytical results are fairly in good agreement with the experiments.

CONCLUSION

The Heat Pipe DHR system, which is fully passive up to the final heat sink, has a sufficient capability to maintain the reactor vessel temperature under its safety limit during accidents. Especially, by adoption of the variable conductance heat pipe, it is possible to design a fully passive DHR system, which can minimize the heat loss during normal operation and can achieve a lower reactor vessel temperature during accidents.

The experiments to obtain the fundamental characteristics data of the variable conductance heat pipe were carried out for the parametric changes of the heat input under the various pressures of non-condensing gas initially enclosed, including the experiments without enclosing the gas for comparison. Water and methanol were used as the working fluids and nitrogen gas was used as the non-condensible gas.

The experimental results showed very clear variable conductance features, i.e., self control characteristics. The working temperature of the system was clearly dependent on the pressure of the non-condensible gas initially enclosed, with higher system temperature with higher gas pressure enclosed. The mechanism of such features of the self control was made clear.

The analyses were done on water and methanol as the working fluids, which show very good agreement with the experimental results.

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NUCLEAR GRAPHITE AGEING AND TURNAROUND

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Abstract

Graphite moderated reactors are being operated in many countries including, the UK, Russia, Lithuania, Ukraine and Japan. Many of these reactors will operate well into the next century. New designs of High Temperature Graphite Moderated Reactors (HTRS) are being built in China and Japan.

The design life of these graphite-moderated reactors is governed by the ageing of the graphite core due to fast neutron damage, and also, in the case of carbon dioxide cooled reactors by the rate of oxidation of the graphite.

Nuclear graphites are polycrystalline in nature and it is the irradiation-induced damage to the individual graphite crystals that determines the material property changes with age. The life of a graphite component in a nuclear reactor can be related to the graphite irradiation induced dimensional changes. Graphites typically shrink with age until a point is reached where the shrinkage stops and the graphite starts to swell. This change from shrinkage to swelling is known as "turnaround".

It is well known that pre-oxidising graphite specimens caused "turnaround" to be delayed, thus extending the life of the graphite and hence the life of the reactor. However, there was no satisfactory explanation of this behaviour.

This paper presents a numerical crystal based model of dimensional change in graphite, which explains the delay in "turnaround" in the pre-oxidised specimens irradiated in a fast neutron flux, in terms of crystal accommodation and orientation and change in compliance due to radiolytic oxidation.

Introduction

The graphite in a graphite moderated nuclear reactor not only acts as a moderator but is also a major structural component, which can not easily be replaced. The life of the graphite may therefore govern the life of the nuclear plant.

During operation the graphite components change size due to fast neutron damage. In addition there are also significant changes to many other graphite properties, most importantly to the strength and modulus, coefficient of thermal expansion and thermal conductivity. During reactor operation significant irradiation dose and temperature gradients in the large cross-sections of these graphite components leads to the generation of internal stresses, which may eventually result in component failure. However irradiation creep relieves these stresses. This is fortunate because it is irradiation creep coupled with suitable design that allows graphite to be successfully used as a moderator.

Whilst fast neutron induced physical changes to the graphite are of prime concern in helium cooled high temperature reactors and light water cooled graphite moderated reactors, in the case of carbon dioxide cooled graphite moderated reactors radiolytic oxidation leads to significant graphite weight loss. This weight loss occurs fairly uniformly throughout the accessible graphite porosity within the component reducing the graphite density. In addition weight loss also modifies some of the graphite properties, particularly the strength, modulus and thermal conductivity.

The fast neutron induced changes to the physical properties are primarily a function of the dimensional change within the well graphitised crystals that make up typical nuclear graphite components.

These crystal dimensional changes arise due to the formation of interstitial and vacancy loops within the graphite basal planes. The formation of these loops causes 'c' axis expansion and 'a' axis contraction, the magnitude of which is a function of fast neutron dose and irradiation temperature, see Fig. 1.

Typical nuclear graphite is manufactured from coke particles joined together by a pitch binder. After graphitisation the coke particles consists of well graphitised crystallites which contain a number of cracks of various sizes parallel to the basal plane. These filler coke particles are typically either needle shaped or spherical shaped. The first type of filler particles leads to anisotropic graphite component material properties, the second type of filler particle leads to almost isotropic material properties.

The binder consists of a conglomerate of much smaller graphite crystals and less well graphitised material and contains gas evolution pores. These gas evolution pores and the porosity within the graphite coke particles give the graphite component a much lower density than may be expected (~1.7-1.9 g/cm^3) compared with 2.26 g/cm^3 for pure graphite crystals.

At the temperatures of interest in modern graphite moderated reactors (greater than 300°C) a typical nuclear graphite shrinks with irradiation dose, before "turning around" and swelling until the graphite reaches its original volume and beyond. At this point all of the graphite physical properties are rapidly deteriorating.

Various attempts [Simmons, 1965, Brocklehurst and Kelly, 1993, Sutton and Howard, 1962, Jenkins, 1964] have been made to model dimensional and other irradiation induced physical changes in graphite. In the first three of these references an attempt was made to relate the dimensional change to the accommodation offered by the small shrinkage cracks [Mrozowski, 1956] within the graphite crystal structure and the orientation of the crystals. However these models assume graphite to be a loose assembly of unconnected graphite crystals. [Jenkins, 1964] pointed out that this was not the case and attempted to devise a mathematical model which accounted for the compliance of the individual crystals and global polycrystalline compliance as well as the accommodation and crystal orientation. Unfortunately his model was very complex and did not readily lend itself to structural assessment of irradiated graphite behaviour. However, the development of modern computers and non-linear finite element analysis techniques now allow this behaviour to be assessed.

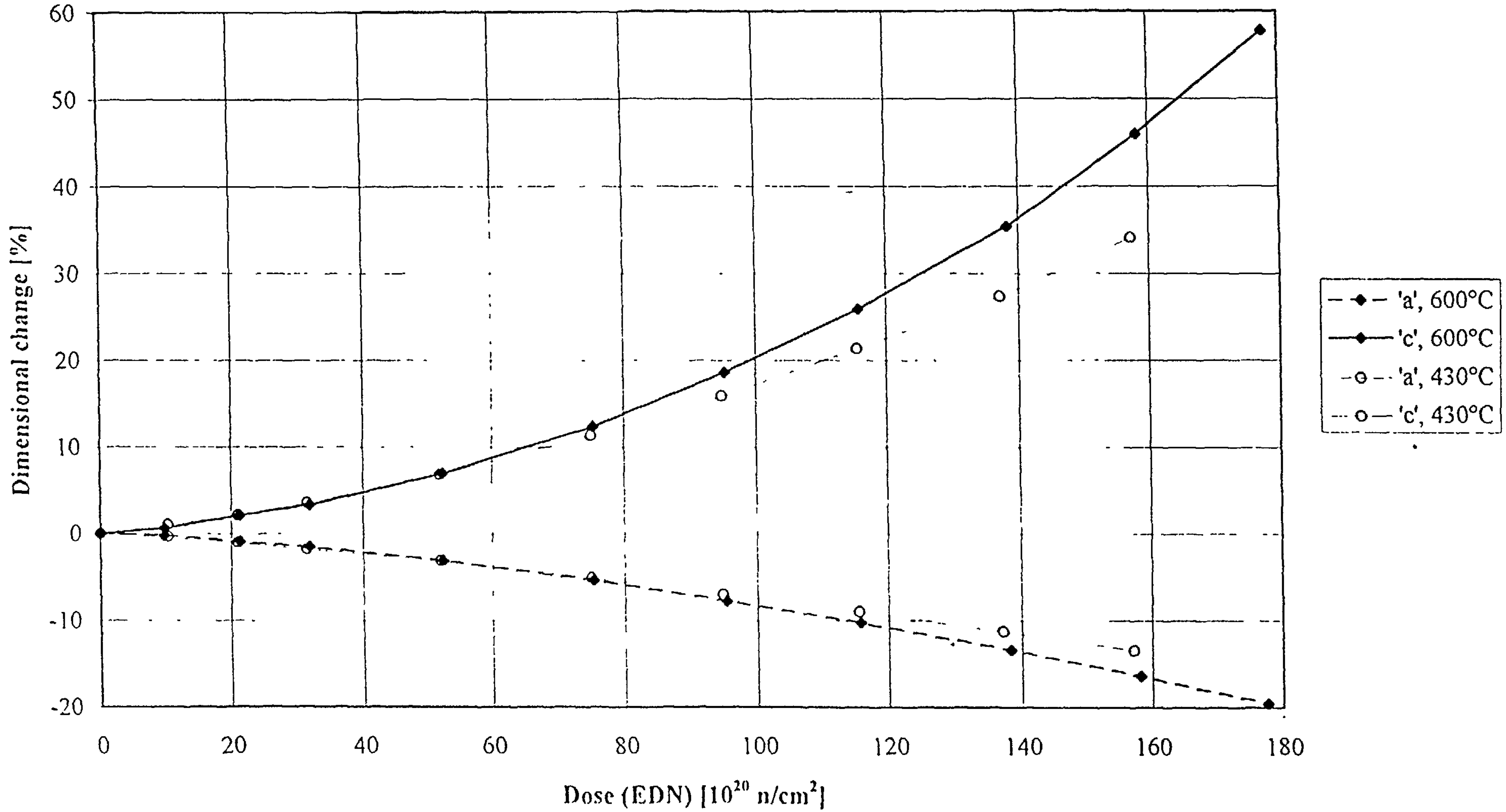

Figure 1 Dimensional changes in highly orientated pyrolytic graphite irradiated at 430°C and 600°C

An early attempt at modelling this behaviour [Marsden, 1998] using very elementary models of graphite crystals, joined together to form a polycrystalline material, clearly demonstrated that "turnaround" could be modelled using finite element methods. As this was the first time that "turnaround" had been modelled in polycrystalline graphite the authors decided to construct a more mathematical rigorous graphite crystal model and polycrystalline model of graphite irradiation ageing.

To test out this new model it was decided to apply it to irradiated graphite data in which the microstructure of various graphites had been changed by radiolytic oxidation [Kelly, 1972]. These experiments appeared to demonstrate that radiolytic oxidation delayed "turnaround" and increased shrinkage. However, all attempts to explain this behaviour in terms of increased porosity and accommodation have previously failed. If [Jenkins, 1964] was right and the change in crystal and polycrystalline compliance were important,.this could be demonstrated using the new models.

This paper describes these new models and their application to modelling dimensional change in graphite and the effect radiolytic oxidation has.

Graphite Crystal Behaviour

Perfect graphite crystals have two main axes 'c' and 'a' and the two properties of interest in this paper for the two principal directions are considerably different as shown below:

Property	'c'	'a'
Modulus (GPa)	36.5	1060
Coefficient of Thermal Expansion (20-120°C) ($K^{-1} \times 10^{-6}$)	26.5	-1.5

The temperature and irradiation behaviour of these properties and the crystal dimensional changes are described below:

Modulus

There is relatively little detailed information on the changes in the crystal modulus and compliance with irradiation. However what is known is that irradiation greatly increases the shear modulus between the basal planes and to a lesser extent reduces the 'c' axis compliance. There is some temperature dependence in modulus.

Thermal Expansion Coefficient (CTE)

It has been shown that at temperatures greater than 300°C the CTE in both the 'c' and 'a' direction is independent of irradiation dose. However the CTE is temperature dependent in both directions.

Dimensional Change

Dimensional change data for graphite crystals has been obtained in material test reactors over a wide range of doses and temperatures. However, some of this data is inconsistent and in some temperature ranges the data is limited. It has been established that dimensional change rates are much larger in graphite crystals irradiated below 300°C than above this temperature and that above ~600°C the dimensional change rate is dependent on crystal size. For this study dimensional change data for irradiation temperatures of 430°C and 600°C was used [Brocklehurst and Kelly, 1992].

The finite element graphite crystal model

This new model was developed, using the ABAQUS finite element program, to synthesise a volume of well-crystallised cracked material, surrounded by a volume of binder. The model was implemented in ABAQUS using a Fortran subroutine facility known as a UMAT that includes the anisotropic, irradiation dependent constitutive relationships for:

a) The crystal dimensional changes $[\Delta a/a]$, $[\Delta c/c]$
b) The crystal CTE (and changes with strain) α_a, α_c
c) The crystal elastic modulus E_a , E_c
d) The crystal Poisson's ratio ν_c
e) Crystal irradiation creep ε_c

The new model was studied in isolation in order to determine the effect of the binder boundary restraint upon:

a) The apparent crystallite dimensional change $[\Delta L_a/L_a]_{app}$ and $[\Delta L_c/L_c]_{app}$
b) The apparent crystallite CTE αL_a and αL_c

To examine the behaviour of a single graphite crystal, a finite element model of an idealised crystal based on work by [Sutton and Howard, 1962], 1.5 μm wide by 6 μm long, was created with three internal cracks 0.025 μm wide. This work took account of the small "Mrozowski" type cracks. The surrounding binder was represented by a 0.75 μm border around the whole crystal, see Fig. 2.

The ABAQUS graphite material property UMAT was utilised, using crystal data for irradiation at 450°C. The coefficient of thermal expansion for the crystal was calculated using the mean instantaneous value of the crystal CTE between the room temperature and the irradiation temperature. For the purposes of this initial analysis, it was decided to ignore the effect of creep.

The modulus in the basal plane 'a' is ~30 times greater than that for the 'c' direction, which has implications for the stresses and deformations within the crystallite. Large stresses are generated if the crystal is restrained in the 'a' direction, due to increases in dimensional change and thermal strains The implications of this Poisson's effect are the subject of ongoing studies.

By cycling the graphite temperature between 20°C and 450°C at several intervals of dose, it is possible to attain the apparent dimensional change and change in CTE with irradiation.

To represent the restraint imposed on the crystal by the surrounding crystals and binder, a volume of material was evenly placed around the outer edges of the crystal.

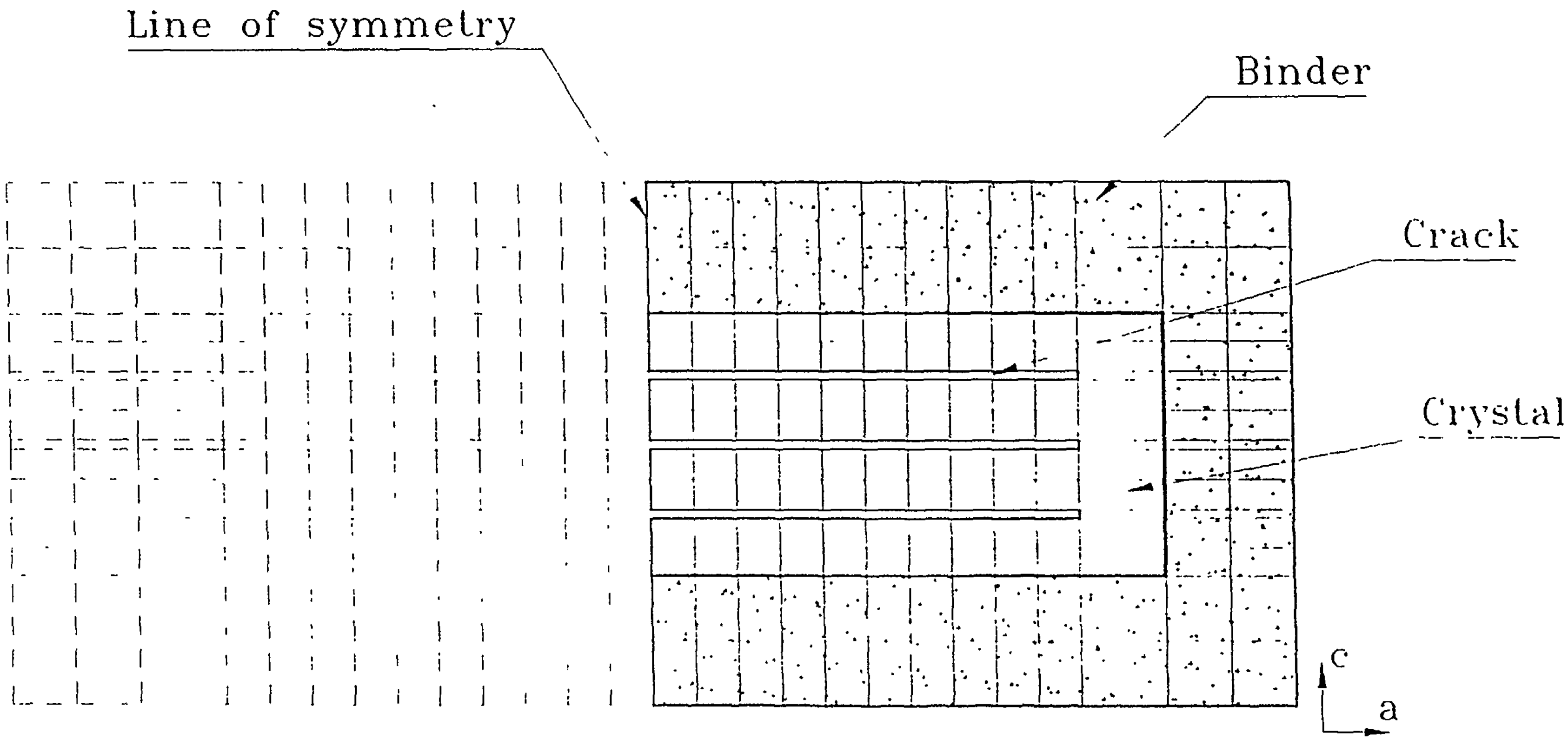

Figure 2 The finite element model of the single graphite crystal and binder

This "binder" volume was given a modulus, that would allow it to simulate the effect of the surroundings within the polycrystalline material. Without this, the crystal would change according to the aspect ratio defined by the respective dimensional change and CTE for each direction. For the initial analysis, the binder was given a modulus that would represent a typical binder with 0% weight loss (10 GPa). The modulus was then adjusted in order to reflect the weight loss, using an empirical relationship between weight loss (x) and modulus; $E = e^{-3.6x}$. Therefore, for a weight loss of 30% the modulus of the binder was 3.4 GPa. The analyses were run to a dose of 80 x 10^{20} n/cm^2 EDN at 450°C, returning to 20°C at various intervals. The resultant displacements for the varying parameters are given in Fig. 3 and Fig. 4. The change in the 'a' direction was the horizontal displacement of the centre of the crystal's right hand edge, and the change in the 'c' direction was the vertical displacement of the centre of the crystal's top edge.

The displacement at the end of the crystal is representative of the graphite crystal dimensional changes and crystal dimensional changes due to the CTE, as it is essentially solid material at this point. However, in the centre of the crystal there is a delay in the onset of dimensional change due to the absorption of growth by the cracks. The delayed change in the rate of displacement in the 'c' direction highlights the points at which the cracks have closed. Comparison of dimensional changes between varying weight losses (see Fig. 4) indicates that an increase in weight loss, increases the dimensional change. This was to be expected as an increase in weight loss reduces the binder's modulus, thus making the restraining material more compliant.

The apparent mean CTE for the 'c' direction at 20°C was obtained using the vertical displacement of the centre of the crystal (see Fig. 5). For the crystal surrounded by a binder with 0% weight loss, the apparent CTE is approximately zero at low doses, with a steady increase starting at 20 x 10^{20} n/cm^2 EDN, reaching a maximum positive value of ~21 x 10^{-6} K^{-1} at 60 x 10^{20} n/cm^2 EDN. The value then remains approximately constant. When the binder has 30% weight loss, the apparent CTE in the 'c' direction has the same basic shape as the 0% weight loss, except the initial increase begins at 30 x 10^{20} n/cm^2 EDN, peaking at ~24.5 x 10^{-6} K^{-1} at 65 x 10^{20} n/cm^2 EDN.

The apparent mean CTE for the 'a' direction was based upon the horizontal displacement of the end of the crystal (see Fig. 6). With respect to the binder with 0% weight loss, the apparent CTE starts at a positive value (0.43 x 10^{-6} K^{-1}) and then gradually increases to 0.71 x 10^{-6} K^{-1} at 40 x 10^{20} n/cm^2 EDN. The apparent CTE then decreases to -0.09 x 10^{-6} K^{-1} at 60 x 10^{20} n/cm^2 EDN, remains constant until 80 x 10^{20} n/cm^2 EDN, where there is another decrease to -0.17 x 10^{-6} K^{-1}. When the binder has 30% weight loss, the apparent CTE in the 'a' direction is of the same basic shape with the peaks and troughs occurring at approximately the same dosage, but being of a different amplitude. The initial value is 0.34 x 10^{-6} K^{-1}, rising to a maximum of 0.48 x 10^{-6} K^{-1} at 40 x 10^{20} n/cm^2 EDN. The value then drops to -0.25 x 10^{-6} K^{-1} at 65 x 10^{20} n/cm^2 EDN, from which it remains constant.

From this analysis it is obvious that the crystal and binder modulus are an important factor in the way crystallites will interact in an overall polycrystalline structure, especially when the latter undergoes some degree of weight loss. The large difference

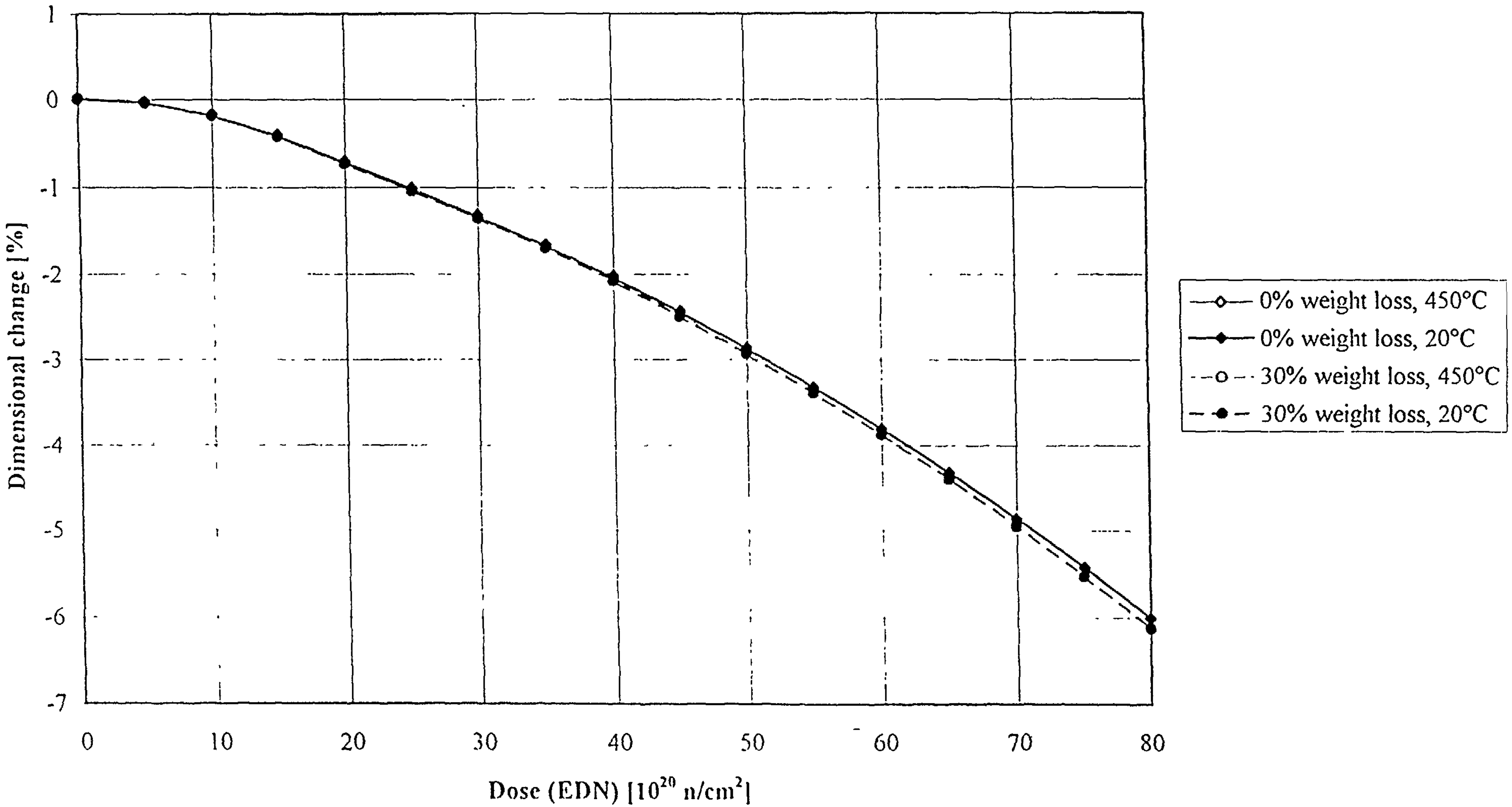

Figure 3 Displacement of graphite crystal in the horizontal ('a') direction

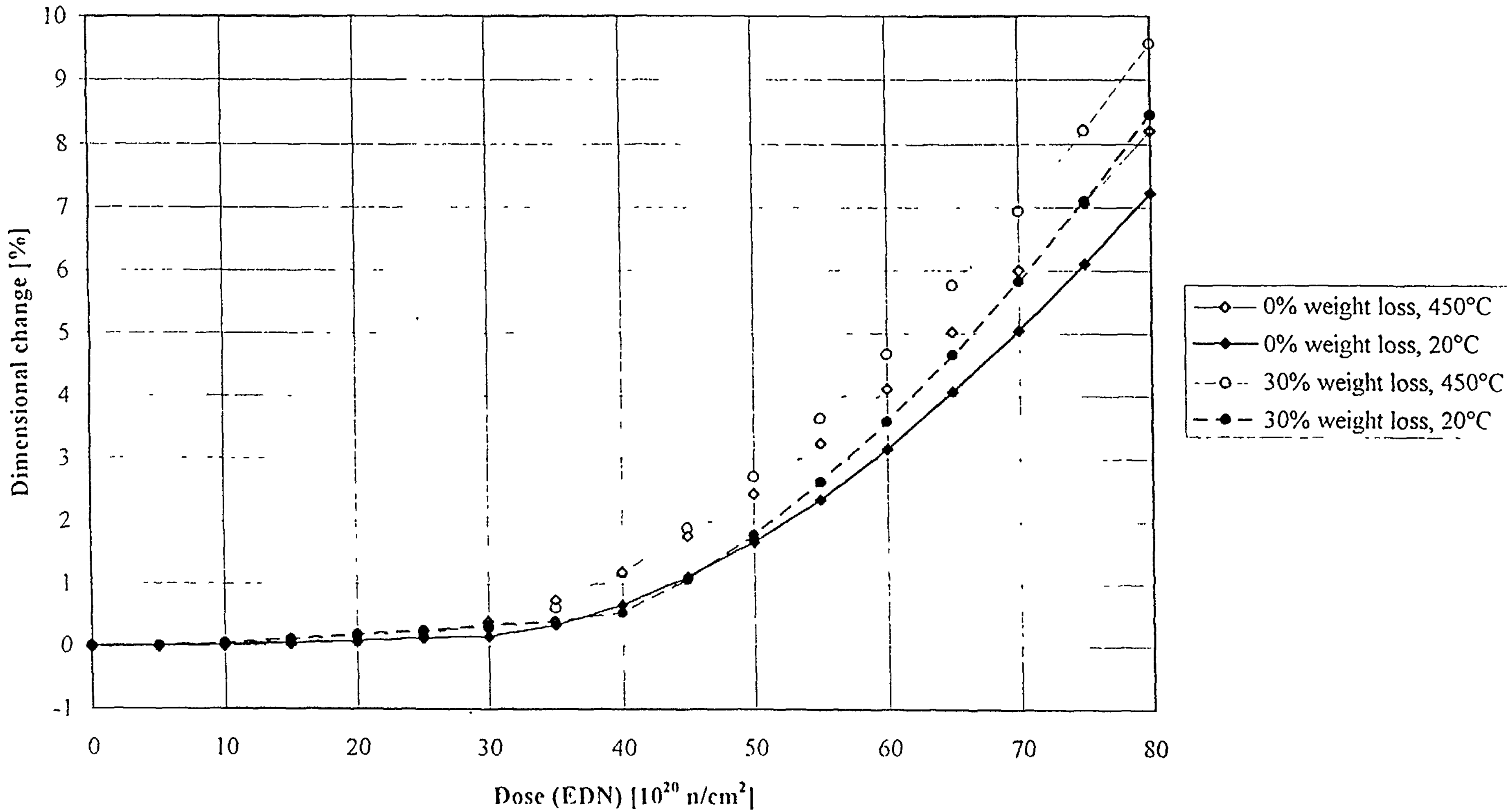

Figure 4 Displacement of graphite crystal in the vertical ('c') direction

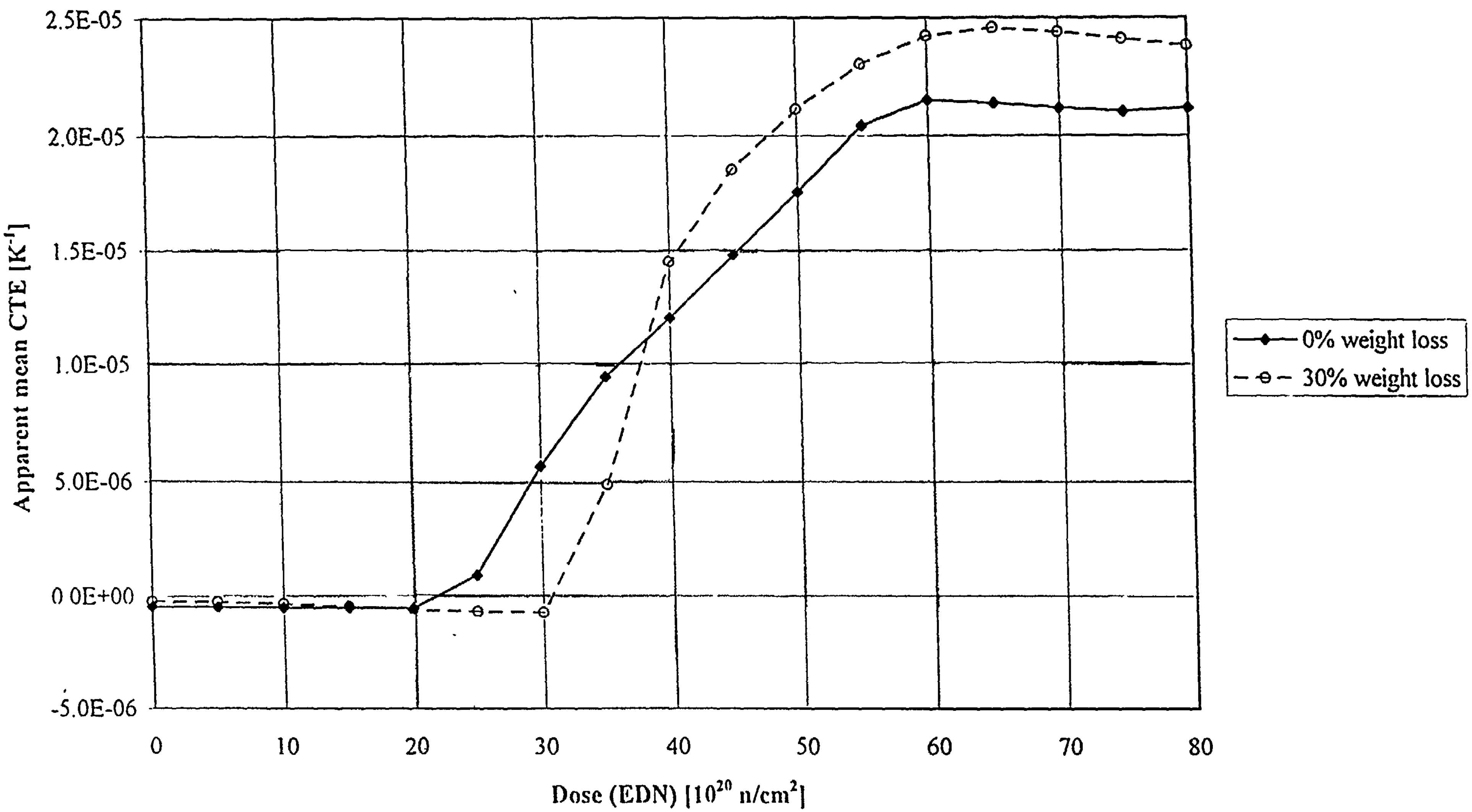

Figure 5 Apparent mean CTE in the vertical ('c') direction

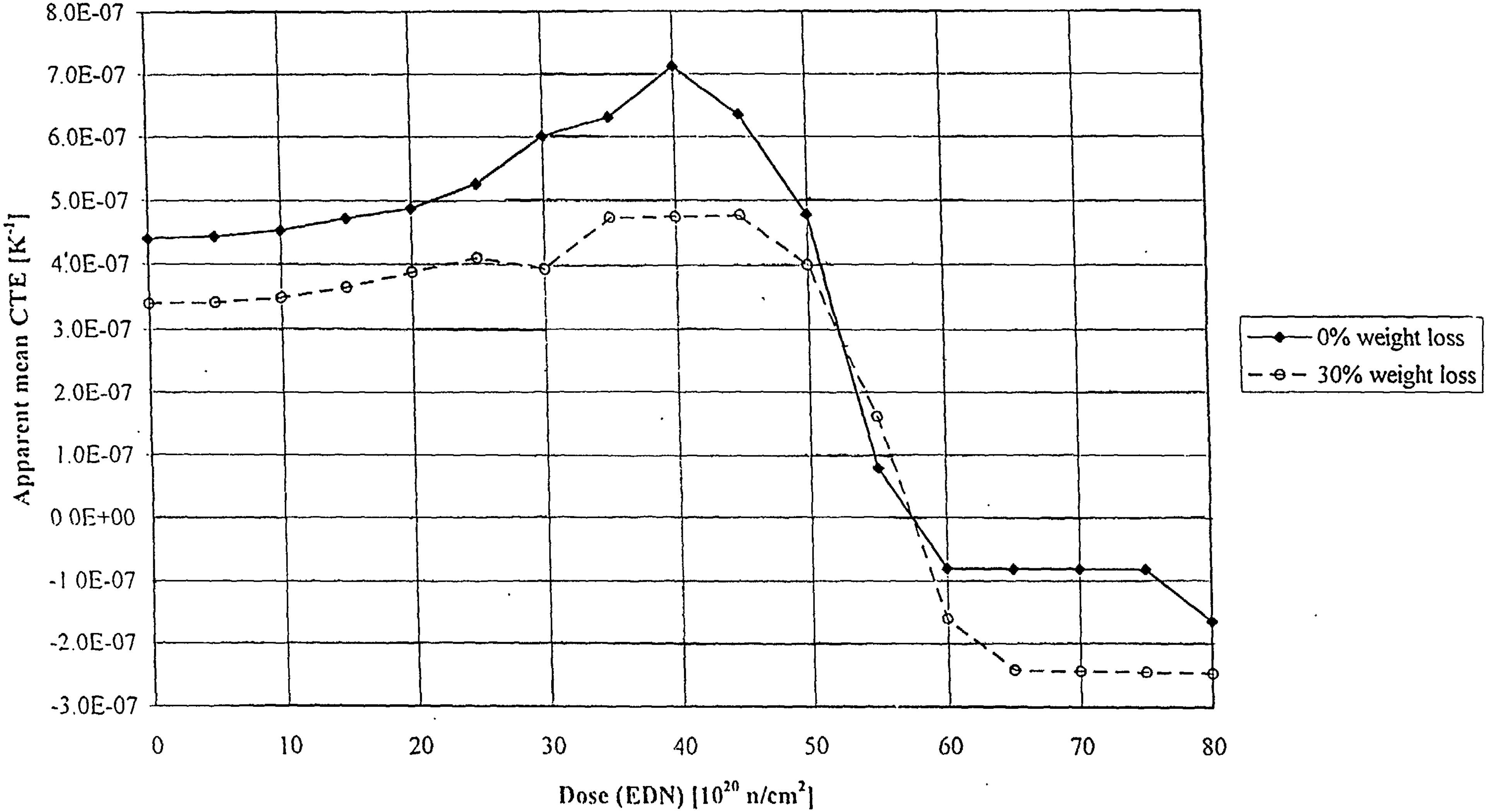

Figure 6 Apparent mean CTE in the horizontal ('a') direction

in crystal modulus between the 'a' and 'c' directions have significant implications for the creation of stress and displacements due to the effect of Poisson's ratio. These stresses and displacements may lead to the creation of basal plane cracks with irradiation and thermal cycling.

The polycrystalline graphite model

The data obtained from the analysis of the single crystal model was then used in a polycrystalline model. This model comprises of a rectangle with two axes of symmetry, made up from individual elements, see Fig. 7. Each individual element may represent:

a) Crystalline graphite material containing cracks, surrounded by binder material, giving an apparent CTE and dimensional change
b) Holes to represent inter-crystalline porosity and gas evolution pores

To simulate the possibility of varying orientation within the structure, it was possible to orient the individual crystallites (the 'c' crystallite axis of a non-orientated crystal corresponds to the vertical axis of the mesh). In this study only 0° and 90° orientation was considered.

The distribution of the graphitic materials and the porosity can be generated randomly, although there is the possibility to generate groups of similar crystal, porosity or other binder type material.

Two assessments were conducted; the first having 0% weight loss and 60% - 40% orientation (0° - 90°), and the second having 30% weight loss and 60% - 40% orientation (0° - 90°). The apparent displacement plots for these are given in Fig. 8 and Fig. 9. The apparent dimensional changes in the 'a' direction exhibit "turnaround" for both weight losses, with the 30% weight loss having greater displacements and a delayed "turnaround" with respect to the 0% weight loss. In the 'c' direction, the apparent dimensional changes for the 0% weight loss initially shrink at the lower doses (up to ~30 x 10^{20} n/cm^2 EDN), and then grow at the remaining doses. The 30% weight loss case has similar trends with initially no movement until ~30 x 10^{20} n/cm^2 EDN, at which point it begins to grow, reaching a greater displacement than the 0% weight loss model. This is behaviour that had been observed experimentally in virgin and pre-oxidised polycrystalline anisotropic needle coke graphite specimens irradiated in the Dounreay Fast Reactor, see Fig. 10, giving some confidence in the new methodology.

The apparent mean CTE in the 'a' and 'c' directions (see Fig. 11 and Fig. 12 respectively) highlights a delayed response in the increase of CTE for the 30% weight loss (~30 x 10^{20} n/cm^2 EDN), when compared to the 0% weight loss (~20 x 10^{20} n/cm^2 EDN). In the 'a' direction the apparent CTE of both cases are initially the same, there is then an increase in the 0% and then 30% model. Between 40 and 50 x 10^{20} n/cm^2 EDN the 30% weight loss model is approximately equivalent to that of the 0% weight loss model, and then slightly less at higher doses. In the 'c' direction, a similar process occurs at the lower doses and then after 40 x 10^{20} n/cm^2 EDN the 30% weight loss case has a greater apparent CTE. In a true polycrystalline graphite there would also be a component of CTE associated with the binder phase. If this binder

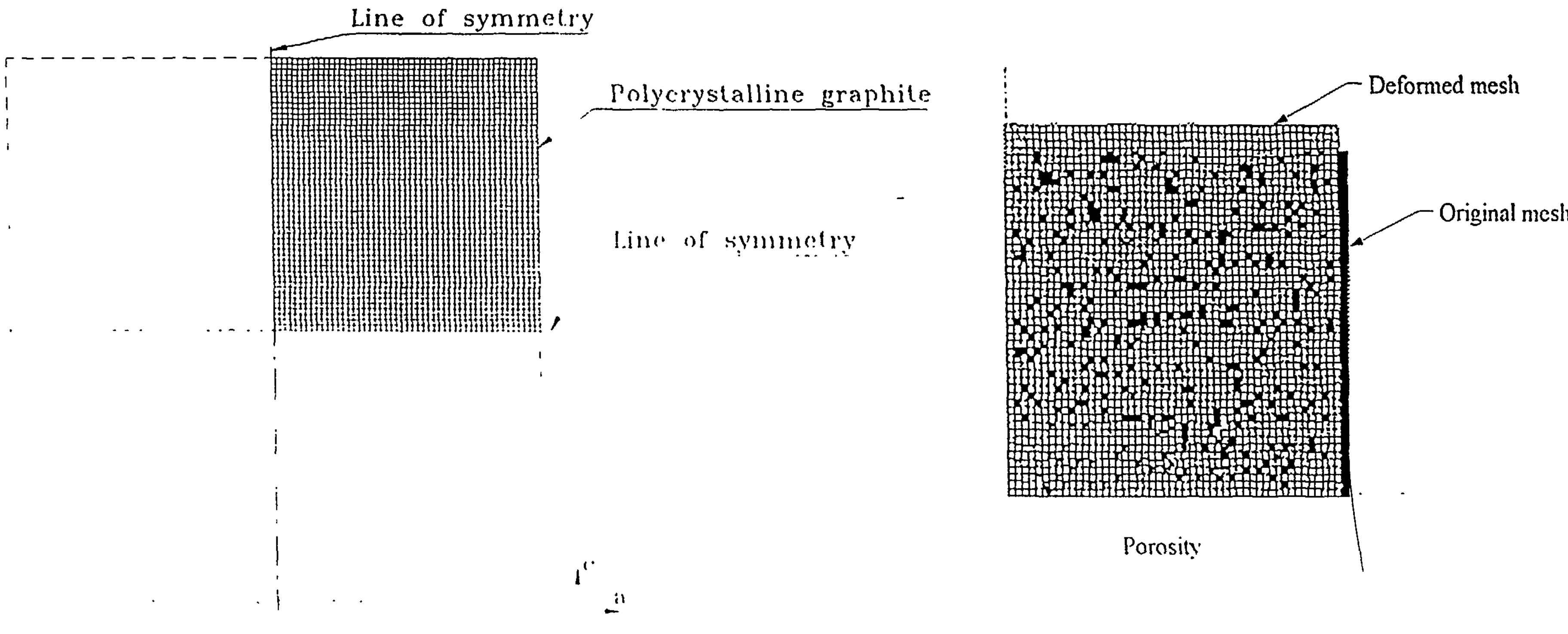

Figure 7 The finite element model of the polycrystalline graphite

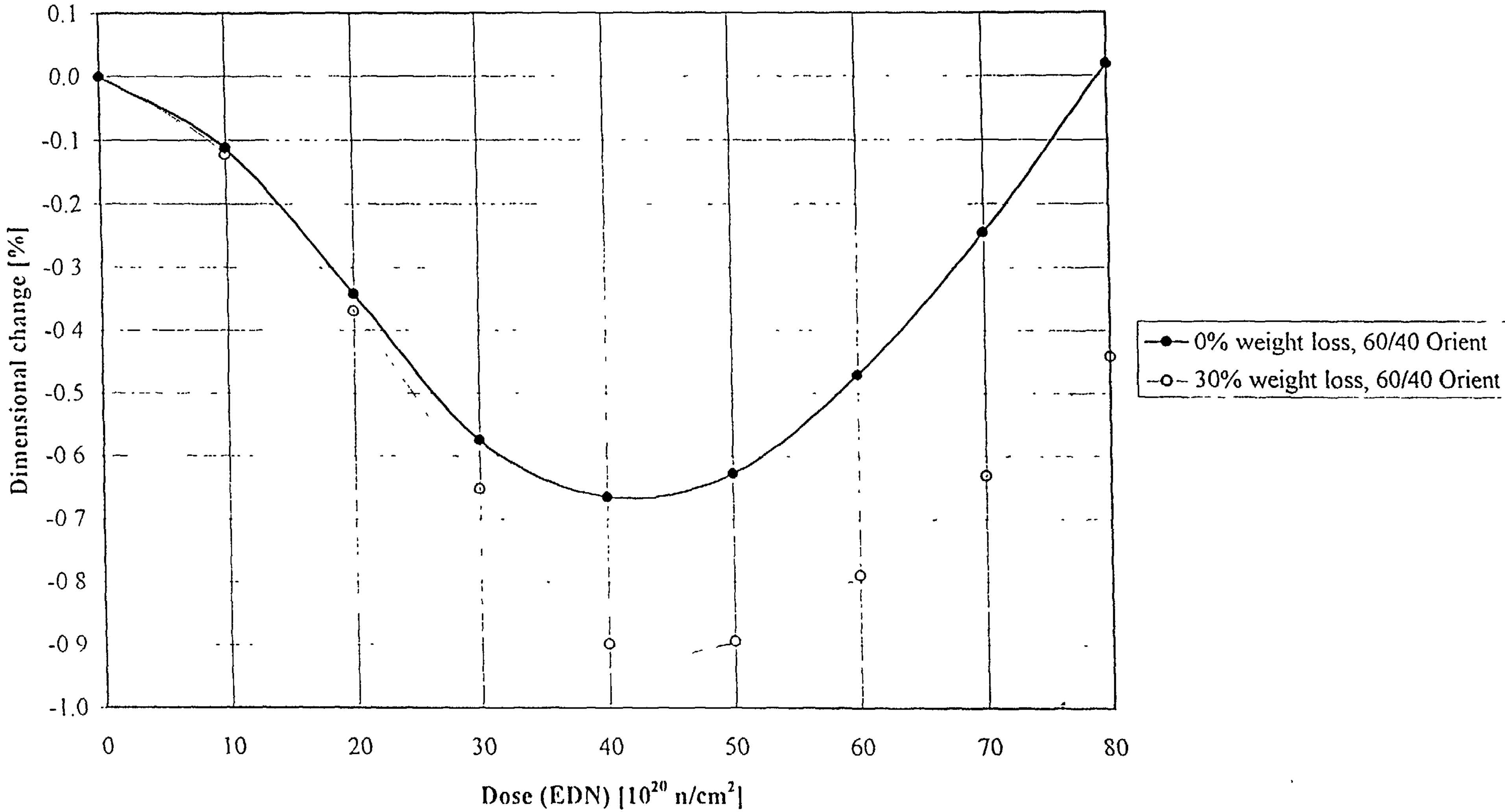

Figure 8 Apparent horizontal displacement of a polycrystalline graphite block (mostly 'a')

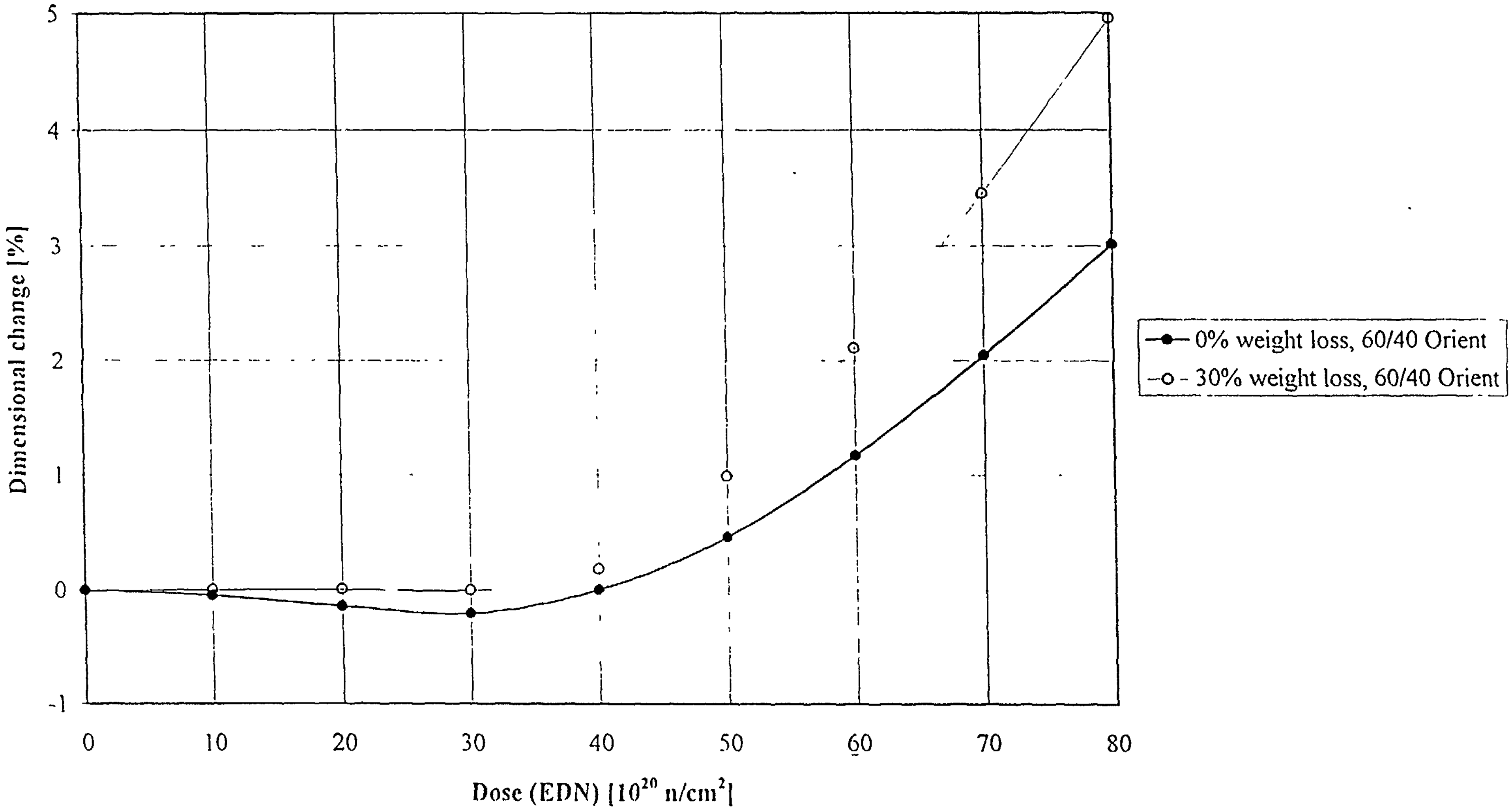

Figure 9 Apparent vertical displacement of a polycrystalline graphite block (mostly 'c')

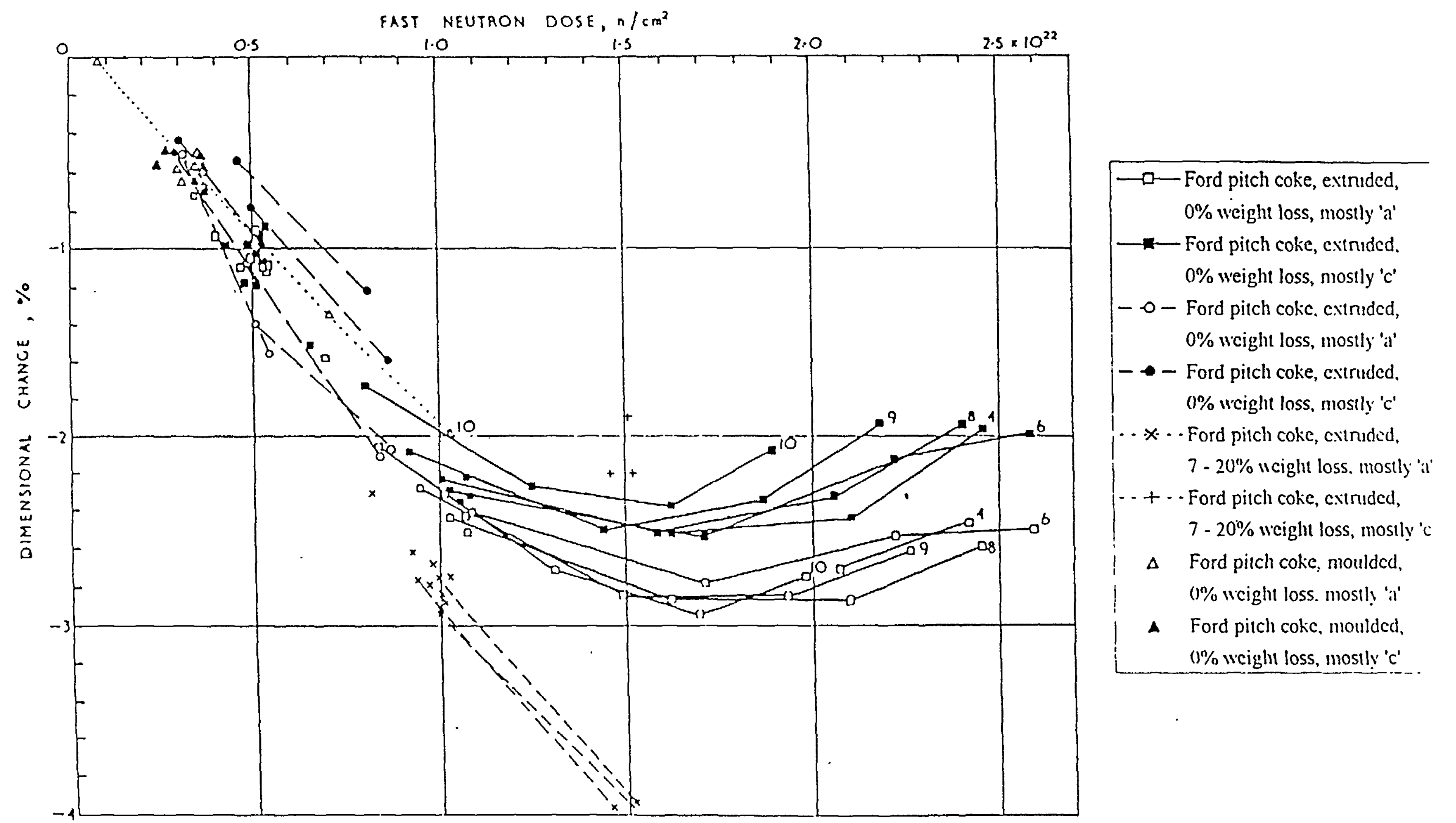

Figure 10 Dimensional changes in BAEL pitch coke graphite irradiated in DFR

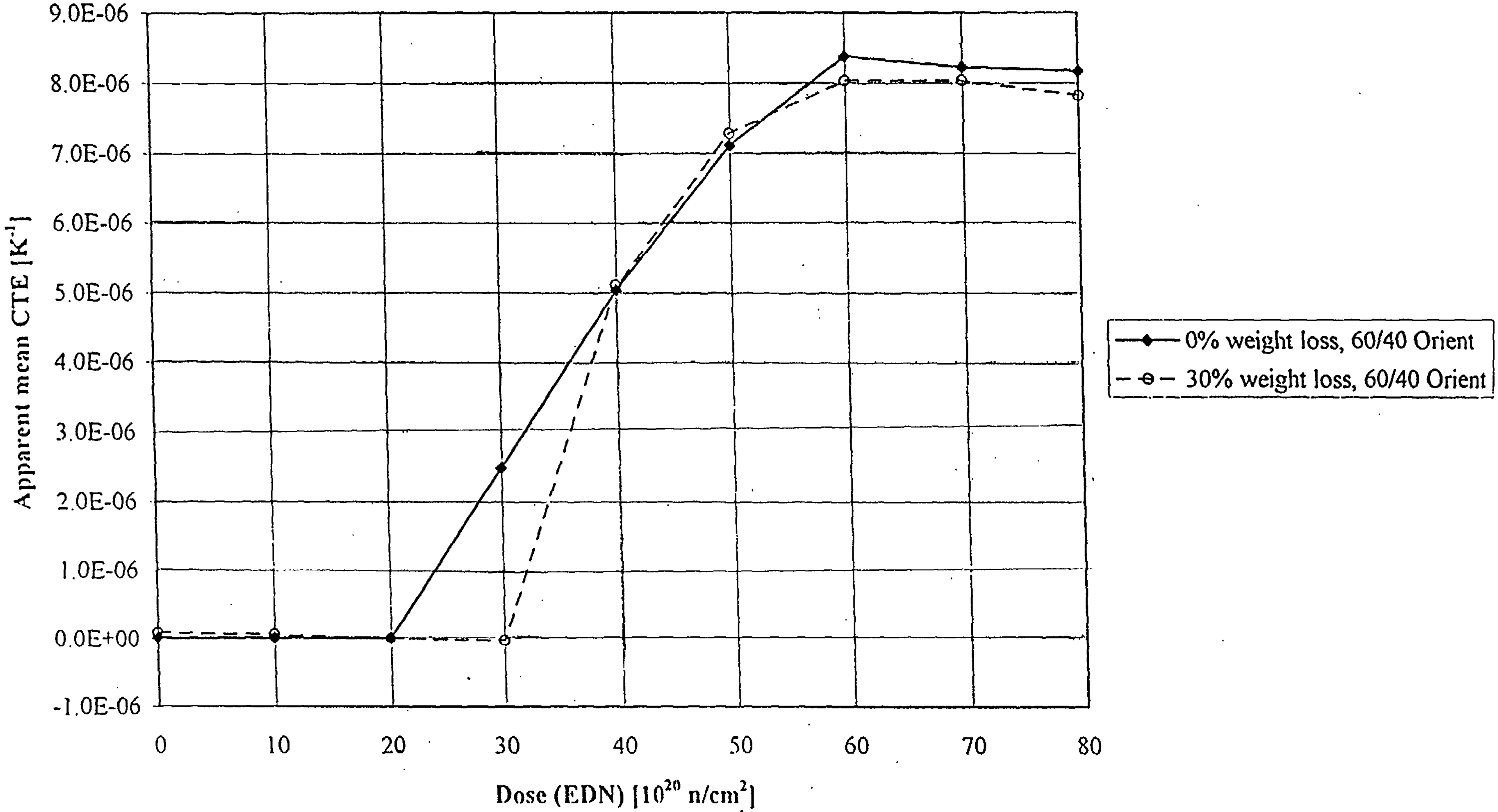

Figure 11 Apparent horizontal mean CTE of a polycrystalline graphite block (mostly 'a')

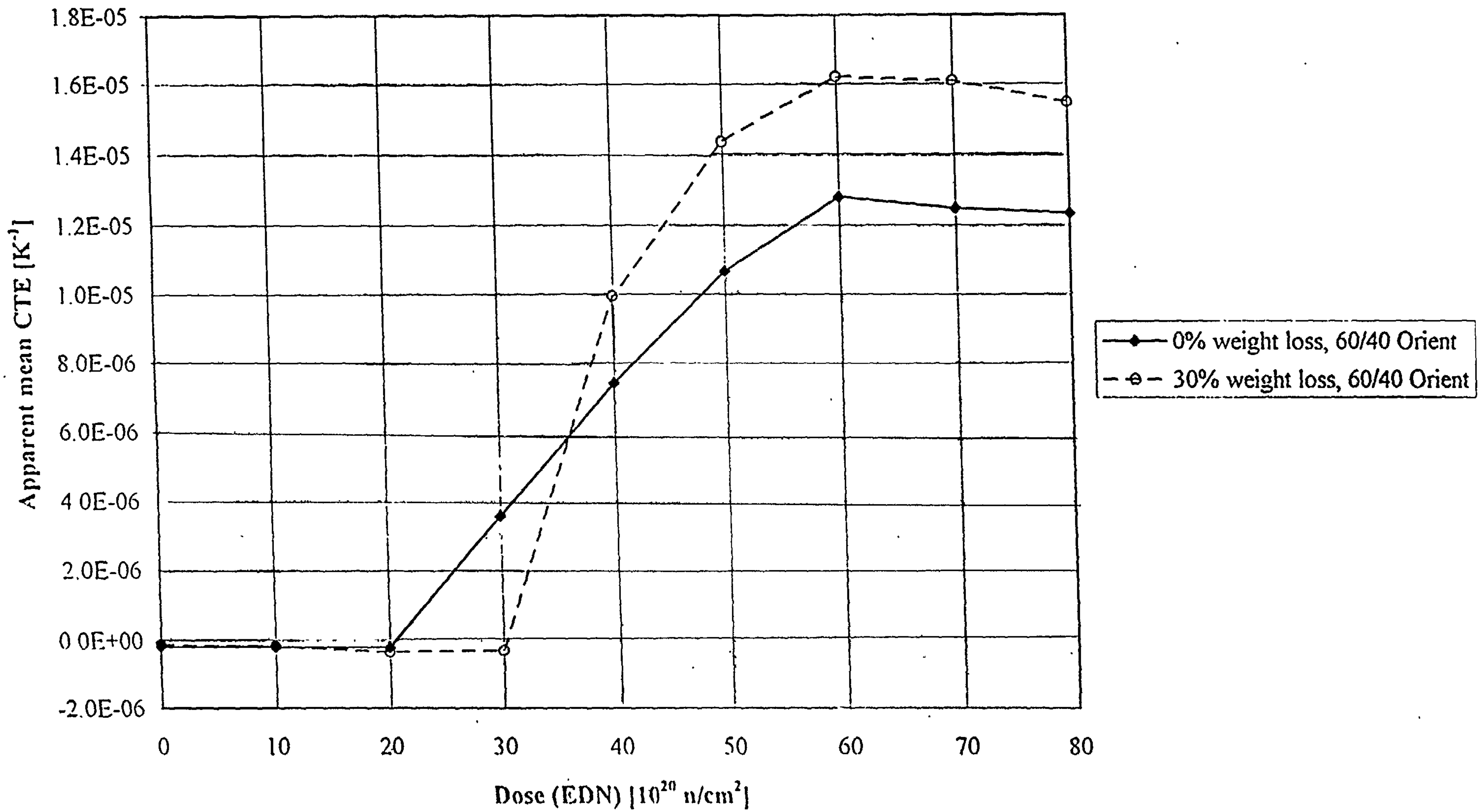

Figure 12 Apparent vertical mean CTE of a polycrystalline graphite block (mostly 'c')

phase CTE were included similar changes to CTE as predicted in Pile Grade A graphite by [Brocklehurst and Kelly, 1993], would be expected, again giving some confidence to the method.

The results indicate that "turnaround" can be explained by crystal growth, accommodation and compliance without the necessity of simulating the generation of new cracks. The phenomenon of delay in "turnaround" may be explained by the decrease of compliance within the structure, caused by weight loss.

Discussion

Previously, the models considered graphite as a single-phase, well crystalline graphitic material. These were only applicable to relatively low irradiation doses, as at higher doses binder filler interaction becomes more predominant. Therefore, it was suggested that a two-phase analytical model would simulate the graphite more accurately.

Initially the polycrystalline graphite's dimensional change is dominated by the 'a' direction shrinkage, as the 'c' direction growth is absorbed by the inter-crystalline accommodation.

At "turnaround" the 'c' direction expansion becomes the dominant driving force, and the polycrystalline graphite begins to expand in both directions. This growth will continue until the final failure of the structure through inter crystalline cracking.

The delay in "turnaround" with weight loss has been demonstrated to be primarily a function of the increase in compliance due to weight loss. Earlier work [Marsden, 1998] has shown that changing the modulus by adding porosity in the polycrystalline model did not significantly affect the change in dimension behaviour. The reason for this is that it did not account for the change in the individual crystal behaviour due the change in the global compliance.

After "turnaround" large stresses due to crystal growth will lead to inter crystalline cracks and deterioration of the properties such as thermal and electrical conductivity, modulus, and failure strength [Shtrombakh, 1995].

It is also true that the overall modulus will change with irradiation due to pinning of the dislocations in the basal plane. However this will apply to both virgin and non-oxidised material. In addition irradiation creep will also have an effect on "turnaround" and property changes as this will affect the inter crystalline stresses. Both of these phenomena are currently under further investigation (the model has the capability of simulating both these types of behaviour)

The model is also capable of assessing three-dimensional needle coke particles and spheroidal graphite particles, see Fig. 13.

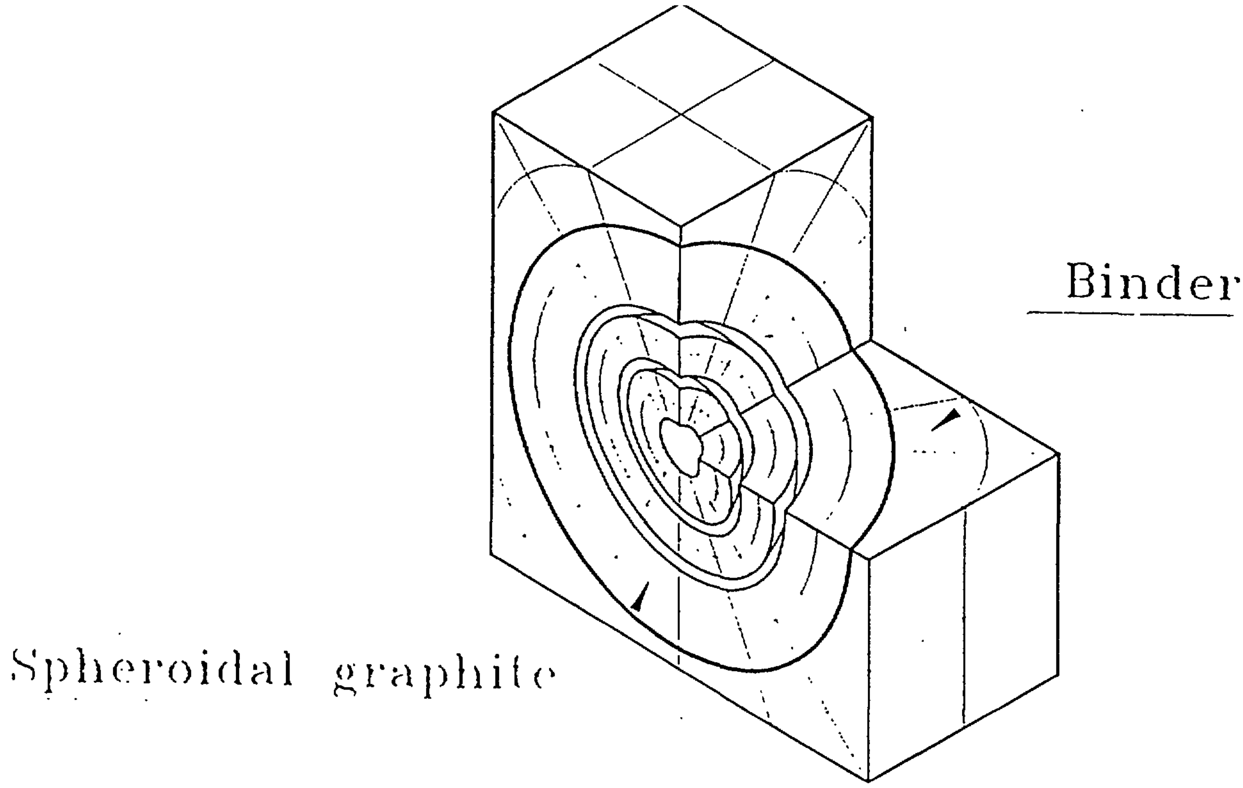

Figure 13 Finite element model of spheroidal graphite

Conclusions

1 A two-phase finite element model of a single graphite crystal containing Mrozowski type cracks, surrounding by a binder region, has been developed. This model can account for crystallite behaviour including apparent changes in CTE and dimensional changes due to fast neutron irradiation.

2 The single crystallite model has been included in a larger polycrystalline finite element model, which has demonstrated the mechanism of shrinkage and "turnaround".

3 The model has been used to explain experimental results of virgin and pre-oxidised graphites.

4 Dimensional changes in graphite are initially governed by:
   a) Crystal dimensional change in the 'a' and 'c' directions, where 'a' is predominant
   b) Accommodation within inter-crystalline cracking
   c) Crystal orientation
   d) The restraint of the crystallite within the polycrystalline graphite
   e) The crystallite compliance
   f) The polycrystalline compliance, which can be affected by the weight loss

5 Dimensional changes at and after "turnaround" are governed by:
   a) Crystal dimensional change in the 'a' and 'c' directions, where 'c' becomes predominant

b) The loss of inter-crystalline porosity
c) Crystal orientation
d) The crystallite compliance
e) The polycrystalline compliance, which can be affected by the weight loss

6 Generation of inter-crystalline stresses at and after "turnaround" may also be governed by the generation of new cracks between crystals and particles. This will lead to a deterioration of thermal conductivity, modulus and failure strength.

7 The change in polycrystalline compliance due to radiolytic weight loss is a major contributing factor to the delay in "turnaround" and the increase in dimensional change.

8 The model can be further enhanced to account for other polycrystalline behaviours such as dimensional and CTE changes for three dimensional needle and spheroid geometries, caused by fast neutron damage and radiolytic oxidation.

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A STUDY ON COST REDUCTION EFFECT DUE TO CHANGES OF SAFETY GRADE OF HTGR

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Abstract

The safety potentiality of a light water reactor (LWR), a fast breeder reactor (FBR) and a high temperature gas cooled reactor (HTGR) was compared from the aspect of the reactor-shutdown, decay heat removal and containment of radioactive material at the accident, The HTGR has the higher safety potential because it doesn't require the instant reactor shutdown, active decay heat removal with coolant and containment vessel.

The HTGR has the possibility to apply the different safety criteria. The cost reduction effect has been studied due to changes of safety grade.

1. INTRODUCTION

The significance of nuclear energy is increasing because it has the potential to solve a number of problems concerned with shortages of fuels and environmental degradation in 21st century. We believe the HTGR is most promising candidate to meet various needs which are not only electrical generation but also utilizing nuclear thermal energy because of safety potential and high temperature output.

MHI has been conducting the research and development to investigate the feasibility of HTGR commercialization in future.

In Japan, Key point in having the HTGR commercialization is economy. MHI has been studying cost reduction method of HTGR to be competitive with the commercialized LWR such as the large sized HTGR, high performance HTGR and small sized HTGR. These systems are shown in Fig-1～3.

The other cost reduction method due to changes of safety grade has been studied according to the small sized HTGR. In order to show the possibility that the HTGR can apply the different safety criteria, the safety potentiality of LWR, FBR and HTGR was compared and the safety classification of HTGR was studied.

1. Reactor	Size
Thermal power	1150MWt
Reactor vessel OD	9.05m
Active core configuration	Annular
Effective active core diameter	3.62m/6.17m
Effective active core height	9.52m
Average power density	6.3MW/m³
2. Primary cooling system	
Coolant outlet/inlet temperature	850℃/490℃
Coolant inlet pressure	72ata
Coolant flow rate	2214ton/h
3. Fuel	
Type	UCO
U enrichment	19.8%
Average fuel burnup	131000 MWD/MTU

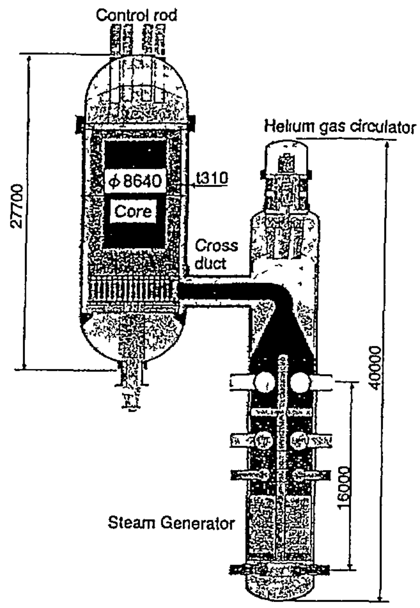

Fig-1 Specification & Concept of Large-sized HTGR

Thermal power	500MWt
Coolant inlet temp.	400℃
Coolant outlet temp.	700℃
Core O.D	281.1cm
Core I.D	100.3cm
Core height	830cm
Reflecter O.D	240cm
Ave. fuel burnup	*15000 MWd/t*
Ave. power density	*10.8 W/cc*
Fuel elements per column	48
Inner reflecter number	7
Fuel particle type	*Multi hole*
Fuel particle O.D	0.094cm
Packing ratio of fuel particle	30%
Operating cycle	11 months
Refueling method	2 butch

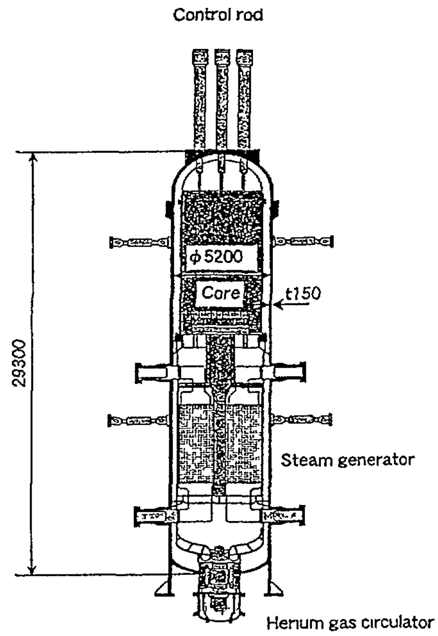

Fig-2 Specification & Concept of High performance HTGR

Thermal power	230MWt
Coolant inlet temp.	550℃
Coolant outlet temp.	850℃
Coolant inlet pressure	5.5MPa
Core O.D	361.6cm
Core I.D	165.2cm
Core height	707.6cm
Ave. fuel burn up	110GWd/t
Ave. power density	4W/cc
Fuel particle type	Pin-in-block
Operating cycle	950days
Refueling method	2butch

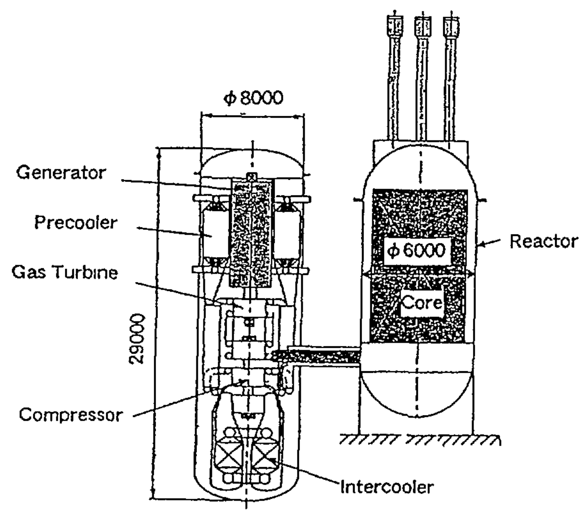

Fig-3 Specification & Concept of Small-sized HTGR

2. SAFETY POTENTIAL

2.1 Safety Design

The engineering safety feature is designed in the nuclear plant to perform the following function according to the philosophy of defense in depth.

(1) Reactor shutdown function which makes the nuclear reaction subcritical.

(2) Reactor cooling function which removes residual heat from the core.

(3) Radioactive material containment function which prevents from releasing radioactive material to environment.

2.2 Comparison of safety potential of LWR, FBR and HTGR

The safety potential of LWR, FBR and HTGR was compared from the aspect of safety design. Comparisons of safety function, accidental event frequency and safety characteristic are shown in Fig-4～6.

(1) LWR

The failure of reactor coolant pressure boundary (RCPB) could cause a loss of reactor coolant and core damage. The prevention and control of core damage is achieved by the reactor shutdown and core cooling function.

<u>Reactor shutdown function</u>

- Reactor shutdown by control rod
- Boron injection

<u>Core cooling function</u>

- Coolant injection to core by ECCS
- Recirculation by ECCS
- Isolation of leak point
- Heat removal from secondary system

Above functions are achieved by active equipments. The accident is kept within design base event (DBE). A loss of active equipment function could cause the core damage and reactor vessel failure. The accident is beyond DBE. Radioactive material is released to the containment vessel (C/V). To keep the containment function is important to prevent from releasing radioactive material to environment.

<u>Containment function</u>

- C/V heat removal
- Prevention of direct heat and contact to C/V
- Isolation of leak point

Above function is achieved by active equipment. A loss of active equipment could cause C/V failure and release of radioactive materials.

(2) FBR

In the case of FBR, the prevention and control of core damage is achieved by the passive and active system.

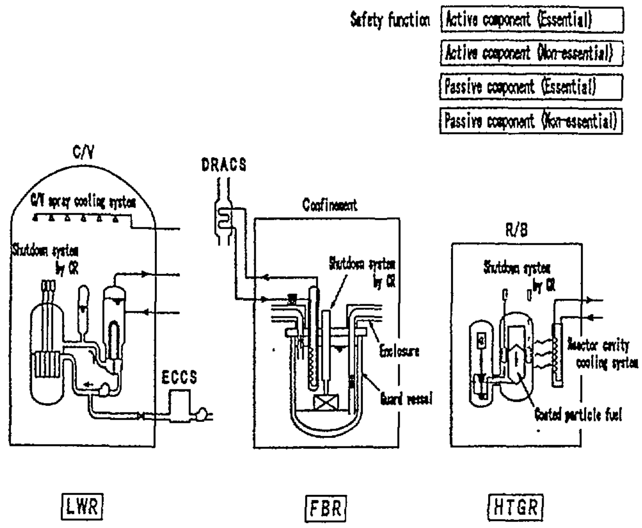

Fig-4 Comparison of safety function of LWR,FBR,HTGR

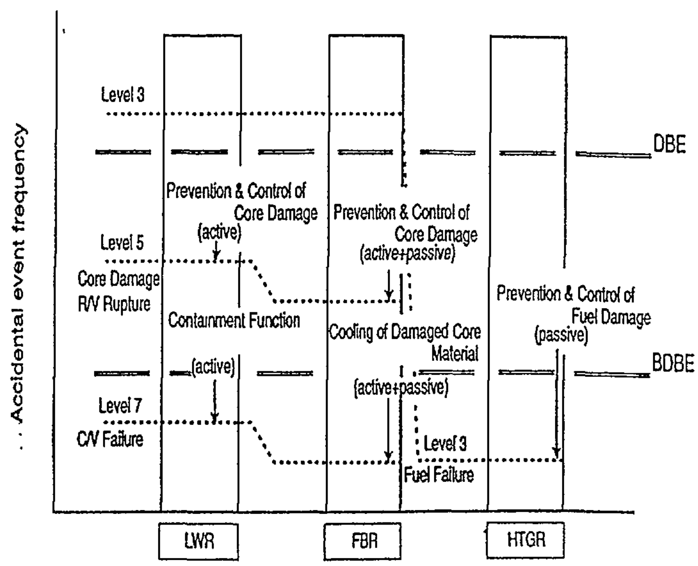

Fig-5 Comparison of Accidental event frequency of LWR,FBR,HTC

Reactor shutdown function

- Reactor shutdown by control rod (active)
- Self-actuated shutdown system (passive)

Core cooling function

- Guard Vessel (passive)
- Auxiliary cooling system (ACS) (active and passive)

A loss of reactor shutdown function could cause the core damage. But core damaged materials are maintained in the guard vessel and cooled by ACS which is natural draft cooling. The guard vessel is very important for the prevention of radioactive material release.

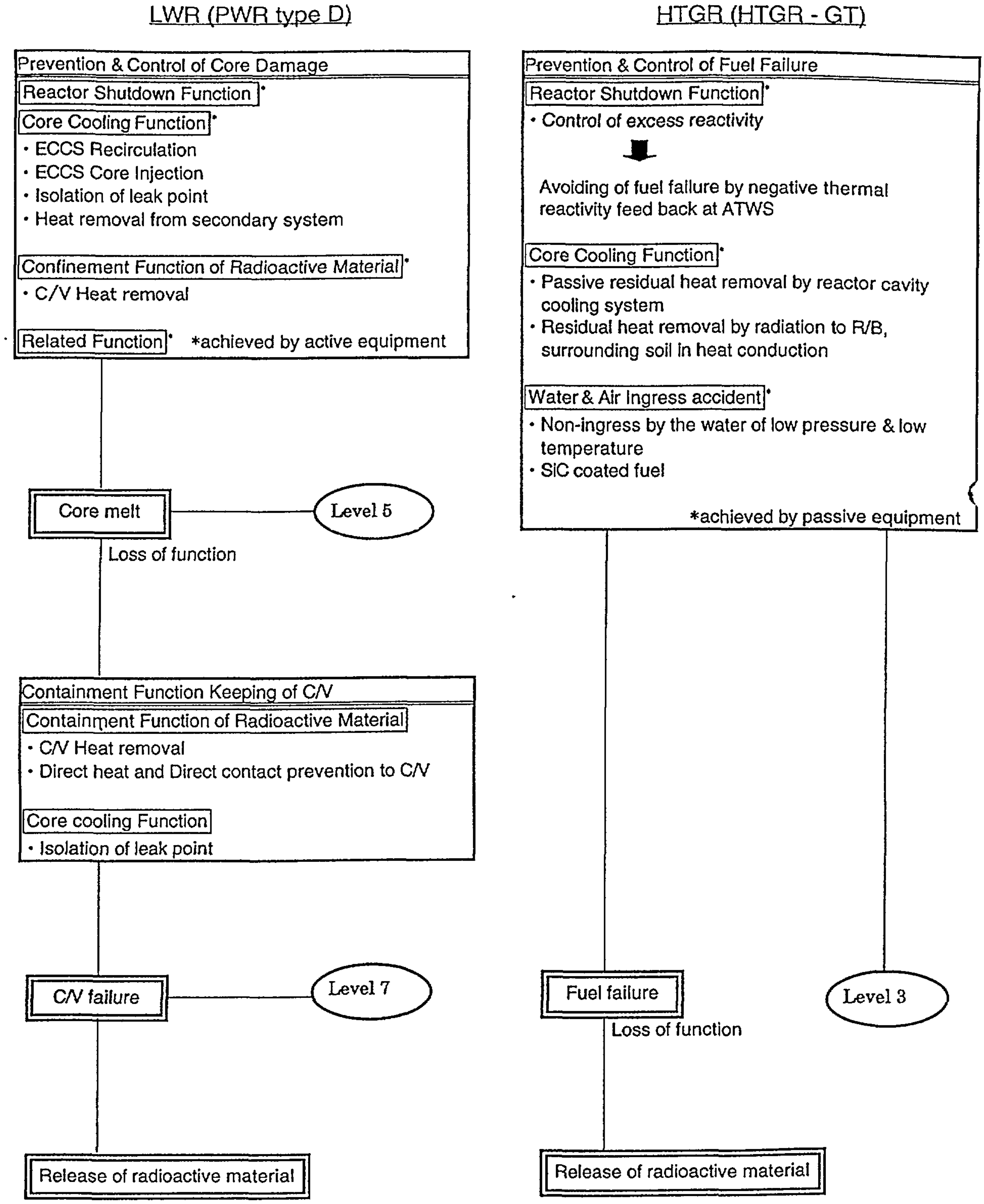

Fig-6 Safety characteristic comparison between LWR and HTGR

(3) HTGR

The safety feature of HTGR was compared with that of LWR.

The failure of reactor coolant pressure boundary could not cause the core damage and fuel failure because of the following reason. The prevention and control of fuel damage is achieved by the reactor shutdown and core cooling function.

Reactor shutdown function

- A loss of coolant accident

  Reactor becomes subcritical by the negative thermal reactivity feed back.
- Withdrawal of a control rod

  Reactor could avoid excess power rise by negative thermal reactivity feed back at ATWS.

Fig-7 shows the analysis of reactor power at the control rod withdrawal accident.

Core cooling function

- Heat removal by the reactor cavity cooling system (RCCS).

Fig-8 shows the analysis of decay heat removal at a loss of coolant accident. Core cooling is achieved by passive equipment. Fuel temperature is less than 1600℃. If RCCS were not available, residual heat might be removed to surrounding soil by heat conduction, keeping fuel temperature less than 1600℃. Further study is necessary.

Water and air ingress is specific accident of HTGR. Water ingress could not cause fuel damage in the case of gas turbine generator, because the cooling water pressure is lower than primary helium and ingress water is maintained in the cold part of power conversion unit. Air ingress from a raptured part is limited and could not cause the fuel damage. SiC coated fuel might be stronger to these accident.

The accident sinario of HTGR is different from that of LWR and FBR. Accidents could not cause the fuel failure in HTGR.The occurrence frequency of fuel damage in HTGR could be less than that of C/V failure in LWR and FBR.

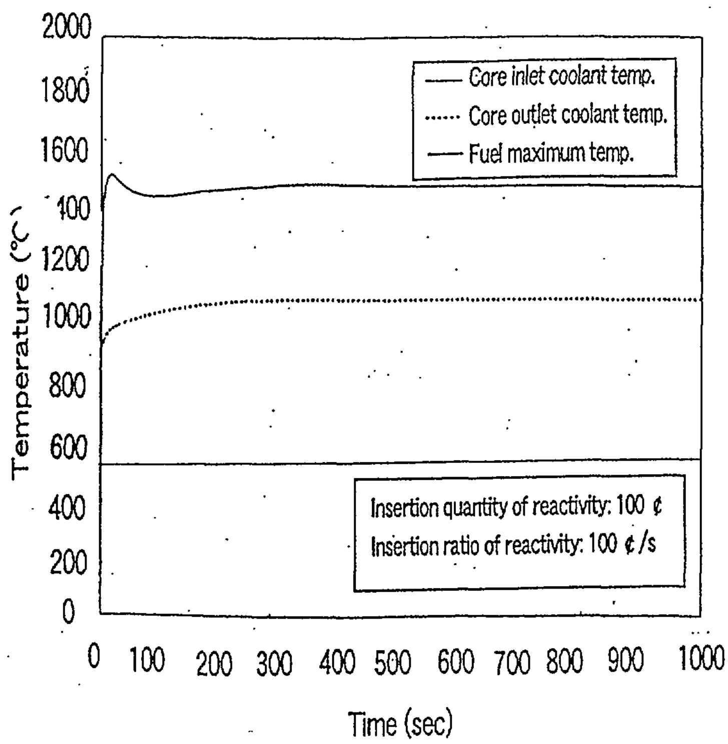

Fig-7 Analysis of reactor power at control rod withdrawal accident

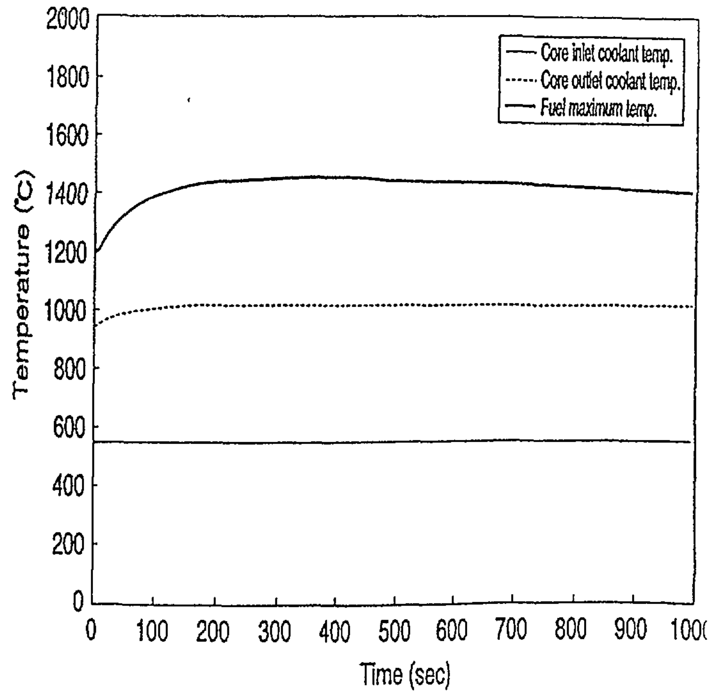

Fig-8 Analysis of decay heat removal at loss of coolant accident

3. EVALUATION ON THE POSSIBILITY OF SAFETY GRADE CHANGES

The possibility of safety grade changes was evaluated from the aspect of different safety features.

3.1 Application of ANSI to HTGR

ANSI/ANS-51.1-1983 defines safety classification for LWR.

We tried to apply its standard taking into consideration about difference of safety potential. The application to HTGR is shown in Table-1.

(1) Safety class 1 (SC-1)

The equipment that forms the part of the reactor coolant pressure boundary (RCPB) could cause a loss of reactor coolant, the large core damage and mount of fuel failure in LWR. In HTGR, the failure of RCPB could not cause the fuel failure and the radiation dose is lower than allowable limit without evacuation at site boundary (shown in Fig-9). RCPB is not classified into SC-1.

(2) Safety class 2 (SC-2)

SC-2 system shown Table-1 is as follows.

- Reactor cavity cooling system

(3) Safety class 3 (SC-3)

SC-3 structures and systems shown Table-1 are as follows.

- Reactor shutdown system
- Core internal structure and reactor vessel
- Support structures of above equipments
- Fuel storage and cooling system

(4) Non-nuclear safety (NNS)

Equipments except SC-1, SC-2 and SC-3 are classified into NNS.

3.2 Application of NSC to HTGR

Japan has the nuclear safety classification standard (NSC). Classification of safety class consists of SC-1, SC-2 and SC-3. The following design targets have to be achieved to classified structures, systems and equipments.

Safety class 1	Ensuring and maintaining the best reliability
Safety class 2	Ensuring and maintaining the best reliability
Safety class 3	Ensuring and maintaining the best reliability which is equal to or better than the general industry level.

Table-1 Definition of Safety Grade and Application of ANSI to HTGR

Safety Class	Definition*	Application to HTGR	
1	Safety Class 1 (SC-1)shall apply to pressure-retaining portions and supports of mechanical equipment that form part of the RCPB whose failure could cause a loss of reactor coolant in excess of the reactor coolant normal makeup capability and whose requirements are within the scope of the ASME Boiler and Pressure Vessel Code, Section III	×	In case of LWR Failure of Safety Class 1 component could cause the large core damage and a mount of fuel failure. In case of HTGR Failure of RCPB could not cause the core damage and fuel failure. There is no SC-1 components in HTGR.
2	Safety Class 2 (SC-2) shall apply to pressure-retaining portions and supports of primary containment and other mechanical equipment, requirements for which are within the scope of the ASME Boiler and Pressure Vessel Code, Section III, that is not included in SC-1 and is designed and relied upon to accomplish the following nuclear safety functions:		
	(a) Provide fission product barrier or primary containment radioactive material holdup or isolation,	×	
	(b) Provide emergency heat removal for the primary containment atmosphere to an intermediate heat sink, or emergency removal of radioactive material from the primary containment atmosphere (e.g., containment spray),	×	Containment is unnecessary because radiation dose is low at site boundary.
	(c) Introduce emergency negative reactivity to make the reactor subcritical (e.g., boron injection system), or restrict the addition of positive reactivity via pressure boundary equipment,	×	Reactor is subcritical by negative thermal reactivity feed back, so emergency shut down system is unnecessary.

* extracted from American National Standard ANSI/ANS-51.1-1983

Table-1 (cont.)

Safety Class	Definition*		Application to HTGR
2	(d) Ensure emergency core cooling where the equipment provides coolant directly to the core (e.g., residual heat removal and emergency core cooling), or	○	HTGR has the reactor cavity cooling system for decay heat removal.
	(e) Provide or maintain sufficient reactor coolant inventory for emergency core cooling (e.g., refueling water storage tank).	×	Coolant injection to core is unnecessary.
3	Safety Class 3 (SC-3) shall apply to equipment, not included in SC-1 or -2, that is designed and relied upon to accomplish the following nuclear safety functions:		
	(a) Provide for functions defined in SC-2 where equipment, or portions thereof, is not within the scope of the ASME Boiler and Pressure Vessel Code, Section III,	–	
	(b) Provide secondary containment radioactive material holdup, isolation, or heat removal,	×	Containment is unnecessary because radiation dose is low at site boundary.
	(c) Except for primary containment boundary extension function, ensure hydrogen concentration control of the primary containment atmosphere to acceptable limits,	×	HTGR could not cause the hydrogen concentration.
	(d) Remove radioactive material from the atmosphere of confined spaces outside primary containment (e.g., control room or fuel building) containing SC-1, -2, or -3 equipment,	×	Primary containment is unnecessary in HTGR.
	(e) Introduce negative reactivity to achieve or maintain subcritical reactor conditions (e.g., boron makeup),	○	The reactor shut down system maintain subcritical reactor conditions.
	(f) Provide or maintain sufficient reactor coolant inventory for core cooling (e.g., reactor coolant normal makeup system),	×	Coolant injection to core is unnecessary.
	(g) Maintain geometry within the reactor to ensure core reactivity control or core cooling capability (e.g., core support structures),	○	The core internal structures and the reactor vessel maintain core geometry.
	(h) Structurally load-bear or protect SC-1, -2, or -3 equipment,	○	The support structures support the reactor shut down system, RCCS, core structures and reactor vessel.
	(i) Provide radiation shielding for the control room or offsite personnel,	×	Release of radioactive materials is low.

* extracted from American National Standard ANSI/ANS-51.1-1983

Table-1 (cont.)

Safety Class	Definition*		Application to HTGR
3	(j) Ensure required cooling for liquid-cooled stored fuel (e.g., spent fuel storage pool and cooling system),	○	The fuel storage and cooling system ensure cooling of fuel.
	(k) Ensure nuclear safety functions provided by SC-1, -2, or -3 equipment (e.g., provide heat removal for SC-1, -2 or -3 heat exchangers, provide lubrication of SC-2 or -3 pumps, provide fuel oil to the emergency diesel engine),	×	
	(l) Provide actuation or motive power for SC-1, -2, or -3 equipment,	×	
	(m) Provide information or controls to ensure capability for manual or automatic actuation of nuclear safety functions required of SC-1, -2, or -3 equipment,	×	Each safety function is achieved by passive components
	(n) Supply or process signals or supply power required for SC-1, -2, or -3 equipment to perform their required nuclear safety functions,	×	
	(o) Provide a manual or automatic interlock function to ensure or maintain proper performance of nuclear safety functions required of SC-1, -2, or -3 equipment, or	×	
	(p) Provide an acceptable environment for SC-1, -2, or -3 equipment and operating personnel.	×	

* extracted from American National Standard ANSI/ANS-51.1-1983

The application of NSC to HTGR was studied. Results of application are as follows.

(1) Safety class 1

- Core internal structure and reactor vessel
- Reactor shutdown system
- Reactor cavity cooling system
- Support structures of above equipment

(2) Safety class 2

- Fuel storage and cooling system

(3) Safety class 3

The other structures, systems and equipments are classified into safety class 3.

4. COST REDUCTION EFFECT

The cost reduction effect was surveyed about following cases according to preliminary study of the small sized HTGR.

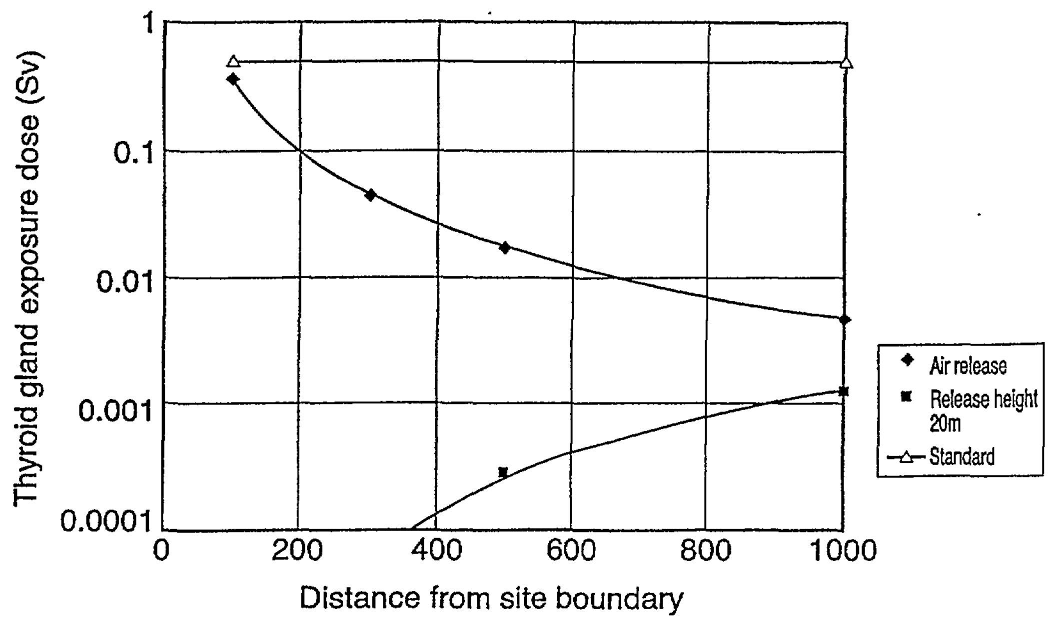

Fig-9 Thyroid gland exposure dose

(1) Case 1

The LWR safety criteria is applied to case 1. Case 1 has a containment vessel like LWR, because the radioactive materials in fuel are assumed to be released to containment vessel at the accident.

(2) Case 2

Case 2 does not have the containment vessel. All equipments are manufactured by nuclear grade.

(3) Case 3

Most of equipments are manufactured by non-nuclear grade except following components.

- RCCS
- Reactor vessel and core internal structure
- Reactor shutdown system
- Support structures of above equipments

Non nuclear grade equipments are manufactured according to fired power plant code and standard. The cost reduction effect is shown in Fig-10.

Case 1 cost is 10% higher than reference case (case 2).

Case 3 cost is 32% lower than reference case.

Changes of safety grade are very effective to cost reduction.

5. LICENSABILITY IN JAPAN

ASME code class is classified according to safety class.

In Japan, seismic class, and MITI-501 (equivalent to ASME) code class are not classified from safety class. Because these classifications are based on the

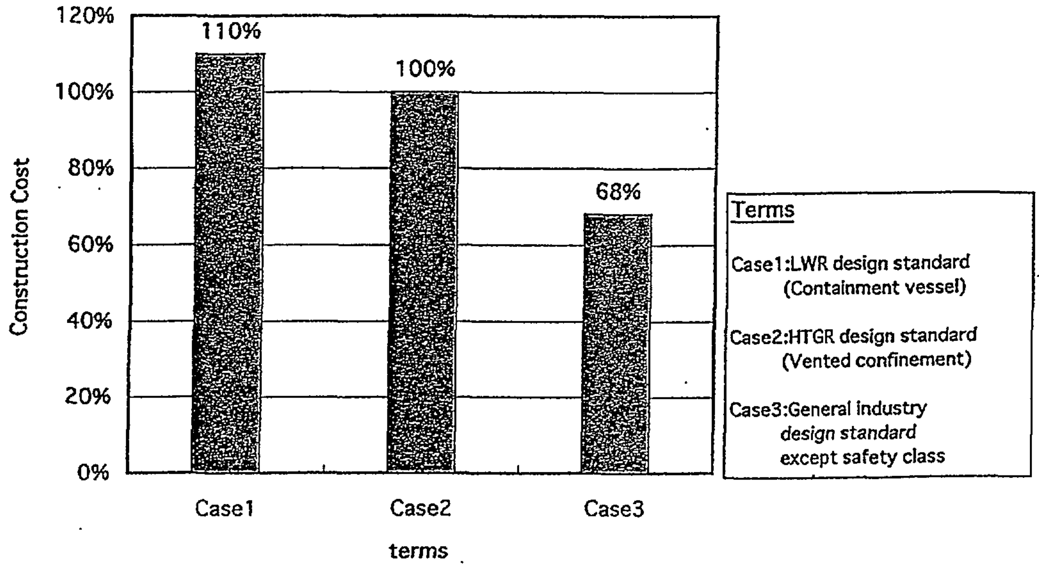

Fig-10 Revaluation for the safety grade at Small-sized HTGR-GT

commercialized LWR and have been established independently. Each equipment is classified by MITI-501 code class and most of them are classified into MITI-501 code class 1～4 which means nuclear grade in spite of SC-3 equipments.

To change the standard and code would require a lot of efforts in Japan, but HTGR has the potentiality to change them.

6. CONCLUSION

HTGR has the higher safety potential in comparison with LWR and FBR, and the possibility of changing safety class and code class. The expansion of non-nuclear grade equipments is very effective to the cost reduction due to the safety class changes. But it requires a lot of efforts to change the standard and code based on classification of seismic class, code class and safety class.

We heard ESKOM has the same ambitious plan of PBMR-SA. We should support it, and we need more quantitative study and international cooperation to establish a new safety criteria for the cost reduction of HTGR.

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CHARACTERISTICS OF DC ELECTRICAL BRAKING METHOD OF THE GAS CIRCULATOR TO LIMIT THE TEMPERATURE RISE AT THE HEAT TRANSFER PIPES IN THE HTTR

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Abstract

In the safety evaluation of a High Temperature Engineering Test Reactor (HTTR), it must be confirmed that the core has no chance to be damaged and the barrier against the FP release is designed properly not to be affecting the influence of radiation around the reactor site. Especially, the maximum temperature of the reactor pressure boundary such as the heat transfer pipes of pressurized water cooler (PWC) must not exceed the permissible values under an anticipated accident such as pipe of rupture in PWC.

A requirement for the gas circulator which circulate helium gas in the primary cooling line and the secondary cooling line is to be braked within 10 second by an electrical braking method after the HTTR reactor has scrammed under the accident in PWC. The reason is why the temperature rise of the heat transfer pipe at PWC has to be suppressed when the gas circulator has stopped, the revolution of the gas circulator is decreases like the free coast down that it takes about 90 second to be zero and the temperature rise of the pipe in the PWC exceeds the permissible value.

By braking within 10 sec, the temperature of the pipe in the PWC is reached about 368 ℃ that is less than the permissible value. Using a simplified equivalent circuit of an induction motor, braking time analysis was performed with obtained electrical resistance and inductance. The obtained braking time is about 10 sec that shows close agreement with analysis values.

1.Introduction

The High Temperature engineering Test Reactor (HTTR) has been constructed at the Oarai Research Establishment of Japan Atomic Energy Research Institute. Its first criticality will be attained in 1998. The HTTR is a graphite moderated and helium gas-cooled reactor with 30 MW in thermal and 950 ℃ of reactor outlet coolant temperature for high temperature test operation.

The HTTR was designing to be applicable to several operations for the high temperature utilization and the safety demonstration tests including the primary coolant flow decrease and the control rod withdrawal at power operation.

The gas circulators in the primary cooling line and the secondary cooling line are required to brake within 10 sec. It is important to suppress the temperature rise of heat transfer tube at the pressurized water cooler (PWC) with in the permissible values using electrical braking under the reactor scram and the accident such as pipe of rupture in PWC.

The design of braking method for the gas circulator is changed from the regenerative braking to the DC electrical braking method. Advantages of the use of DC electrical braking method are (1) braking miss probability is about 10^{-5} frequency/demand and (2) braking force is generated without commercial electrical power.

This paper describes the scenario of an accident condition such as pipe rupture in PWC, the requirements of the braking action, the description of DC electrical braking method; comparison between calculated results and the test results.

2.Outlines

2.1 Outline of the HTTR

The HTTR consists of 30 Mt reactor core, primary helium cooling line, a secondary cooling line, an auxiliary cooling line and the related system. Major specifications of the HTTR are shown in Table 1.

Table 1 Major specifications of the HTTR

Reactor thermal power	30 MW
Reactor active core Equivalent diameter Effective height Average power density	 2.3 m 2.9 m 2.5 W/cm^3
Fuel loading	Off load batch
Fuel	Low enriched UO2
Fuel element type	Pin in block
Number of fuel elements	30 columns
Reactor coolant Temperature at reactor outlet Temperature at reactor inlet Direction of coolant flow Pressure	 850 ℃ (in rated mode) 950 ℃ (in high temp. mode) 395 ℃ Downward flow 4 Mpa
Containment type	Pressure retained steel vessel

A reactor pressure vessel is 13.2 m in height and 5.5 m in diameter. The core located in the reactor pressure vessel is a contained graphite reflector, core support structures and other internal components. The bird eye view of the reactor pressure vessel and the core is shown in Fig.1.

The core is cooled by helium gas with the reactor inlet temperature of 395 ℃ and the downward flow through the core. The maximum temperature in the fuel at the normal operational condition is about 1500℃ at the reactor outlet temperature of 950℃.

The reactor cooling system is composed of the primary cooling line, the secondary cooling line, the auxiliary cooling system and two-reactor pressure vessel cooling system. The reactor cooling system is shown in Fig.2, schematically. The auxiliary cooling line (ACS) is the stand by condition during a normal reactor operation and operated to remove the residual heat from the core at the reactor scram.

The primary cooling line has two heat exchanger. One is a He/He intermediate heat exchanger (IHX) and the other is a pressurized water cooler (PWC). The primary cooling line has four gas circulators and the secondary cooling line of the IHX has one gas circulator. The reactor

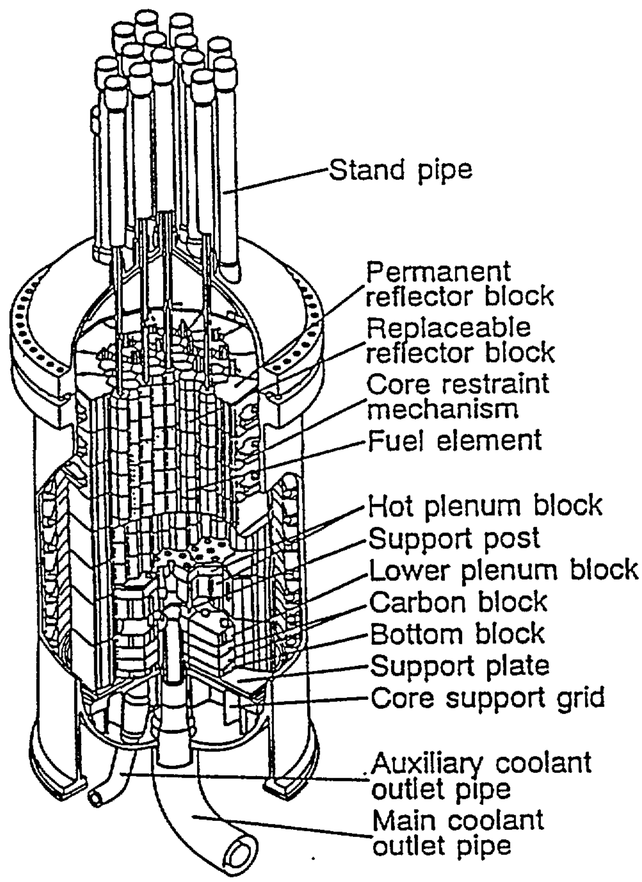


Fig. 1 The bird's-eye view of reactor vessel and core

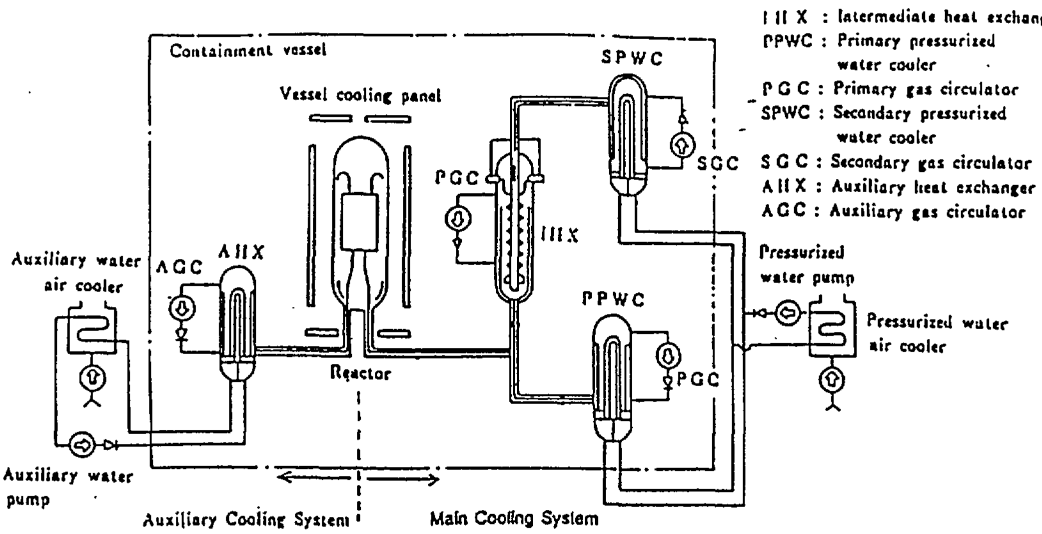


Fig. 2 Cooling line of the HTTR

pressure vessel cooling lines (VCS) are operated in normal operation to cool the biological shielding concrete wall.

2.2 Configurations of gas circulator

A cross-sectional view of gas circulator is shown in Fig.3. Gas bearings are used to maintain such a condition that the relative motion between the rotor and its stator obtains the lift. They acts on the principle that a pressure field, which depends on the viscosity, the linear speed, the load and the size of the bearing, builds up between two convergent surfaces in relative movement due to the viscosity forces that originate in the moving fluid.

The rotating part consists of a stiff and a hollow shaft. It supports the motor; the single stage centrifugal impeller at its end and the motor compartment separates from the impeller compartment by a heat screen. The heat exchange with the outside cooling water system takes place at the location of the inner casing on which the motor is shrunk.

The gas bearing system consists of radial bearings mounted on either end of the motor and titling pad thrust bearings mounted on gimbals system.

3. Accident scenario of a pipe rupture in PWC

3.1 Accident scenario

In this accident, the pressurized water flow is lost and the heat removal of PWC can not maintained and the fuel temperature rises. In the accident, the reactor is scrammed by the reactor protection system and the ACS is started automatically to remove the residual heat of the core.

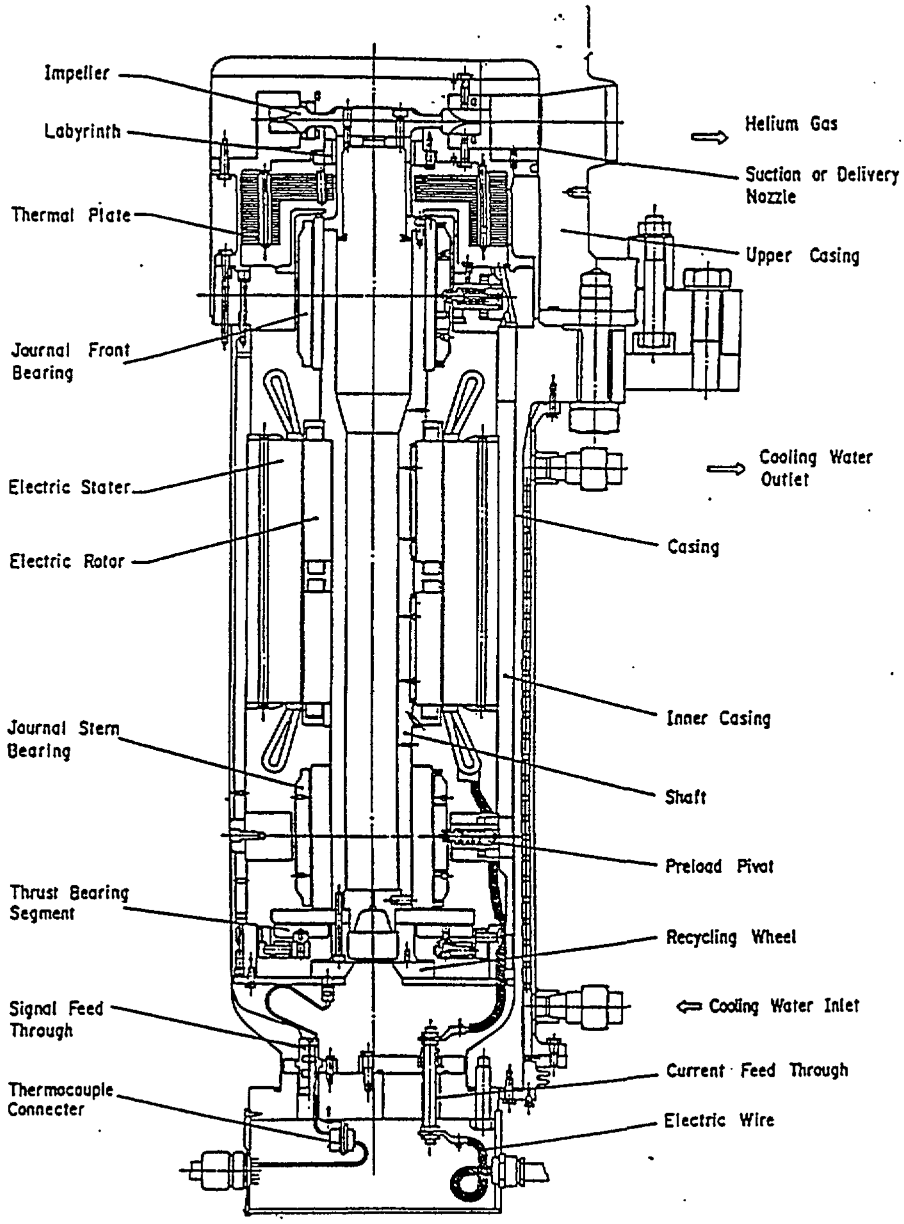


Fig. 3 Cross-sectional view of the gas circulator

The thermal and hydraulic behavior during this accident was evaluated with the THYED-HTGR code. In this analysis, this accident was analyzed as the transient of the piping broken at the outlet of the pressurized water pump. Analytical conditions are as follows:

(1) The reactor is operated at the single loaded operation (Tandoku).
(2) Reactivity feedback by the temperature changes in the fuel and moderator is neglected.
(3) Start-up failure of one out of two axially gas circulator is considered as a single failure condition.
(4) Loss of off-site electric power supply is considered at the time of a scrammed.

Changing the braking time of the gas circulator in the primary cooling line, figure 4 shows the analytical results of the maximum temperature at the heat transfer pipe in the PWC.

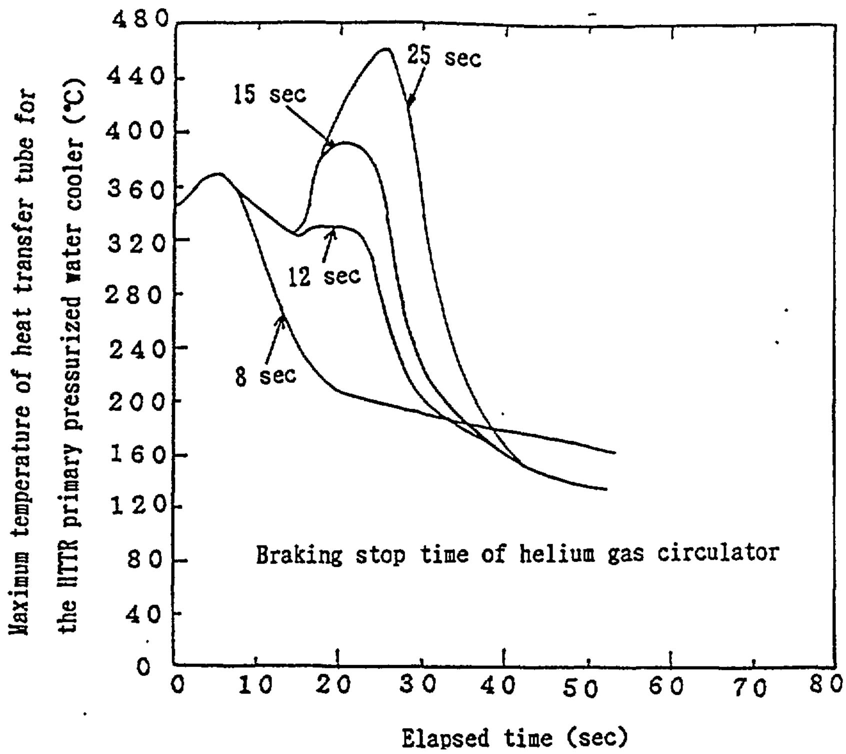


Fig. 4 Analytical result of the maximum temperature rise at the heat tube in the PWC

The maximum temperature of the PWC heat transfer pipe depends on the braking time, its temperature rises up to 460 ℃ from its 368℃ under the braking time from 10 sec to 25 sec and the maximum fuel temperature does not rise up to a extreme value .

3.2 Requirements to brake the gas circulator

Even if the anticipated operational occurrences occur in the reactor facilities, the event must be terminated before the core is damaged and the reactor facilities shall be recovered to the normal operation condition. Criterion on the reactor pressure boundary has been established to judge the integrity against the events assumed in the HTTR. The safety criteria are that the maximum temperature of the reactor pressure boundary is 600 ℃ under the anticipated operational transients and 650℃ under accidents.

Requirements to brake the gas circulator are due to suppress the temperature rise of the reactor coolant pressure boundary at the PWC. Detailed requirements are summarized as follows.

(1) Temperature limit of the PWC

Temperature rises of the PWC transfer pipe depend on braking time of the gas circulators. In case of loss of coolant at the pressurized water cooling line, the reactor is scrammed, the gas circulators are operated with the free coast down flow and the residual core heat is transferred to the PWC. Therefore, due to the loss of coolant at the pressurized water cooling line, the cooling ability is decreased and the temperature at PWC transfer pipe is increased.

If the gas circulators are able to stop as fast as the free coast down, the maximum temperature of the PWC transfer pipe is restricted less than the permissible one. The gas circulator is designed to brake within 10 sec after the reactor scram from the temperature analysis.

(2) Braking without commercial electric power

The safety analysis for the anticipated event is to be carried out without commercial electric power. So, the braking of the gas circulators has to act without commercial electric power.

Under the conditions of the commercial electric power lost, regenerative braking needs an external electrical power to produce the magnetic fields for about 2.5 second because it can make a closed electrical circuit for regenerative action. From the economical and reliability standpoints, a battery charger unit is applicable to the supply from an external electric power.

(3) Reliability of braking

The rotation of the gas circulator is controlled by adjusting power frequency in proportion to the helium flow rate using an invertor-divertor unit with GTO (Gate Turn Off) thyrister named VVVF (Variable Voltage Variable Frequency).

Required braking miss probability of the gas circulator is about 10^{-5} frequency/demand from the safety analysis and it leads to conservative estimations for the regenerative braking used VVVF.

4. Design

4.1 Operation of the gas circulator

It is designed to circulate the primary cooling gas of the HTTR with a constant flow rate that is specified by the consideration of the heat removal from the core. Nominal characteristics of the gas circulator are as follows;

- Intake pressure: 4.0 MPa
- Discharge pressure: 4.09MPa
- Helium flow rate: 15.1 t/h
- Speed of rotation: 11250 rpm

- Electric output: 242 kW
- Inlet temperature:395℃

Operational conditions for the various HTTR operation are summarized in Table 2.

Table 2 Operational conditions for the HTTR gas circulator

Operation mode		Flow rate (t/h)	Pressure rise (bar)	Rotation (rpm)
Tandoku-rated	Max Min	15.1 15.1	0.96 0.62	11170 9700
Heiretu-rated	Max Min	9.9 9.9	0.65 0.41	8900 7400
Tandoku-high.temp	Max Min	12.34 12.34	0.69 0.43	9420 8110
Heiretu-high.temp	Max Min	8.1 8.1	0.48 0.29	7630 6320

4.2 Selection of braking method for the gas circulator

A safety requirement for the braking of the gas circulator is to be braked within 10 sec after the reactor scram without commercial electrical power.

According the detailed design, the providing electrical power for the regenerative braking method which is necessary to generate the magnetic field about 250 kW x 2.5 sec x 5 units. The reliability of braking for the gas circulator contained regenerative braking components is lead to conservative from the probability safety analysis.

DC electrical braking that acts like an inductive generator has much advantage. The schematic diagram of DC electrical braking is shown in Fig. 5. It is conducted as follows. Providing external current is produced the steady state magnetic field, then the eddy current is induced by the magnetic interaction. Therefore, braking torque in proportion to B (Intensity of magnetic field) x I (Induced current) are produced and the rotation of the gas circulator is decreased faster than the free coast down.

Evaluation of reliability of DC electrical braking was carried out. The main components of DC electrical braking are (1) battery and (2) current

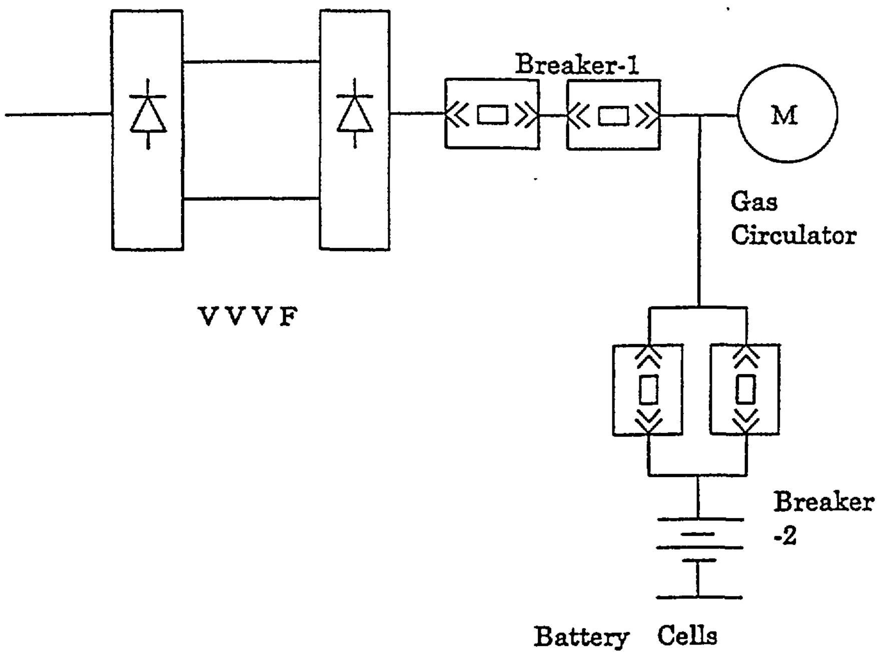


Fig. 5 Schematic diagram of DC electrical braking method

breakers. The current breaker-1 and breaker-2 are designed to have double current breakers and fault probability for braking action is decreasing down to 10^{-5} frequency/demand. According of the MTBF for the battery and the current breaker is about 10^{-7} hours; therefore, the calculated fault probability of DC electrical braking leads to about 10^{-5} frequency/demand.

Finally, DC electrical braking for the induction motor is selected for the braking method of the gas circulator that has satisfied the requirement such as the reliability and to perform DC electrical braking action without commercial electric power.

5. Braking time analysis

5.1 Fundamental equation

DC electrical braking action is based on the inductive generator action. The equivalent circuit of the inductive generator is shown in Fig.6. The magnetic forces are converted the joule heating at the rotor. Therefore, the joule heating, P is producing the resistance at the rotor and the braking torque, Tb, is transformed by using this relation as follows;

$$Tb= P/w=2xr_2xI^2/s$$

When, P: joule heating in the rotor

w: angular speed

r_2: resistance of the rotor

I: current of the rotor

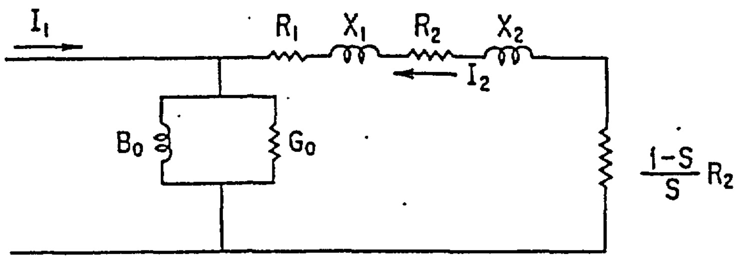
s: slipping ratio

The dynamic equation of the rotor is expressed as follows;

$$I \frac{dw}{dt} = -102xL/w - \mu xMxR - Tb$$

$I = GD^2/4g$

$L = L0x\ (w/w0)^3$



Primary resistance	R1=0.0172	Ω	
Secondary resistance	R2=0.0242	Ω	
Primary reactance	X1=0.1341	Ω	
Seconadry reactance	X2=0.1305	Ω	
Exciting condactance	G0=0.2	1 /Ω	
Exciting saceptance	B0=0.25	1 /Ω	

Fig. 6 Equivalent circuit of the inductive generator

Where, I: moment of inertia at the rotor

L: axial power

g: gravitational constant

R: radius of the rotor

M: weight of the rotor

GD^2: flywheel effect of the rotor

μ :friction coefficient

and zero suffixes mean the initial values before the reactor scram. The magnetic force for the braking torque in the induction motor is solved.

5.2 Performance Test results

Figure 7 shows the braking time for gas circulator. The linear line is predicted curve and dot point is measured ones. All of test results show that the measured breaking time is about one second less than the predicted ones. Between from 1.0 and 0.3 of normalized angular speed w, the measured rotation curve due to DC electrical breaking provides an equally good fit to the predicted ones.

The normalized angular speed w is less than 0.6 that the decreasing rotation speed due to DC electrical breaking is reached to zero faster than the measured ones.

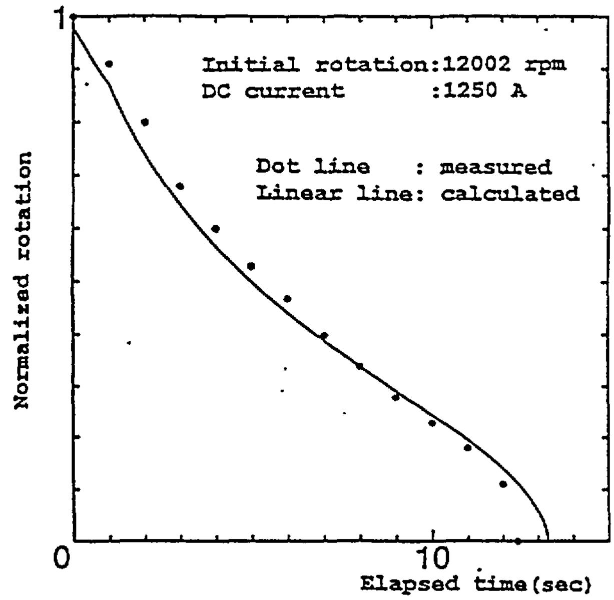


Fig. 7 Comparison between measured value and calculated one

5.3 Operational limitation

The HTTR has various operational modes. Therefore, the providing DC current that which change the DC current from 1000 A to 1400 A, must be changed to keep the braking time within 10 sec.

In the case of under 10000 rpm, it is found that the providing DC current 1200 A is required to brake the gas circulator within 10 sec. Other hand, in case of over 10000 rpm, the providing DC current 1400 A is necessary to brake within 10 sec.

Due to the operation schedule of the HTTR, the providing DC current must be changed to keep the braking time within 10 sec. At the reactor shutdown, it is changed the battery cells that is possible to provide required the DC current. Table 3 shows the recommended DC current against the HTTR operation mode and calculated results is shown in Fig. 8. It is found that the gas circulator is to be braked within 10 sec.

Table 3 Recommended providing DC current for HTTR operational mode

Operation mode	Rotation (rpm)	Recommended DC current (A)	Braking time (second)
Tandoku-rated	11700	1400	9.8
Heiretu-rated	8900	1200	9.3
Tandoku-high.temp	9420	1200	9.5
Heiretu-high.temp	7630	1000	8.7

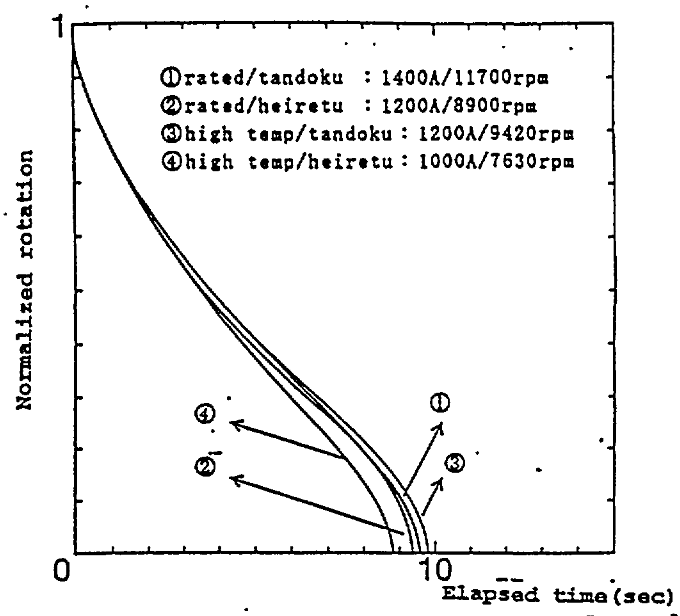


Fig. 8 Recommended DC current against various operational mode

6. Conclusion

We experimented to brake the gas circulator using the DC electrical braking in the helium gas field under high temperature around 400℃ and high-pressure around 4.0MPa.

Its performance is summarized as follows;

(1) The maximum temperature criterion on the reactor pressure boundary which is 600 ℃ under the anticipated operational transients and 650℃ under accidents, are satisfied when the gas circulator is braked within 10 sec .

(2) Analyzed method for the DC electrical braking of the gas circulator is useful to combine the braking torque that is obtained from the equivalent electrical circuit and the mechanical dynamics.
(3) Measured DC braking time is very agreement with predicted one.
(4) Recommended DC current for the various operation mode of the HTTR is cleared.

ACKNOWLEDGEMENTS

The authors would like to acknowledge Messrs. T.TANAKA, M.OHKUBO and H.SUZUKI, HTTR Project, JAERI for their continuing guidance and encouragement. Many thanks are also due to the members of division HTTR Test Development.

ANALYSIS OF STRESS IN REACTOR CORE VESSEL UNDER EFFECT OF PRESSURE LOSE SHOCK WAVE

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Abstract

High Temperature gas cooled Reactor (HTR-10) is a modular High Temperature gas cooled Reactor of the new generation. In order to analyze the safety characteristics of its core vessel in case of large rupture accident, the transient performance of its core vessel under the effect of pressure lose shock wave is studied, and the transient pressure difference between the two sides of the core vessel and the transient stresses in the core vessel is presented in this paper, these results can be used in the safety analysis and safety design of the core vessel of HTR-10.

Key Words: High Temperature Gas cooled Reactor Large Rupture Accident Pressure Lose Shock Wave Stress

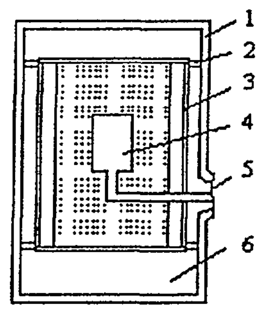
1 Preface

HTR-10 is a module High Temperature Gas cooled tested Reactor designed by institute of Nuclear Energy Technology (INET). Its double ends of the heat duct break accident is one of hypothesis ultimate accidents. The helium in the primary loop will jet from the rupture. When the accident happens and the heavily pressure loss shock wave are rapidly spread into the core. The safe characters of the core vessel in the shock wave define the safe stability of the structure component sin the core because the structure components are contained in the vessel of the core. So it is important to analyze the dynamic behavior of the vessel in the shock wave.

2 Simplified calculating model

Because the structure of the reactor core is very complex, we have several simplified hypothesises below.

The pebble-bed core is a simple gas cavity, and the effect of the coolant channel of the helium is ignored. Because the pressure drop in the core is spread around through gap of the coolant among the graphite components, we assumed the drop reach the inner wall of the core in a instant and the Graphite components are ideal rigid media in order to analyze the core vessel. In order to simplify the calculated region, the pressure vessel was dealt with as a cylinder vessel and the components on its top and bottom were neglected. The simplified analyzing model of reactor of HTR-10 is shown in figure 1. The component 3 is the core vessel which is cylinder shape, 0.30 meter thick, 6.3 meter high and mean diameter 3.8 meter in the figure. We can consider it as a shell because its thickness is much smaller than its height and diameter.



1. Vessel shell 2. Position pad 3. Core shell 4. Core 5. The rupture 6. Gas plenum
Figure 1. Simple model of HTR-10

3 mathematical model

We deal with the core shell in columnar coordinate(r, Φ, Z). The value of radius r is a constant "a" equaling the central radius of the cylinder shell.

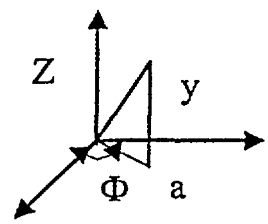
3.1 fluid region

3.1.1 Control equations

$$\frac{\partial \rho}{\partial t} + \nabla \cdot (\rho \vec{U}) = 0$$

$$\frac{\partial (\rho \vec{U})}{\partial t} + \nabla \cdot (\rho \vec{U}\vec{U}) = -\nabla p$$

$$\frac{\partial \rho I}{\partial t} + \vec{U} \cdot \nabla (\rho I \vec{U}) = -p \nabla \cdot \vec{U}$$



p=ρRT
I=C_vT

3.1.2 Boundary conditions
The boundaries of fluid are classified to four categories as follows:
(1) No sliding rigid wall
(2) Free sliding rigid wall
(3) Determined outflow
(4) Determined inflow

3.2 Solid region

3.2.1 Three dimensions elasticity shell equations

$$\rho h \ddot{U} = N_{\phi}^{\circ} + N_{\phi z}^{'} - M_{\phi}^{\circ} / a + M_{z\phi}^{'} / a$$

$$\rho h \ddot{V} = N_{z}^{'} + N_{\phi z}^{\circ}$$

$$\rho h \ddot{W} = q - M_{z}^{''} + 2 M_{z\phi}^{\circ '} - M_{\phi}^{\circ\circ} - N_{\phi} / a$$

$$N_z = C(V' + \nu U^0 + \nu W / a) \qquad N_{\phi} = C(U^0 + W / a + \nu V')$$

$$N_{\phi z} = C(1-\nu)(V^0 + U')/2 \qquad M_z = D(W'' + \nu W^{00} - \nu U^0 / a)$$

$$M_{\phi} = D(W^{00} + \nu W'' - U^0 / a) \qquad M_{z\phi} = -D(1-\nu)(W^{0'} - U' / a)$$

C=E•h/(1-ν^2) D=E•h^3/12(1-ν^2)

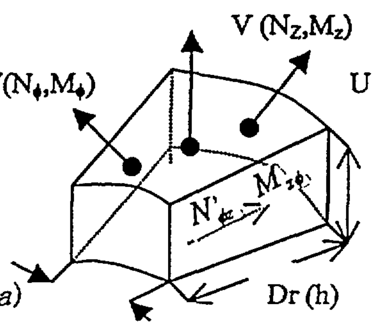


Figure 2 Control volume of fluid

3.2.2 The stress relations
The stresses are determined by the force and torque per length along the thickness of the vessel.

$$\sigma_{zm} = N_z / h \qquad \sigma_{zb} = 6 M_z / h^2$$

$$\sigma_{\phi m} = N_{\phi} / h \qquad \sigma_{\phi b} = 6 M_{\phi} / h^2$$

$$\sigma_{\phi zm} = N_{\phi z} / h \qquad \sigma_{z\phi b} = 6 M_{z\phi} / h^2$$

3.2.3 Boundary conditions
Both ends of the core vessel are considered as fixed boundaries, so,
U=V=W=W' =0

3.3 Pipe region

3.3.1 In this code, we use one dimension control volume and its control equations are:

$$\frac{\partial \rho}{\partial t} + \frac{1}{A}\frac{\partial \rho u A}{\partial x} = 0$$

$$\frac{\partial u}{\partial t} + u\frac{\partial u}{\partial x} = -\frac{1}{\rho}\frac{\partial p}{\partial x} - \frac{f}{2D}u^2 - \frac{k}{2L}u^2$$

$$\frac{\partial I}{\partial t} + u\frac{\partial I}{\partial x} = H_{fric}$$

I=C_vT
p=ρRT

3.3.2 Boundary conditions

The boundary of the upper reach is the thermodynamic condition of the juncture unit of the vessel and pipe. The last expanded unit is surrounding atmosphere. The pressure of surrounding is the boundary of the lower reach.

4 Numerical method

The codes, K-FIX and FLX in this paper were designed by Los Alamos National Science Lab in U.S.A. K-FIX is composite of two parts and a two phases fluid dynamic code. FLX code is used in structure dynamic calculation, specially, study of the dynamic behavior of the supporting ring . The codes were redesigned by our institute (INET), and applied in the investigation of HTR. The figure 2 shows the control volume of fluid (solid shell).The calculation region and grid are shown in figure 3,4,5.

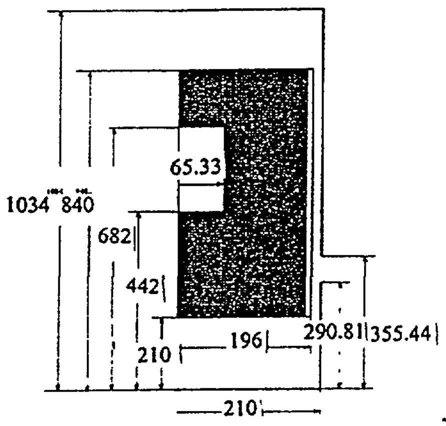


Figure 3 calculation region

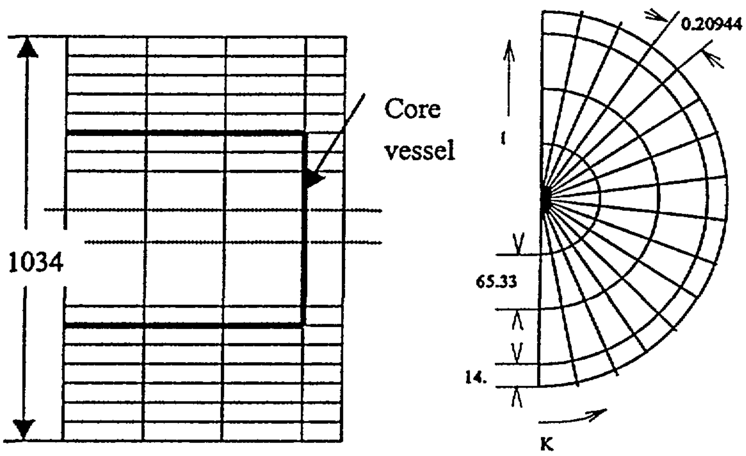


Figure 4 calculation grid for core

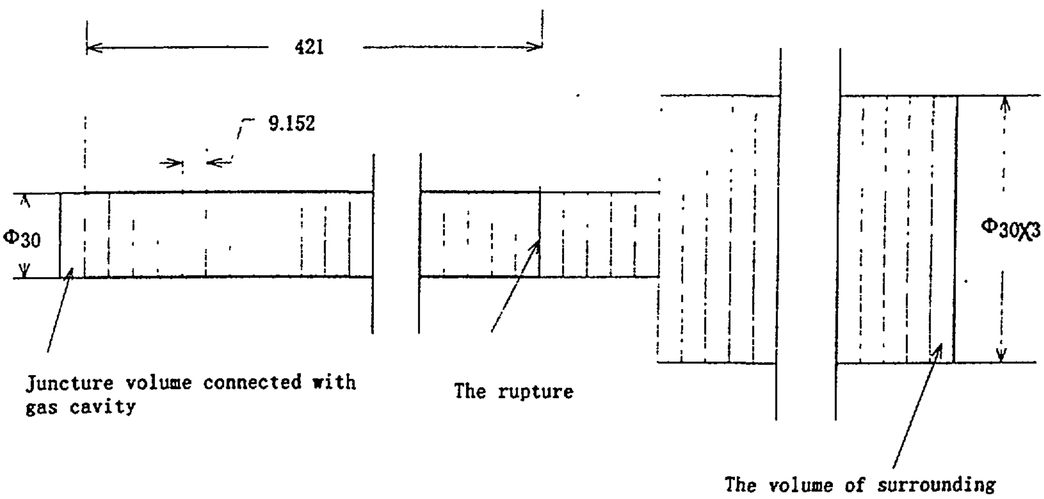


Figure 5 Pipe calculation grid

4.1 Fluid equations

The difference equations are solved through the point-relaxing method. First, the momentum equation (ignoring pressure item) is calculated to get the approximate velocity vector; then the continuity, momentum and energy equations are done by using the similar value for resulting in the exact velocity field. Whether or not the equations converge is controlled by the remaining of the continue equation. All iteration progresses are proceeded by the regulating pressure, calculating point by point till all points approach the exact results. At last, for getting the gas energy,.the exact velocity will be put in the energy equation.

4.2 Solid equations

The equations are solved by the difference method, and the numeric results are reached through explicit integrating of limited difference equations. First, we used the displacements of the last step to calculate the force and torque per length, then we got the accelerate velocity in all direction that will be integrated to become new velocities and displacements. The next calculating step will go on.

4.3 Coupling of solid and fluid

The fluid code and solid code are coupled explicitly. In every iteration, the fluid pressures decide the distribution of pressure difference between one side and the other of the solid structure. The motion of the core vessel is driven by the accelerate velocity (force) in radius direction. And in the next step, new fluid pressure is decided, using the radius direction motion of the shell.

5 Results and analysis

We calculated the mechanic performance of the shell based on the simple core model of HTR-10. Results show that the eruption lasts no more than 1s. Within 2ms after the eruption, the pressure difference between inner side and outer of the core vessel rapidly goes up to 1.81Mpa. after 2ms, the pressure in the rupture and the pressure difference will go down slowly. The instant change of the pressure difference is shown in figure 6. The time axes is logarithm coordination. The distortion and stress of the core vessel near the rupture is lager than far from it. We choose the point of the largest stress near the rupture and calculate its stress. Its sum stress (the sum of membrane and bend stress) is respectively shown in figure 7,8,9. In this paper, we set the pull stress positive and press negative. The stress of any position in the thick section of the core vessel is the sum of membrane and bend stress.

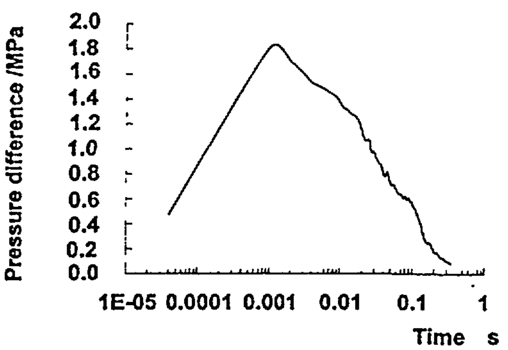


Figure 6 The rate of pressure difference between inner and outer side of the core vessel

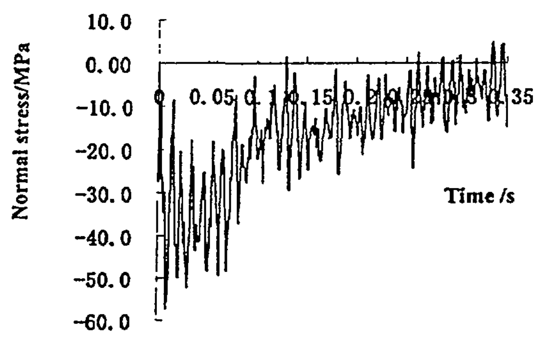


Figure 7 Normal stress in the axial section

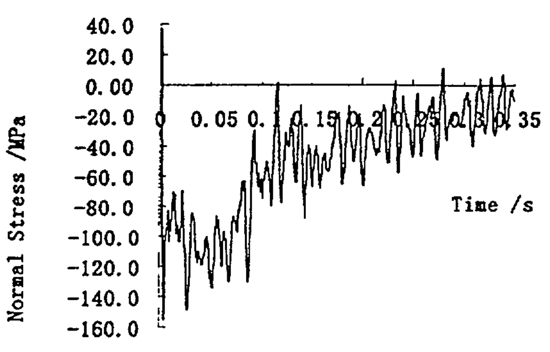


Figure 8 Normal stress in the peripheral section

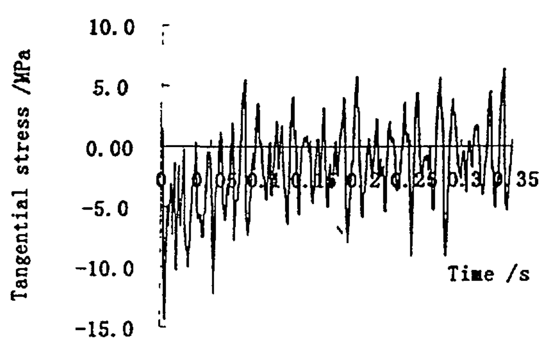


Figure 9 Tangential stress in the peripheral section

We can find that absolute normal stress in the peripheral section is the largest one, 100MPa order. It accords with the behavior of HTR under this kind of accident. Because of the chose point, which has the maximum stress, this point is the most dangerous one. As shown in figure. 7,8,9, the stress occurs while the helium goes out. After this, the altitude of the stress gradually decreases down to 0. Furthermore, when the double ends break accident occurs, it might damage the core, so strength of the shell must be checked. From the results, we can find that the maximum difference pressure between inner side and outer is 1.86MPa under the shock wave in the accident of the core vessel. The normal stress of the vessel in the axial section is 57MPa. The stress in the peripheral section is 158MPa, and the maximum tangential stress is 14.5MPa.

The material made of the core vessel of HTR-10 is 15CrMn and its yielding stress is bigger than 266Mpa at 255℃ in the accident. The calculating results show that the most dangerous point in the peripheral section has the maximum normal stress, about 15MPa. This value is smaller than 226MPa. According to the third strength theory[4]:

$$\bar{\sigma} = \sqrt{\sigma^2 + 4\tau^2} = \sqrt{158^2 + 4 \times 14.9^2} = 161 MPa < 226 MPa.$$

We can conclude that the core shell will not be destroyed and unstabilized. It is safe.

6 conclusion

High Temperature Gas cooled Reactor (HTR-10) is a modular High Temperature gas cooled Reactor of the new generation. It is designed and developed by INET, Tsinghua University. The double ends break accident of the heat gas duct is the most severe accident assumed. The safety character in the accident is important to the development of HTR. It is shown in the calculated outcomes. Even under this accident, HTR-10 still remains safe. HTR-10 has fine safety characters, at the same time, these data provide the basis to the core vessel analyzing and designing.

Symbol tabulation

U: Peripheral displacement (m);
V: Axial displacement (m);
W: Radius displacement (m);
ρ: Density (kg/m^3);
h: Thickness of core vessel (m);
a: The central radius of the cylinder (m);
v: Poisson ratio;
q: The pressure difference of the inner and outer side of the vessel (Pa);
E: Yang modulus (Pa);
N: The force acting on the microelement of the vessel (N);
M: The torque acting on the microelement of the vessel (N•m);
I: Internal energy;
P: Gas pressure (Pa);
R: Gas constant;
T: Gas temperature (K);
C_v: Specific heat (J/kg);
H_{fric}: Dissipation of energy (J);
σ: Shell stress (Pa);
Subscripts:
' : Derivative of valuable with respect to z;
° : Derivative of valuable with respect to aϕ;
, : Derivative of valuable with respect to t;
Subscripts:
Z: Part of the valuable in Z direction;
Φ: Part of the valuable in aΦ direction ;
m: Menbrane stress;
b: bend stress;

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FABRICATION OF HTTR FIRST LOADING FUEL

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Abstract

This paper summarizes the fabrication of the first loading fuel for HTTR, High Temperature engineering Test Reactor constructed by JAERI, Japan Atomic Energy Research Institute. The fuel fabrication started at the HTR fuel facility of NFI, Nuclear Fuel Industries, Ltd., June 1995. 4,770 fuel rods were fabricated through the fuel kernel, coated fuel particle and fuel compaction process, then 150 fuel elements were assembled in the reactor building December 1997.

Fabrication technology for the fuel was established through a lot of R&D activities and fabrication experience of irradiation examination samples spread over about 30 years. Most of all, very high quality and production efficiency of fuel were achieved by the development of the fuel kernel process using the vibration dropping technology, the continuous 4-layer coating process and the automatic compaction process. As for the inspection technology, the development of the automatic measurement equipment for coated layer thickness of a coated fuel particle and uranium content of a fuel compact contributed to the higher reliability and rationalization of the inspection process. The data processing system for the fabrication and quality control, which was originally developed by NFI, made possible not only quick feedback of statistical quality data to the fabrication processes, but also automatic document preparation, such as inspection certificates and accountability control reports.

The quality of the first loading fuel fully satisfied the design specifications for the fuel. In particular, average bare uranium fraction and SiC defective fraction of fuel compacts were 2×10^{-6} and 8×10^{-5} respectively. According to the preceding irradiation examinations being performed at JMTR, Japan Materials Testing Reactor of JAERI, the specimen sampled from the first loading fuel shows good irradiation performance.

1. Introduction

Table 1 and Figure 1 show the primary specification and structure of the first loading fuel for the HTTR. The fuel assembly represents a hexahedral prism of 360mm across flats and 580mm in length, and consists of a graphite block, fuel rods to be inserted and retained in a graphite block and so on. 31 or 33 fuel rods are inserted in a fuel assembly. The fuel rod accommodates 14 fuel compacts in a graphite sleeve of 34mm in outer diameter. The fuel compact has a shape of annular cylinder which consists of graphite matrix and Tri-isotropic (TRISO)-coated fuel particles of 0.92 mm in diameter, dispersed within the matrix body. A coated fuel particle consists of a UO_2 kernel, 600 μm in diameter, and 4 layers of pyrolytic carbon(PyC) and silicon carbide(SiC) [1].

2. Fabrication Schedule

As is shown in Figure 2, the fuel fabrication started June 1995. 4,770 fuel rods with 12 kinds of enrichment in total, which correspond to 66780 fuel compacts, were fabricated

TABLE 1 MAJOR SPECIFICATION OF HTTR FUEL

Item	Diameter or Thickness (μm)	Density (g/cm^3)
UO_2kernel	600	10.63
Low density PyC (1st)	60	1.10
High density PyC (2nd)	30	1.85
SiC (3rd)	25	3.20
High density PyC (4th)	45	1.85
Coated fuel particle	920	(4.45)

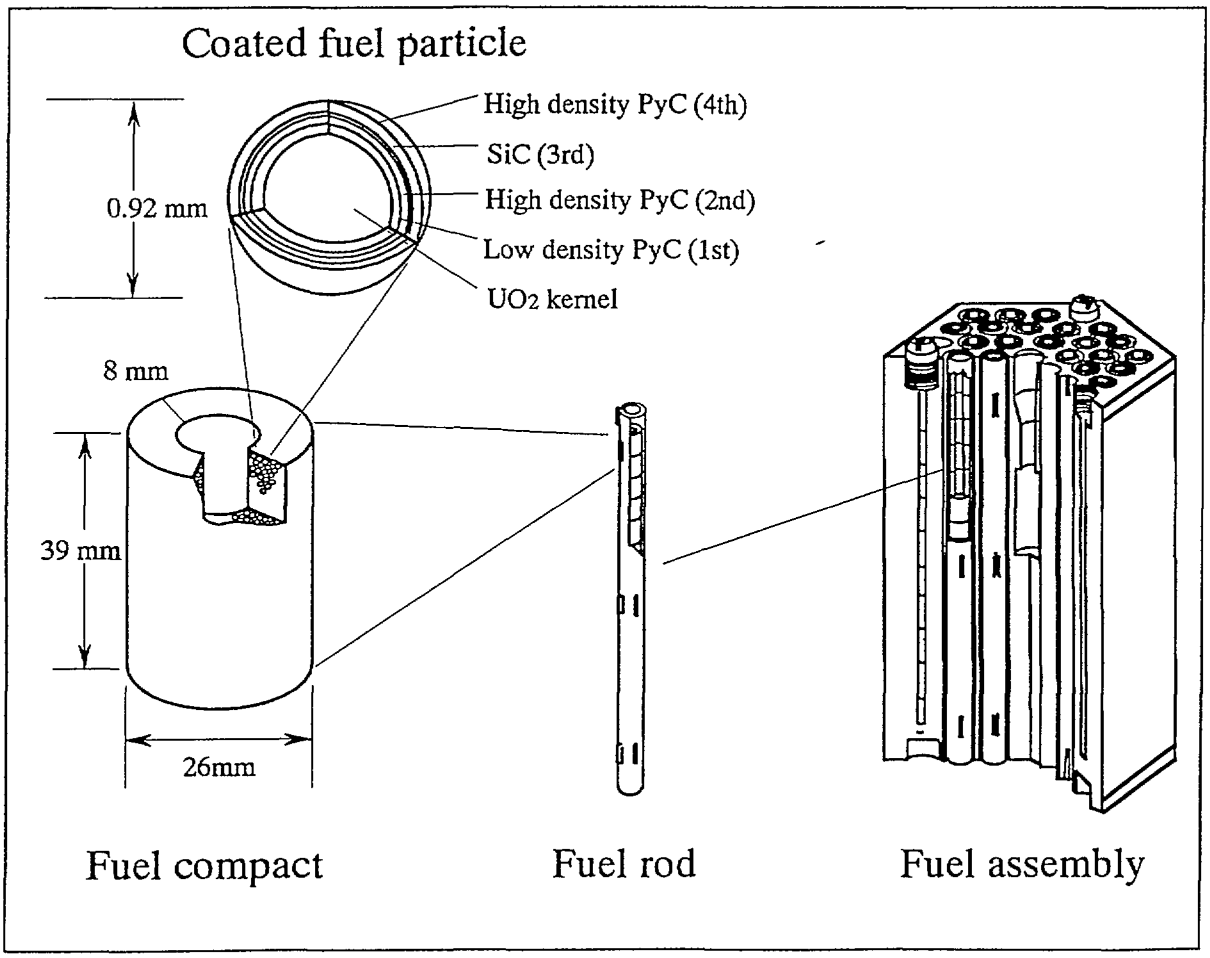


FIG. 1 STRUCTURE OF HTTR FUEL ASSEMBLY

through the fuel kernel, coated fuel particle and fuel compact process. The fuel rods were transported to the reactor building of the HTTR for three times. Then 150 fuel elements were assembled there December 1997. All fuel assemblies, sorting by a column of 5 fuel assemblies, were stored in the new storage cells under helium atmosphere. At present, fuel loading is under way.

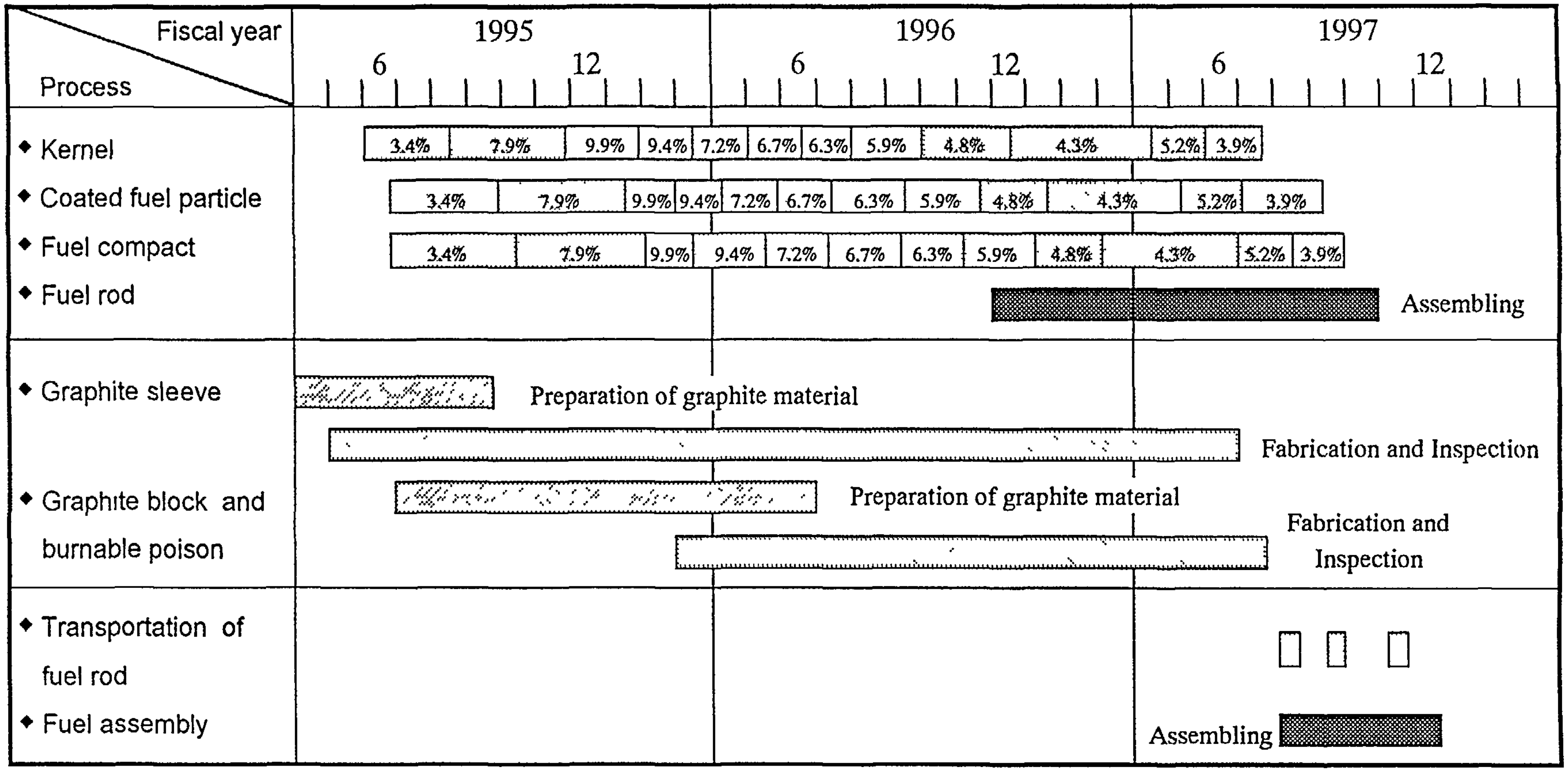


FIG. 2 SCHEDULE OF HTTR FIRST LOADING FUEL FABRICATION

3. Fabrication Technology and the Quality

In late 1980s, NFI set up semi-mass-production equipment with licensed capacity of 200 kgU/year and fabricated 260 kgU of fuels for Very High Temperature Reactor Critical Assembly in JAERI. After that, NFI fabricated the fuels for the irradiation tests in Oarai Gas Loop−1 of JMTR etc., continuing not only to study the fabrication technology to achieve better sphericity of coated fuel particle and less defective fraction of coated layer, but also develop the rational fabrication technology and equipment. In 1992, NFI launched the HTR fuel plant with the licensed capacity of 400 kgU/year, which incorporated all the results obtained until then. Finally, the fabrication of 900 kgU of the HTTR first loading fuel successfully completed last December. Fabrication technology and the quality of the HTTR first loading fuel are described below.

3.1 Fuel Kernel

Fuel kernels are fabricated by, so called, "Gel Precipitation Process". The process is shown in Figure 3. Metal solution is prepared as the mixture of the starting material of uranyl nitrate solution and additives to control the viscosity of the solution. Droplets of the metal solution are generated at the vibrating nozzles and fall into ammonia water to be aged to ammonium diuranate(ADU) particles. The reaction products of ammonium nitrate etc. are washed off, then the particles are dried and calcinated to UO_3 particles at 500 ℃ in air. The UO_3 particles are reduced and sintered to UO_2 particles with about 97 % T.D. at 1600℃ under hydrogen atmosphere.

Mainly, coated layers of the coated fuel particles have a function to enclose fission products generated in fuel kernel. The mechanical strength of the coated layers depends on their thickness and sphericity. These characteristics strongly depend on the diameter and sphericity of the kernel. Therefore it is essential to establish the fabrication technology to obtain the kernels with more uniform diameter and excellent sphericity. The vibrating nozzles from which droplets are emitted with high speed were developed for such kernels fabrication. The vibrating nozzles can emit droplets with uniform diameter continuously since the diameter of a droplet is determined by the combination of the flow rate of metal solution and the frequency of the nozzles as indicated in the equation (1). The diameter of a droplet is controlled to have the same uranium content as that of a fuel kernel.

$$Q = \frac{\pi \cdot D^3}{6} \cdot f \qquad (1)$$

where Q : flow rate of metal solution
D : diameter of droplet
f : frequency of vibrating nozzle

Most of the degradation of kernel sphericity is caused by the deformation at the stage of droplet formation and wet-ADU particle. At the stage of droplet formation, a process was applied to prevent the deformation of droplets when landing on the ammonia water. In the process, droplets are solidified while falling in ammonia gas blown against the droplets. At the stage of wet-ADU particle, a process was applied in which aging, washing and drying are carried out in the same conical dryer. It is possible to mitigate the impact against the particles by retaining very soft wet-ADU particles in the same conical dryer during the operations, and came to prevent the deformation.

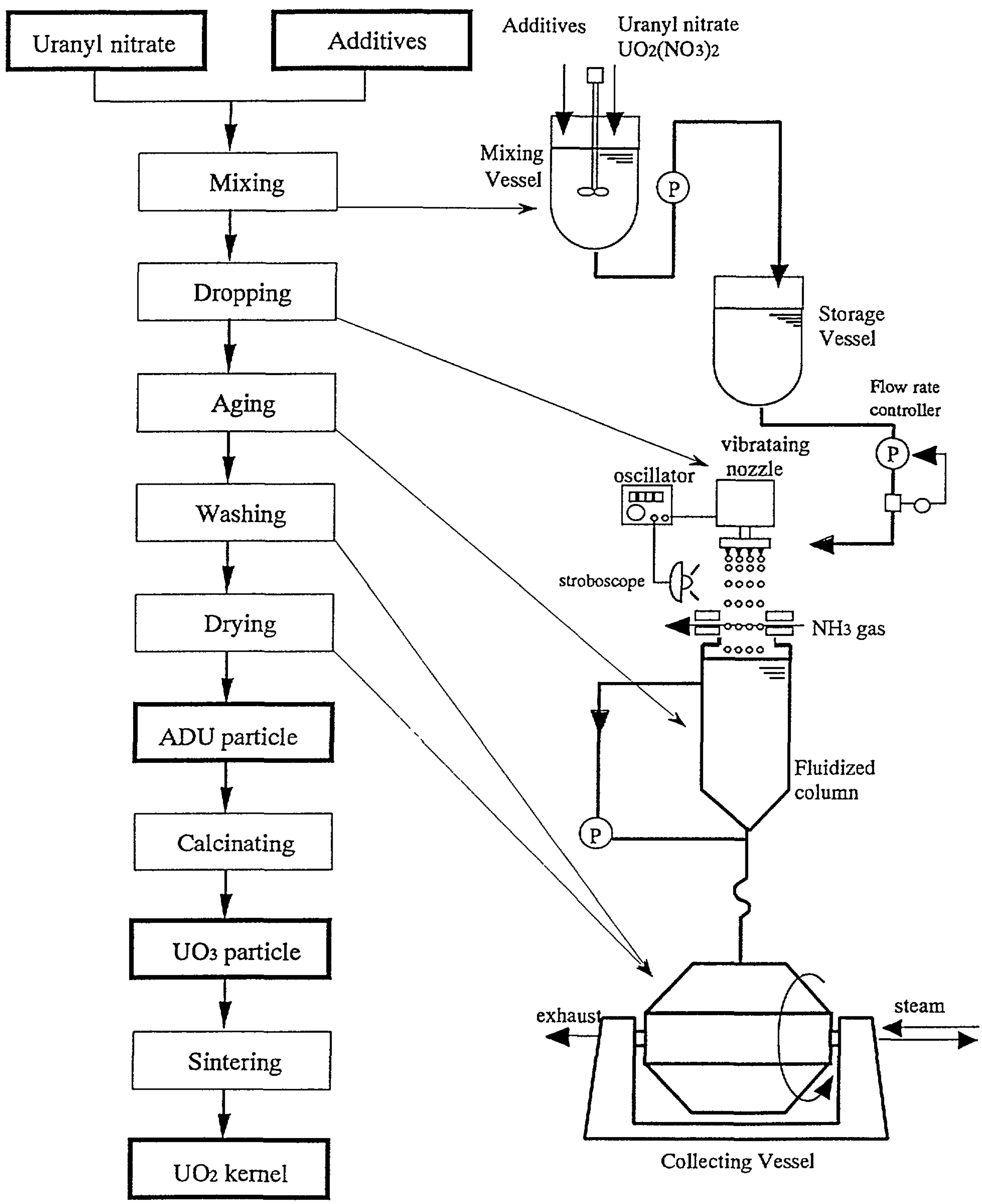


FIG. 3 FABRICATION PROCESS OF UO_2 KERNELS

Figure 4 shows the inspection result of kernel diameter for enrichment lots. Almost all standard deviations were less than 10 μ m and uniform diameter of kernels were obtained. Figure 5 shows the inspection result of sphericity of kernel for enrichment lots. The average of each lot was about 1.05 which indicated the sphericity was very much excellent.

3.2 Coated Fuel Particle

Figure 6 shows the fabrication flow diagram of coated fuel particles. A coated fuel particle consists of a kernel and four coating layers formed around the kernel by vapor-deposition technology using a fluiodized bed type of coater. Mixing gases of acetylene(C_2H_2) and argon are used for the deposition of porous and low density pyrolytic carbon(PyC) for the first layer ; propylene(C_3H_6) and argon for the deposition of dense pyrolytic carbon for the second and fourth layer; methyl-trichloro-silane(MTS) and hydrogen for the deposition of silicone carbide(SiC) for the third layer. After first and second layer coating, test specimen for density measurement etc. are sampled.

At the fabrication stage of coated fuel particles, failure fraction of coating layers other than the thickness and density is most important to ensure the function to retain fission products inside the particles. Causes of coating layer failure might be broadly classified into two groups. One is to be the mechanical impact against the particles caused by the intermediate loading and unloading of the particles for the previous coating process. The other is to be the random strong collisions between particles themselves during coating. With respect to the former one, developing the continuous coating process to eliminate intermediate loading and unloading, the failure fraction was reduced. A sampling method

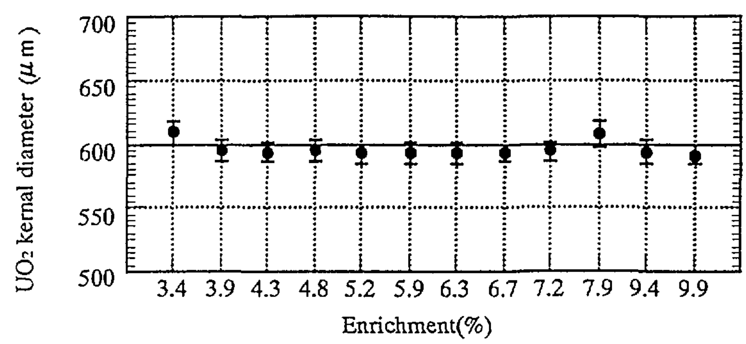


FIG. 4 DIAMETER OF UO_2 KERNEL

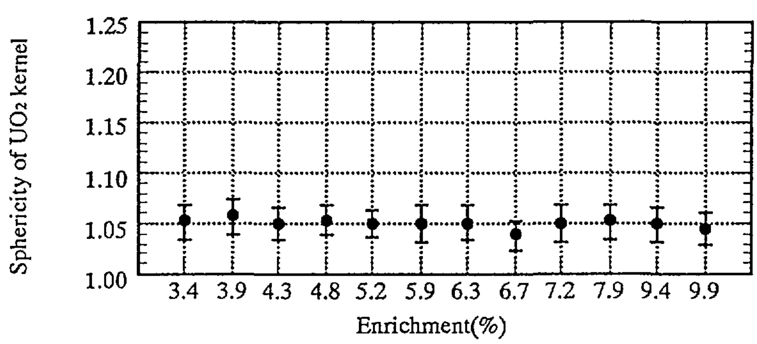


FIG. 5 SPHERICITY OF UO_2 KERNEL

UO2 kernels

↓

First layer coating
Low density pyrocarbon)
C_2H_2+Ar

↓

Second layer coating
(high density pyrocarbon)
C_3H_6+Ar

↓

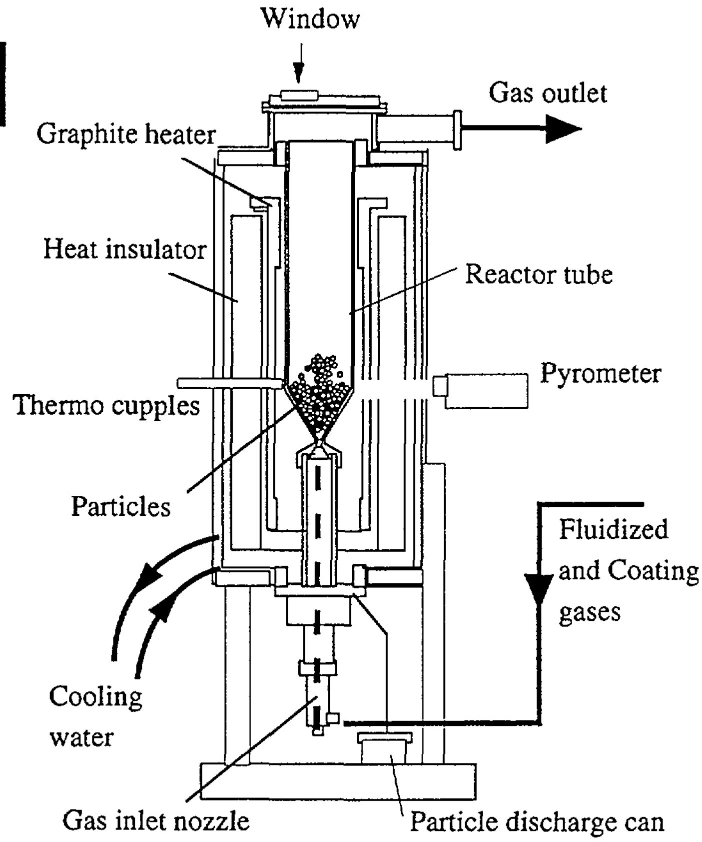
Third layer coating
(Silicon carbide)
CH_3SiCl_3+H_2

↓

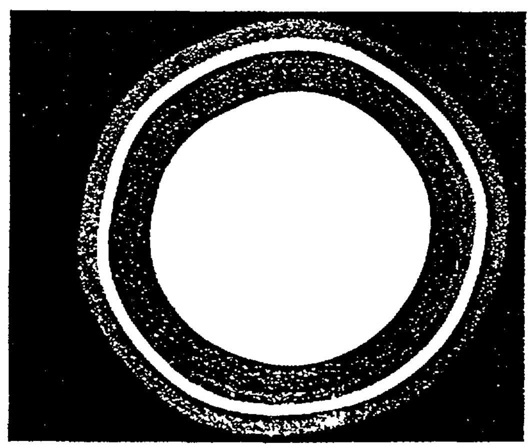
Fourth layer coating
(high density pyrocarbon)
C_3H_6+Ar

↓

TRISO type coated fuel particles



Coater



Polished cross section of coated fuel particle

FIG. 6 FABRICATION PROCESS OF COATED FUEL PARTICLE

was also developed to take out a little amount of particles by stopping the gases supply in a moment. With respect to the latter one, gases flow rate, mixing ratio of gases and coating temperatures were optimized empirically to mitigate the random collisions and realize mild flowing condition of particles. As the result, SiC defective fraction came down to the level of 10^{-6}. Figure 7, 8 and 9 show the inspection results of diameter, layer thickness and density of the coated particles respectively, which are more than satisfactory compared with the specifications.

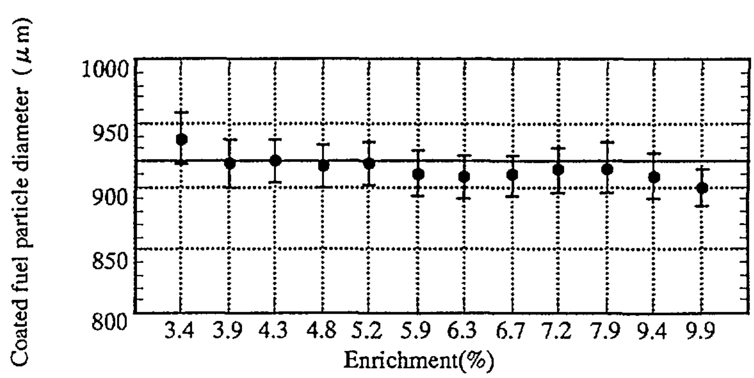


FIG. 7 DIAMETER OF COATED FUEL PARTICLE

In addition to the development of the fabrication technology, NFI developed the automatic inspection system to measure the coating layer thickness to enhance the reliability as well as productivity of the coated fuel particles. The system is composed of a video-microscope and an image processor, which can provide coated layer thickness of 100 particles on X-ray film in a short period of time. The image processor scans the film in X- and Y-direction to collect data of density (darkness) and detect the borders of coating layers of a coated particle.

3.3 Fuel Compact

The fabrication flow diagram is shown in Fig. 10. First, natural graphite powder, electro-graphite powder and a binder are mixed, then the mixture makes graphite matrix after fine grinding process. Coated fuel particles are over-coated with the graphite matrix and warm-pressed to make annular cylinder of green compacts. Green compacts are preliminarily heat treated for carbonization at 800℃ under nitrogen atmosphere, then sintered at 1800℃ under vacuum to make fuel compacts.

Among those processes, the green compact pressing is most complicated, which involves weighing, pre-heating, loading and unloading of over-coated particles and printing a identification number on a green compact and so on. As shown in Fig. 10, the full-automatic pressing system was introduced to rationalize the process. Certain amount of over-coated particles for a green compact are automatically sampled and weighed by the automatic weighing instruments. Weighing accuracy is better than ±0.05 % of the set weight. Weighed over-coated particles are heated in a furnace to soften the graphite matrix preliminary, then poured into dies of the warm-pressing machine which is a rotary hydraulic pressing machine with 8 sets of die and punch. After warm-pressing for about 15 minutes, green compacts are taken out to the printing station where a mark of ^{235}U enrichment and a serial number are printed on a green compact using a ink jet printer. The ink was selected so that the mark and number should be legible even after the following sintering process at 1800℃ under vacuum and produce very little nuclear effect on the reactor operation. The system can make 230 green compacts in about 8 hours.

In order to improve the failure fraction of coating layers at the stage of a fuel compact, it is necessary to disperse coated fuel particles in a green compact as uniformly as possible.

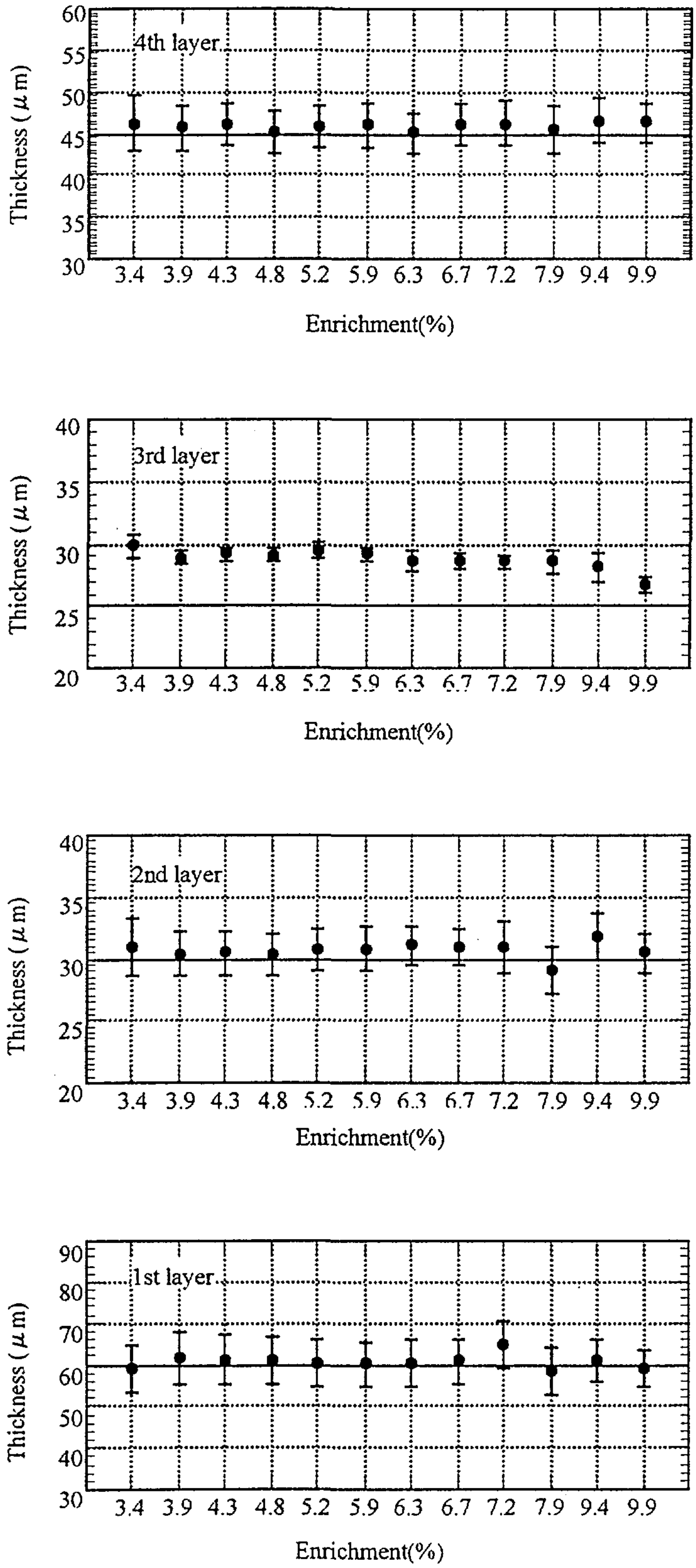


FIG. 8 LAYER THICKNESS OF COATED FUEL PARTICLE

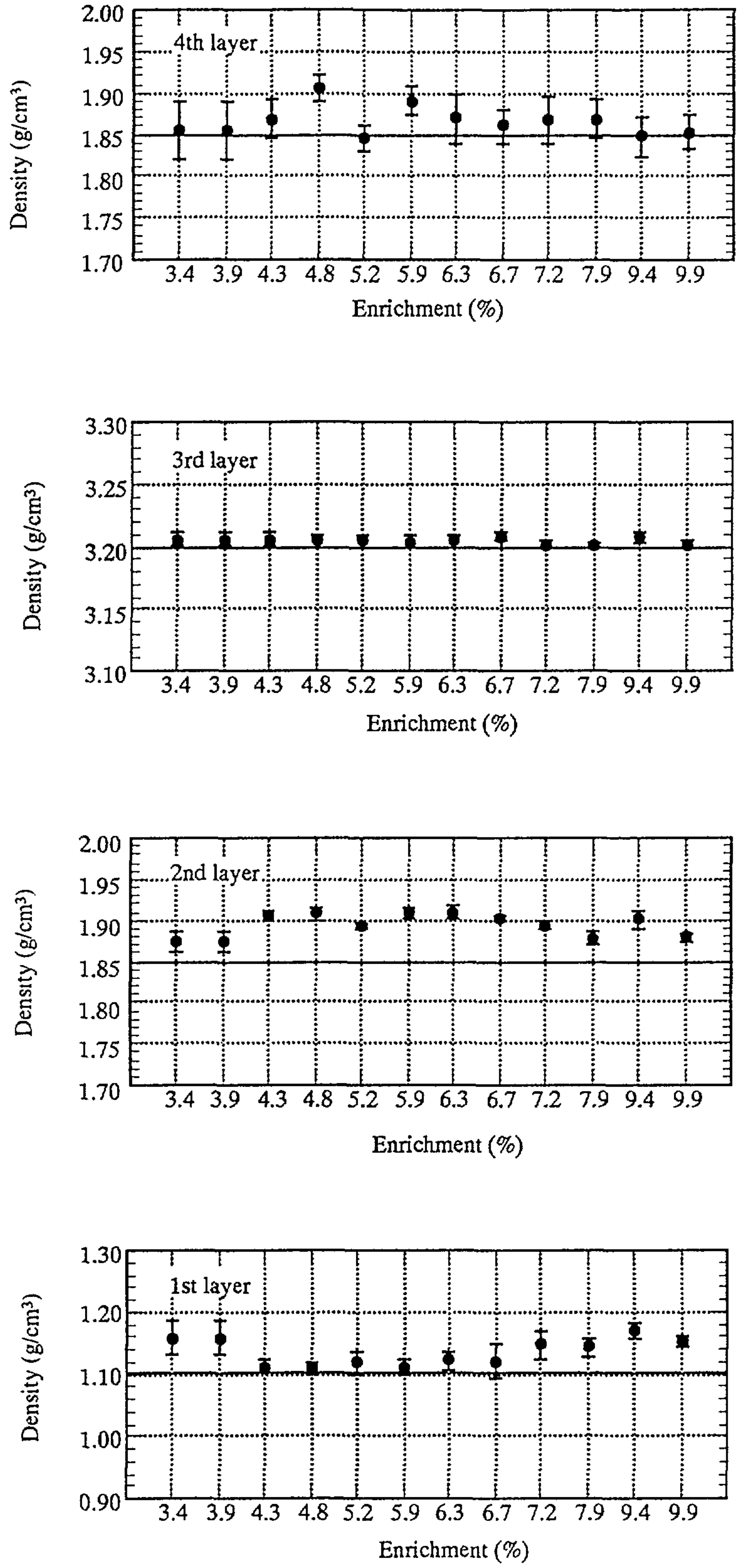


FIG. 9 LAYER DENSITY OF COATED FUEL PARTICLE

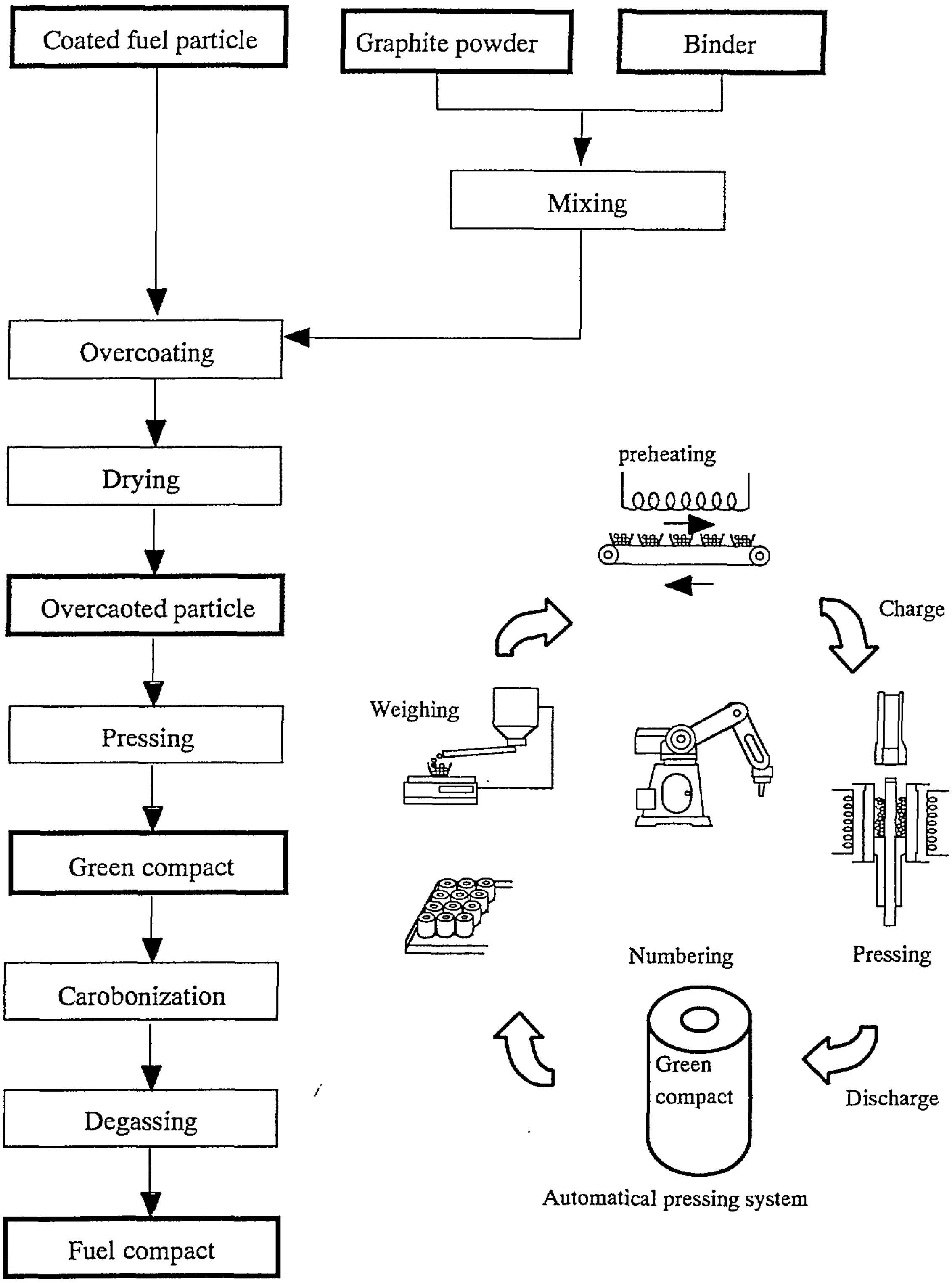


FIG. 10 FABRICATION PROCESS OF FUEL COMPACT

From this point of view, the punching speed and die temperature during the warm-pressing were optimized to fix the best timing of the softening by heating and the plastic flow of matrix graphite by pressing[2]. As the result, it was realized that average bare uranium fraction and SiC defective fraction for all fuel compact lots were 2×10^{-6} and 8×10^{-5} respectively[3,4]. Fig. 11 shows the final result of SiC defective fraction for all lots.

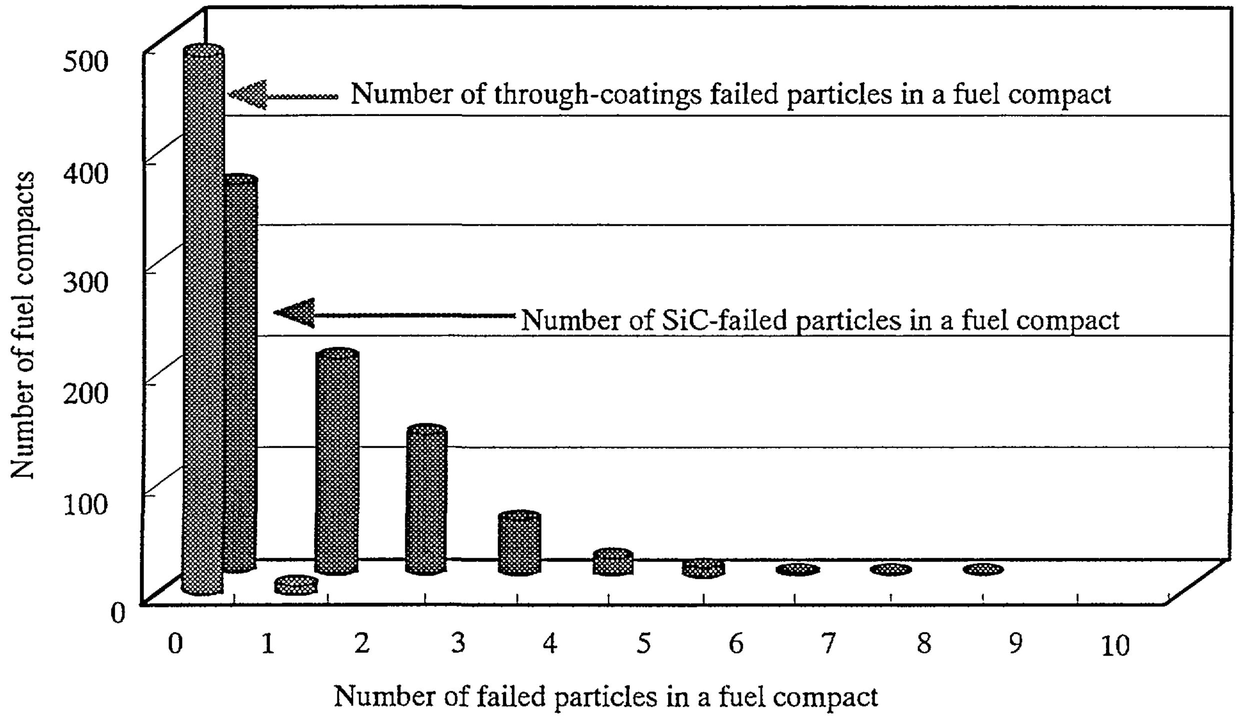


FIG. 11 NUMBER OF FAILED PARTICLES IN A FUEL COMPACT

4. Commercial Production of HTR Fuel

Fuel fabrication cost depends largely on the rate of plant operation and the production. According to the information based on the fabrication experience with pebble type of fuels in former West Germany, they said that the production scale over one million of pebbles per year (about 10 tU/y) would have the effect of mass production.

The HTR fuel plant of NFI were designed and constructed so that the plant could fabricate the first loading fuel and reload fuels just for the HTTR in accordance with the schedule following the construction and operation plan for the HTTR. The fabrication equipment including incidental facilities installed were developed based on the HTR fuel fabrication technology accumulated over 30 years in the past and NFI's plant design technology for the fabrication of light water reactor (LWR) fuels. The equipment could be easily introduced to the above-mentioned scale of the production. For example, the main equipment of two lines of kernel and coated fuel particle fabrication could produce 5 tU/y of coated fuel particles as the case may be. Then, as a module, they could be installed as they are to a 10 tU/y scale of plant. On the other hand, if even a 5 tU/y scale of HTR fuel plant like NFI's plant with such equipment came to continuously produce 5 tU/y of fuels by taking measures listed below, it could be also said that HTR fuel fabrication substantially got into the commercial stage.

- reinforcement of a part of the facility for such as fuel storage, intermediate storage for semi-finished products, and so on
- automatic handling for fuels and semi-finished products

—reductions in cost for raw materials, consumables and tools
—reductions in the consumption of raw materials and consumables
—rationalization of the fuel specification
—reductions of inspection items and frequency
—rationalization of the inspection to be made by the client or competent authority

5. Conclusion

After establishing the highly developed fabrication technology for HTR fuel, NFI successfully completed the fabrication of about 900 kgU of the HTTR first loading fuel last December. The quality of the fuel was entirely satisfactory. In particular, average bare uranium fraction and SiC defective fraction for all fuel compact lots were 2×10^{-6} and 8×10^{-5} respectively.

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AN HTR COGENERATION SYSTEM FOR INDUSTRIAL APPLICATIONS
(Session 3)

AN HTR COGENERATION SYSTEM FOR INDUSTRIAL APPLICATIONS

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Abstract

Because of its favourable characteristics of safety and simplicity the high-temperature reactor (HTR) could become a competitive heat source for a cogeneration unit. The Netherlands is a world leading country in the field of cogeneration. As nuclear energy remains an option for the medium and long term in this country, systems for nuclear cogeneration should be explored and developed. Hence, ECN Nuclear Research is developing a conceptual design of an HTR for Combined generation of Heat and Power (CHP) for the industry in and outside the Netherlands.

The design of this small CHP-unit for industrial applications is mainly based on a pre-feasibility study in 1996, performed by a joint working group of five Dutch organisations, in which technical feasibility was shown. The concept that was 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 ACACIA. The output of this installation is 14 MW electricity and 17 tonnes of steam per hour, with a pressure of 10 bar and a temperature of 220 ° C. The economic characteristics of this installation turned out to be much more favourable using modern data.

The research work for this installation is embedded in a programme that has links to the major HTR projects in the world. Accordingly ECN participates in several IAEA Co-ordinated Research Programmes (CRPs). Beside this ECN is involved in the South African PBMR-project. Finally, ECN participates in the European Concerted Action on Innovative HTR.

1. Introduction

Background

The development of the helium cooled graphite moderated high-temperature reactor (HTR) is taking place for more than thirty years now. Safety studies have indicated that an HTR with spherical fuel elements has very favourable safety characteristics: the loss of coolant as well as graphite fires will not result in any significant fuel damage. Because of the relatively high temperatures of the system, the HTR would be very suitable as a heat source for combined generation of heat and power (CHP). This application complements major HTR developments elsewhere in the world: the South-African electric utility ESKOM is investigating the use of modular HTR reactors for expanding their electric generating capacity [1], and INET of China and JAERI of Japan are building an HTR test reactor for the development of nuclear process heat systems [2, 3].

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In 1996, a joint working group in the Netherlands evaluated the technical and economical feasibilities of a nuclear cogeneration installation with a thermal power of 40 MW [4, 5, 6][2], which was based on the (Dutch) market demand for CHP-units.

Cogeneration of Heat and Power

Forecasts on the future consumption and production of energy indicate an expanding world market for the combined generation of heat and power. 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 percent of the total electricity supply in the Netherlands has been generated by decentralised units and autoproducers. Another 33 percent has been generated by large natural gas fired power plants. Given the expected further depletion of the indigenous resources of natural gas (the fuel for CHP), a potential market could emerge for an alternate primary energy source within the next two decades, see figure 1. Nuclear energy could be one of the substitutes, if competitive prices and public acceptance for this new nuclear application can be achieved.

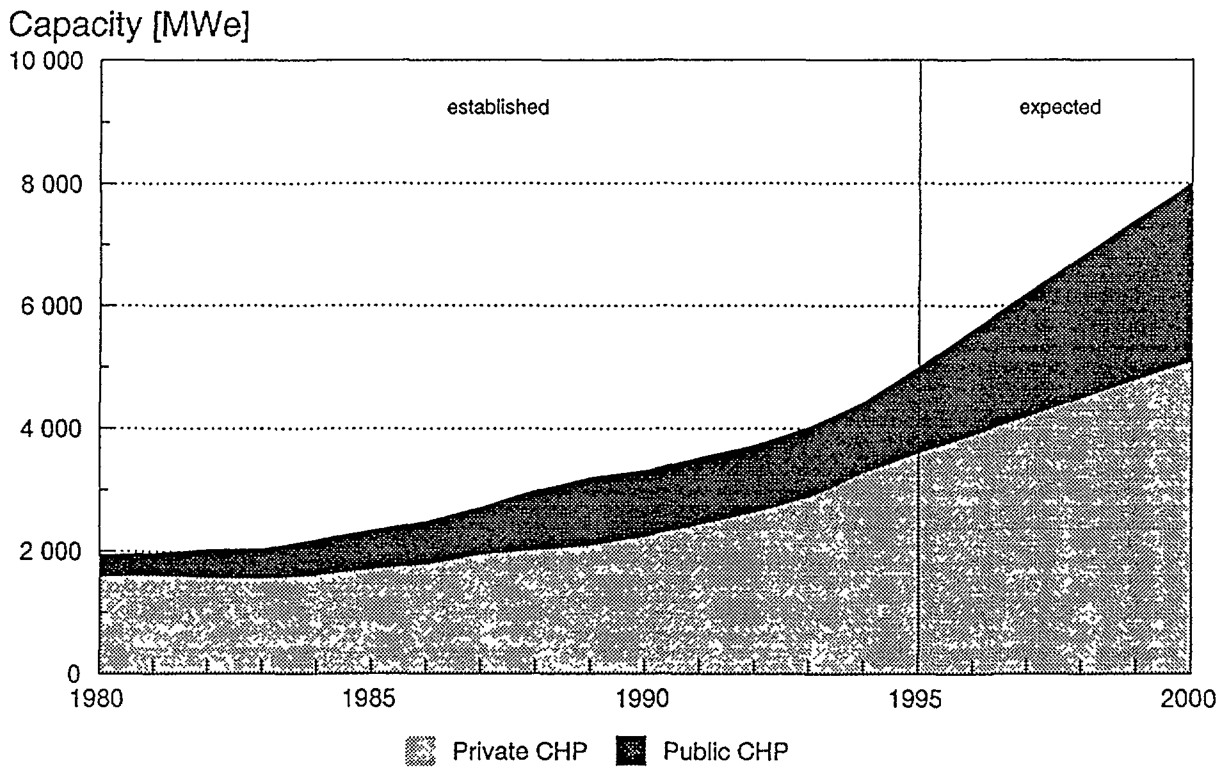

Figure 1: *Development of CHP in the Netherlands.*

Reference Design for an Industrial Cogeneration Unit

The reference configuration of the CHP-unit is based on a reactor design by KFA Jülich, Germany [7], and an energy conversion system design by Longmark Power International (LPI) of Cambridge, USA [8]. One of the major results of the pre-feasibility study was, that there are no technological barriers to build a 40 MW thermal cogeneration unit. However a second look at economics, market potential and licensing aspects is highly recommended to establish this new breed of innovative reactor technology.

Consequently, the design features of the pre-feasibility study concept has been changed slightly to improve a small cogeneration unit for industrial applications with a heat output corresponding more to

[2] The summary report of the pre-feasibility study is also available on Internet: http://www.ecn.nl/unit_nuc/research/htr/main.html

the needs of the (Dutch) industry in terms of steam amount and conditions. Beside this an electrical output of 13.6 MW will be delivered by the CHP-unit, named ACACIA, which is an acronym for AdvanCed Atomic Cogenerator for Industrial Applications.

The design features of the ACACIA unit include:
- HTR based versatile heat source combined with a closed (Brayton) cycle energy conversion system,
- design thermal power of the reactor core of 40 MW,
- spherical fuel elements with a 60 mm diameter containing small (0.9 mm diameter) fuel particles according to the German TRISO design,
- loading mechanism of fuel into the core - as long as space allows - during operation, which is generally called the 'peu à peu' fuelling concept,
- resistance against accidents characterised by the two scenarios loss of flow and loss of coolant, both combined with an anticipated transient without scram (ATWS) condition,
- economic and simplified design by reduction of the systems, structures and components that need to meet nuclear qualification requirements.

2. Basis Configuration

The basis configuration of the ACACIA unit is a helium-cooled graphite moderated nuclear reactor with a thermal power of 40 MW and a core exit temperature of 800 °C. The nuclear power is generated in the reactor vessel which contains standard pebble bed HTR spherical fuel elements with a diameter of 60 mm, see figure 2.

The nuclear power is transferred to helium cooling gas which is pressurised to 2.3 MPa (23 bar). The helium is expanded to 1.0 MPa in a helium turbine. The turbine drives a helium compressor and generates electrical power in the attached generator. The expanded helium will be used to preheat compressed helium to increase the efficiency of the thermodynamic cycle and also for heat generation, through an intermediate helium circuit, for industrial applications. The remaining low-temperature waste heat will be rejected to the environment. The cooled-down helium will be compressed and directed to the reactor vessel where it will be heated again. The main design parameters of the conceptual ACACIA configuration are presented in table 1.

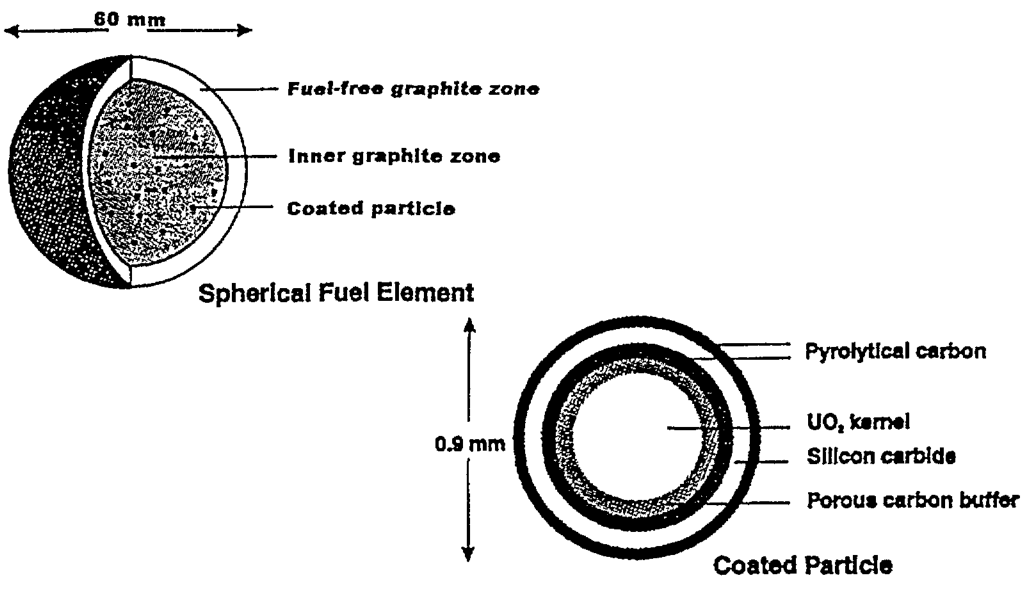

Figure 2: *Fuel particles in spherical fuel elements [9]. About 11000 of these particles are embedded in the graphite matrix of a fuel element.*

Table 1: *Main design parameters of ACACIA configuration.*

Characteristics of Main Components		Design Value
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	12
Generator		
Electrical Power	MW	13.6
Overall Performance		
Electric Efficiency ($P_{electric}/P_{reactor}$)	-	0.34
Heat Efficiency ($P_{heat}/P_{reactor}$)	-	0.30
Net Thermal Efficiency	-	0.64

The applied energy conversion system comprises a single-shaft turbine-compressor with a directly coupled electrical generator, see figure 3. The generator electrical power is 13.6 MW. An intermediate helium circuit heats a closed heat generation loop to deliver 17 tonness per hour superheated steam of 1 MPa and 220 °C. The remaining heat is rejected to the environment through a circulation loop of cooling water.

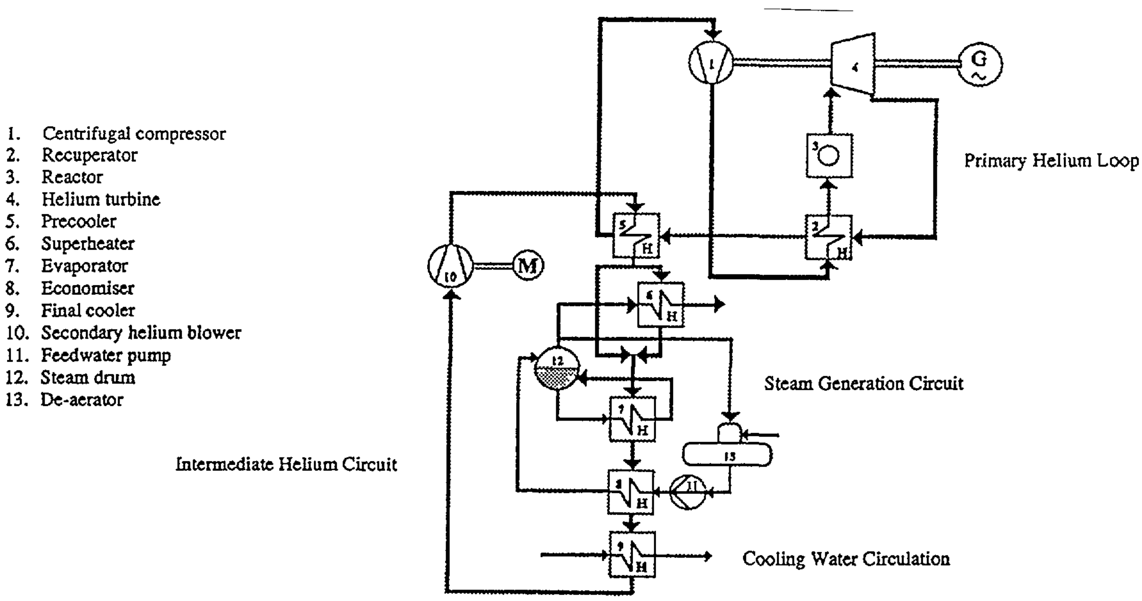

Figure 3: *Main components of ACACIA cogeneration system.*

3. Technical Analyses

During the study to the feasibility of a cogeneration unit, in 1996, scoping analyses in the field of reactor physics, safety and graphite oxidation have been achieved, and extensively outlined in public reports [4, 5, 6], as well as during the previous IAEA Technical Committee Meeting at ECN, Petten [10]. Therefore, this paper will only describe a summary of the differences between the INCOGEN[3] configuration of the pre-feasibility study and the ACACIA system, arising from the in this study given recommendations.

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 defuelling. Because of the foreseen period for inspection and maintenance of the power conversion system and the reactor pressure vessel a defuelling interval of 3 years has been chosen for the ACACIA system (versus 10 years in the INCOGEN study). This reduction of the defuelling interval has advantages for the core dimensions, in particular a significant decrease of the core height is expected.

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, for the time being the ACACIA core safety behaviour is expected similar.

Energy Conversion System

One of the major changes of the ACACIA installation related to INCOGEN is the energy conversion system. The INCOGEN concept is optimised for almost maximum 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 through market potential reasons. A consequence of this optimisation to the heat production is less electrical power and a lower overall performance (see table 2). But, these disadvantages makes no odds against the much higher market potential of the cogeneration unit for industrial applications, and that therefore becomes a more realistic opponent to the conventional (natural gas fired) industrial CHP systems.

Table 2: *Energy Conversion System comparison.*

Main Energy Conversion System Design Parameters		Design Value ACACIA	Design Value INCOGEN
Heat Generation			
Inlet / Outlet Temperatures	°C	80 / 220	40 / 150
Pressure	MPa	1	1
Transferred Heat	MW	12	18
Cogeneration Medium	-	Superheated steam	Hot water
Generator			
Electrical Power	MW	13.6	16.5
Overall Performance			
Heat Efficiency	-	0.30	0.45
Electric Efficiency	-	0.34	0.41
Power to Heat Ratio	-	0.88	0.92

[3] 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 is 16.5 MW electricity. The remaining heat will be used for cogeneration applications, which heats an external pressurised water circuit from 40 °C to 150 °C, useable for low temperature industrial processes or district heating networks.

The flow diagram of the ACACIA unit is shown in figure 3. 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 more expensive turbo-machinery. After the recuperator, the helium passes through the reactor-core (3), where it is heated. In the turbine (4), which drives the compressor and the generator, the helium is expanded. 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 with free power turbine. Then a high-pressure turbine is 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 drum (12), the intermediate helium first flows partly through a superheater (6). The bulk of the flow by-passes this heat-exchanger, because the steam has to be only 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 influences on the plant efficiency of some variation have been established with a steady-state model [11]. In order to test the control-structure, a dynamic model is currently under construction[4].

Plant Lay-Out

In figure 4 a cross section of the reactor and turbine building of an ACACIA unit is drawn[5]. The primary cycle of the energy conversion system is completely contained in a pressure vessel as shown in figure 5. The steam drum and final cooler are not shown in figure 4. This equipment is located on the same level as the gas-gas heat-exchangers and the turbo-machinery. The design of the secondary cycle is currently not integrated, but it is a simple series connection of tube and shell heat-exchangers.

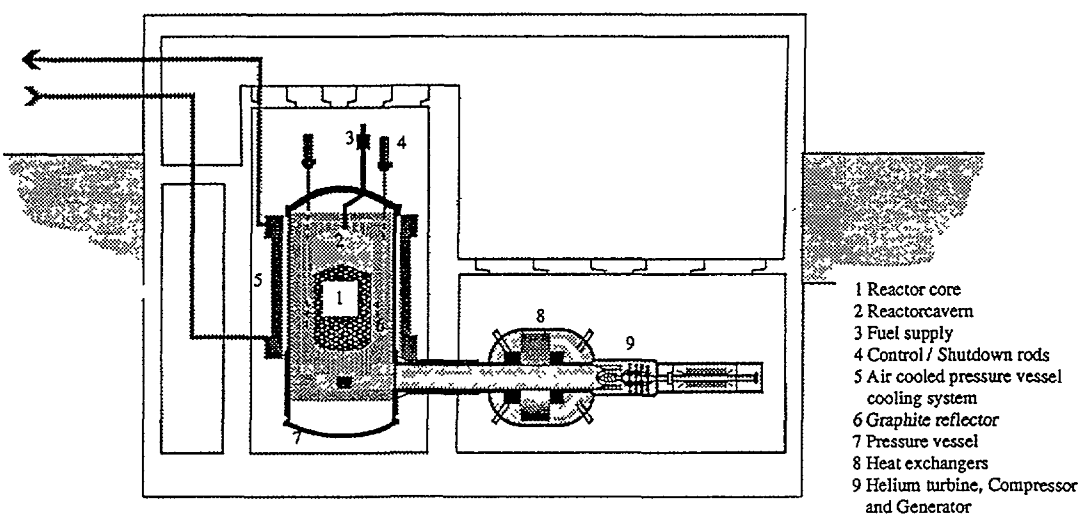

Figure 4: *Cross Section Reactor and Turbine Building of ACACIA unit.*

4 See paper *Transient Analysis for the HTR Coupled to the Energy Conversion System* of E.C. Verkerk during this Technical Committee Meeting.

5 Due to the somewhat shorter reactor pressure vessel the reactor building of the ACACIA plant is lower than for INCOGEN system.

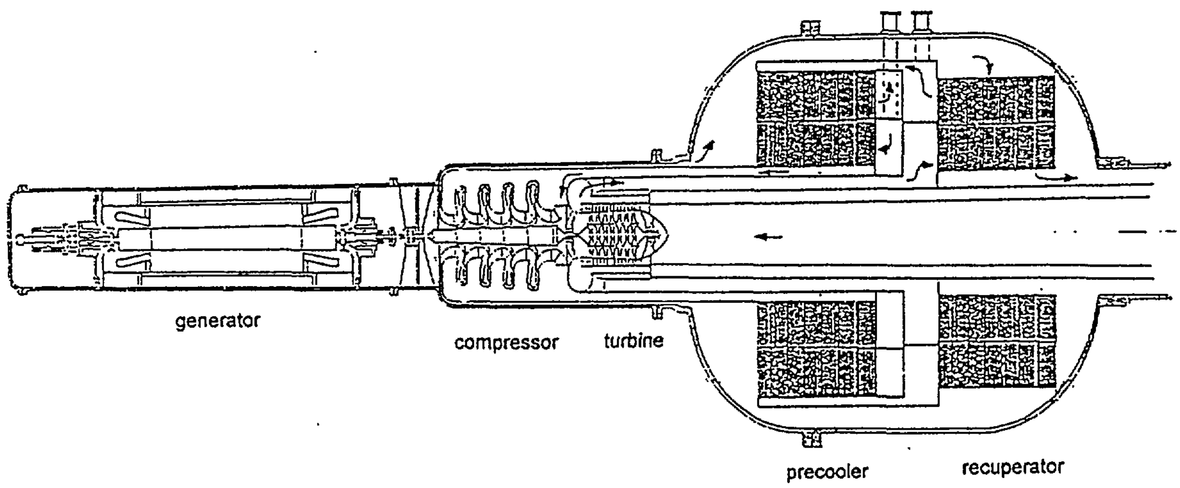

Figure 5: *Schematic View Primary Cycle of the Energy Conversion System [11].*

4. Economic Assessment

A reassessment of the economics was highly recommended in the INCOGEN pre-feasibility reports [4, 5], because the cost figures for INCOGEN were based on 'old fashioned' down-scaled Siemens data for the reactor and from down-scaled General Atomics data for the closed cycle helium turbine. This approach leads for the time being to the conclusion that the costs are too high compared to natural gas fired CHP units.

Hence, a second look to the economics has been carried out. This chapter will focus 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-design[6], that have been presented during the previous IAEA Technical Committee Meeting [12].

In the INCOGEN pre-feasibility study the investment cost and production cost have been estimated at 8909 Netherlands Guilders (NLG) per produced kWe 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/kWe and 0.106 NLG/kWh, respectively [13][7]. Due to the neglect of the simplified design of the smaller ACACIA system versus the PBMR an additional price reduction to be 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.

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6 Pebble Bed Modular Reactor system of 226 MWth for (100 MW) electricity production.

7 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.

5. Links with International HTR-programmes

To establish an ACACIA unit, in the beginning of the next century, it is essential that the economical and technological research work is embedded in an international HTR network. Figure 6 shows the links of the Dutch HTR-project with the major HTR-activities in the world.

ECN participates in several IAEA Co-ordinated Research Programmes (CRPs) for safety calculations, defined by the IWGGCR[8], on the Japanese HTTR[9] and the American / Russian GT-MHR[10] design. Besides this ECN is involved in the South African PBMR-project. Finally, ECN takes part, with partners from Germany, France, United Kingdom and Italy, in the European Concerted Action (CA) on Innovative HTR to define co-operative developments in the HTR field for the EU Fifth Framework Programme.

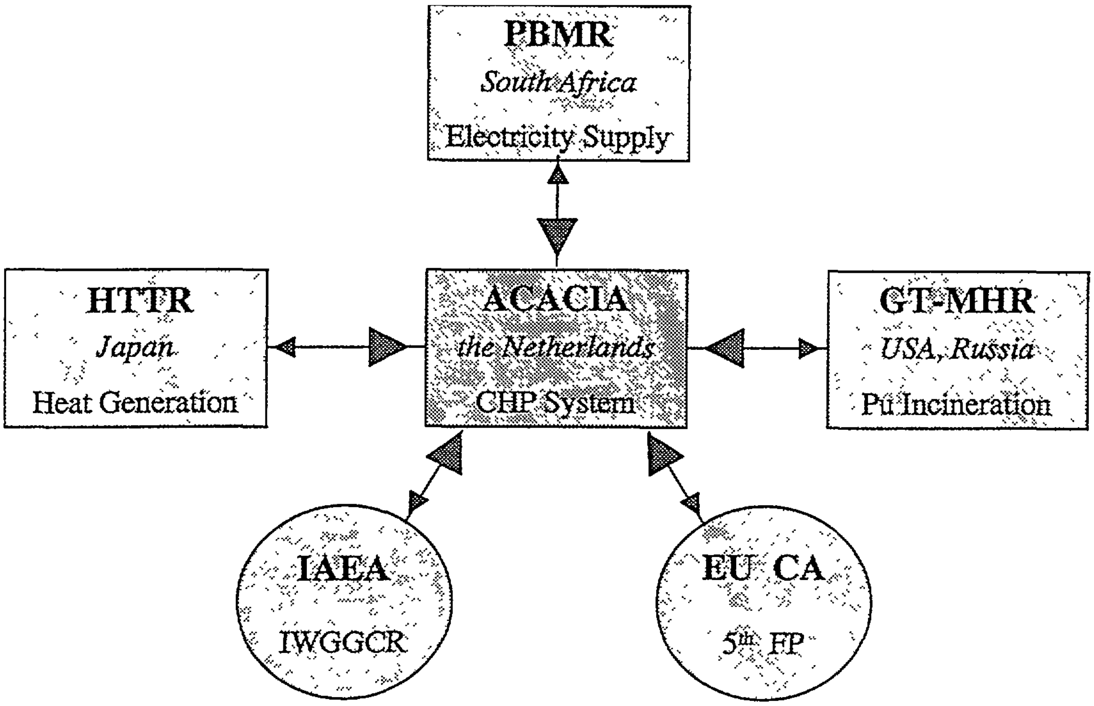

Figure 6: *Links with major HTR projects in the world.*

6. Conclusions

During the INCOGEN study, in 1996, technical feasibility of a 40 MW thermal cogeneration unit was shown. A reassessment of economics with modern data and a heat output corresponding more to the needs of the industry was highly recommended. Consequently, the INCOGEN concept has been changed slightly and has been renamed ACACIA.

ACACIA enters the competitive area with economics based on PBMR cost data translated to the Dutch situation. The superheated steam output (of 220 ^{0}C and 10 bar) is identified as interesting for a variety of heat consuming branches of industry.

8 The IAEA International Working Group on Gas Cooled Reactors (IWGGCR) aims for the exchange of information between member states regarding their Gas Cooled Reactor (GCR) programme, and advises the IAEA on major research activities in the GCR field.

9 High Temperature Test Reactor of 30 MW for heat generation [14].

10 Gas Turbine - Modular Helium Reactor [15].

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DEVELOPMENT PROGRAMME ON HYDROGEN PRODUCTION IN HTTR

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Abstract

In Japan Atomic Energy Research Institute, a hydrogen production system is being designed to produce hydrogen by means of a steam reforming process of natural gas using nuclear heat (10MW, 905℃) supplied by the High Temperature Engineering Test Reactor (HTTR). The safety principle and criteria are also being investigated in the HTTR hydrogen production system. A facility for an out-of-pile test prior to the demonstration test with the HTTR hydrogen production system is under manufacturing. The out-of-pile test facility simulates key components downstream an intermediate heat exchanger of the HTTR hydrogen production system on a scale of 1 to 30. The test on safety, controllability and performance of the hydrogen production system will be started in 2001 and continued for 4 years or longer. In parallel to this, a hydrogen permeation test and a corrosion test of a catalyst tube of a steam reformer are being carried out to obtain data necessary for the design of the system.

As for the HTTR hydrogen production system, a conceptual design is in progress, and check and review for the demonstration program will be made in 2000 from a financial point of view as well as technical view.

Following a brief overview of the program, the design achievements including safety philosophy so far and technical issues to be resolved are to be summarized in the paper.

I. INTRODUCTION

Research and development (R&D) for clean, economical, stable, safe and abundant energy should be promoted from a viewpoint of technology as a potential measure to mitigate the global warming issue as well as for massive and stable energy supply and utilization. We have various options as alternative energy for fossil fuels: solar, geothermal, hydropower and nuclear energy and so on. While available natural energy is limited due to its stability, quality, quantity and density, it is sure that nuclear energy by high-temperature gas-cooled reactors (HTGRs) has the potential to come up with a share as regards a satiable energy supply and utilization. Nuclear energy has been exclusively utilized for electric power generation, but the direct utilization of nuclear thermal energy is necessary and indispensable so that the energy efficiency can be increased and energy savings can be promoted in the near future. The hydrogen production is one of the key technologies for direct utilization of nuclear thermal energy.

The Japan Atomic Energy Research Institute (JAERI) has carried out the R&D on the High Temperature Gas-cooled Reactors (HTGRs), and started the construction of the High Temperature Engineering Test Reactor [1] (HTTR) in the Oarai Research Establishment in March 1991. The HTTR is a test reactor with thermal output of 30 MW and outlet coolant temperature of 950 ℃, and has capability to demonstrate a nuclear heat utilization system. An intermediate heat exchanger (IHX) installed in a reactor containment vessel can supply thermal heat of 10 MW to the heat utilization system.

A hydrogen production system by steam reforming of natural gas, chemical reaction; $CH_4+H_2O=3H_2+CO$, is to be the first heat utilization system of the HTTR. It is the reason that its technology matured in fossil-fired plant enables the coupling with the HTTR in the early 2000's and technical solutions demonstrated by the coupling will contribute to other hydrogen production systems such as water splitting by a thermochemical method [2]. From Science and Technology Agency (STA) of Japan, the JAERI has been entrusted with the R&D of the HTTR hydrogen production system in January 1997. The R&D of the HTTR hydrogen production system consists of studies of design and safety of the HTTR hydrogen production system, an out-of-pile test, and components tests, which are a hydrogen/tritium permeation test and a corrosion test, necessary for the construction of the HTTR hydrogen production system. The outline of the R&D and safety-related technology for the hydrogen production system in HTTR are reported in the following.

Ⅱ. DEMONSTRATION PROGRAM OF HYDROGEN PRODUCTION IN HTTR

Figure 1 shows the development schedule of the demonstration program of the hydrogen production in the HTTR. The first half of the demonstration program boxed by solid lines in Fig.1 is from 1997 to 2004, and the last half boxed by dotted lines is planned until 2010. The execution of the last half is decided after taking the check and review (C&R) in 2000. The design and safety studies of the HTTR hydrogen production system are carried out until 2000. The facility of the out-of-pile test for the HTTR hydrogen production system is designed and constructed in 1997-2000, then the out-of-pile test is performed until 2004. In the component tests, two experimental apparatus were completed in 1997, and the hydrogen/tritium permeation test and the corrosion test are being carried out in 1998-2000. After the C&R of the HTTR hydrogen production system, the construction of the HTTR hydrogen production system will be started from 2001 and the demonstration test will be started in around 2005-2010.

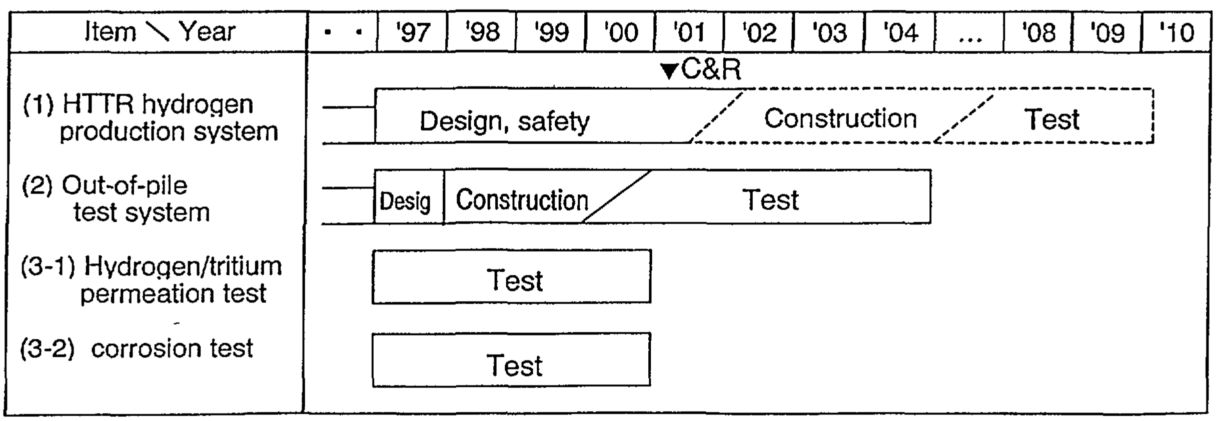

Fig.1 Schedule of demonstration program.

Ⅲ. HTTR HYDROGEN PRODUCTION SYSTEM

1. Overview of HTTR hydrogen production system

The HTTR hydrogen production system is designed to utilize the nuclear heat effectively and achieve hydrogen productivity competitive to that of a fossil-fired plant with operability, controllability and safety acceptable enough to commercialization. Figure 2 shows an arrangement of the main components. The HTTR reactor supplies nuclear heat of 10MW with 950℃ to the IHX in the reactor cooling loop, and then the nuclear heat is transferred from the IHX to the secondary helium loop to be utilized for the production of hydrogen. Due to heat loss along the secondary helium piping from the IHX to a steam reformer (SR), the secondary helium temperature is reduced to 880℃ at the SR inlet, whereas the IHX outlet temperature is 905℃. Design specifications of the HTTR hydrogen production system is shown in Table Ⅰ. The key components, such as the

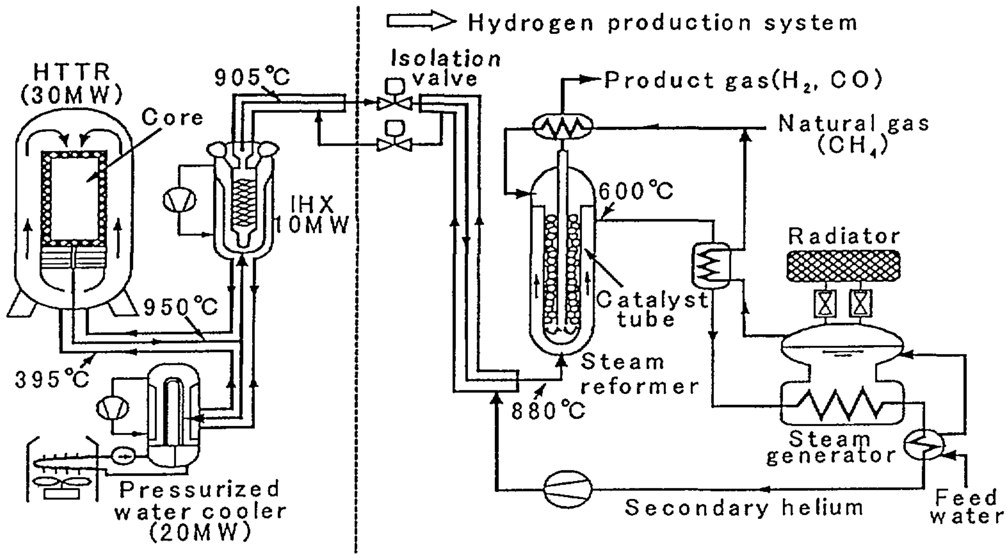

Fig. 2 Flow scheme of HTTR hydrogen production system.

Table Ⅰ Design specifications of HTTR hydrogen production system and out-of-pile test facility.

Items	HTTR	Out-of-pile
Pressure		
Process-gas / helium-gas	4.5 / 4.1 MPa	
Temperature inlet at steam reformer		
Process-gas / helium-gas	450 / 880℃	
Temperature outlet at steam reformer		
Process-gas / helium-gas	600 / 600℃	600 / 650℃
Natural gas feed	1296kg/h (81kmol/h)	43.2kg/h (2.7kmol/h)
Helium gas feed	8748kg/h	327.6 kg/h
Steam-carbon ratio (S/C)	3.5	2~4
Hydrogen product	3800Nm³/h	110Nm³/h
Heat source	Reactor (10MW)	Electric heater (380kW)

SR and a steam generator (SG), and their arrangement were designed to achieve hydrogen productivity competitive to that of a fossil-fired system with operability, controllability and safety acceptable enough for commercialization [3].

2. Safety-related technology to be developed

Although the hydrogen production system by steam reforming is matured in fossil-fired plants, some safety-related technology should be developed for coupling with HTGRs as well as the HTTR as shown in Fig. 3.

(1) Mitigation of thermal disturbance to reactor

The SG supplies steam to the SR, and can also stabilize the inlet temperature of the IHX in the secondary helium coolant loop. Even if the helium temperature at the SR outlet, that is, the SG inlet is increased by some thermal disturbance such a malfunction in the reforming process gas line, the helium temperature at the SG outlet can be kept constant at the saturation temperature of steam by controlling the pressure in the SG. It is possible that the nuclear reactor can be stopped according to a normal operation procedure but not with a reactor scram for some malfunction or accident at the heat utilization system by this performance of the SG working as an absorber of thermal disturbance. We aim to limit the temperature fluctuation of the secondary helium gas within 10℃ at the SG outlet, because the temperature rise above 15℃ compared with the normal temperature at the reactor inlet causes the HTTR reactor scram.

(2) Assurance of structural integrity of catalyst tube

(a) Control of pressure difference between helium and process gases at catalyst tube

The catalyst tube in the SR is a component important to safety, because it forms a pressure boundary between helium and process gases. In design of the catalyst tube, its wall thickness should be decided considering both outer pressure of helium gas at 4.1MPa and inner pressure of process gas at 4.5MPa to assure the structural integrity in all conditions such as not only normal start-up and shut-down but also malfunction and accident at the heat utilization system. This design, however, makes the wall thickness too heavy; for example, the wall thickness becomes about 130mm as against an inner diameter of 128mm using Alloy 800H. To realize the reasonable wall thickness, 10-mm level, it should be decided considering the pressure difference between helium and process gases, and a control system is required to keep the pressure difference within an allowable value.

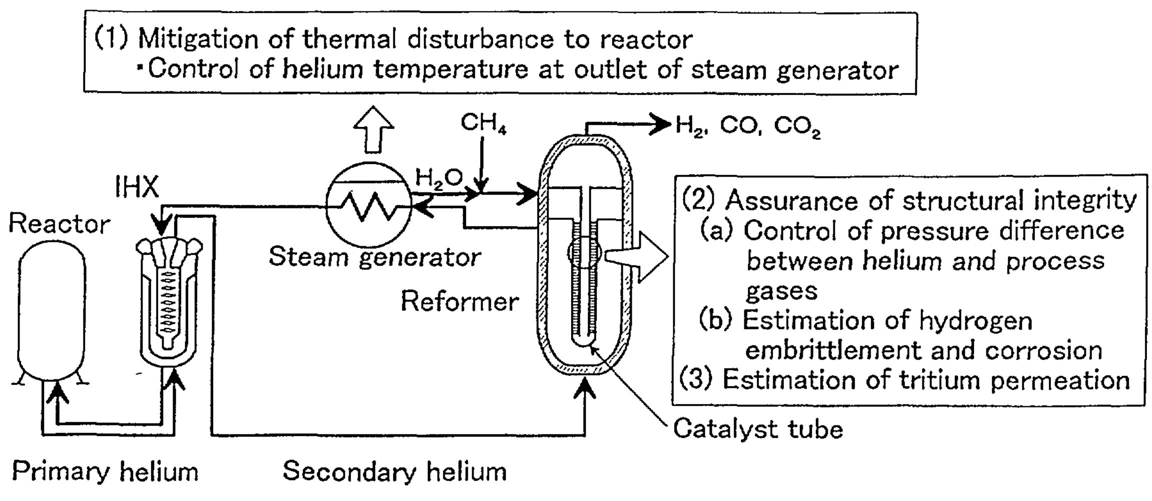

Fig. 3 Safety related technology to be developed for HTTR hydrogen production system.

In concrete terms, the control system makes the process gas pressure follow pressure change of helium gas.

(b) Estimation of hydrogen embrittlement and corrosion of catalyst tube

The catalyst tube will be made of Hastelloy XR, which is a nickel-base, helium corrosion- and heat-resistance super alloy developed for the HTTR in the JAERI, and its strength in high temperature is nearly equal to that of Alloy 800H. In order to verify the validity of the structural design of the HTTR hydrogen production system, characteristics of corrosion due to metal dusting and oxidation and strength reduction due to hydrogen embrittlement should be estimated in the corrosive gases such as CH_4, CO, H_2O and H_2 simulating the SR condition.

(3) Estimation of tritium permeation

Tritium produced in the HTTR core flows with the primary helium coolant to the IHX, then permeates through the Hastelloy XR tube of the IHX to the secondary helium coolant and through the Hastelloy XR tube of the SR, at last, mixes with the process. Therefore tritium concentration in the process gas should be estimated because tritium can not be perfectly removed by a purification system in the HTTR.

Ⅳ. OUT-OF-PILE TEST

1. Objective and test facility

The main objectives are investigation of transient behavior and establishment of operation and control technology, focussing on establishment of the safety-related technology (1) and (2-a) described in section Ⅲ.2, as well as design verification of performance of high temperature components, such as the SR and SG.

The test facility has an hydrogen production capacity of 110Nm^3/h and simulates key components downstream the IHX of the HTTR hydrogen production system on a scale of 1 to 30 [4]. Design specifications of this test system is also shown in Table Ⅰ. Figure 4 shows a schematic flow diagram of the test facility. An electric heater with 380kW is used

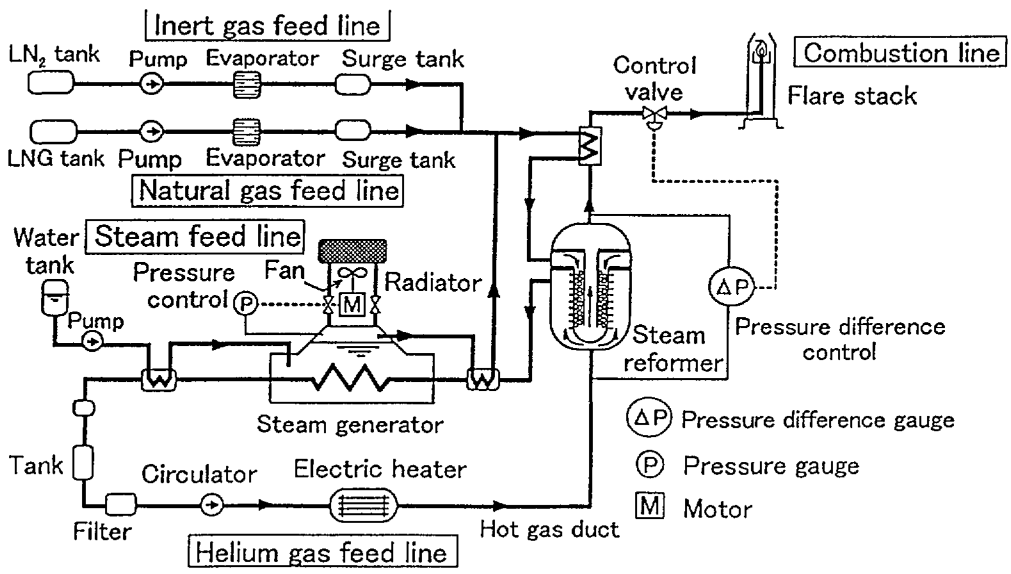

Fig. 4 Schematic flow diagram of the out-of-pile test facility.

as a heat source instead of the nuclear heat in order to heat helium gas up to 880℃ at the SR inlet of the same conditions as the HTTR hydrogen production system. The process gas pressure is controlled by a control valve installed downstream from the SR, monitoring the pressure difference between helium and process gases.

Figure 5 shows the schematic view of the SR which has one bayonet-type catalyst tube made of Alloy 800H. Dimensions such as diameter and length is approximately the same as those of the catalyst tube of the HTTR hydrogen production system. By the way, the SR of the HTTR hydrogen production system has 30 bayonet-type catalyst tubes made of Hastelloy XR. Process gas flows in the catalyst tube at inlet temperature of 450℃ and helium gas flows in a channel between catalyst and guide tubes at inlet temperature of 880℃. In the fossil-fired plant, the process gas receives heat from combustion air of about 1200℃ by radiation, and the heat flux at the outer surface of the catalyst tube reaches 70,000-87,000 W/m^2. In order to achieve the same heat flux as that of the fossil-fired plant, it is very important to promote heat transfer of helium gas by forced convection because the temperature of helium gas, that is, the temperature of heat source is too low compared with that of the fossil-fired plant. So, disc-type fins, 2mm in height, 1mm in width and 3mm in pitch, are arranged around outer surface of the catalyst tube in the test facility in order to increase a heat transfer coefficient of helium gas by 2.7 times, 2150W/m^{2}K with the fins, and a heat transfer area by 2.3 times larger than those of smooth surface, respectively. As the result, the heat transfer performance of the catalyst tube in the test facility becomes competitive to that of the fossil-fired plant.

2. Test plan

The tests are considered on three categories, that is, (i) normal start-up and shut-down test to establish the operation method, (ii) safety-related test dealing with malfunction and accident at process and helium gas lines, and (iii) high temperature components test to investigate the performance.

(1) Normal start-up and shut-down test

The normal start-up and shut-down of the hydrogen production system is carried out following in those of the HTTR reactor. The change of temperature and pressure of helium gas causes fluctuation of the reforming reaction in the SR attended with fluctuation of both helium gas temperature at the SR outlet and process gas pressure in the catalyst tube. The objective of the test is optimization of feed of natural gas and steam according to change of the temperature and pressure of helium gas supplied from the HTTR reactor in order to restrain the above fluctuation within allowable range.

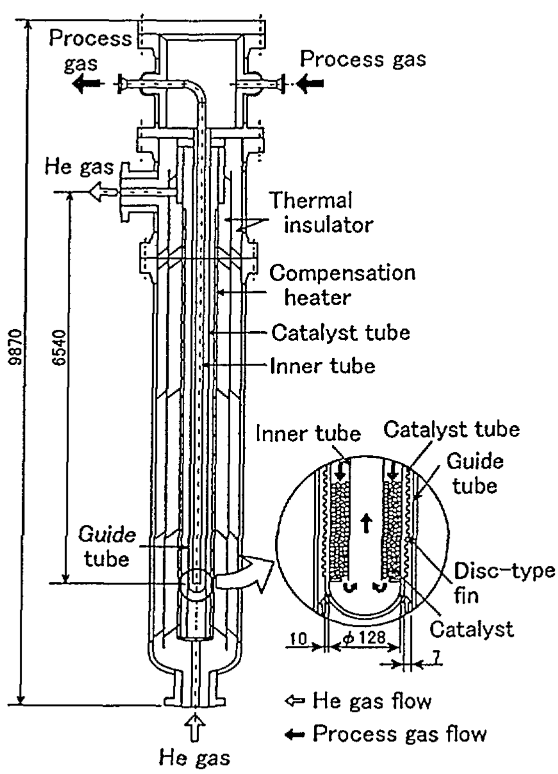

Fig. 5 Schematic view of steam reformer.

(2) Safety-related test

(a) Malfunction and accident at process gas line

The objective is establishment of the safety-related technology (1) and (2-a) described in the section Ⅲ.2, dealing with malfunction and accident at the process gas line. The test is carried out by step-change of feed amount of natural gas and steam and so on, and then the control system of the pressure difference is optimized and the performance of the SG for mitigation of fluctuation of helium gas is investigated. Even if the process gas feed is stopped completely, we aim to stop the HTTR reactor by the normal operation procedure but not with the reactor scram. At this time, heat of the helium gas can not be removed at the SR because the reforming reaction is curtailed. A cooling system of helium gas by the SG, whose detail is described in the section (3) in this chapter, is examined to investigate the cooling performance and controllability.

(b) Malfunction and accident at helium gas line

The reactor scram is conducted in this case. The emergency shut-down method of the hydrogen production system is established to assure the safety, especially structural integrity of the catalyst tube, by the experiment.

(3) High temperature components test

Thermal and hydraulic performance of the SR and SG is clarified. The SR is investigated focussing on reaction characteristics which is very important to predict transient behavior and hydrogen productivity of the hydrogen production system. The cooling system of helium gas at the trouble has been designed, using the SG and an air-cooled radiator which is installed above the SG to increase cooling power of the SG. Steam produced in the SG is condensed into water at the radiator, and steam and water circulate between the SG and radiator by natural convection. In order to limit the temperature fluctuation of helium gas at the SG outlet within an allowable value, it is important to control saturation temperature of steam in the SG, that is, to control the pressure in the SG. The pressure is controlled by a fan at the radiator adjusting flow rate of the cooling air as shown in Fig. 4. The pressure controllability, transient behavior of temperature of helium gas and steam, steam production rate and natural convection of steam and condensed water are investigated in detail.

Ⅴ. COMPONENT TESTS

In parallel to the out-of-pile test described above, the following tests are planned out with other small testing apparatus to establish the safety-related technology (2-b) and (3) described in section Ⅲ.2 and to obtain detailed data for a safety review of the HTTR hydrogen production system and development of calculation codes; (i) corrosion test and (ii) hydrogen/tritium permeation test.

The objective of the corrosion test is estimation of the effect of corrosion, oxidation and hydrogen embrittlement on strength reduction of Hastelloy XR. Metallography and material tests on strength and creep are in progress with test specimens exposed in the corrosive gases at temperature up to 900℃.

While tritium produced in the HTTR core permeates in to the hydrogen production system, hydrogen in the product gases also permeates in the opposite direction from the SR to the primary helium coolant loop. The aims of the hydrogen/tritium permeation test are to obtain the data of permeation coefficient in the very low tritium partial pressure less than 10 Pa, to examine the effect of an isotope of hydrogen simultaneously existing in the gas, and the effect of protection for hydrogen/tritium permeation by the coating film on the reforming tube such as oxidation film, calorizing film and so on [5]. In such a region of low tritium partial pressure, chemisorption phenomenon at the tube surface during dissociation and adsorption is more dominant than diffusion in the tube. In this region, it is known that tritium penetration rate is proportional to a square root of tritium partial pressure not to a linear of it, therefore hydrogen and deuterium are used instead of tritium in the test.

VI. ECONOMIC ASPECT

One of the biggest problems is absolutely economy for the commercial nuclear process heat utilization system. It is said that the economy of the total system depends on the capital cost for the HTGRs which supply heat to the heat application system via intermediate heat exchanger, because it is presumed that the fraction of the heat application system downstream the intermediate heat exchanger and the hot duct is relatively small in comparison with the HTGRs themselves. According to a private communication, a German simple estimation suggests the fraction is less than one-thirds, maybe one-fifth. Thus, the economy improvement of the reactor is inevitable for the success of the commercial plant.

It is obvious that the HTGR safety is achieved by a large core with low power density. In comparison to the current LWRs, the power density is less than one-tenth in the HTGRs. Such low power density yields the inherent safety aspect, whereas it requires more capital due to the scale demerit. For example, the size of the pressure vessel of the HTTR with 30 MW thermal output is as large as that of medium size of LWR with 500 MW electrical output. The HTGRs are apparently disadvantageous in economy in comparison with the current LWRs.

On the other hand, the inherent safety aspect in the HTGRs could make it possible that no or quite limited engineering safeguards of reactor grade quality are needed. The only safety elements in the entire system are the fuel element and graphite core components which can be checked in running operation, while the safety of LWRs with high power density is ensured by extensive, active and passive safeguards and the reactor grade quality of the components and materials. Sophistication and expensive reactor grade quality is particularly required for all components of LWRs, but, in the case of HTGRs, ultimately only for the fuel element and graphite core components. Thus, the HTGRs would provide a new, qualitatively different safety, resulting in decreasing the cost.

This safety aspect can also make the heat application system designed in a general industrial safety grade, not nuclear grade, resulting in the significant cost reduction of the system. JAERI is now under developing a new safety criteria applicable to the future commercial heat application system, including countermeasures against possible fire or explosion by combustion gasses like methane and hydrogen.

VII. CONCLUDING REMARKS

Under an understanding that HTGRs can play an important role to expand the nuclear heat application to chemical industries against the current environmental issue of the CO_2, JAERI proceeds with the development of the nuclear process heat application system coupling to the HTTR. Global eyes are kept by not only nuclear persons of interest but also the public upon the development of the HTTR heat application system, since its successful achievement may enhance the possibility to solve the environmental issue of CO_2 emission as well as a possible energy crisis which might happen in the future.

Finally it should be emphasized that an overall support and understanding from the overseas countries of concern are needed and wished for the success of the Project. The Project is highly expected to contribute so much to promoting international cooperation on the development of HTGRs and its process heat application.

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TECHNO-ECONOMIC ANALYSIS OF SEAWATER DESALINATION USING HIGH TEMPERATURE GAS COOLED REACTOR

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Abstract

Our world is facing the increased global shortage of potable water and pollution of water, so is China, especially in big cities and foreland, which is facing the power shortage as well. It is ideal to promote seawater desalination to satisfy the potable water demand in these areas. Among the various processes, MED, RO and VC have proved well developed and promising. Due to the inherent safety and its vapor produced with high parameters and features of small size and modular design, HTGR (High Temperature Gas-cooled Reactor) of 2x200MW is chosen as the energy source for the desalination in dual production of clean water and power.

This paper is to discuss the techno-economic feasibility of different seawater desalting systems using 2x200MW HTGR in the areas mentioned above, that is, ST-MED (Steam Turbine Cycle), RO, MED/TVC, RO/MED and GT-MED (Gas Turbine Cycle). The exergy concept is used in calculating availability to get cost of energy in desalination, and power credit method is used in economic assessment of different systems to get reasonable evaluating, while economic-life levelized cost method is adopted for calculating electricity cost of referred HTGR plant. In addition, sensitivity analysis on ST-MED economy is also presented.

Introduction

Seawater desalination is a well-established and commercially available technology. Among the various processes, RO, MED and VC have been in large-scale commercialization and proved promising in the future[1] [2]. Nuclear power and fossil-fired plants are the two main sources to generate electricity, heat or both, but nuclear plants have advantages on economy and environment protection over the latter, which indicate its competence in desalination markets.

High Temperature Gas-cooled Reactor (HTGR) of 2x200 MW is an advanced reactor featured by inherent safety and high temperature gas. Its small size and modular design are favored in such process applications as desalination. In addition, it can also provide better economy and safety over reactors of other type while serving as the energy source for desalination.

In spite of its well establishment, there are still some to be improved in desalination industry, such as reducing of energy consumption and high capital investment especially when combined with nuclear power plant for the latter. Moreover, the expense of operation and maintenance is also high. Therefore it is necessary to perform techno-economic analysis, resorting to optimal design with improvement in technology, process and material property, and finally increasing desalting capacity and reducing both energy consumption and product cost.

1.Coupling of desalination plant with HTGR

Inherent safety, small size and modular design feature HTGR of 2x200MW, thus providing flexibility in siting and coupling with desalination plant. The main design parameters of HTGR is shown in Table I. In this paper, the following coupling cases are to be considered:

ST-MED: MED coupling with steam turbine cycle plant (backpressure turbine).
TVC: thermal vapor compression using jet with backpressure turbine.
RO: Reverse Osmosis (two stages).
RO/MED: hybrid desalination of MED and RO.
GT-MED: MED coupling with indirect gas turbine cycle.

TABLE I Technical and Economic Data of HTGR

	ST Cycle	GT Cycle
Reactor thermal power(MWt)	2x200	600
Maximum power output(MWe)	2x80	287
Primary loop temperature(℃)	250/700	NA
Primary loop pressure(MPa)	6	NA
Secondary steam temperature(℃)	200/530	NA
Secondary loop pressure(MPa)	19	NA
Base investment($/kWe)	2587	2020
Investment addition	10%	
Fuel cost($/MWe.h)	13.08	9.20
O&M cost($/MWe.h)	12.3 (M$/a)	8.65
Economic life(a)	30	
Lead time(a)	4	
Plant self-used power	5%	
Annual load factor	80%	
Decommission factor	5%	NA
Referred money date	1993	
Amortization time(a)	30	

Based on the model and data above, the calculated results are as follows:

Electricity cost: 0.080$/(kWh) (ST cycle);
0.065$/(kWh) (GT cycle)

2. Method and model of techno-economc analysis

The cost of desalination or water cost ($\$/m^3$) is chosen as the evaluating criteria in the economic assessment. It is between 0.7 and 2.0 $\$/ m^3$ in general at present. In addition, overall investment, specific investment in unit desalting capacity ($\$/m^3/d$) and specific energy consumption are also to be considered.

For RO system, the water cost is cost of all the expenses divided by the corresponding desalting capacity. While for dual production system, evaluating methods could be classified as apportioning methods [1] and power credit methods[1] [3] .

The apportioning methods divide the total costs between the electricity and heat in a certain ratio selected on the bases of various criteria. But it is difficult to ensure that the ratio employed is truly representative for the ratio can be somewhat arbitrary. The power credit method selects a predetermined value for one of the products based on the cost of that from an alternative source. Using that value as the cost of the product, the cost of the second product can be determined. Therefore cost credit of the first product will affect that of the second. The alternative can be a single production plant (either existing or conceptual). This method can effectively give the upper limit of either cost.

In this paper, the condensing HTGR plant is chosen as the alternative plant with steam turbine cycle. In order to achieve more precise water cost, availability analysis is applied to evaluate the equivalence relation between the heat supplied to desalting plant and electricity. The availability of the heat defined as following is suggested to be equivalent with the electricity generated from referred condensing plant. That is, assuming the process steam expanded in the lower pressure turbine in the referred plant, the generated electricity is equivalent with that generated from condensing plant. This definition indicates the actual capacity of electricity production, or electricity loss due to heat supply to the desalination plant. All the calculations of dual production plants in this paper are based on this modified method, and defined as equivalent power credit method hereafter.

According to the above analysis, the cost of electricity of 2x200 MW HTGR should be acquired first for the calculation of the water cost.

2.1 Electricity cost of 2x200MW HTGR

The cost of electricity is calculated on the base of levelized electricity cost method, which consists of the following components in Fig.1. The technical and economic data[1] [4] of HTGR is listed in Table I. In the table, the referred power plant of gas turbine cycle is GT-MHR of 600MW instead of 2x200 HTGR because of lack of cost data of HTGR gas cycle.

$$\textit{Electricity Cost}\left\{\begin{array}{l} \textit{Investment Discount}\left\{\begin{array}{l} \textit{Base Investment} \\ \textit{Inflation \& Escalation} \\ \textit{Interest during Construction} \end{array}\right\} \\ \textit{Fuel Cost} \\ \textit{Operation \& Maintance Cost} \\ \textit{Decommision Cost} \end{array}\right\}$$

Fig.1 Components of electricity cost

2.2 Cost of desalination

Components of desalination cost are shown in the following figure. According to the equivalent power credit method, the levelized water cost is given by:

$$C_{Desal} = \frac{C_{DCapt} + C_{Heat} + C_{DOnM} + C_{Ee}}{D_a}$$

Where: C_{Dcapt}, C_{Heat}, C_{DOnM} and C_{Ee} are the investment discount, energy cost, O&M cost and electricity consumption cost in US dollars respectively; and D_a is annual desalination plant capacity in m^3/a. The energy cost is given on the basis of availability analysis described above.

3. Results and analysis

A computational program has been developed on the techno-economic analysis of the desalination systems using 2x200 MW HTGR, including ST-MED, GT-MED, TVC, RO and

- Desalination Cost
  - Energy Cost
  - Investment Discount
    - Base Investment
    - Seawater Intake / Outfall Cost
    - Owners Cost
    - Contigency
    - Inflation & Escalation Cost
    - Interest during Construction
  - O & M Cost
    - Labor Cost
    - Spare Parts Cost
    - Chemical Cost
    - Membrane Replacement Cost(RO Plant)
    - Insurance Cost
  - Electricity Consumption Cost
    - Internal Use
    - Seawater Pumps
    - Booster & High – pressure Pumps(RO Plant)
    - Others

Fig.2 Components of desalination cost

RO/MED. In addition, sensitivity analysis on ST-MED water cost has also been performed. The results are illustrated in Table II and Table III, where in Table II the investment of HTGR is not included, which is 0.6032 billion US$ of ST-HTGR and 0.4821 billion US$ of GT-HTGR.

TABLE II Calculation Results of Various Desalination Systems

ITEM	ST-MED	TVC	RO	RO/MED	GT-MED
Water cost ($\$/m^3$)	1.089	1.222	1.078	1.032	0.783
(1) Energy cost	26%	33%	NA	13%	NA
(2) Desalination plant investment discount	47%	43%	42%	45%	66%
(3) O&M cost	11%	8%	18%	16%	16%
(4) Electric power consumption	16%	16%	40%	26%	18%
Production capacity (m^3/d)	9.31E4	2.87E5	2.34E5	1.86E5	8.49E4
Power output (MWe)	149	93.3	160	149	166
Gain-output Ratio	10.6	18/21.5	NA	10.6	10.6
Overall investment (billion $)	0.1801	0.5676	0.4013	0.3434	0.1637
Experience availability	High	Medium	medium	low	Low

Concluded from the results above, GT-MED shows the best economy, followed by RO/MED, RO, ST-MED and TVC. GT-MED also has the lowest overall investment, followed by ST-MED, RO/MED, RO and TVC. Desalination of distilling such as MED and TVC can produce fresh water of high quality, and the latter can achieve high production capacity, though its water cost and thermal-economy are both lower than those of MED. For RO, achievement of low water cost needs large amount of cheap electricity. RO/MED combines both advantages of both, however the coupling may be more complex. GT-MED will be the strongest competitor against RO for its high thermal efficiency and economy if technology of gas turbine is improved.

Factors that will have effect on technical and economic competence of various desalination systems are illustrated in Table III, in which the interest, load factor of the system, specific investment of either desalination plant or power plant rank the most important on production cost, while specific investment of either plant and interest are those on overall investment.

TABLE III Results of Sensitivity Analysis of ST-MED

Importance	Desalination cost(variation)	Overall investment(variation)
1	Load factor(1.200)	Specific investment of HTGR plant(0.770)
2	Annual interest(0.653)	Specific investment of desalination plant(0.217)
3	Specific investment of desalination plant(0.460)	Annual interest(0.146)
4	Specific investment of HTGR plant(0.280)	Escalation(0.091)
5	Economic life(0.126)	Lead time(0.091)
6	Escalation(0.048)	—
7	Lead time(0.026)	—

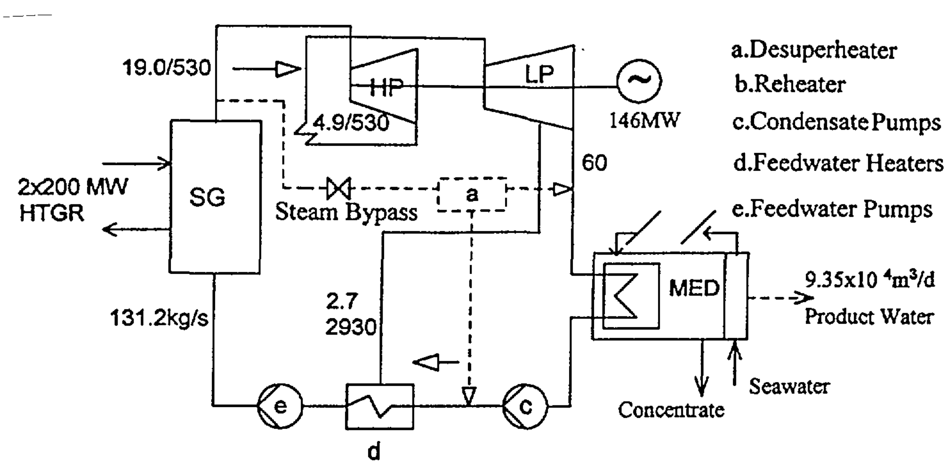

Fig.3 Scheme of ST-MED for 2x200MW HTGR (Steam Side)

4. Conclusion

- Desalination using 2x200MW HTGR is techno-economic feasible, and has advantages over those using other energy sources.
- In view of technology of the state-of-the-art, economy, complexity, need of dual production of potable water and electricity in the foreland, ST-MED is considered as the best choice. RO of one-stage will be the best if there lies a large amount of cheap electricity and potable water is the only need.
- The optimized design characters of ST-MED are illustrated in Fig.3 and the final product cost is 1.071 \$/ m^3 with net power output of 122MWe.
- For desalination in distillation method such as MED and TVC, the key to reduce product cost lies in reducing energy consumption and investment of plant equipment, while for RO, it lies in reducing energy consumption and O&M cost (such as RO membrane).

Advanced features in technology and economy of advanced reactors such as high load factor, inherent safety will also bring competency in technology and economy of overall desalination system.

- In addition, system of hybrid processes will also gain high product capacity, low cost or both, for example, RO/MED mentioned above. If pretreated seawater is mixed with product water from MED, better economy will be achieved at some expense of quality product.

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USE OF PLUTONIUM IN PEBBLE BED HTGRs

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Abstract

This paper provides a summary of the current status of world-wide inventories of weapon-grade plutonium and plutonium from reprocessing of power reactor fuel. It addresses the use of pebble bed HTGRs for consumption of the plutonium in terms of the fuel cycle options. The requirements and neutronics aspects, and results from parameter studies conducted using pebble bed reactor types, are discussed, along with proliferation and waste disposal aspects.

Summary

More than 200 t of weapon-grade plutonium and about 150 t of plutonium from reprocessing have been available world-wide in 1995. Additionally, 60-80 t of plutonium are being produced each year in LWR plants. For the handling of these materials different strategies are under discussion, each showing specific characteristics with respect to the transmutation rates and disposal features.

Vitrification mainly represents a dilution of plutonium within a corrosion and leach-resistant glass matrix without using the energy content of the fissile material. 'Classical' mixed-oxide (MOX) fuel does not allow for significant reductions of plutonium as 'new' plutonium is being generated by the U238 within that type of fuel.

The use of coated particle technology together with innovative loading schemes by separated feed and breed fuel elements in pebble bed cores allow for optimisation of burn-up for both fuel types and enhanced plutonium burning rates. Due to differences in the neutronic behaviour of weapon-grade or reactor-grade plutonium the fuel compositions of the breed elements have to be adjusted for reaching self-stabilising safety features of the reactor core especially by negative feed-back coefficients. Thorium-based breed fuel elements with some addition of highly enriched uranium from dismantled weapons as driver fuel cope with the objectives for the use of reactor-grade plutonium, whereas uranium based breed elements provide better neutronic stability for weapon-grade plutonium burning as long as no burnable poisons are taken into account. It is also shown that the necessary heavy metal content in the elements can be optimised in reasonable ranges. The comparison to other once through cycles of different reactor types illustrate the benefits of this approach and of the thorium-based coated-particle fuel in case of reactor-grade plutonium burning.

The corrosion and leaching characteristics of coated-particles are comparable or even superior compared to vitrification. Beyond that, this type of fuel has the capability of denaturating the plutonium already in a once-through cycle, thus avoiding additional intermediate reprocessing steps. After burning of the weapon-grade material in this way, the residual fissile materials in the coated-particles are equally 'diluted' as in the case of vitrification. It would be necessary to divert about a million fuel pebbles and to destroy about 10^{10} coated particles for collecting a critical mass for a plutonium bomb. As the technology for these processes is not available, this fuel composition is rather proliferation resistant after having extracted the energy with high efficiency.

The characteristics of such fuel is mainly covered by available experimental verification of the past HTGR test programmes and test or demonstration reactors, thus providing a profound basis for the fuel fabrication and fuel qualification by irradiation tests. Due to the general

importance for the destruction of plutonium in once-through cycles, this development should be part of an international R&D programme.

Approaches to Solve the Plutonium Issue

Weapon-grade fissile material from the dismantling of war heads should be denaturated and/or used as nuclear fuel to prevent proliferation or future misuse from final deposits. This problem has not only to be addressed for military plutonium but also for plutonium from civil use of nuclear energy and for the large quantities of highly enriched military uranium, as well. Whereas the latter could in principle be diluted with natural or depleted uranium, plutonium should be transmuted to avoid separation by purely chemical means. Table 1 shows a listing of the world-wide available military plutonium and the permanent production from civil reactors.

Table 1: Amount of Plutonium World-wide (Status 1995)

100 t weapon-Pu in Russia 100 t weapon-Pu in USA 2.6 t weapon-Pu in GB 6 t weapon-Pu in F 3.5 t weapon-Pu rest of the world	from military production
150 t Pu from reprocessing world-wide	*cumulated from civil production*
60...80 t Pu/year in LWR-plants	from permanent production

As a conclusion from these figures, two approaches have to be undertaken for solving the plutonium problem:

- highly effective burning or denaturation of existing plutonium and
- minimisation of the generation of additional plutonium from the operation of existing reactors.

Both measures are necessary to reduce proliferation risk and to stop the 'spiral' of ever increasing amounts of transuranium isotopes.

It is well known from neutron cross section comparison and conversion chains (figure 1) of uranium- and thorium-based fuels that the relation of the probabilities for fission against absorption of all the derivatives of thorium in a thermal reactor is much higher than for low-enriched uranium. In other words, thorium-based fuel produces with about 2% significantly less minor actinides than uranium-based fuel (11%).

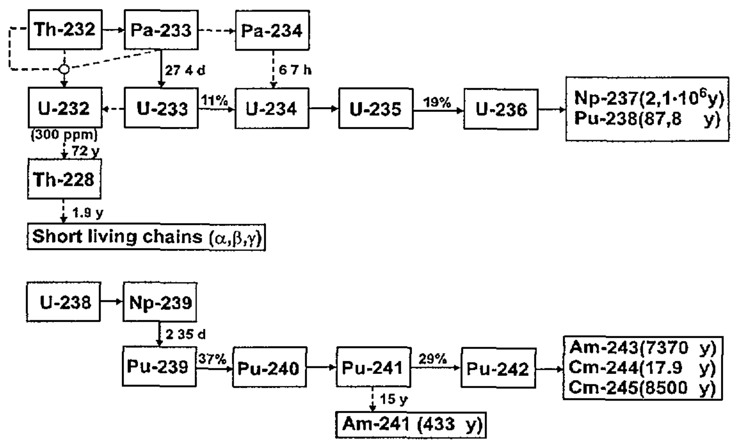

Figure 1: Conversion Chains for Th and U

In this sense, it would be advised to mix plutonium with thorium - as described in this report - for coping with the objective of minimised generation of minor actinides and effective burning of plutonium. The plutonium reduction of uranium based MOX for mixtures with weapon- or reactor-grade plutonium only reaches about 25-30% at burn-up rates of about 40.000 MWd/t (as usual in current LWR). So, there is a large potential of further reductions and the option to use once through cycles if the fissile plutonium content could be minimised by optimised fuel compositions and higher burn-up.

The coated-particle technology represents a crucial key to reach high burn-up and destruction rates for plutonium in a once-through cycle and without multiple intermediate reprocessing steps as they are necessary for most other options.

The actual different possibilities of handling the plutonium issue are illustrated in figure 3. Vitrification mainly provides a dilution of plutonium and a rather effective enclosure by corrosion and leaching resistant glass. Mixing this material with highly radioactive fission products would only provide a shelter against misuse for some hundred years and leave a 'plutonium mine' for the far future just as in the case of direct disposal of nuclear fuel that still contains significant amounts of fissile plutonium as shown in figure 2. This could only be avoided by multiple recycling or highly efficient burning / transmutation techniques.

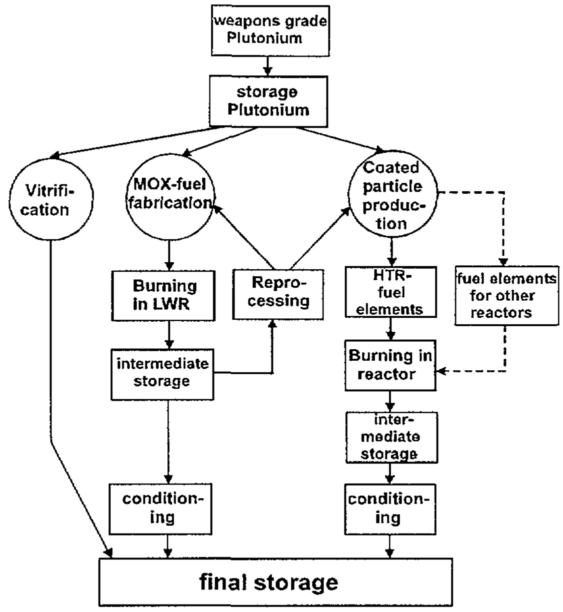

Fig. 2: Handling Options for Plutonium

Assumed that weapon-grade plutonium would be converted to fuel, there is no reason to exclude the handling and use of highly enriched military uranium for an optimisation of the burning capabilities of a specialised fuel. This provides an additional parameter for the fuel design of plutonium burners and contributes as well to the destruction of weapon-grade uranium. However it has to be taken into account that the neutronic behaviour of military plutonium is very different from that of civil plutonium from reprocessing due to the different isotope vectors. This strongly influences the nuclear stability of the core as well as the design and composition of the breed fuel elements as discussed later.

Requirements and Neutronic Aspects

General requirements for a nuclear power plant being optimised for plutonium burning should be the following:

- plutonium residuals in the spent fuel as low as possible (once through)
- highest standards of safety (no danger of catastrophic releases from the reactor nor from the spent fuel storage)
- suitability for direct disposal
- low radiotoxicity of long-lived wastes
- proliferation resistance of the fuel cycle
- maximum production of energy / economic operation
- minimal amounts of high level waste
- flexibility of the fuel cycle
- use of well established technical solutions

These requirements have been put into consideration for a study made by Khorochev [1] on the basis of 350-500 MWth annular core pebble bed HTGR designs with emphasis on the cross-correlation of plutonium and thorium fuel with self-acting (passive) safety features by negative temperature coefficients both for civilian and military plutonium.

The main differences in these fuels are to be seen in the different isotope vectors. Military plutonium mainly contains plutonium 239 having a large resonance at 0.3 eV and only small amounts of plutonium 240 which possesses a huge resonance at 1.02 eV. The presence or absence of plutonium 240 strongly influences the neutronic behaviour when plutonium should be burned in thermal reactors.

The energy dependence of the neutron flux is also rather different for variations in heavy metal contents and moderation ratios. For reaching a negative moderator coefficient by a harder spectrum, higher heavy-metal and plutonium contents are more favourable. But for an under-moderated design the potential for water ingress has to be excluded by gas-turbine technology.

Normally, the fuel and the breeding material is mixed in one fuel element. But for reaching a high burn-up of the plutonium it could be better to have different elements that either contain plutonium (feed) or thorium respectively uranium 238 as breed materials. Feed and breed fuel elements could be circulated differently in a pebble bed core and must not necessarily reach identical burn-up as an additional optimisation parameter.

The Doppler coefficient expresses the change in criticality by temperature change of the resonance absorbers due to the broadening of the resonance. For the above mentioned two pebble concept this has to be calculated separately for feed and breed elements including potential time delays with respect to the temperature interaction of the different fuel types during transients.

Having separate feed particles also allows for having rather small coated particles containing the plutonium. Smaller particles about 0.25 mm in diameter may be better suited for very high burn-up in the range of 750.000 MWd/t because they are more robust than larger particles.

Results from Parameter Studies

The two-pebble strategy that could only be realised by pebble bed reactor types has to be optimised by five main parameters influencing the fuel design:

- civil and military plutonium need different approaches
- the choice of using thorium or uranium as breeding material influences the generation of minor actinides <u>and</u> the safety characteristics
- the heavy metal content in the feed and breed elements have some impact on burn-up and economy
- the relation of feed and breed elements determines the moderation ratio and the energy spectrum, but needs additional considerations for water ingress accidents
- the number of reshuffling cycles can be optimised for both fuel types separately.

It has been shown for uranium-based fuel cycles that the two-pebble strategy has significant benefits compared to the mixed-oxide fuel or mixture of feed and breed particles in one elements as mainly been used and studied in the past.

For assessing the following tables, we have to mention some simplification, which was applied for the numerical simulation:

- The fraction of Pu241 of the real LWR-Pu was treated as if it were Pu239; Pu242 was treated as Pu240. Thus, the Pu was assumed to be 70 % Pu239 and 30 % Pu240 at the start of burnup.
- The „Minor Actinides“ were not included in the burnup chains, i.e. the parasitic neutron absorption caused by the build-up of Am and Cm was neglected.

This leads to somewhat optimistic values of the achievable burn-up and consequently of the fraction of Pu, which is transmuted (see table 2):

Table 2: Plutonium burn-up for different heavy metal loading of pebble bed fuel elements.

Relation Feed / Breed		MOX	Feed/Breed Particles	Feed/Breed Elements
heavy metal	g / pebble	1.5 / 10	1.5 / 10	3.0 / 20
burn-up	1000MWd/t		776 / 65	793 / 105
average burn-up	1000 MWd/t	100	158	200
total Pu burning	%	61.3	90.3	96.2

It can be seen that in the two-pebble concept the breed pebbles can be longer exposed within the reactor and that the whole cycle offers some additional flexibility for optimisation or adjusting to the allowable burn-up in the feed elements.

The potential benefits of using thorium as breeding material has already been mentioned but in conjunction with plutonium burning it has to be noticed that the Doppler coefficient of thorium is less negative than for uranium 238 and that plutonium 239 provides a positive contribution to the moderator temperature coefficient as shown in the next table 3.

Table 3: Temperature coefficient of pebble bed core with different heavy metal fuel contents

Heavy Metal	3g W-Pu / 20g Th	3g W-Pu / 20 g U238
Reshuffling	4 / 6	4 / 6
Doppler Coefficient	$-2.0\cdot10^{-5}\ K^{-1}$	$-3.4\cdot10^{-5}\ K^{-1}$
Fuel Coefficient	$-1.6\cdot10^{-5}\ K^{-1}$	$-2.72\cdot10^{-5}\ K^{-1}$
Moderator Coefficient	$+1.2\cdot10^{-5}\ K^{-1}$	$-2.34\cdot10^{-5}\ K^{-1}$

This shows that for weapon-grade Plutonium the negative contribution of Pu240 to the moderator coefficient is not sufficient due to the rather low content of this isotope. Such a fuel composition of weapon-grade Plutonium and Thorium would only have a very small negative temperature coefficient in total and would not cope with the safety requirements. U238 and the additional Pu240 being produced by absorption show much more a self-stabilising behaviour.

In case of burning reactor plutonium there is already enough Pu240 in the feed elements thus allowing for using thorium as breed material. High enriched uranium e.g. from the dismantling of weapons has also been taken into account as driver for optimising the whole cycle.

It is important to mention that the value of the moderator coefficient is very sensitive not only to the isotope composition of the Pu, but also to the degree of neutron moderation and to the resulting shape of the neutron spectrum, respectively. The central graphite column of the core, which has been intended in this study, represents a strong source of optimally thermalized neutrons for the surrounding active core. First results of an additional study being currently performed for a cylindrical active core (no central graphite column) indicate a sufficient negative moderator coefficient also for the use of weapons-grade Pu together with thorium as breed material [3].

The reference cases finally chosen for the central column-core regarded here are a compromise between the different requirements and they are illustrated in figure 3. As a result it can be seen, that these cycles allow for a reduction to 11-12% of total plutonium in the feed elements. For reactor grade plutonium the thorium-based breed elements practically contain no transuranium actinides and only a low amount of fissile uranium. In case of weapon- grade plutonium there is about 0.7g additional plutonium being generated in the breed elements but the isotope vector of this plutonium is not suited for military use. The total plutonium reduction in that cycle would be a factor of three and would leave room for further minimisation if the use of burnable poison is taken into account.

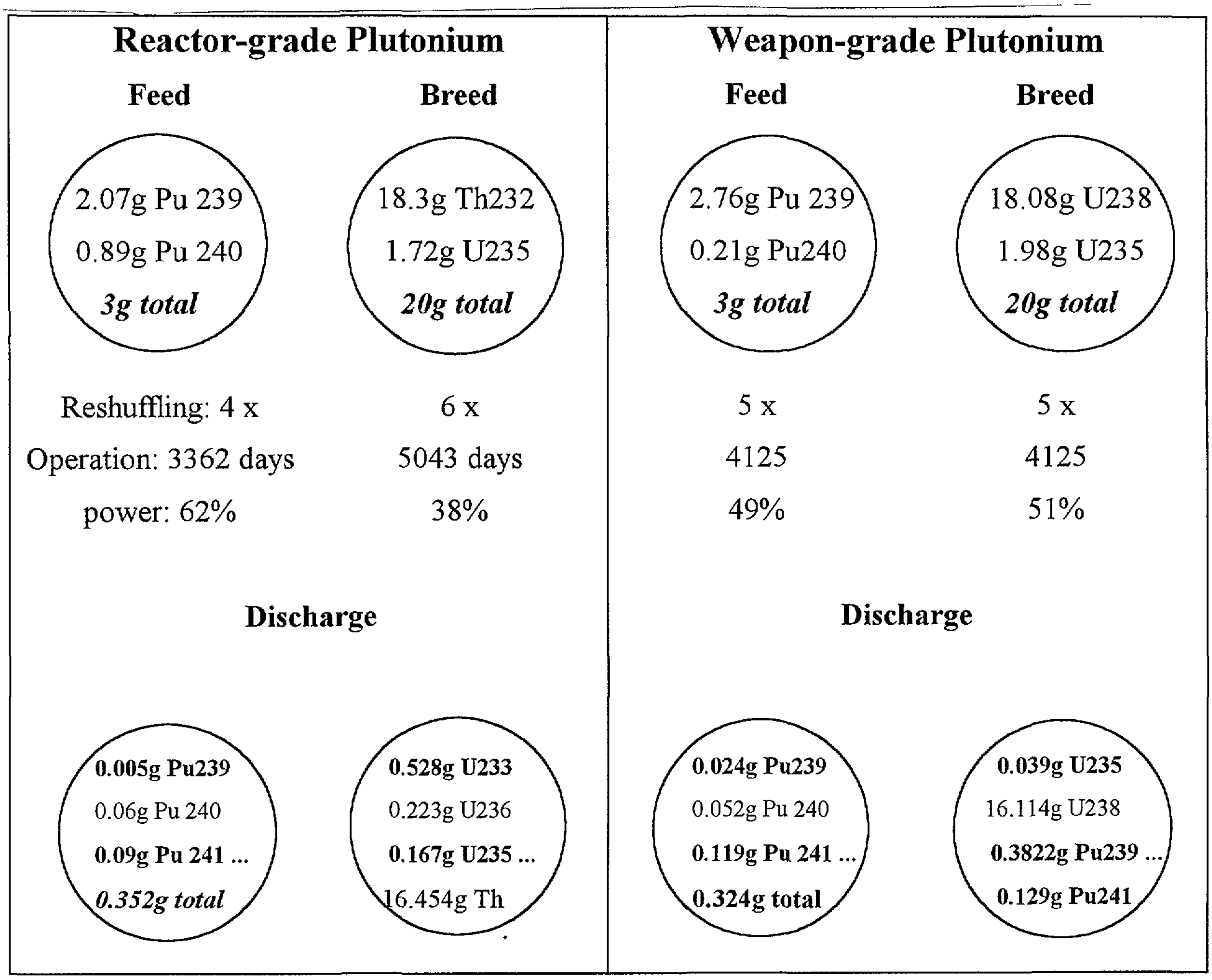

Figure 3: Results of burn-up of plutonium containing pebble bed fuel elements

As a result it can be seen, that these cycles allow for a reduction to 11-12% of total plutonium in the feed elements. For reactor grade plutonium the thorium-based breed elements practically contain no transuranium actinides and only a low amount of fissile uranium. In case of weapon- grade plutonium there is about 0.7g additional plutonium being generated in the breed elements but the isotope vector of this plutonium is not suited for military use. The total plutonium reduction in that cycle would be a factor of three and would leave room for further minimisation if the use of burnable poison is taken into account.

The table 4 shows a comparison for different reactor types in comparison to HTGR Pebble Bed Reactors (PBR) illustrating the significant reduction of plutonium in PBR designs both for weapon-grade [2] and reactor-grade plutonium.

Table 4: Plutonium conversion in different reactor types

	Initial loading	Discharged plutonium (once through)			
Reactor Type		**ALMR**	**ABWR**	**PBR[2] (w-grade)**	**PBR (r-grade)**
Total Pu (kg)	**1000**	**618**	**648**	**199**	**43**
Pu 239 (kg)	**940**	**476**	**279**	**6**	**0,6**
Pu 240 (kg)	**60**	**130**	**233**	**72**	**1**
Pu 241 (kg)	**0**	**12**	**97**	**55**	**6,4**
Pu 242 (kg)	**0**	**0**	**39**	**66**	**35**

The technology for the coated-particle fuel containing plutonium is mainly available and has been demonstrated in several tests as e.g.:

- fabrication of plutonium containing particles (Mol, 1964):
  irradiation in Studsvik reactor:
  burn-up: 360 000 MWd/t (at 1200 °C)
  particles showed good behaviour
- plutonium containing particles have been inserted into the DRAGON reactor:
  burn-up: 600,000 MWd/t,
  dose: 10^{21} n/cm², 700...1100 °C
  fission product release: Xe 137, Kr 85, Xe 135 between 10^{-5} and 10^{-6}
  no damage of particles
- fabrication of (U, Pu), O_2-kernels after the Sol-Gel-Process and corresponding TRISO-concept:
  irradiation in FRJ-2 (1970):
  burn-up: 17...18,7 % FIMA, T ≈ 1050...1200 °C,
  fast dose: $5.3 \cdot 10^{19}$ n/cm² (EDN)
  fission product release: Xe 133, Kr 85: $4 \cdot 10^{-7}$
  no damage of particles
- test of plutonium fuel in Peach Bottom

This experience must of course be complemented by the coating of plutonium kernels with today's technology and extensive testing in test reactors to demonstrate the extremely high burnup that nearly converts the whole fuel into other chemical elements by quasi-total fission. The potential of the coated particle technology in this respect is unique and should inspire collaborative international R&D programmes.

Proliferation and Disposal Aspects

The proliferation resistance of HTGR fuel can be shortly discussed on the example of the status of burnt fuel at MEU-elements (50 % Th + U with < 20 % enrichment). The necessary steps to be able to gain e.g. 10 kg of plutonium 239 would be:

1. diversion of 10^6 fuel elements (despite of international control)
2. head end for 10^6 fuel elements (technology today not available)
3. destruction of $1 \cdot 10^{10}$ coated particles (technology not available)
4. reprocessing of the Pu-U-mixtures

It is evident that this cannot be done in an unobserved manner and that other ways of diversion of weapon-grade materials would need much less efforts.

Applying the thorium-based fuel cycle for the reactor-grade material would lead to a reduction by 10-20 times in terms of plutonium 239 compared to direct disposal of the plutonium. The radiotoxicity of the discharged fuel would also be significantly reduced compared to the standard MOX fuel.

The coated-particles provide an excellent shelter against migration of fission products even under very long time scales of millions of years as can be seen from the leaching experiments that have been performed in FZJ. Such experiments should be extended to other storage conditions to prove this behaviour also for other geologic disposal sites than salt. The ceramic graphite shell around the coated-particle fuel provide an additional chemically and mechanically resistant enclosure that can even be reinforced by further conditioning and disposal provisions.

In contrast to other fuel types, HTGR fuel provides excellent final disposal features already with the fuel production step and is thus well suited for direct disposal strategies combined with a credible exclusion of catastrophic releases during operation in the reactor and for final disposal.

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SUSTAINABILITY AND ACCEPTANCE — NEW CHALLENGES FOR NUCLEAR ENERGY

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Abstract

This paper discusses the concept of sustainability in relation to acceptance of nuclear energy. Acceptance is viewed in terms of public acceptance, industrial acceptance, and internal acceptance/consensus within the nuclear community. It addresses sustainability criteria, the need for innovation, and the different levels of acceptability. The mechanisms of risk perception are discussed along with the technological consequences from risk perception mechanisms leading to specific objections against nuclear energy.

Summary

Sustainability can be seen as a principle of intergenerational equity and of environmental stewardship with a long-term scope. Due to the danger of an over-stressing of environmental resources by still strongly increasing world population and of the average standards of living, the change towards 'Sustainable Developments' has got the leading political and technological orientation for the 21st century. Criteria for sustainable energy supply systems, however, still have to be defined in a consensual and general way for establishing a guidance to judge the compatibility of different options like nuclear or others.

In the past Climate Protection Conferences nuclear energy has not been considered as a sustainable energy supply option although the nuclear community is strongly advocating with the argument of CO_2-free electricity production. This may be due to the increasing anti-nuclear movements in many countries and the anticipated safety, waste and economic problems.

Sustainability does not only address environmental but also the mental compatibility. The problem of acceptability of nuclear energy and its inter-relationship with technological innovation and evolution of the boundary conditions is discussed on three levels

- public acceptance
- industrial acceptance
- internal acceptance / consensus within the nuclear community.

'Risk Perception' is being understood as a subject / result of complex cultural and social evolution as well as a human sense that does - obviously - not obey to the risk formula being used in nuclear engineering. An attempt is being made to illustrate the mismatch of the two-parameter linear product in the definition of a technical measure for risk and the human sentiment for the acceptance of risk in a multi-parameter interdependence of different additional aspects. A better understanding and 'acceptance' of these mechanisms might lead to an improved communication culture between public, industry and nuclear community.

The socio-political dimension of nuclear energy has to be recognised by the nuclear community by a pro-active attitude in responding to the public awareness for safety and waste issues as well as to the changes in markets and competitiveness also by the development of innovative nuclear technologies as a long-term scope for overcoming these actual problems.

1. Sustainability Criteria

The principle of 'Sustainable Development' can be interpreted as an approach to solve the conflict between ecology, economy and intergenerational social responsibility. This terminology evolved already in the 18^{th} century from forestry that also needs a long-term planning and cultivation. In the 1980^{th}, the concept of 'a development that meets the needs of the present without compromising the ability of future generations to meet their own needs' had been taken up again by the general assembly of the United Nations and in the 'Brundtland Report'. It was also taken as a mayor guideline in the discussions and conferences on climate protection (e.g. Rio, Kyoto) and seems to be a robust and consensual orientation being applied to a broad spectrum of human activities also for the next future [1].

In former times, mankind has always been a 'Solar Community' mainly taking its energy from regenerative sources (e.g. wood, water, wind) and physical labour from humans or animals. This kind of energy supply also created social structures (e.g. bondage, slavery) and hierarchies that have been overcome after entering the 'Fossil Energy Era' about 150 years ago. The impact of using fossil energy on the climate, flora and fauna is obvious and cannot be continued or even enhanced in the long run, thus offering the main argumentation to establish sustainable energy supply structures with reduced emissions of climate gases or toxic substances / wastes endangering present and future generations.

Although nuclear energy is a proven CO_2-free energy supply system it has not been judged as a future tool at the Global Climate Protection Conferences. Recently, the European Parliament rejected only by a very narrow vote (225 against 218) a statement that 'nuclear power could not be considered a safe and sustainable method of energy production and did not, therefore, belong in the policy against climate change' [2]. This highlights the total discrepancy between the argumentation of proponents on the one hand and ignorance of the potential of nuclear energy on the other.

There is a broad consensus in many countries all over the world that nuclear is not acceptable from a socio-political point of view and a different judgement on the criteria on sustainability that should be valid for all energy supply systems and should allow for neutral evaluations beyond different convictions and dualism of alternatives. So, there is a real need for establishing such criteria that would probably address a spectrum of aspects like [3]:

- Degradation of Resources (e.g. fuel, materials, land losses)
- Environmental Impacts (CO_2, SO_x etc.)
- Human Health Effects (acute and latent fatalities in operation and after accidents)
- Social Aspects (acceptance, risk perception, proliferation risk etc.)
- Economic Competitiveness (inclusion of external cost?)
- Avoidance of Non-Degradable Waste (e.g., amount, confinement time)
- Robustness (supply security, failure friendliness, grace periods etc.)

Strategies towards sustainable developments have not only to include the innovation potentials of the energy systems and the cost effectiveness [4] for CO_2-mitigation, but also to concentrate on the socio-political dimension of nuclear power [5] and to adopt the technology to the mechanisms for acceptance in a pro-active way.

2. Need for Innovation

The mechanism of 'evolution of species' is mainly driven by changes / evolution of the boundary conditions and those of other competitors to which the species have to respond by viable mutation or change of attitudes. The boundary conditions for nuclear energy have evoluted drastically in contrast to the expectations in the past decades. It is unrealistic to expect an adaptation of the social and economic environment to the intrinsic technical evolution of nuclear technology as often claimed by the nuclear community. In the contrary, nuclear technology also has to 'mutate' quick enough to keep pace for coping with the evolution of the requirements and that of competing techniques.

Nuclear energy still possesses a huge innovation potential like any other applied technique. But this potential has not been introduced strong enough into the discussion on the future of nuclear power and into the improvement of the market situation. There is a real danger that the future of nuclear power might be decided on the basis of the anticipated deficits of the nuclear technology established some decades ago not taking this potential for improvement into account. Successfully applied technical solutions, convincing operational experiences, advanced commercial projects and innovative approaches complement each other for a long-term option on nuclear power and for a progressive argumentation on future perspectives.

The reasons against nuclear energy are partially known from investigations / interrogations and the question arises whether technological innovations can respond to these fears, changed views and trends as being discussed for some selected examples in this paper. This applies for the public as well as for politic and industrial decision makers.

'Market' compatibility is a pre-condition for industrial engagement, acceptance and risk perception concerning the safety of investments and return of capital. **'Structural' compatibility** is needed to apply nuclear energy under effective & rational licensing, manufacturing and operation procedures. **'Environmental' compatibility** of nuclear power is often referred for defending the need for nuclear energy. **'Mental' compatibility** of nuclear technology might even be more important for reconciling nuclear energy with public opinion again.

These goals - in a long-term view - also represent a challenge for R&D on innovative technological approaches and should be fostered by a consensus within the nuclear community despite potential different views on technological options [6].

3. Different Levels of Acceptability

The acceptability of nuclear power is normally focused on the perception of the public and the politic level being mainly influenced by safety / waste concerns and other conflicts like credibility / communication problems and different political perceptions on future energy supply systems as it will be later discussed in more detail.

Another level that has to be addressed is the decline of acceptability for nuclear power by industrial decision makers. The actual mechanisms of industrial decisions on investments or engagement in nuclear energy are also a result of evolutionary - possibly irreversible - changes in the attitudes of industrial leadership, management and criteria (e.g. shareholder value, short term return of capital etc.) as well as the 'inverted' competitiveness of nuclear against low-priced and highly effective fossil-fired power plants. The 'safety' of investment is not only judged on a purely technical and financial basis but also by taking into account e.g., the 'risk' of changes in market structures, needs for back-fitting and danger of early shut-down by political influences or even draw-backs in other business areas due to the nuclear engagement. Normally, the risk for loss-of-investment and damages in case of nuclear accidents are also partially not accepted by insurance as it may be for other industrial projects.

The field of industrial acceptability of nuclear power is often assessed with a sentimental view to the 'good old times' when industrial leaders claimed for long-term perspectives and developments as well as for large investments with a long-term return of capital. Monopolistic structures in the electricity generation are also reduced in favour of free markets, competition and the possibilities for independent energy suppliers. The actual situation has to be analysed and 'accepted' as a basis for 'tuning' the technical concepts and economic features of nuclear power plants. The economic attractiveness of conventional power plants is also a result of drastic improvements in efficiencies and cost reductions using the benefits of series production and low-cost manufacturing of equipment in the world market. Long and complicated licensing procedures impose additional handicaps and risks on nuclear energy.

Another - but interconnected - area of 'risk perception' results is the still underdeveloped consensus on common strategies within the nuclear community especially with regard to innovations. Evolutionary incremental improvements may reduce the technical risk and the needs for extensive demonstrations, but they may possibly not keep pace with the evolution of requirements. Addressing innovative approaches in a dualistic view by the attribute 'revolutionary', indicates that innovations are judged to endanger existing technologies and installations.

It is rather essential that the nuclear community defines a consensus on the coexistence of different generations of established technologies and different options on innovative technologies and concepts. Such an 'internal' consensus and a visible willingness also to respond to the evolution of the social environment (e.g., by flexible innovation strategies) may be a pre-condition for an effective communication towards an 'external' public and political consensus.

The drastic beneficial changes by the globalisation and e.g., by the European Unification also have to be accepted and actively transposed into new collaboration structures on the industrial, administrative and R&D side instead of the acquainted former structures on national levels. The proposed 'Global HTGR R&D Network (GHTRN)' is also one step into this direction and could be a model for other technology developments.

4. Mechanisms of Risk Perception

Although probabilistic risk assessments (PRA) have proven as an adequate tool for balanced technical improvements of safety, the argumentation with the results of PRAs did not lead to improved public acceptance. And even much smaller probabilities or comparisons of risks associated to different energy supply scenarios might not really change the situation.

The reason might be that the risk defined as a linear product of damage and probability does not respond to the functional dependencies of the very essential human sense for the handling / acceptance of danger that permitted a survival of the human species since millions of years even under more dangerous and hostile conditions. All other human senses behave mainly in a non-linear way, so why this one?

It is a proven fact that the hemispheres of the human brain operate in different ways. Whereas the left side mainly balances rational intellectual input and experiences, the right side mainly reacts in an associative way. Both judgements influence the acceptability. This cannot be changed by the proponents of nuclear power and must be accepted as a God-given natural fact to which the use of nuclear energy must comply.

The following hypothesis for elaborating on an 'Acceptability Formula' has not to be taken too serious in a mathematical, analytical or even psychological sense. It should only indicate that the risk formula and their results may even be counter-productive in the communication with 'normal human beings'.

Everybody can check by himself whether the perception of danger has the same power than the probability of its occurrence. Nobody would play a lottery if he would judge it like a nuclear expert in probabilistics. There is a large personal potential benefit, with a limited amount of money as venture against a negligible probability for a significant prize. Obviously, the probability for that chance is strongly overestimated in the expectation of a huge benefit although a smaller gain is more probable. If the potential financial risk would be very much higher - e.g. the income of a whole month or even a year - the participation in lotteries would be reduced only to some individuals accepting that game.

Even in case of very tempting benefits there will always be an upper limit for the venture or damage otherwise it will not be accepted at all.

Higher potential damages - even the loss of life - are accepted, if there is the judgement of being able to influence the procedure by own skills either in games or e.g., by driving a car.

This is also willingly accepted even for a higher potential number of death by e.g., flying with a large aeroplane due to the believe in the skills of the pilot and the reliability of the plane and the airline. But panic reactions might result in the very moment when minor unusual disturbances occur.

Accordingly the 'Acceptance Formula' has to be enlarged by some other parameters describing the voluntariness, reliance in personnel / equipment and control of both.

The Institute for Decision Research in Oregon [7] referred to some more comprehensive qualitative risk and benefit characteristics as shown in Table 1.

Kind of risk consequence	***nature of risk source***
• voluntariness • possibility for avoidance • distribution over time • geographic distribution • degree of being affected • potential for control • social control • active influence • degree of familiarising / knowledge • scientific/technical maturity	• human, social, artificial risk • possibility for fleeing • sentiment for danger • reversibility / irreversibility • alternate possibilities • distance to the source of danger
dimension of risk consequence	***nature of benefit***
• fatal vs. limited damage	• exclusivity of benefit

• delayed vs. prompt damage • catastrophic vs. continuous damage • extreme catastrophes	• public vs. private benefit • distribution of benefit

Normally, only some of these parameters are being selected for simplified statistical reductions but for a complex acceptability problem like nuclear the full set of influences with different weights has to be kept in mind. Cultural / national influences may lead to some different weights or variations but the principle mechanisms are the same at least in a latent manner.

It is obvious that the risk formula does not at all respond to this complexity of judgements that human being are capable to perform in an indigenous way. This excellent result of human evolution also has to be respected within the argumentation on nuclear energy and to be transposed into technical and structural requirements for reconciling nuclear with the mechanisms of acceptability.

5. Technological Consequences from Risk Perception Mechanisms

Some of the 'Factors in the Acceptability Formula' can indeed be addressed by technological / structural convergence with these parameters as e.g.

Qualitative Risk / Benefit Characteristics	Potential Response
possibility for avoidance	independent barriers, self-acting safety systems
potential for control	Accident Management Measures (Fire Brigade)
social control	transparent' technology and surveillance
degree of familiarising / knowledge	neutral competent information
scientific / technical maturity	open information on the technological progress
fatal vs. limited damage	restriction of max. releases
delayed vs. prompt damage	larger grace periods
catastrophic vs. continuous damage	transmutation of waste into short-lived material
extreme catastrophes	deterministic exclusion by design
possibility for fleeing	slow accident progression
sentiment for danger	open information on all abnormal situations
reversibility / irreversibility	restricted max. release
alternate possibilities	neutral comparisons, sustainability criteria
distance to the source of danger	siting, underground construction
distribution of benefit	lower local electricity cost near to NPP

These few examples show that the trends for advanced and innovative designs are on the right orientation for coping with these anticipated parameters and that recent licensing requirements requesting restriction of maximum releases and inclusion of core melt accidents in the design as well as the need for a cooperative communication can be correlated logically.

The technological innovation potential for addressing these factors are not being discussed in an open way mainly because of the 'risk' perceived by the nuclear community itself with regard to potential negative consequences for older designs.

The distribution of the benefits to the whole community while concentrating the risk on the neighbourhood of the power stations is in disagreement to the assessment of social justice. The price for fuel is often lower near to the refineries. Why should the surrounding municipalities of NPPs not benefit by lower electricity cost?

6. Specific Objections against Nuclear Energy

The reasons to reject nuclear energy that are mainly mentioned in public debates and inquiries are referred to in [8]. They can be categorised according to their degree of relevance and cross checked whether technological innovations or alternate approaches or structures can have an influence on that according to the personal judgement of the author.

Category A: very important objections / doubts

Objection	Technological / Structural Response
Health damage / death by radiotoxic emissions	improved designs, barriers and filters
Equal chances for energy alternatives	support emission-free alternatives by low-cost nuclear energy
Restrictions/burdens for future generations	improved decommissioning, waste reduction, waste transmutation
Democratic control of the 'Atomlobby'	dissolving the confrontation on nuclear energy
Interweaved spheres of interest	transparent separate structures
low efficiency of public control	less formal, more pragmatic 'open' procedures

Category B: important objections / doubts

Objection	Technological / Structural Response
Enlarged damage by terrorist attack	reinforced structures, grace periods for AM
Technical risk of catastrophic dimension	Exclusion of catastrophic releases by design
Environmental damage by radiotoxic emissions	Exclusion of catastrophic releases by design, reduced operational releases
Impediment of better alternatives	'Coalition' & Support for introduction / subvention of alternatives
Potential for extortion of governments	reinforced structures, supervision
Proliferation of atomic weapons/materials	modified fuel cycles, burning of actinides

Category C: minor important objections / doubts

Objection	Technological / Structural Response
Enlarged damage by war	De-fuelling, underground siting
Environmental damage by waste heat	Co-generation (CHP), improved efficiencies
Tendency for environmental damage	'good-neighbour design'
Large electricity deficit after accidents	Reduced size of units, SMR
Dependency from Uranium supply	Stockpiles, high conversion, Thorium use
Restrictions in personnel rights (terrorists)	Robust designs, underground siting

Surveillance System ('Atom-State')	general problem (e.g. 'Oeko-State')
Restricted ability for defence	De-fuelling, underground siting
Restricted control of nuclear techniques	'Transparent' structures
Small participation chances	Open response on public concerns
High degree of estrangement	improved information / education

The examples show again that technological, structural and communicative responses are in principle also possible concerning most of the objections and fears. The evolutionary trends for advanced and innovative reactor designs and technologies are mainly coherent with these requirements from public judgement. In some cases the congruence of innovations or design improvements is even given by other reasons e.g. coping with internal impacts (hydrogen / steam explosions, missile impact) leads to robust containment structures as improved shelter against terrorist attack.

Even an underground siting for future NPPs as proposed for some American designs (MHTGR, PRISM) could be reconsidered if the aspects of improved shelter against terrorism and potential war would be decisive for acceptance (e.g., Sacharow and von Weizsäcker turned from pro-nuclear to anti-nuclear mainly due to this reason).

The security of electricity supply in times of shortages in energy supply or other crisis lead in some countries to enormous efforts on keeping own primary energy sources (e.g. hard coal mining, gas and oil storage). The judgement that they are safe is taken from the fact that they are underground-sited and always available even in critical situations. NPPs in such locations would - in contrary to the above mentioned assumption - represent a safe and reliable energy source even in a crisis.

Those reasons are of course strongly influenced by the general political situation and may be less important in times of globalisation, global inter-dependencies and safe / sufficient energy supply.

Other aspects have possibly also inverted since the end of the 'cold war' as the danger from proliferation is more due to the disarmament of weapon heads and huge amounts from that materials. It is not yet commonly known or accepted that nuclear reactors are able to destroy weapon-grade material but the nuclear community itself has not yet concluded on adequate techniques being optimised for that purpose or on the co-existence of different solutions. So, why should the public accept this as an argument in favour of nuclear reactors?

This situation of internal dissent of nuclear actors is given for many technological alternatives or responses as indicated above hindering consequent and flexible innovation approaches on public acceptance matters. A pro-active technological response based on a consensus within the nuclear community would possibly help to create new arguments for the nuclear case instead of blocking persuasive measures for destroying this sensitive material by ever-lasting discussions on technical alternatives. This attitude is not very much different from that of opponents arguing for some future alternate energy supply systems instead of nuclear.

The credibility of the nuclear community and its arguments would significantly increase if 'quasi-religious wars' and dualistic views on specific reactor lines and technical preferences from the past would be stopped in favour of credible innovation efforts coping with the actual and up-coming challenges for the nuclear case. It would also increase the credibility if the motivation for innovation and technological evolution would be based on the adaptation of reasons that are stemming from the psychological mechanisms for acceptability and from clearly defined reasons for objection and fear.

7. Conclusions / Deficits

Decisive for the acceptance of specific technologies may be the fact [9] whether they are judged as a 'limited risk' with a balancing of benefits and disadvantages or as a 'real fundamental danger' that has to be eliminated as such. This also influences the role either to use a technology in an active way or only being passively afflicted by it. Several other parameters influence the mechanism of acceptance in a rather complex way so that the - probabilistic - argumentation by the 'Risk Formula' does not seem to be adequate at all, especially for non-experts. The limitation of the plausible threat on the one side and of the quasi-eternal threat by nuclear waste - also by technological innovations concerning the safety of NPPs and the reduction / transmutation / safety of waste - may be a precondition for re-starting a rational acceptance debate.

Other objection against a technology may additionally result from

- pure conservatism
- negative cost /benefit relation
- exclusion from control and participation in technology development and application /use
- ecological goals
- reduced believe in technical progress
- social development into the direction of a 'Risk Society'

The industrial acceptance of nuclear energy is also threat by the 'risks' from these influences and reduced economic margins whereas the nuclear community still has to define a consensual strategy for responding to the overall acceptance situation and for the role of innovations.

Technologies are always judged as a socio-technical system within the social context of their development, control and application as well as their incorporation in political decision processes and the credibility of control.

The actual acceptance problems do not represent a general objection against innovations and are also not specific for certain countries. In some variation they are existing in all highly industrialised societies with some variation of the general scheme and mechanisms. Federalism, decentralisation and low national identification may influence the difference of interest between those who develop and apply a technology and those who feel threat by it. Mass media generate enhanced attention and especially offer a public forum for criticism and politicisation of conflicts with regard to specific technologies.

The risk / technology controversy is not a temporary pathologic effect but a characteristic of modern societies. It is very important to handle this conflict in a positive way and to develop - as a 'social innovation' - a culture of communication and dispute in conjunction with technological innovations that respond to the mechanisms and needs of acceptability.

Such an environment may help to get nuclear energy back into the row of candidates for sustainable technologies.

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