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High temperature gas cooled reactor technology development

Proceedings of a Technical Committee meeting held in Johannesburg, South Africa, 13–15 November 1996

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FOREWORD

The successful introduction of an advanced nuclear power plant programme depends on many key elements. It must be economically competitive with alternative sources of energy, its technical development must assure operational dependability, the support of society requires that it be safe and environmentally acceptable, and it must meet the regulatory standards developed for its use and application. These factors interrelate with each other, and the ability to satisfy the established goals and criteria of all of these requirements is mandatory if a country or a specific industry is to proceed with a new, advanced nuclear power system. It was with the focus on commercializing the high temperature gas cooled reactor (HTGR) that the IAEA's International Working Group on Gas Cooled Reactors recommended this Technical Committee Meeting (TCM) on HTGR Technology Development.

Over the past few years, many Member States have instituted a re-examination of their nuclear power policies and programmes. It has become evident that the only realistic way to introduce an advanced nuclear power programme in today's world is through international co-operation between countries. The sharing of expertise and technical facilities for the common development of the HTGR is the goal of the Member States comprising the IAEA's International Working Group on Gas Cooled Reactors.

This meeting brought together key representatives and experts on the HTGR from the national organizations and industries of ten countries and the European Commission. The state electric utility of South Africa, Eskom, hosted this TCM in Johannesburg, from 13 to 15 November 1996. This TCM provided the opportunity to review the status of HTGR design and development activities, and especially to identify international co-operation which could be utilized to bring about the commercialization of the HTGR.

EDITORIAL NOTE

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SUMMARY

The Technical Committee meeting (TCM) on High Temperature Gas Cooled Reactor (HTGR) Development was held in Johannesburg, South Africa, from 13 to 15 November 1996. The meeting was convened by the IAEA on the recommendation of its International Working Group on Gas Cooled Reactors (IWGGCR), and was hosted by the state electric utility of South Africa, Eskom. Approximately eighty participants and observers from ten countries (China, France, Germany, Indonesia, Japan, Netherlands, the Russian Federation, South Africa, the United Kingdom and the United States of America) and the European Commission attended the TCM. Thirty-two papers were presented covering a large range of topics including:

- Summaries of HTGR national development and commercialization programmes,
- Safety and Management,
- Development of the pebble bed modular reactor in South Africa,
- Status of HTGR test reactor programmes,
- HTGR plant component and system design development.

Significant activities are occurring in the advancement of HTGR development, particularly with regard to the utilization of the gas cooled reactor to achieve high efficiency in the generation of electricity and in process heat applications. Technological advances in component design and processes such as heat exchangers, turbo machinery, reformers and magnetic bearings coupled with the international capability to fabricate, test and procure these components provides an excellent opportunity for achieving economic commercialization of the HTGR. Eskom is in the process of performing a technical and economic evaluation of a helium cooled pebble bed modular reactor directly coupled to a gas turbine power conversion system for consideration in increasing the capacity of their electrical system. In China and Japan, test reactors are currently under construction which will have the capability of achieving core outlet temperatures of 950°C for the evaluation of nuclear powered heat process applications. This TCM provided the forum to share the recent international advances in HTGR technology, and to identify technical and economic development pathways where international co-operation will support and encourage commercialization of this advanced nuclear power plant.

The TCM was opened by J. de Beer, Executive Director for Technology, Eskom, G.P.N. Venter, Director General of Minerals and Energy for South Africa, and L. Brey for the IAEA. J. de Beer provided an overview of the technical and economic evaluation of the HTGR currently in progress by Eskom. He stressed the factors influencing Eskom's interest in the HTGR as a possible future source of electrical generation and the long term trends in system growth which tend to support this advanced nuclear power plant. G.P.N. Venter addressed the South African government's plan to uplift the economy, stressing the societal requirements of the nation. Abundant and inexpensive electricity is basic to South Africa's economic and social growth. He emphasized the development and application of high technology as being the ultimate long term wealth creator in any country, and that it was the creation of a technology based future that was important to the decision by South Africa to host this TCM. L. Brey spoke about the timeliness and importance of this meeting based on the current and anticipated HTGR related activities of the Member States comprising the IWGGCR. He indicated that the TCM was being convened as an activity of the IAEA in the performance of its mission to foster the international exchange of scientific and technical information in the peaceful application of atomic energy throughout the world.

The TCM provided the opportunity for the representatives of national organizations and industry to address their individual HTGR related programmes and plans. Although the structure of the meeting was divided into individual sessions based on specific topics, there exists a strong interrelationship between countries in the sharing of technical expertise, test facilities and in the design, fabrication and procurement of systems and components.

South Africa provided an excellent location for this TCM. In mid 1995, Eskom initiated a detailed economic and technical evaluation of the Pebble Bed Modular Reactor (PBMR-SA) as a potential candidate for future additions to its electric generation system. This utility, with an installed generation capacity of ~38,000 MW(e), is experiencing a number of significant changes in load growth. Eskom's electrical system is currently concentrated in the heavily industrial Johannesburg region with minimal electrical interties to neighboring countries and limited transmission capacity to the coastal regions of the country. This north central area of South Africa is also the location of South Africa's large coal deposits which provide the fuel for about 90% of the utility's electrical generating capability. Recent societal changes which reflect an opening of South African industry to the world market are causing a shift of the nation's new industrial plant to the coastal regions of the country. This, coupled with the need for future generating capacity, the lack of transmission capability and the wish to diversify its energy sources are the primary reasons for this interest in the HTGR.

The requirements set by Eskom for the installation of new generation capacity include a capital and operation cost which must match (or improve upon) that being achieved by their large coal stations. This currently represents a retail power cost to the customer of approximately two US cents per KW•h. Other requirements for the plant include an availability approaching 90%, location and plant size to match the load, public acceptance and environmental cleanliness.

Of paramount importance in the design and public acceptance of this advanced nuclear power plant are the safety considerations and corresponding licensing requirements established for its implementation. The South African Nuclear Regulatory Authority (CNS) is the responsible government agency in nuclear licensing and regulatory matters. The licensing process envisioned for the PBMR-SA will be based on quantitative risk assessment to demonstrate compliance with fundamental safety criteria laid down by CNS. This is similar to the licensing process established for the Koeberg Nuclear Power Station and other nuclear installations in South Africa. The design approach utilized by Eskom is to establish conservative boundary limitations on the PBMR-SA to assure safety under all operational and transient conditions.

The conceptual design of PBMR-SA features a helium cooled pebble bed reactor with a power output of approximately 220 MW(th) coupled to a closed cycle gas turbine power conversion system consisting of two turbo compressors, a turbo generator, a recuperator, precooler and an intercooler, all located within three steel pressure vessels The three turbo machines are equipped with magnetic bearings and the recuperator is of a fin plate design for compactness. The overall net efficiency of this Brayton cycle system is expected to exceed 45% based on a reactor outlet helium temperature of 900°C and a maximum system pressure of 70 bar.

The South African engineering firm Integrators of Systems Technology (IST) is performing the detailed evaluation for Eskom with support from fifteen industry and national organizations worldwide. Although the design of the PBMR-SA is quite simple with only the need for a minimum of support systems, major portions of the plant have been modelled in an engineering simulator to allow investigation of system and component interaction, to study the plant's power response capabilities and to develop its control philosophy.

The PBMR-SA reactor basically builds on German designs utilizing the experience from the Thorium High Temperature Reactor and the AVR as well as the technical developments associated with the 200 MW(th) HTR-MODUL and the 250 MW(th) HTR-100 plants. These plants utilize a steam cycle in contrast to the Eskom design for a direct cycle helium turbine. Although this modifies certain boundary conditions, the nuclear and thermal hydraulic investigations performed at the Kernforschungszentrum Jülich GmbH (KFA) to determine the optimum power of the core have reinforced the conservative design originally defined for the HTR-MODUL. The choice for a core design limited to 220 MW(th) with a diameter of 3.5 meters and the use of graphite constrictions for nuclear control and shutdown outside of the pebble bed provides conservatism in maintaining the maximum accident fuel temperature to 1600°C. Also, the German designs have in the past used a multiple pass fuelling regime in their reactors. A possible change in fuelling for the PBMR-SA to a once-through-then-out technique with boron-carbide coated particles for controlling the neutron flux profile is currently under consideration. While this technique provides the advantages of simplification in overall fuel management, allowing off-line defuelling and considerable capital savings due to simplified equipment requirements, consideration must be given to the possibility that this fuelling method may cause a limitation on core power to under 200 MW(th).

Other design considerations for the PBMR-SA include the components for the power conversion unit (PCU). The Atomic Energy Corporation of South Africa (AEC) is supporting the preliminary design for the two turbo compressors, the power turbine, precooler, intercooler, and the recuperator. GEC Alsthom of France is providing the preliminary design of the electric generator and the associated exciter. The three rotating shafts of the PCU are to be supported by magnetic bearings which are also being designed by AEC. The recuperator is of the compact perforated fin plate type with the precooler and intercooler being of conventional finned tube design. The thermodynamic loadings of the precooler and intercooler are anticipated to be nearly identical, which may allow for interchangeability of components. The electrical generator will basically be of standard design with the additional requirements of operating in a vertical configuration on magnetic bearings and with a high pressure helium atmosphere. However, these requirements are not considered to represent significant design concerns. Both AEC and GEC have had previous experience related to the design of similar PCU components.

Additional PBMR-SA related presentations focused on the plants' seismic and structural considerations, the design of the helium storage and control system, and the simulation of the passive natural air convection heat removal system. Based on conceptual design considerations, it is projected that the use of natural air convection cooling of the outer vessel surface will limit the reactor vessel temperature to below 380°C for a loss of forced cooling event. An alternative study by KFA provides for the possibility of vessel cooling through the use of a self-acting heat removal system based on the boiling of water.

Five companies and institutions in the Netherlands are currently investigating the application of a small HTGR for heat and electricity cogeneration. The base design chosen for this study is the 40 MW(th) helium cooled pebble bed Inherently Safe Nuclear Cogeneration (INCOGEN) plant featuring a closed cycle gas turbine to generate about 16.5 MW(e) with a precooler/heat exchanger system to provide about 18 MW(th) for process heat applications. This project, under the co-ordination of the Netherlands Energy Research Foundation (ECN), is evaluating the use of small nuclear installations which are tailored to the heat requirements of local industry to determine the power rating of the plant. Their evaluation supports a nuclear energy source with inherent safety features which become more prominent with smaller plant size; and with cogeneration applications where the economic savings is in a very simplified plant with a high thermal efficiency. The market potential for this plant includes both new industrial users and the replacement of existing diesel or natural gas turbine driven systems. A major technical development effort currently in progress for the INCOGEN plant is to extend the ECN reactor physics code system to perform combined neutronics and thermal hydraulics steady state, burnup and transient calculations on the pebble bed HTGR.

Indonesia is undergoing substantial growth in its energy requirements and, although it has significant oil and gas resources in production, these are diminishing rapidly. An assessment currently underway by the National Atomic Energy Agency of Indonesia (BATAN) with support from the IAEA, is the evaluation of the HTGR as a cogenerator heat source to be used in a reforming process to convert natural gas into syngas as feed material in producing automotive fuels. A specific application of carbon dioxide reforming of low hydrocarbons is the vast Natuna gas field, with the unusually high CO_2 content of approximately 71%. The Natuna field, as well as other gas field projects on Indonesia's many islands, is being investigated by BATAN for the promotion of syngas production through the application of the HTGR. Cost projections indicate that there exists a potential for economic competitiveness for exploitation of the Natuna gas field through HTGR cogeneration (reforming plus electrolytic hydrogen) in the production of energy alcohol. This product is of high market value as a substitute for gasoline and diesel fuel. Another process under consideration for the HTGR is in the recovery of heavy oil. In many cases more then 50% of the original oil remains after primary and secondary recovery processes have been completed. The feasibility of using the HTGR for recovery of this oil through the utilization of steam flooding has been the focus of a study for the Duri oil field. A recent reassessment of this HTGR application to the Duri field has shown that a two-fold reduction in thermal power requirements can be achieved through a modification in the steam injection well pattern.

Construction of China's High Temperature Reactor (HTR-10) continues, with initial criticality anticipated for 1999. This pebble bed reactor of 10 MW(th) will be utilized to test and demonstrate the technology and safety features of the HTGR. Development of the HTGR by China's Institute of Nuclear Energy Technology (INET) is being undertaken to evaluate a wide range of applications such as electricity generation, steam and district heat production, combined steam and gas turbine cycle operation, and as process heat generation for methane reforming. The HTR-10 is the first HTGR to be licensed and constructed in China. The safety design criteria implemented for this plant includes a negative temperature coefficient for all operating conditions, a maximum fuel temperature limit of 1600°C for all events, two independent and diverse reactor shutdown systems, accident avoidance resulting from control rod ejection, a maximum equivalent diameter of 65 mm for the rupture of any primary coolant boundary line, separate pressure and heat resisting functions for the system structure, and a passive residual heat removal system able to transfer decay heat from the reactor vessel at a rate which will maintain the fuel and pressure vessel within design conditions. INET has undertaken a programme of engineering experiments to verify the design characteristics and performance of the HTR-10 components and systems. These include depressurization tests of the hot gas duct from operating pressure, tests of the control rod drive mechanism, performance validation of the hot gas duct under operating pressure and temperature conditions, two phase flow stability testing for the once through steam generator and performance testing of the fuel handling system.

HTGR development in the USA currently centres around General Atomics (GA). The principle focus of this activity has been in the design of the Gas Turbine-Modular Helium Reactor (GT-MHR). This plant features a 600 MW(th) helium cooled reactor with prismatic fuel elements as the energy source coupled to a closed cycle gas turbine power conversion system (PCS). The net efficiency of the GT-MHR is 47.7% with an electrical output of 286 MW(e). The integration of a HTGR into a closed Brayton cycle PCS has long been recognized as a highly efficient means of producing electricity. Substantial progress in the development of this plant has occurred through the technological advances of components such as magnetic bearings and fin plate recuperators. In 1993, GA and MINATOM combined their efforts for the cooperative development of the GT-MHR for commercial deployment. This was subsequently expanded in 1994 to include deployment of the GT-MHR for the disposition of weapons plutonium and included joint funding of conceptual design preparation. In early 1996, FRAMATOME joined with MINATOM and GA as a participant in this co-operative programme. This programme currently addresses US and Russian conditions for the GT-MHR including safety, regulatory, operational, economic, and performance requirements.

The Russian gas cooled reactor development programme includes a pebble bed modular HTGR with a thermal output of 200 MW(th) aimed at process heat applications and the above mentioned 600 MW(th) GT-MHR for electricity production. Ongoing and future research and development work in support of the GT-MHR includes fuel assembly and fabrication design for fuel particles, compacts, and blocks; tests of the reactor and power conversion system including evaluation of the traveling seals to reduce helium leakage; and studies of vessel material mechanical properties to 500°C and large scale vessel production techniques.

Japan provided a comprehensive overview of their HTGR research and development programme. The principle focus of this programme is completion of the High Temperature

Engineering Test Reactor (HTTR) at the Japan Atomic Energy Research Institute (JAERI) site in Oarai, Japan. This 30 MW(th) helium cooled reactor will be utilized to establish and upgrade the technology for advanced HTGR development and to demonstrate the effectiveness of selected high temperature heat utilization systems. Initial criticality of the HTTR is scheduled for October 1997, with commencement of the startup physics test programme beginning with fuel loading (mid 1997) and continuing throughout 1998. This comprehensive programme will include physics tests associated with approach to criticality, testing of the preliminary annular core consisting of 18 fuel columns and zero power tests of the full core of 30 fuel columns. Among these start-up tests will be measurements of control rod reactivity worth, scram reactivity, excess reactivity, neutron flux distribution and reactor noise analysis for both the annular and the full core. The HTTR evaluation programme will then continue with rise-to-power tests to full power with a core outlet temperature of 950°C.

Demonstration of the effectiveness of selected high temperature heat utilization systems will then begin with the testing of hydrogen production by natural gas steam reforming. The HTTR steam reforming system is expected to demonstrate the ability of nuclear heat to achieve hydrogen production at costs competitive with a fossil fired heat source. An out-of-pile testing program to confirm the safety, controllability and performance of this system will be performed prior to its actual evaluation with the HTTR. Other studies being performed by JAERI include the design evaluation of gas turbine systems for the production of electricity and laboratory scale development of the thermochemical IS hydrogen production process. Selected future processes for demonstration with the HTTR are anticipated to include these heat utilization systems.

Development, fabrication and testing programmes for specific HTTR components were also reviewed. The fuel assembly for this reactor is the pin-in-block type of hexagonal graphite block which uses Triso coated fuel particles. To date, half of the initial HTTR fuel has been fabricated with a current fuel failure fraction of ~5 x 10^{-5} as measured by the burn and leach method. This Trisco coated fuel utilizes a silicon carbide (SiC) coating for mechanical strength and as a barrier to the diffusion of metallic fission products. Because of gradual lose of mechanical strength above 1700°C, research into replacing the SiC coating with zirconium-carbide (ZrC) has shown that the ZrC Triso fuel particles have better caesium retention, but provide a less effective barrier to ruthenium than the Triso particles which incorporate the SiC layer. Also, the HTTR utilizes graphite as the moderator and to provide strength to the reactor. This material is subjected to a wide range of temperatures and high neutron fluence. Understanding the residual strain and/or stress accumulated in the graphite components is very important and is the subject of the development by JAERI of a non-destructive examination technique utilizing micro-indentation to estimate residual graphite strain.

Pressure tests on the HTTR primary and secondary cooling system were successfully performed in March 1996. This included the pressure testing of the reactor pressure vessel (RPV). The selection of 2 1/4 Cr-1 Mo steel for the fabrication of this critical component was determined by its relatively high temperature, low neutron fluence environment. The technology incorporated into the design and fabrication of the HTTR RPV is expected to be readily applicable to future commercial HTGR vessel requirements.

Future deployment of the HTTR receives considerable interest from key industry, national and academic organizations from Japan. Without exception, these organizations look to the HTGR as a very important future energy source for process heat applications. The HTR Heat Utilization Core Group is considering the use of the HTGR in three fields: as a replacement for fossil fuels, in the production of clean fuels from fossil resources and in the production of hydrogen from water as the ultimate clean energy carrier. An important aspect of the efforts under way by the Research Association on HTGR Plants is to make people aware of this safe and efficient nuclear power source so that the many benefits it can provide in meeting society's energy needs of the future are clearly understood by the decision makers. Mitsubishi Heavy Industries is investigating the feasibility of HTGR commercialization for the future. All of these TCM contributors acknowledged the positive environmental aspects of the HTGR in helping to meet the energy needs of the future.

Significant in the international co-operation for HTGR technology development is the offer by Japan and China for use of the HTTR and HTR-10 test reactors in a proposed new Co-ordinated Research programme (CRP) on Evaluation of HTGR Performance. Chief Scientific Investigators will utilize these facilities for performance based tests and HTGR code and model verification throughout startup, steady state and transient operational conditions. The areas to be evaluated in the HTTR and HTR-10 testing programmes include core physics, safety characteristics of the reactors, fission product release and transportation behaviour, thermal hydraulics, control response and high temperature component performance. This proposed CRP is in addition to a previously established CRP which is utilizing the HTTR for the investigation of high temperature heat utilization systems.

Development of the HTGR is of principal importance to the Member States comprising the IWGGCR. A major recommendation resulting from the 1996 International Working Group meeting is to proceed with the creation of a central function for the overall co-ordination of national research and technology development programmes, including establishment of a HTGR information and data archiving system and for the international advancement of this advanced nuclear power source. This has led to the formation of the Global HTGR R&D Network (GHTRN). Although the general charter and details for the scope of the GHTRN are still under development, it has been determined by the IWGGCR that co-ordination of the HTGR R&D related resources of each country by a central c-oordination function is the most effective and efficient method available to support and advance the HTGR.

In summary, two areas of particular significance stand out for commercializing the HTGR. These included initial plant capital costs and the current state of plant system and component development. It was acknowledged that the on-going economic evaluations for both the PBMR-SA and the GT-MHR programmes are estimating plant capital costs approaching US$ 1000/installed KW(e). The reasons provided for this low cost include a very simplified modular plant design, licensing requirements commensurate with the safety attributes of the HTGR, and competitive procurement of components on an international basis. Also acknowledged for the PBMR-SA is that the current international technology base allows for the design and construction of a full scale prototype plant without the requirement for additional extended research and development.

This TCM provided the opportunity for sharing of recent international advances in HTGR technology. This included a review of the elements which collectively provide for the successful introduction of this advanced source of nuclear power. Major considerations for commercializing the HTGR were explored including the state of technology development, economic competitiveness, environmental acceptance and safety, public acceptance and the regulatory standards for its use and application.

OPENING ADDRESS

P. Maduna
Minister of Minerals and Energy Affairs,
Government of South Africa,
Pretoria, South Africa

Presented by G.P.N. Venter

I am very pleased and feel honoured to be here to address the opening remarks of the IAEA Technical Committee Meeting on "High Temperature Gas-cooled Reactor Technology Development". It is an honour for South Africa to be the host country for the TCM. The history of the High Temperature Reactor dates back to 1966, when a 15MW prototype reactor was commissioned in Juelich, Germany. It is reported that our coal reserves will last for another 100-120 years. Therefore we need to investigate all possible alternative power generation and supply options that could be made available for South African in the medium to long term. This implies that our research efforts into energy supply should, in addition to the nuclear option, also vigorously pursue interest in renewable energy sources such as the solar, wind, hydro and tidal options.

South Africa has been blessed with extensive natural resources which are vital to improving the well-being of the citizens of this country. The need for electricity is basic to our economic and social growth.

We are a mixture of a developed and developing nation. A prime example of this is the electricity supply industry.

At the distribution end of the industry, the biggest task is that of providing electricity to the majority of the population that has no main supply. In this area, the targets set down by the Reconstruction and Development Program are being successfully met. I must give credit to Eskom in this regard, as they are currently reportedly meeting their commitment of 300,000 new connections per year.

The other end of the electricity supply industry, the power stations, are clearly part of the developed economy. South Africa's power stations include Kendal, which at over 4000MW, is the largest hard coal power station in the world; and Koeberg, the only nuclear power station on the African continent. In total, the country has over 36000MW of installed plant, with Eskom being the fourth largest utility in the world by this measure.

The government's macro-economic plan is aimed at uplifting the economy, and to ensure that the social and economic inequalities that exist are overcome. If this is to be done, it must be based, among other things on the development and application of high technology, as this is the ultimate long term wealth creator in any country.

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

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

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

This current low cost of electricity in South Africa is due to the highly successful construction program of coal stations in the 1970s and 1980s, and the reduction of costs. This building program was, in fact, somewhat over ambitious as it left the country with an overcapacity of plant, It is only now, with several years of good growth, that this excess is being gradually reduced. Eskom is therefore investigating a number of options for potential future generating plant, with specific emphasis on competitive costing.

These investigations include further coal based plant, imports of hydroelectric power from neighbouring countries, underground pumped storage systems, solar power stations, gas fired combined cycle plants and nuclear power, as I have already mentioned.

It is pleasing to see that a long term vision is being increasingly taken over the electrical energy needs of the country. It is only with detailed and thorough studies that the correct decisions can be made when the time comes to invest in new plant. The economic targets for these technologies are particularly tough, given the cost of power they have to match.

The nuclear studies originally considered stations similar to the existing nuclear plant at Koeberg, but given the condition of cost-competitiveness and the ongoing debate in the international arena, particularly in relation to environmental constraints, it was concluded that there is more potential in high temperature gas-cooled reactors, based on proven German technology, at least at the level of the prototype reactor.

This is not to say that South Africa has had anything but good service from the Koeberg power station near Cape Town over the last twelve years, and there is no reason to believe that it will not continue until the end of its design life of forty years for reactors of this type, i.e. the pressurized water reactor, and other current generation designs, can be seen as the workhorses of the nuclear industry.

I am told that it was while reviewing the potential of the high temperature gas-cooled reactor, particularly in terms of its inherent safety, that Eskom, in conjunction with local and overseas industry, entered into the current investigations. In reviewing the targets that these studies are aiming to demonstrate, it is clear that they are quite demanding. If they are met, it will only be because of the hard work and dedication of all the scientists and engineers involved, both in South Africa and overseas.

I should also mention that in terms of government policy, my department expects that mainstream involvement of all of our scientists and engineers in these endeavours, and not only those from the white population group, will be a reality in the execution of these investigations. In other words, black economic empowerment must be vigorously pursued and encouraged through projects of this nature. As a country and as an economy, it behoves us to do everything in our power to ensure that blacks are brought into the centre of our economy where they will play roles above the realm of purveyors of cheap unskilled labour that were assigned to them in the past.

If an improvement can be made in the peaceful commercial performance of nuclear power, the potential benefits would be significant, both in terms of the low cost of power and the improvements in environmental emissions.

Finally, let me express my wish for the success of this technical committee meeting. It is my sincere hope that this meeting will result in a quality programme of international exchange and co-operation toward further analysis and evaluation of the HTGR as a possible energy source for future world electricity supply.

GAS COOLED REACTOR PROGRAMME DEVELOPMENT

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UTILITY REQUIREMENTS FOR HTGRs

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Abstract

Eskom, the state utility of South Africa, is currently evaluating the technical and economic feasibility of the helium cooled Pebble Bed Modular Reactor with a closed cycle gas turbine power conversion system for future power generating additions to its electric system. This paper provides an overview of the Eskom system including the needs of the utility for future generation capacity and the key performance requirements necessary for incorporation of this gas cooled reactor plant.

INTRODUCTION

Eskom, the South African state utility, is currently investigating the viability of HTGRs as part of its future generating mix. This paper explains the background to Eskom's investigation and key requirements for any possible HTGR program.

SOUTH AFRICA'S ELECTRICITY SUPPLY INDUSTRY

Eskom generates over 95% of the electricity in South Africa with peak sent out demand in 1996 of 27 967 MW. Over 90% of this capacity is represented by large coal-fired stations (mainly 6 units of 600-700MW per station) fed by dedicated low cost coal mines. Eskom's current cost of coal is approximately R33/ton (US$7.50). This has led to a very low average retail power cost of approximately two US cents per kWh. This low cost structure has been achieved by a successful construction program of these coal stations in the 1970s and 1980s. This program achieved low costs because of its scale, both in terms of the size of the stations and the number of units ordered.

The program, in fact, proved to be over optimistic due to a fall in load growth from 8%/y to 5%/y in the early 1980's and Eskom over built, achieving a 50% reserve margin in the late 1980s and early 1990s. This led to the mothballing of some of the older coal stations (4513MW) and the delaying of the last station of the construction series, Majuba, at 4117MW, by eight years. With the growth of demand since that time this excess has been reduced and the operating reserve margin (excluding mothballed stations) for the utility at the peak demand in 1996 was below 15%. The issue facing the utility is therefore whether the current growth in demand will continue and how, if it does, to construct power stations in line with our current cost structure. These issues relate to total installed capacity, lead time and cost. There are also other factors which are starting to impact the utility. They are as follows.

The South African industry is very heavily dependent on primary industry: this is reflected in the load profile. Most of the extensive deep mining operations and heavy industry (steel, ferrochrome, aluminium etc.) operate on a 24 hour basis. The domestic market has, historically, been quite small as the black population had very limited access to electrical supply. The result was that Eskom installed very limited load following plants. The only ones that are used operationally are the two pumped storage schemes (1400MW) and limited hydro (600MW). As the drive to electrify the black townships continues (300 000 connections a year by Eskom alone, plus 150 000 by other authorities) and the secondary manufacturing and service industry grows, the need for load following plant is growing significantly.

In the apartheid era the South African industry became very inwardly focused. As the majority of the economically active population, and therefore customers, are close to Johannesburg the industry tended to be based close to this area, and away from the coast (on average 1000km away). The heavy industrial electrical load was therefore in this area and this was good for Eskom as the country's large coal deposits are in the same area. It is of note that the only non-coal base load station is the nuclear station (Koeberg 1800MW) near Cape Town, which is some 1400km from the next station (Tutuka, a 4000MW coal fired station). As South Africa has opened itself up to the world market after the fall of apartheid the need to base export orientated industrial plant near the coast has become apparent. Recent examples include the Hillside Smelter in Richards Bay (850MW load) and the Saldanha Steel Mill near Cape Town (200MW). There are plans for several more such major plants. This concentration of load on the coast will place severe demands on the Eskom transmission system as well as increasing the problems of quality of supply.

UTILITY NEEDS

Given this scenario, what are the outline requirements for new capacity for Eskom? The present analysis would indicate the following priorities:-

Cost	The capital and operating cost must match (or improve on) that achieved by large (4000MW+) coal stations.
Lead Time	The lead time must be as short as possible to avoid the type of over-capacity Eskom experienced at the end of the 1980s.
Load Following	The station must be able to load follow to compensate for the limitations on Eskom's current capacity.
Availability	This must be as high as possible. Eskom's current target for existing stations is 90% (7% planned outage and 3% forced).
Location	The plant should be able to be located where the load is without impacting the overall costs.
Environment	It is vital that any new plant must be environmentally (and publicly) acceptable.
Fuel Diversity	Although not vital it would be valuable to increase Eskom's diversity of fuel supply (currently 92% coal).

KEY PERFORMANCE FOR PBMR

The HTGR Eskom is investigating is the Pebble Bed Modular Reactor, a HTGR based on German fuel design using a closed cycle gas turbine for power conversion. Each of the above requirements has been reflected in the performance targets set for plant. It is accepted that these targets are EXTREMELY ambitious, but it is believed that they are achievable.

Unit Size	As a reference the module size is intended to be 100MW electrical output (220MW thermal). This size is dictated by the limits of the German Modul style core (without internal structures). The size is also very appropriate for sites as it allows multiple modules to support a single city.
Cost	The target capital cost, including interest and owner's costs, is under $1000/kW, which is in line with the costs of a coal fired station. It is assumed that this cost will only be achieved under series construction conditions (10 or more modules). The fuel cost is aimed to be comparable with current fuel costs at Koeberg (Eskom's PWR), which is in line with past German studies. The O&M costs are difficult to determine but, given the inherent simplicity of the system, a target figure of R3-6/MWh is expected. This just below Eskom's best figures for current coal stations.
Construction Time	The construction time for the series construction is targeted at 24 months, with 36 months for the lead module.
Load Following	The current reference design is a 10% per minute ramp increase in power from 20% to 100% power. A 10% step of current power level inside 1 second (e.g. 40% to 44%) and any load reduction (up to 100% load rejection) without trip. The plant is intended to operate at any power level (0-100%) on a continuous basis. Plant efficiency should not materially change between 20% and 100%.
Availability	Eskom's current plant is achieving over 90% availability. The target for the PBMR is a single planned outage of 30 days every six years. The trip rate is intended to be 1 per 70 000 hours, which is in line with current best practice for power reactors.
Location	The restrictions on siting for a nuclear plant are principally defined by the emergency plan requirements. The target for the PBMR is to have no planning restrictions beyond 400 meters from the reactor. In terms of seismic design the reference design is currently being aimed at the existing Koeberg site (with an SSE defined as horizontal acceleration of 0.3g) but the requirement is to be able to meet 0.5g for future versions without significant cost impact.
Environment	The current criteria for normal off site releases are seen, from a technical point of view, to be adequate. It should be readily possible to meet current severe South African criterion (250 uSv/y) for

Environment (cont.)	effluent releases during normal operation. The other related technical issue is that of waste, both process waste and spent fuel. The current strategy is to limit the processes which generate waste and to provide adequate storage in the plant systems to store all the spent fuel generated during the operating lifetime of the plant and to provide storage facilities until final disposal.
Fuel Diversity	In Eskom's terms the use of nuclear fuel could be seen to be advantageous as it diversifies away from the present heavy reliance of coal.

The item missing from this discussion is public acceptance. It is possible to build a station which meets the technical (and licensing) requirements but which does not have public acceptance. In many ways this has been the "Achilles Heel" of the current generation of nuclear plant. Public concern is based on two perceived problems, disposal of waste and accidents. The waste problem needs to be addressed in a wider context than a single design but it is vital that it is seen to be resolvable. The issue of accidents also must be seen to have been solved. The classic question is "Can the nuclear plant have an accident which could affect the public?". The answer for all current generation is "Yes, but it is such a remote possibility that ....". The only part of this answer that is heard is the first word, the rest is only limited mitigation! To be acceptable the answer must be "No". There must be no physically credible event which can necessitate offsite action.

As can be seen, the current targets for the PBMR are based in the needs of the utility. They are seen as very tough but if they can be demonstrated to be achievable during the current phase of investigation, the way may be open for further work on the project.

ANNEX A - ESKOM STATISTICS

ESKOM Peak Demands(1986-1996)

1986	18 278 MW
1987	20 001 MW
1988	20 587 MW
1989	21 871 MW
1990	21 863 MW
1991	22 342 MW
1992	22 460 MW
1993	23 169 MW
1994	24 789 MW
1995	25 133 MW
1996	27 967 MW

ESKOM Installed Capacity (gross) - 1996

Coal	34 125MW
Gas Turbine (oil fired)	342MW
Hydro	600MW
Pumped Storage	1 400MW
Nuclear	1 930MW

This equates to a total sent out capacity of 36 563MW of which 4 531MW is in reserve storage

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INCOGEN: NUCLEAR COGENERATION IN THE NETHERLANDS

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Abstract

A small heat and power cogeneration plant with a pebble bed high temperature reactor (HTR) is discussed. Cogeneration could be a new market for nuclear power and the HTR could be very suitable. The 40 MWth INCOGEN system is presented. Philosophy, layout, characteristics and performance are described. The lower power level, advanced component technologies and inherent safety features are used to obtain a maximally simplified system. Static and dynamic cycle analyses of the energy conversion system are discussed, as well as the behaviour of the reactor cavity cooling system. Although the cost study has not been finished yet, cost reduction trends are indicated.

1. Introduction

In Western Europe and the United States, the traditional market for nuclear power plants, large scale base load power generation, is stagnating and sometimes even shrinking. A healthy nuclear power industry has three branches of activities: plant construction, maintenance and decommissioning. If the size of the first activity is shrinking, other markets have to be found.

The market for autoproducers is growing fastly. In this branch a lot of power is generated in combination with heat: cogeneration. This market is fully served with fossil fuels now, mainly natural gas. It is expected that the natural gas price will not remain as stable and low as it is now. In principle, there should be a place for nuclear power on this market. In that case, three requirements should be fulfilled:

1. repair of social support
2. economic performance
3. acceptable prototype investment

The High Temperature Reactor can, if properly designed, serve as the nuclear heat source for such a cogeneration plant.

From the beginning of 1996, a consortium of five companies and institutions are working together on a cogeneration system with a pebble bed HTR of the peu-à-peu concept as a heat source. The name of the system is INCOGEN: Inherently safe Nuclear cogeneration. Sponsor is the Dutch Ministry of Economic Affairs, both through direct funding of ECN as their "house institute" and through the Programme to Intensify Nuclear Knowledge (PINK). The five companies/institutions are listed in Table 1.

For a product like a cogeneration plant, it is necessary to get an idea of the requirements of potential customers and the possibilities of the component manufacturing industry. Therefore in July 1996, an introduction day was organised on which the philosophy of the nuclear cogeneration was presented. The response was quite positive. About ninety people attended the meeting, and apart from a massive, for a nuclear project relatively positive press and media coverage, about ten companies were interested to become more involved in the project. The newsletter INCOGEN Bulletin [1] gives a description of the event and it's follow-up.

Table 1. Companies and institutions participating in the INCOGEN project.

company/institution	task
Netherlands Energy Research Foundation (ECN), Petten	project coordination, reactor dynamics, shielding
Stork NUCON, Amsterdam	plant layout, energy conversion
KEMA, Arnhem	cycle optimisation, licensing
Interfaculty Reactor Institute (IRI), Delft	reactor statics, burnup calculations
ROMAWA, Voorschoten	market analysis, analysis of Dutch industry potential

In this paper, philosophy, layout, characteristics and performance of the INCOGEN plant is described. A description of the reactor physical behaviour van be found in the paper of J.C. Kuijper [2], and an analysis of the potential markets and the management system in the paper of G.A.K. Crommelin [3].

2. Philosophy

Unlike for traditional nuclear power systems, the economy-of-scale philosophy cannot be applied at cogeneration. Heat requirements of the local industry will determine the power level, which in most cases will be no more than several tens of megawatts. Economic performance therefore must be obtained by simplicity. Three forms of simplification can be identified:

1. power level: In comparison with older, larger pebble bed reactor designs, the low power level only gives opportunities for simplification. An example is the defuelling machinery, which can be omitted. From a certain power level on, the vessel size could become unacceptably large. For a cogeneration plant, the peu-à-peu concept can be used instead of the recirculation (MEDUL) concept of the HTR Module [4].
2. new or advanced technologies: because of advances in fuel development (TRISO coated particles vs. BISO in THTR), and in gas turbine technology, application of the direct Brayton cycle comes within the realm of possibility.
3. inherent safety features: C. Goetzmann [5] gives a number of design guidelines for small reactors, amongst them: Use inherent safety features - that become more prominent with smaller plant sizes - for lowering capital cost. Major cost drivers are nuclear grade components, i.e. components with a nuclear safety function, Using inherent safety features, their number could be minimized. An example is the core behaviour at core heatup accidents combined with stuck control rods. As is shown in the paper of J.C. Kuijper [2], the fuel never reaches temperatures where the coated particles are damaged, even not after reaching recriticality.

These features are used to attain a maximum of transparency. It should be possible to explain to anybody, laypeople included, why e.g. an emergency core cooling system is not needed. The core damage grace period is to be infinite, not some days or even a week. To exclude any discussion about large radioactivity releases, the fission products are contained at their source by the fuel grain coatings, but also protected against attacks from outside by a silicon carbide coating around the fuel element.

It is expected, that with the use of these simplification opportunities, the disadvantage of the lack of economy-of-scale can be more than compensated.

3. Layout and characteristics

Although the parameters of the INCOGEN plant are not fixed yet, a first study has been performed to get an idea of main parameters and costs. By means of a market survey, a final power level, to be expected between 20 and 200 MWth, will be determined. In the base study the power level has been set to 40 MWth. Main parameters are listed in Table 2.

Table 2. Main parameters of INCOGEN nuclear cogeneration plant.

Core thermal power (MWth)	40
Core outlet temperature (°C)	800
Core inlet temperature (°C)	494
Helium pressure (bar)	23
Net electric power generation (MWe)	16.5
Net heat cogeneration (MWth)	18.0
Net electric generation efficiency (%)	41.2
Net heat generation efficiency (%)	44.9
Net plant thermal efficiency (%)	86.1

A drawing of the INCOGEN cycle design is shown in Figure 1. The heat source of the plant is a 40 MWth helium cooled graphite moderated pebble bed HTR, called HR1. The reactor is directly coupled to a helium turbine. After leaving the turbine the still 522°C hot helium enters a recuperator and a precooler consecutively. In the precooler a water circuit for process heat applications is heated. After the precooler the helium enters a compressor that is mounted on the same shaft as the turbine and the generator. Via the recuperator the helium enters the reactor core again. The INCOGEN 40 MWth base plant overall arrangement is shown is Figure 2. Plant building layout is shown in Fig. 3.

The HR1-reactor has some novel safety features compared to other reactor concepts, mostly due to it's limited size. For example, after a total loss of coolant the fuel remains fully intact, even if the reactor shutdown system fails and the reactor goes critical again after a number of hours. More about this is explained in the paper of Kuijper et al. [2]. The reactor is fuelled with the German type spherical fuel elements (pebbles), following the so-called Peu-à-peu fuelling concept [6]. Through continuously fuelling, the reactor contains just a small amount of overreactivity. Only once every ten years the reactor is defuelled completely. In the primary cell, a passive air-cooled afterheat removal system is located.

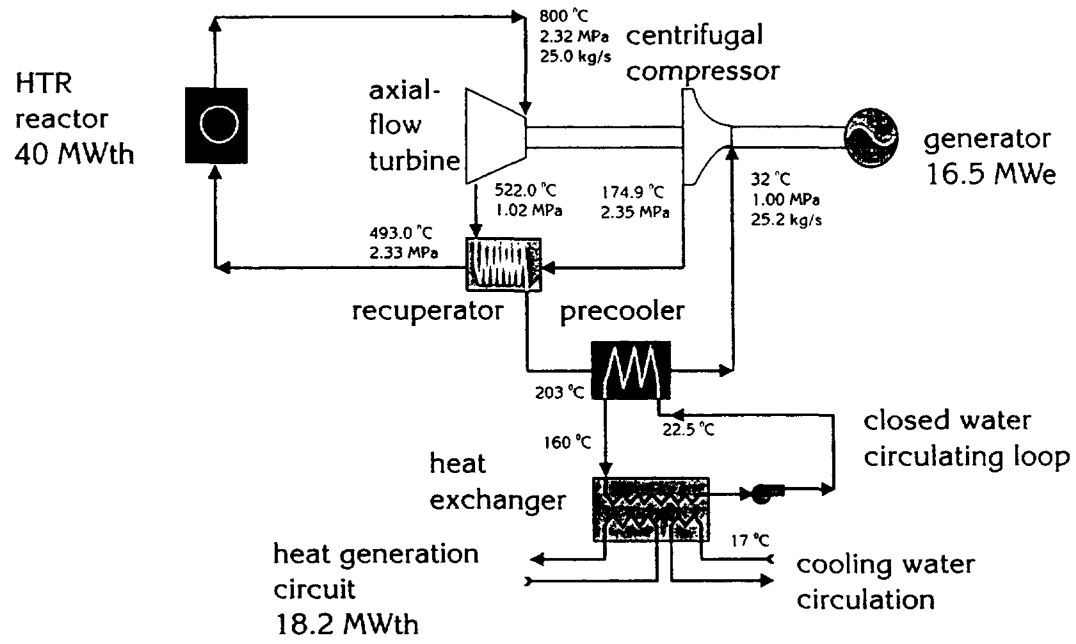

Fig.1 INCOGEN 40 MWth cycle design.

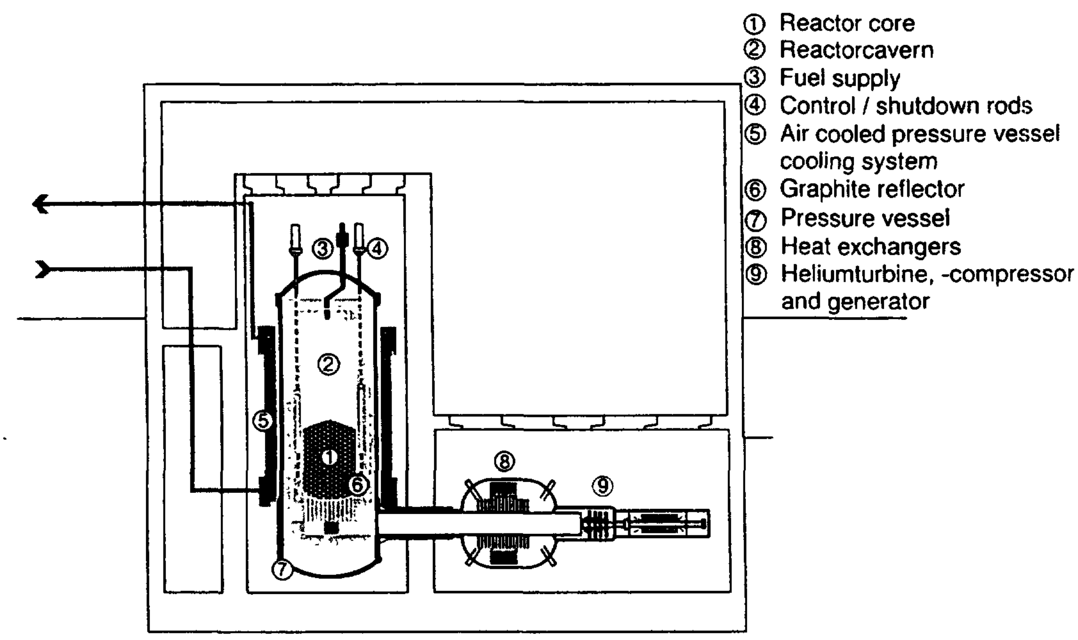

Fig. 2. INCOGEN 40 MWth base plant overall arrangement.

4. Performance

The performance of three (groups of) components is researched in detail. The behaviour of the reactor is being analysed by ECN with the code PANTHER-THERMIX [2]. The energy conversion system performance is modelled by KEMA for cycle optimisation with their code SPENCE [7]. The reactor cavity cooling system is researched by ECN with the code FLOW3D [8].

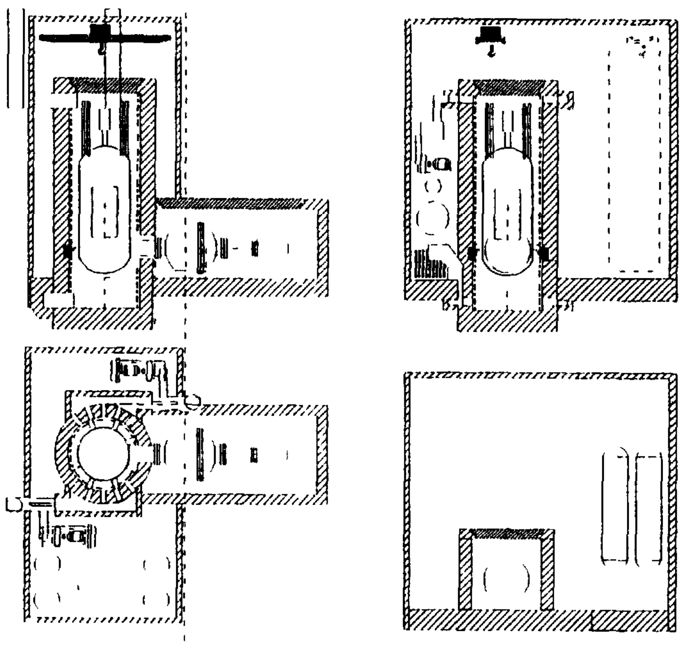

Fig. 3. INCOGEN 40 MWth building layout.

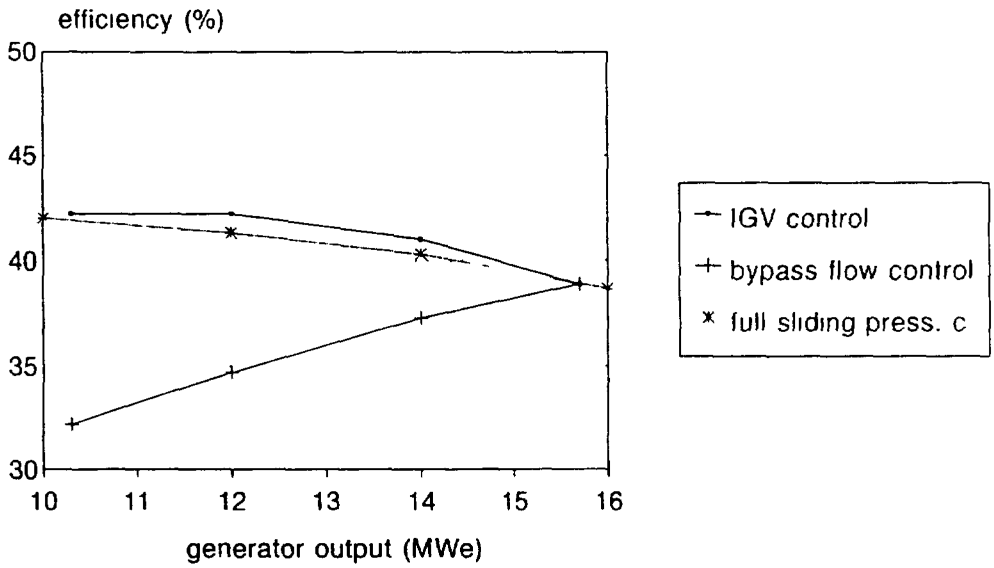

Fig. 4. Efficiency at part load.

For the energy conversion system, both static and dynamic analyses have been performed. For example, the net efficiency appears to be fairly insensitive to the gas compressor inlet temperature. For part load operation, three methods of control have been analysed: 1. full sliding pressure control, 2. reactor bypass control and 3. compressor inlet guide vane control (IGV control). Fig. 4. shows the efficiency for part load operation for the three modes of control. The most favourable type of operation is the full sliding pressure control. The net efficiency increases slightly and the thermal transients are moderate at part load. A good alternative is the option Inlet Guide Vane control, allowing fixed operational pressure at the compressor suction side. However, the life time of the cycle will be decreased due to the high thermal transients at part load. Therefore it is necessary to level off the primary inlet temperature of the recuperator. Very dramatic is the loss in net efficiency of the bypass control mode at part load. Although the buffering of helium is very low at part load, the influence on the reactor is enormous.

Dynamic plant performance has been analysed as well. The effects of varying the heat cogeneration and the load following behaviour will be treated here. Fig. 5 shows the heat cogeneration response by decreasing the inlet temperature of the compressor. It can be seen that the heat cogeneration decreases when the inlet temperature of the compressor decreases. The response of the reactor is very smooth. The load following behaviour is presented in Fig. 6. Again it can be seen that the response of the reactor is very smooth. The load demand ramps are from 100-45-100% in five minutes.

The performance of a first design of reactor cavity cooling system (RCCS) has been analysed. For this, the RCCS of the American modular high temperature reactor design, the MHTGR [9], has been transferred and adapted to the INCOGEN system.
Cold air from the environment flows through two cold ducts to the downcomer, which is situated against the cavity wall. At the bottom of the cavity, the air is distributed among rectangular tubes. The air in the tubes is heated mainly by radiation from the vessel and

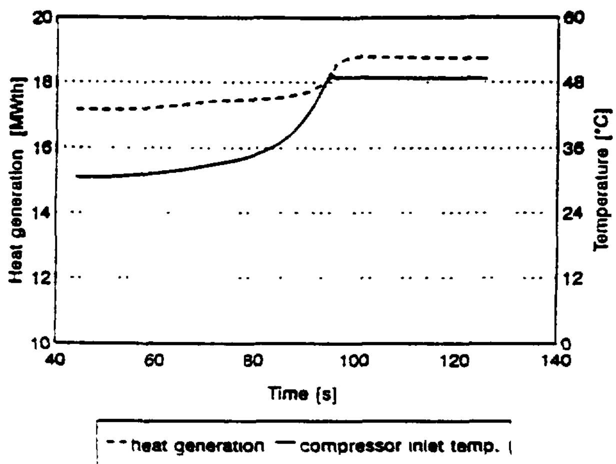

Fig. 5. Heat cogeneration response by decreasing the inlet temperature of the compressor.

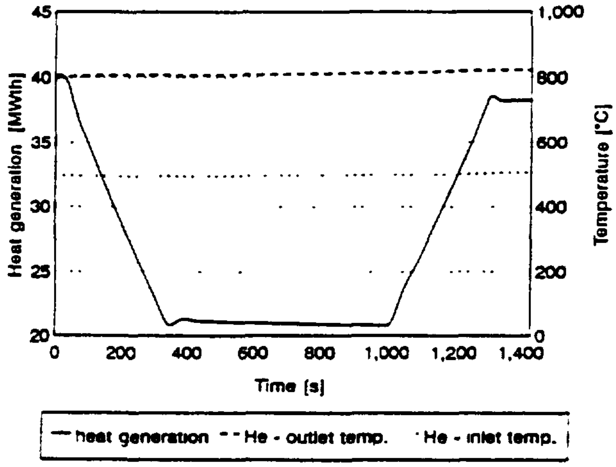

Fig. 6. Load following behaviour.

Table 3. Sensitivity analyses on the reactor cavity cooling system.

Influence of the properties of the inlet/outlet structure.

situation	temperature (K) reactor wall min – max	cavity 1 m – 11 m	outlet duct	total mass flow through ducts (kg/s)	heat transfer convection (kW)	(%)	radiation (kW)	(%)
chimney 30 m high (reference calculation)	623 – 634	376 – 419	375	13.3	208	20.8	792	79.2
inlet duct temperature = 350 K	651 – 657	439 – 473	437	11.3	151	15.1	849	84.9
without chimney	646 – 653	421 – 465	428	7.8	158	15.8	842	84.2
no friction losses in ducts (K = 0)	621 – 631	373 – 414	370	14.2	213	21.3	787	78.7
blockage of one inlet duct and one outlet duct	628 – 637	387 – 428	387	11.5	196	19.6	804	80.4

Influence of the amount of heat release from the reactor vessel wall.

heat release	temperature (K) reactor wall min – max	cavity 1 m – 11 m	outlet duct	total mass flow through ducts (kg/s)	heat transfer convection (kW)	(%)	radiation (kW)	(%)
0.5 MW	519 – 531	346 – 374	345	11.0	147	29.4	353	70.6
1 MW (ref. calc.)	623 – 634	376 – 419	375	13.3	208	20.8	792	79.2
2 MW	756 – 763	434 – 487	427	15.7	251	12.6	1749	87.4

flows through these tubes and afterwards through two hot ducts back to the environment. First a reference calculation has been performed. The reactor vessel temperatures range from 350 to 361 ^{o}C; the cavity wall temperature from 103 to 146 ^{o}C, and the gas temperature in the outlet duct is heat to 102 ^{o}C. A number of sensitivity analyses have been performed to compare different circumstances with the reference case:
- elevated inlet duct gas temperature (by 50K),
- absence of a chimney,
- no friction losses in the ducts,
- blockage of one inlet and one outlet duct,
- change of the amount of heat release from the reactor vessel.

Table 3 gives the results for these cases.

5. Costs

The results of the life cycle cost analysis are not available yet. Therefore here only a small chapter where the trend of cost reduction is indicated, if certain measures are applied. If the INCOGEN installation is treated like a normal electricity generating nuclear plant according to the US Cost Estimate Guidelines for Advanced Nuclear Power Technologies, the investment costs would amount to 7600 $/kWe ('92). Of course this is far too high. With cogeneration some improvement can be obtained. But most is to be expected of the Goetzmann-suggestion [5], to use inherent safety features to reduce capital cost. A first attempt, conventional classification of some components, gave a cost reduction of 800 $/kWe. A higher reduction, including that due to automatisation, is expected to be possible. Table 4 gives some cost figures for the INCOGEN installation.

Table 4. Investment costs of a 40 MWth INCOGEN plant.

	absolute investment costs	investment costs per kWe ('92 US$)
Electricity only, prototype	125	7600
Cogeneration, prototype	125	6000
Cogeneration with some components conventionally classified, prototype	110	5200
Cogeneration with some components conventionally classified, small series production	80	3700

6. Conclusion

A heat and power cogeneration system with an HTR as a heat source would be very suitable for the cogeneration market. The for this market required power level gives the opportunity to use inherent safety features to reduce cost, to outweigh the economy-of-scale disadvantage. Advantages and disadvantages of different control methods are indicated. Reactor response with load following appears to be very smooth. Costs are still too high but there is still room for further reduction.

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IS THERE A CHANCE FOR COMMERCIALIZING THE HTGR IN INDONESIA?

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Abstract

Indonesia is one of the developing countries in Asia-Pacific regions that actively improving or at least continuously maintain its economic growth. For this purpose, to fulfill a domestic energy demand is a vital role to achieve the goals of Indonesian development. Pertamina, the state-owned oil company, has recently called for a significant increase in domestic gas consumption in a bid to delay Indonesia becoming a net oil importer. Therefore, there is good chance for gas industry to increase their roles in generating electricity and producing automotive fuels. The latter is an interesting field of study to be correlated with the utilization of HTGR technology where the heat source could be used in the reforming process to convert natural gas into syngas as feed material in producing automotive fuels. Since the end of 1995 National Atomic Energy Agency of Indonesia (BATAN) has made an effort to increase its role in the national energy program and Batan is also able to involve in the Giant Natuna Project or the other natural gas field projects to promote syngas production applying HTGR technology. A series of meeting with Pertamina and BPPT (the Agency for the Assessment and Application of Technology) had been performed to promote utilization of HTGR technology in the Natuna Project. In this paper governmental policy for natural gas production that may closely relate to syngas production and preliminary study for production of syngas at the Natuna Project will be discussed. It is concluded that to gain the possibility of the HTGR acceptance in the project a scenario for production and distribution should be arranged in orther to achieve the break even point for automotive fuel price at about 10 US$/GJ (fuel price in 1996) in Indonesia.

INTRODUCTION

Indonesia is one of the developing countries in Asia-Pacific regions that actively improving or at least continuously maintain its economic growth. For this purpose, to fulfill a domestic energy demand is a vital role to achieve the goals of Indonesian development. Pertamina, the state-owned oil company, has recently called for a significant increase in domestic gas consumption in a bid to delay Indonesia becoming a net oil importer[1]. Therefore, there is good chance for gas industry to increase their roles in generating electricity and producing automotive fuels. However, gas development for domestic consumption is complicated due to the scattered locations of the gas reserves and scattered demand in Indonesia. Most of the gas resources are remote from the main consumption areas, and therefore projects will require large investment and long lead time.

It is realized that distribution of synthetic fuel to the consumer is easier than for natural gas. Furthermore there substantial tendencies increase in supply-demand gap for oil fuel mainly diesel fuel and gasoline in Indonesia as well as in the Asia-Pacific regions. The HTGR as a heat source could be used in the reforming process to convert natural gas into syngas as feed material in producing synthetic fuels. Therefore, it is important to assess the realizability of the HTR role in developing the natural gas projects. In this paper governmental policy for natural gas production in Indonesia that may closely relate to syngas production and preliminary study for production of syngas at the Natuna Project will be discussed.

GOVERNMENTAL POLICY FOR NATURAL GAS PRODUCTION

In order to speed up the development process, the government of Indonesia is engaged directly in economic activities, through state-owned enterprises operating in various sectors. In the petroleum sector, the government's involvement is based primarily on the constitutional provision that all natural resources belong to the state and that economic activities considered essential to the country be controlled by the state. Pertamina, the state oil and gas enterprise, has a monopoly over all aspecs of oil and natural gas production, and controls all exploration, production, refining, transportation and marketing of oil and gas in Indonesia. However in the upstream activities the Minister may appoint other parties as contractors for the state enterprises, if required, for the execution of upstream operation which can not yet be executed by the state enterprises.

The involvement of Pertamina in the key activities of the oil and gas industry, include:

- Upstream exploration and development of oil and gas fields, through direct participation or through establishment of production sharing contract with national or foreign contractors,
- The operation of the major gas transmission and supply systems,
- Equity participation in downstream oil gas development projects.

The different areas in which the private sector, domestic as well as foreign, has played a significant role, include among others:

- Financing of major oil and gas projects; Pertamina has utilized foreign commercial lending institutions, trading house and other sources to acquire capital for the projects expansion.
- Engineering and construction of major oil and gas projects; the development and construction of the majority of the capital intensive projects has been contracted to private firms.
- International marketing and distribution of crude oil, oil products and natural gas; the marketing, distribution and export oil and gas is carried out by Pertamina through its affiliate companies.

In the downstream petroleum activities, throgh the Presidential Decree no. 42/1989 and its related Minister of Mines and Energy Decree no. 03/1993, the private sectors, foreign as well as domestic, have the opportunity to invest in the refining and processing of oil and gas undertakings. In the future, the private participation will become increasingly important in the development of oil and gas industry in Indonesia. New regulations have been introduced, in particular, the April 1992 revisions of the Foreign Investment Law, which allow foreign firms to have 100% equity in domestic projects, which either have a minimum paid up capital of US$ 50 million or which are implemented in less developed and remote areas outside Java or within areas jointly developed with the neighbouring countries. More recently, the Government Regulation no. 20 issued in May 1994 was intended to allow foreign companies to invest in almost all kind of business activities in Indonesia, with few exception in the very essensial infrastructure.

Pertamina, recently called for an significant increase in domestic gas consumption in a bid to delay Indonesia becoming a net oil importer. With the scenario of optimistic production related to low and high consumption, Indonesia will become a net oil importer in 2010 and 2003 respectively (the present daily production is about 1.6 million barrels per day). Therefore, there is good chance for gas industry to increase their roles in generating electricity and producing sintetic fuels. Noted that currently oil and natural gas reserves (proven and potential) are estimated at 9.1 billion bbls of oil and 123.6 TSCF of gas[2)]. However, gas development for domestic consumption is complicated due to the scattered locations of the gas reserves and scattered demand in Indonesia. Most of the gas resources are remote from the main consumption areas, and therefore projects will require large investment and long lead time.

Distribution of synthetic fuels is easier than for natural gas using pipeline or LNG tankers. Hence, this approach may reduce the problems of natural gas distribution. The high temperature gas coled reactor technology (HTGR) can be applied as a heat source in steam or carbon dioxide reforming for synthetic gas production from natural gas as a feed material in making synthetic fuels. However, due to public acceptance difficulties faced by the nuclear development, there should be more benefit can be achieved by the HTGR that its introduction in Indonesia could be performed successfully.

PRELIMINARY STUDY OF SYNTHETIC FUEL PRODUCTION

Conversion of natural gas to products is always indirect, via synthesis gas. Direct conversion, e.g. by oxydation of methane to methanol or by oxidative coupling to ethylene, is theoretically possible, but practical processes have not been developed and may never be.

	n P mol/s MW_t	S 10^3\$/(mol/s) \$/kW_t	I Mio \$	Y Mio\$/y	X \$/GJ	X/X_Σ %
1. REF	12375	110	1361	163	0.54	9.6
2. HTR,t	3090	500	1373	165	0.55	9.8
3. MES	16299	90	1467	176	0.59	10.5
4. Invest.	annuity = 12%/y		4201	504.1		
T= 8000 h/y	n P GWt	P T 10^6 GJ/y	X_F \$/GJ	Y_F Mio \$/y		
5. CH4 t:	9.9	286	3.8	1086.8	3.62	64.6
6. HTR, t	3.09	89	1	89	0.30	5.3
7. Prod. t	10.4	300		1680	5.6	
8. Cost C	* $C = X_7 \bullet 1.2$ (for loses) =				6.72*	

Fig. 1: A simplified cost calculation for methanol (liquid) as synthetic fuel at the Natuna Project. The annuity is assume to be 12%. The specific cost fraction of HTR and nuclear fuel is 9.8% and 5.3%, respectively.

Technologies for large-scale production of synthetic diesel oil, gasoline and substitutes for these products from synthesis gas are available today. High quality diesel and gasoline can be produced from synthesis gas by Fisher-Tropsch reactions. The technology has been used notably by SASOL LIMITED in South Africa, and a major part of the transport fuels in that country is actually made in Fischer-Tropsch plants on the basis of synthesis gas made from natural gas or coal[3)].

Since 1993/94 there is a gap about 20% or about 8 billion liters per year between supply and demand of oil fuel mainly for diesel fuel and gasoline. It is estimated that in 2008/209 the figure will reach at about 40% or 33 billion liters per year. The current price of gasoline in Indonesia is 9.6 US$/GJ (Rp. 700/liter). In this paper to simplify a case of study in applying the HTGR it is assumed that production capacities of 8 billion liters per year of oil fuel or equivalent to 3.0×10^8 GJ/year of synthetic fuel with energy price 9.6 US$/GJ using natural gas from Natuna gas field as a raw gas should be achieved. It is also assumed that the composition of Natuna raw gas being fed to the plant is 75% of methane and 25% carbon dioxide which have energy price of 3.8 US$/GJ. Assuming fed gas composition as above all of the CO_2 in the fed gas could be converted into the marketable product synthetic fuel. Actually the CO_2 content in the Natuna gas is about 70%, but their content could be adjusted to about 25% and the excess carbon dioxide is reinjected or be treated by some other way.

A simplified method for calculation of cost estimation as proposed by Barnet[4)] has been used in the present study. In the calculation it is assumed that the specific investment for HTR, reformer, and methanol synthesis plant are 500 US$/kWt, 110×10^3 US$/(mol/s), and 90×10^3 US$/(mol/s), respectively. Fuel price for the HTR is assumed 1US$/GJ. Based on his calculational prosedure (an assumption is that reaction turnovers are always 100%), to cover the demand 3.0×10^8 GJ/year the plant need a feed gas about 2.86×10^8 GJ/year or 286 billion CF/year methane and 3090 MWt heat source from HTR which its operating time 8000 hours/year.

A simplified cost calculation for the synthetic fuel production is shown in Fig.1. In the final result of cost estimations an additional cost charges of 20%, C = 1.2 X, is used to compensate an assumption that reaction turnovers are always 100%. Based on the result it is clear there exists a potential of economical competitiveness for the exploitation of the gas of the Natuna Gas Field, in the production of synthetic fuel from the Natuna gas, making use of nuclear energy in the form of high temperature heat from the High Temperature Reactor, HTR. The estimated cost inferred from the calculation is about 6.72 US$/GJ while the oil fuel price (gasoline) in Indonesia is 9.6 US$/GJ. We believe the production cost from the proposed scenario could be reduced that lower than 6.72 US$/GJ by making much effort to optimize the capital investment.The other reasons for existing economical competitiveness are that distribution of methane to domestic market is tremendous problem and synthetic fuel is a readily marketable product in the market of transportation as a subsitute dor gasolin and diesel with their relatively high market values.

SUMMARY

The economical competitiveness for the exploitation of the gas of the Natuna Gas Field, Indonesia in the production of synthetic fuel applying nuclear energy in the form of high temperature heat from the High Temperature Reactor, HTR has been discussed through the present paper. It is concluded that by making much effort there a chance for commercializing the HTR in Indonesia. Further study should be made to obtain the best estimated cost as soon as possible.

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THE ESSENTIAL TRENDS OF HGR DEVELOPMENT IN THE RUSSIAN FEDERATION

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Abstract

The up-to-day Russian consept of HTGR technology development is presented in paper. At present it reduces to the following:

- the modular HTGR measured up to modern safety requirements with pebble bed core and spherical fuel elements of up to 200 MW thermal power have being developed for production of process heat;

- for electricity production with maximal efficiency through gas turbine cycle the plant design with modular HTGR containing prismatic block annular core has being carried out.

Conceptual design of GT-MHR reactor plant of 600 MW thermal, developing with GENERAL ATOMICS cooperation is the base of Russia activity in HTGR technology in current time.

The main characteristic properties of GT-MHR are presented in this paper. It is assumes that reactor construction should be oriented as well to a core with weapon-grade plutonium as to a core with lowenriched (up to 20%) uranium to use in reactor plant for commercial electricity production. The design and construction cost evaluations for plant consisting of 4 modulars are shown, the schedule of demonstration reactor realization is presented as well. In paper the following key R&D problems, wich may be resolved by the best way on the base of international cooperation are presented:

- works associated with refine of fuel fabrication technology and its irradiation examination up to deep burnup (up to 70% fima);

- running tests of the fabrication technology for graphite blocks of necessary dimentions;

- studies of vessel material mechanical properties under the temperature up to 500 °C and large-scale vessels production;

- tests of the core and reactor components in critical test facilities including studies of control rods effectiveness and reactivity temperature effects, investigation of uniformity of helium mixing at the core outlet and fuel colums stability;

- tests of turbine system separate parts including electromagnetic bearings as well as combined tests of full-scall turbine system in the test facility;

-tests of a "hot" gas line and traveling seals to reduce helium bypass leakage inside the power conversion loop , et. so.

1. STRATEGY OF HTGR UTILIZATION

During more than 30-years of HTGR development in Russia the considerable experience has been accumulated. The cooperation betweeen various organization and institutes, sufficient number of facilities for testing of HTGR equipment, irradiation of fuel, graphite and structure materials were created.

However, in the last years, the activities associated with the development of HTGR projects in Russia was sharply reduced due to economic reasons.

Nevertheless, there is weigty background for the statement that HTGR should play significant role in nuclear energetics in Russia. The objective need for HTGR is clearly defined by their unique potential: firstly - of high temperature level of coolant (up to 1000 °C), secondly - of safety properties allowed to allocate the plants with HTGR directly with consumer of high temperature heat and electricity.

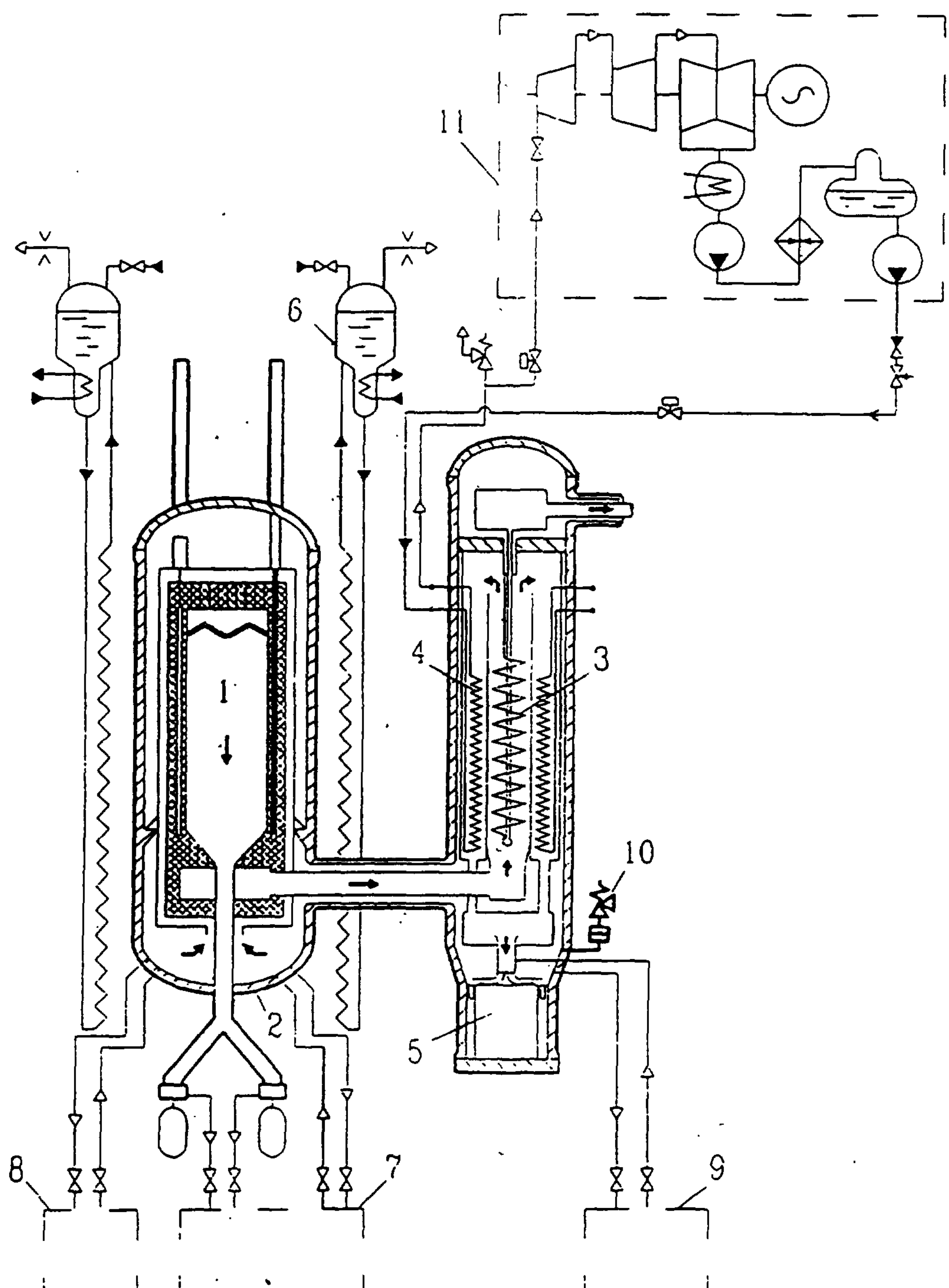
The task is efficiently to use this advantages of HTGR. There are following ways for that:

(1) The production of high temperature heat for verious industrial processes. Basic consumption of heat in industry falls at temperature up to 550-600 °C. This level can be provided by HTGRs with outlet helium temperature of 750 °C. Almost all oil refineries and plants for production of petrol and diesel fuel from coal can be served by HTGR with such a level of temperature. This allows to save only for the oil refineries about 15 % of the oil processed.

It should lay emphasis of attention on additional unique features of HTGR, namely capability of module concept of HTGRs to provide passive removal of residual heat through the reactor vessel surface by natural processes (convection, conductivity, radiation). Unlike of other types of reactors the residual heat is removed not only under loss of flow but if the coolant escaped from primary circuit. The maximum fuel temperature doesn't exceed design basis limit of 1600 °C in any accidents if nominal power of reactor with pebble bed core is about 200 MW and about 600 MW with prismatic blocks core reactor.

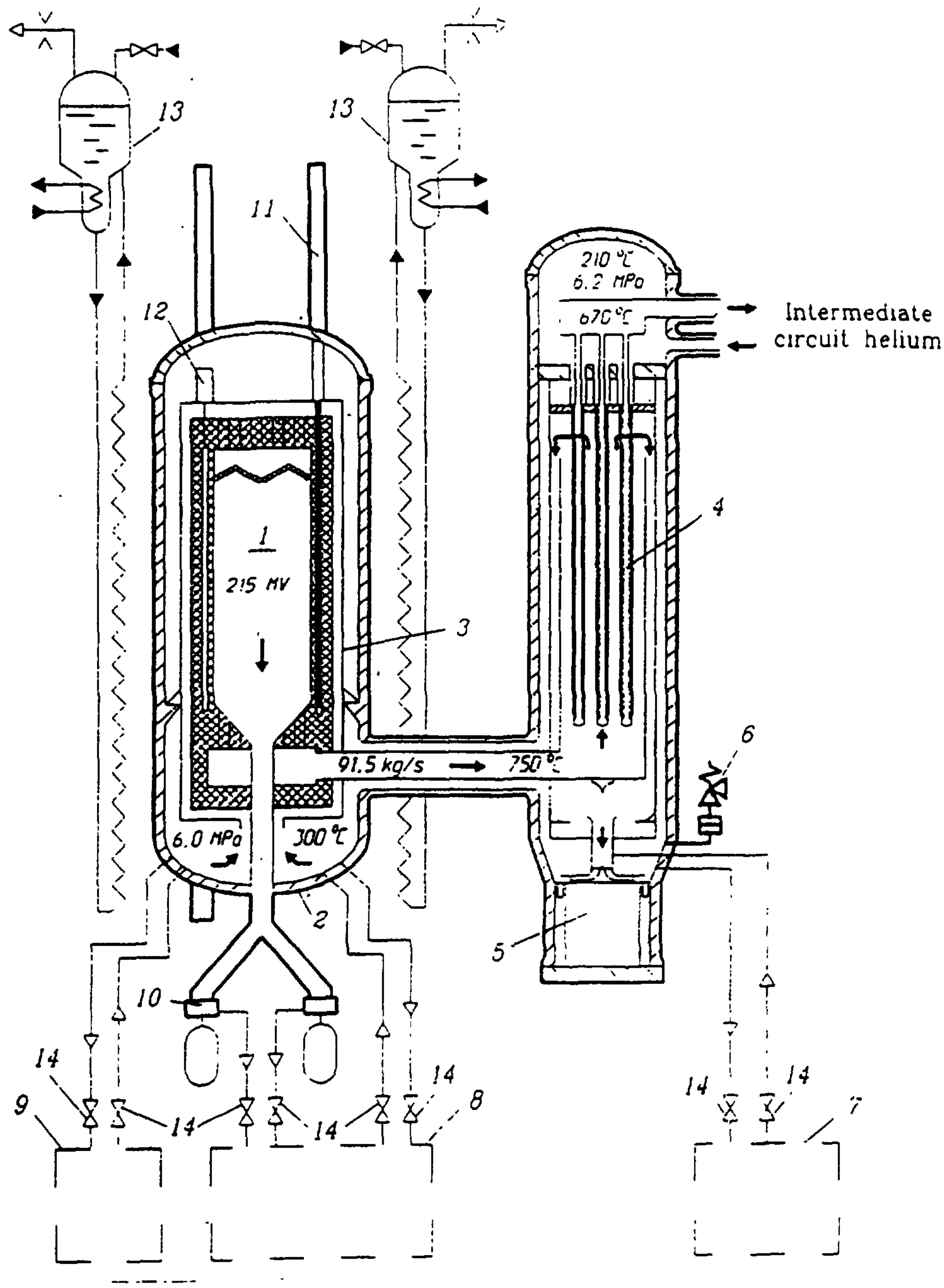
TABLE 1. THE MAIN CHARACTERISTICS OF REACTORS

Parameter	Magnitude	
	VGM	VGMP
Thermal power, MW	200	215
Helium temperature, °C		
reactor inlet	300	300
reactor outlet	750-900	750
Helium flow rate, kg/s	59...85	91,5
Helium pressure, MPa	5	6
Number of loops	1 main and 1 auxiliary	
Core diameter/height, m	3,0/9,4	
Average power density, MW/m^3	3,0	3,2
Fuel enrichment for U-235, %	10	
Average burnup, GWday/t	76,0	90,0



1 - reactor
2 - pressure vessels unit
3 - intermediate heat exchanger
4 - steam generator
5 - gas circulator
6 - surfase cooling system
7 - fuel circulation system
8 - small absorber balls system
9 - helium purification system
10- relief valve
11- steam - turbine plant

FIG. 1 VGM reactor plant.



1 - reactor core	8 - fuel circulation system
2 - vessels block	9 - small absorber balls system
3 - core protective housing	10- fuel element discharge facility
4 - intermediate heat exchanger	11- CPS rod drive
5 - main gas blower with a cut-off valve	12- loading volume of SABS
6 - relief valve	13- emergency cooldown system
7 - helium purification system	14- localized valve

FIG. 2. VGM-P schematic diagram

The possibility of core meltdown and its relocation in competely excluded taking into account very high temperature of graphite sublimation.

VGM project [1] with pebble bed core reactor of 200 MW aimed at combined production of process of heat and electricity. VGMP project [2] with reactor of 215 MW was completely orientated towards production of heat and designed for a standard of oil-refinery plant with accounting of user's requirements.

Table 1 shows the main characteristics of these reactors. Fig.1,2 show the schematic diagrams of these reactor plants.

(2) The second way is use of HTGR's only for electricity production. It is to be noted however, that connection of HTGR with conventional steam-water cycle doesn't correspond to their capabilities and can't provide competitiveness of HTGRs with the other energy sources. Realization of HTGR advantages should look for a combination of HTGRs with new technologies.

Usage of gas-turbine cycle in HTGR, which development was began recently allows to reach a net thermal conversion efficiency up to 48-50 %. This in combination with very deep burnup of fuel and uniqûe level of safety opens the new prospects for HTGR as electricity producer.

Thus, at present the strategy of HTGR utilization has formed, namely:

- for production of only process heat, using modular type of HTGR with thermal power of about 200 MW and pebble bed core;

- for electricity production through gas turbine cycle using modular type of HTGR with thermal power of about 600 MW and a core formed from prismatic blocks.

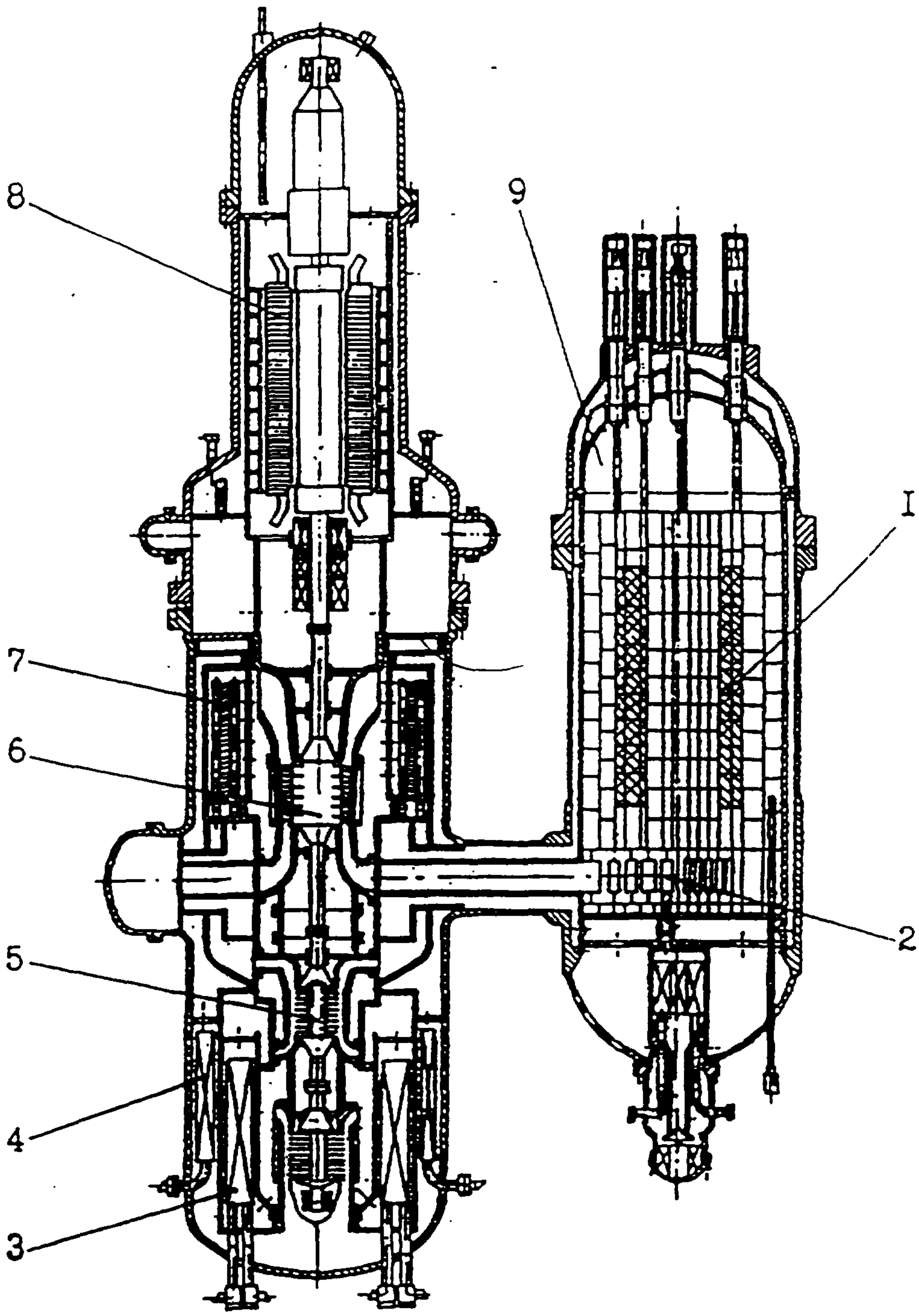
This strategy is assumed to more completely demonstrate the advantages of this type of reactors.

2. THE DESIGN OF HIGH TEMPERATURE HELIUM REACTOR WITH A GAS TURBINE (GT-MHR)

HTGR has a unique ability on reaching overhigh fuel burnup (up to 800 GWday/t in comparison with 150 GWday/t in sodium reactors and 60 -70 GWday/t in VVER) and has a posibility of 90 % burnup of initial fission material. This capacity allows to use of HTGRs for involving of weapon grade plutonium (WGPu) to energy production with subsequent disposition of spent fuel without additional reprocessing.

To realization of this unique property MINATOM (Russian Federation Ministry for Atomic Energy) and leading American firm General Atomics (GA) in February 1995 have written the agreement on joint development of HTGR with direct gas turbine cycle and began to finance the conceptual desing of GT-MHR. FRAMATOM has recently joined this cooperation program.

Russian GT-MHR desing is based on GA reactor concept [3]. GT-MHR plant is a coupling of passively safe modular reactor with helium coolant with up-to-date technology



1 - reactor core
2 - core outlet plenum
3 - intercooler
4 - precooler

5 - compressor
6 - turbine
7 - recuperator
8 - generator
9 - core inlet plenum

FIG. 3. GT-MHR reactor arrangement.

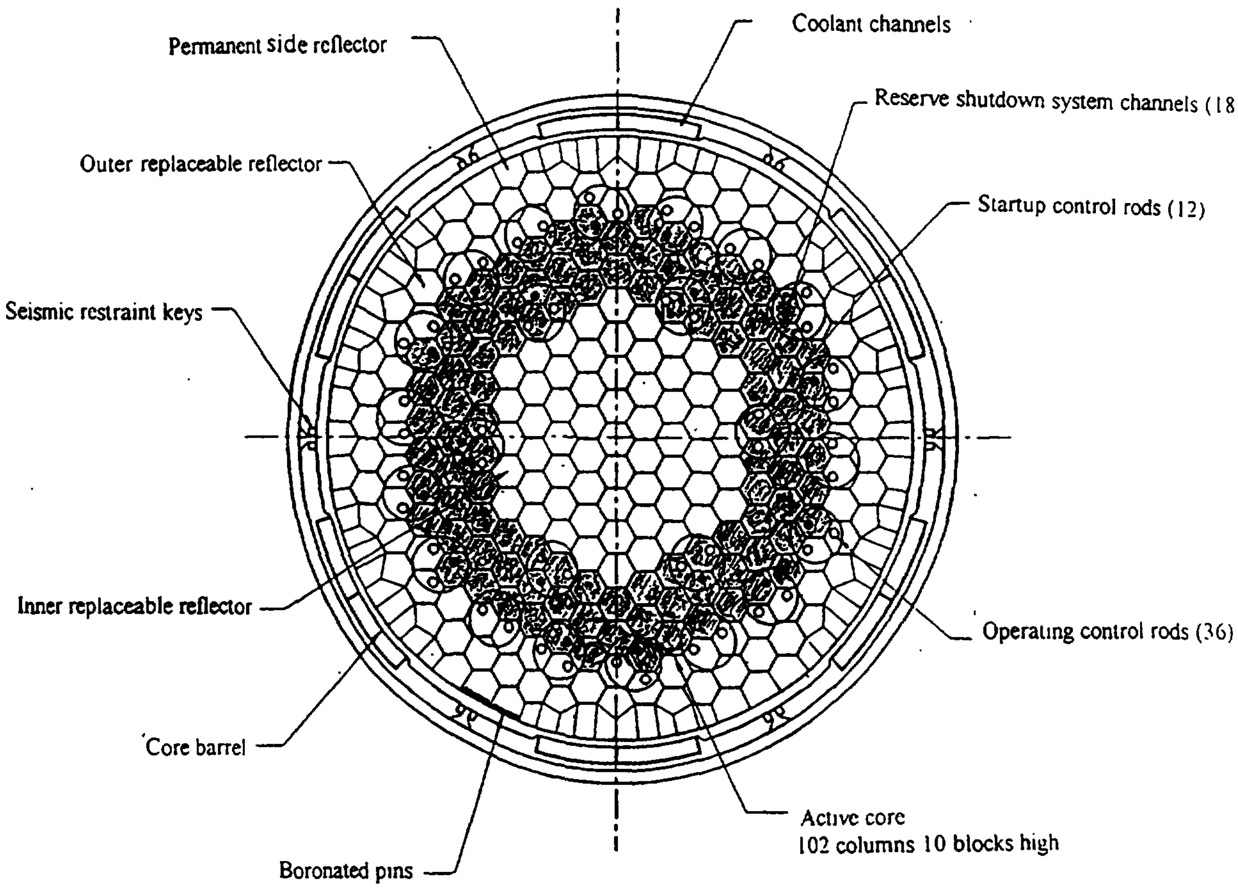


FIG. 4. Reactor system cross section through the core.

developments: compact highly-effective heat exchangers, high-strength high temperature steel alloy vessels.

The works on fuel technology are the most important part of conceptual design. They include the production of test samples of plutonium fuel, and development and justification of requrements to industrial facility on the fuel from weapon grade plutonium production. In the future the GT-MHR project will be destined for commercial electricity production and for sale to third countries. In reactor for this needs the low-enriched up to 20 % UO_2 will use as a fuel in coated particles.

Fig.3 presents the general view of reactor plant (RP). The main components of RP are placed in three steel vessels: reactor vessel, vessel of power conversion system and a cross vessel. All three vessels are located in underground silo. The characterictics of this containment are typical of the contaiments used in current LWR designs. The reactor vessel houses the annular core, core support, control rod drives, heat exchanger and gas circulator of the auxiliary loop. There are the penetrations for fuel reloading as well. The reactor vessel is surrounded by a reactor cavity cooling system, which provides totally passive decay heat removal in any possible accidents included beyond design accidents.

The reactor core (Fig.4) contains hexagonal graphite fuel blocks (with width of 360 mm and height of 800 mm), arranged on 10 in each of 102 columns. In fuel blocks 3191 thousant fuel compacts are placed in the core. The fuel compact of 12,6 mm in diameter and of 50 mm height consists of graphite matrix with uniformly distributed maltilaied coated particles on the basis of pyrocarbon and silicon carbied (PyC-SiC-PyC). For reactor shutdown and control of reactivity there are two systems: control rods system and system containing small absorber balls.

A specific feature of the core is the using of the Er_2O_3 - based poisoning absorber, which is pressed together with the graphite forming cylinder rods similar to fuel compacts. Erbium in the reactor with weapon grade plutonium serves a double purpose: it not only provides compensation of the initial reactivity excess, but it also ensures the negative temperature coefficient of reactivity. It contains ~ 23 % Er-167, the only one of the known poisoning absorbers having a strongly pronounced resonance (> 1000 barn) near 0,5 ev. The presence of this element is essential for obtaining the negative temperature reactivity coefficient when the temperature rises.

There is a graphite reflector inside and outside of the core. The layers of reflector adjacent to the core can be replaced during reactor operation.

The vessel of power convertion system houses turbogenerator (turbine and compressor with one stage of generator cooling) and heatexchanging equipments (recuperator, intercooler and precooler). The generator is placed in separate box and strongly connected with turbocompressor through the coupling. The generator and compressor are rotaited directly by turbine.

The magnetic bearings (three radial and one rested) are used in turbogenerator. Safed mechanical bearings are provided as well.

The hot gas duct to supply of hot helium from reactor to turbine is arranged in a cross vessel. The schematic flow diagram of GT-MHR is shown in Fig.5.

The direct Brayton cycle in GT-MHR is being carried out in following way:

-the helium coolant is being twice pressed in compressor and in intercooler;

-prilimary of helium heating up (before reactor input) is taking place in recuperator which uses of helium energy after turbine;

- in reactor helium is being heated up to 850 ^{o}C;

- the hot helium of high pressure then is being expanded in turbine which moves the compressor and generator;

- the helium of low pressure goes to recuperator where the helium of high pressure (after compressor) is heated up to 490 ^{o}C and goes to the core input;

- the excess of heat is removed in precooler with return water circuit.

The main characteristics of GT-MHR for Pu consumption are shown in Table 2. It should be noticed that the GT-MHR has sufficient less impact on the environment (less thermal release and less need in water).

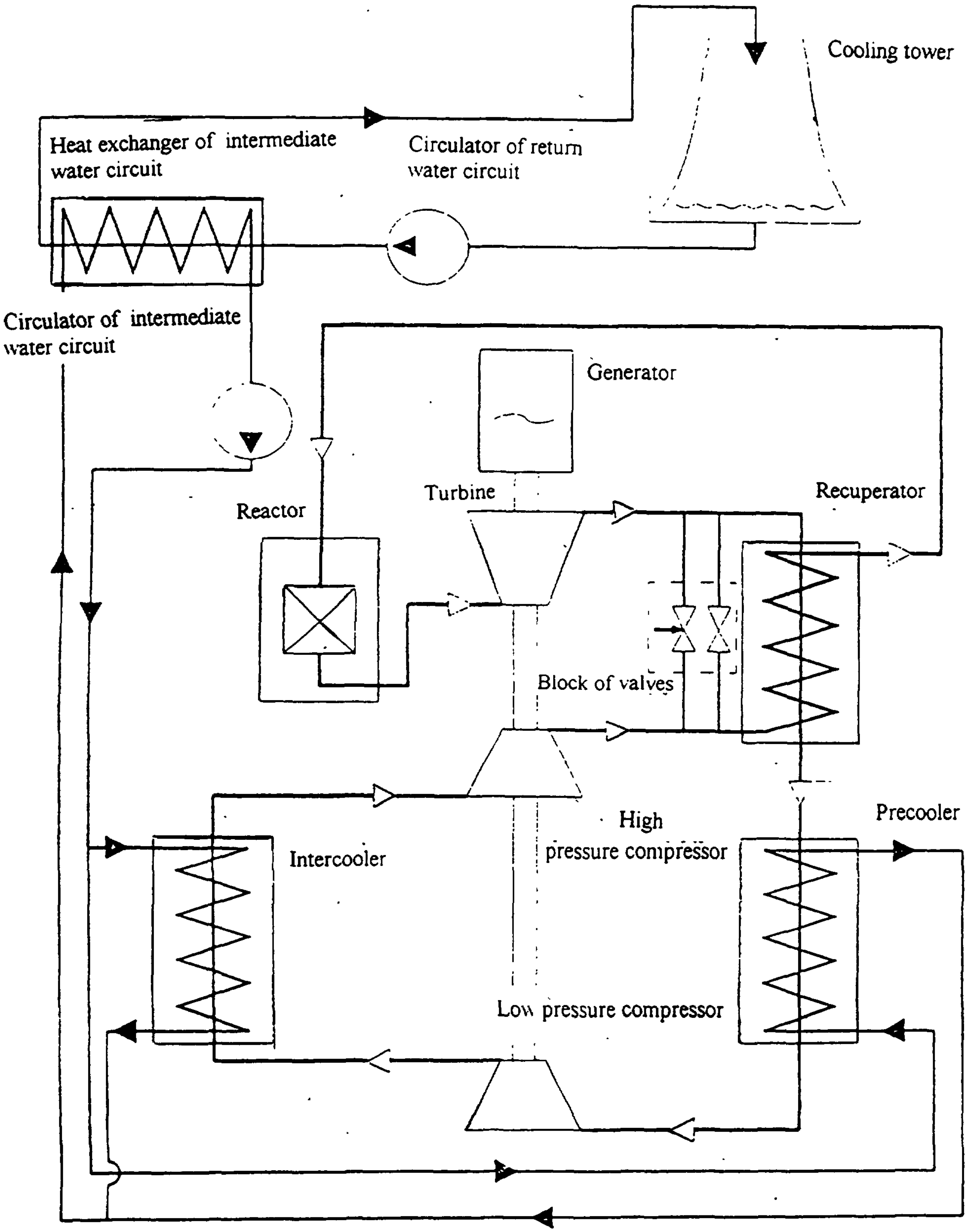


FIG 5 Schematic flow diagram of GT-MHR gas turbine cycle

TABLE 2. THE MAIN CHARACTERISTICS OF GT-MHR FOR Pu CONSUMPTION

Parameter	Value
Reactor power (thermal), MW	550-600
Turbine inlet temperature, °C	850
Reactor inlet temperature, °C	490
Compressor inlet temperature, °C	26
Turbine inlet pressure, MPa	7,02
Plant efficiency, % net	47-48
Core diameter inside/outside, m	2,96/4,84
Core height, m	8
Outside radial reflector diameter, m	7
Fresh WGPu loading, kg	750
Content of Pu-239 in WGPu, %	94
Distraction level of Pu-239, %	90
Amount of loaded WGPu, kg/year	250
Average burnup, MWdays/kg	650
Amount of WGPu involved in fuel cycle during 60 years one reactor service life,t	15
Reactor vessel overall dimensions:	
diameter inside/outside, m	7,3/7,8
height, m	26
Amount of radioactive wastes per 1 MW(e)	lower by 33 % than in VVER
Amount of heat releases into the environment per 1 MW(e)	lower by 50 % than in VVER
Electricity cost (estumated for commercial NNP)	lower by 20 % than for advanced VVER

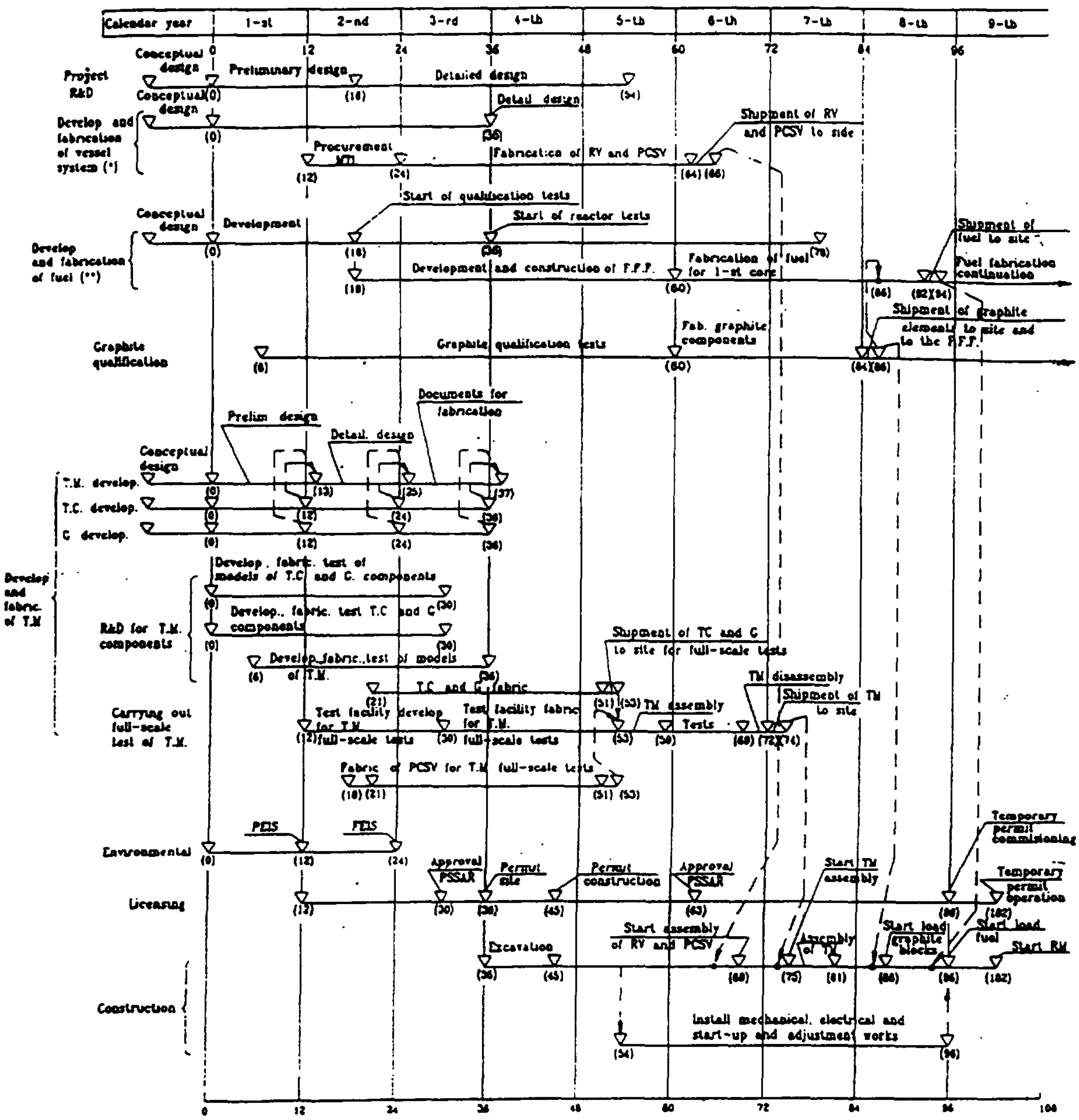
3. THE KEY RαD PROBLEMS FOR DESIGN REALIZATION

The GT-MHR project general schedule is shown in Fig.6. To support developments and license of GT-MHR the following major development tasks will be completed:

(1) Works associated with fuel assembly design and fabrication technology for fuel particles, fuel compacts and fuel blocks. Fuel irradiation in reseach reactors. The fulfilment of RαD on fuel on the base of WGPu and industrial facility creation will extend over 6-7 years.

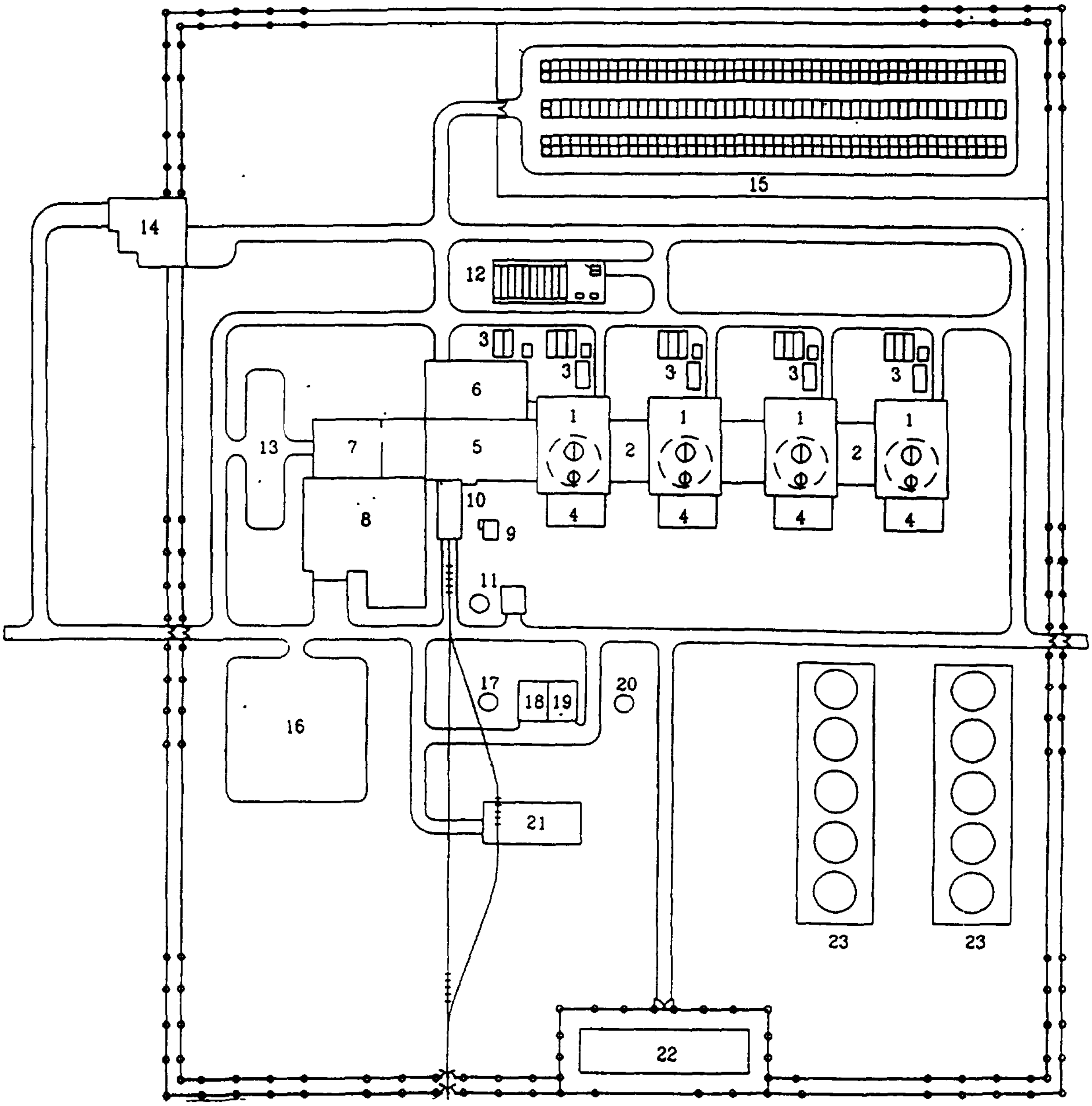
(2) Running tests of the fabrication technology for graphite blocks of necessary dimentions for fuel blocks, reflector blocks, tests of graphite samples including irradiation in research reactors will need the same duration.

(3) Test of the core and reactor components in critical test facilities including studies of control rods effectiveness reactivity temperature effects will need about 3 years. As well as investigation of uniformity of helium mixing at the core outlet and fuel columns stability.



TC - turbocompressor
TM - turbomachine
G - generator
FST - full-scale tests
PSSAR - preliminary standard safety analysis report
FSSAR - final standard safety analysis report
RV - reactor vessel
PCSV - power conversion system vessel
PEIS - preliminary environmental impact statement
FEIS - final environmental impact statement
F.F.F. - fuel fabrication facility

FIG. 6. GT-MHR project general schedule



1 - reactor building
2 - interim spent storage location
3 - power module cooling auxilary systems equipment
4 - power supply equipment building
5 - reactor service building
6 - radioactive waste management building
7 - personnel service building
8 - operations center
9 - remote shutdown building
10- cask washdown bay
11- fire pumphouse
12- helium storage structure
13- parking
14- NI warehouse
15- on-site spent fuel storage area
16- parking
17- demineralized water tank
18- auxilary boiler area
19- standby power system
20- fuel oil storage tank
21- turbomachinery maintenance facility
22- switchyard
23- cooling tower

FIG. 7. Plant arrangement

(4) Test of turbine system separate parts including magnetic bearings as well as combined tests of a turbine system in the test facility including aerodynamic and thermal-mechanical tests of heat-exchanger equipment models and test of a "hot" gas line and traveling seals to reduce helium bypass leakages inside the power conversion loop will extend the duration of 4 years.

(5) Studies of vessel material mechanical properties under the temperature up to 500 °C, licensing of technology and large-scale vessels production will need the duration of 5-6 years.

Taking into account of up-to-day state of named above technologies in Russia, design development and necessary RαD fulfilment are planed during 4-5 years (without extended irradiation test of fuel and fullscale tests of turbomachine). The named problems may be resolved by the best way on the base of international cooperation.

The cost estimations completed in this year are shown that the cost of design development and necessary RαD fulfilment consists of ~ US $275 M.

The general plant arrangement containing four reactor blocks is shown in Fig.7. For design and construction of the first reactor block the duration of 8,5 years was taken, each following block will startup in every half year after the first unit. The expenditures for 4 block reactor plant will consist of ~ US $960 M. Specific cost of electricity will be ~ ¢2,3 /kW.h.

4. CONCLUSION

The strategy of HTGRs utilization in Russia up-to-day is following: the modular HTGR with pebble bed core with thermal power of 215 MW - for production of process heat; for electricity production with maximal efficiency through gas turbine cycle will use modular HTGR with prismatic block core. For this purpose a feasibility study of VGMP reactor has been performed and conceptual design of GT-MHR is under way.

The conceptual design of GT-MHR with gas-turbine cycle is the main trend of Russia activity in HTGR field. The leading organization of Russia : RRC Kurchatov Institute, SIA Lutch, ARSRIIM, OKBM, etc. are involved in this work.

To complete all RαD works in planned duration the all scientific and technical reasons are having place. The problems associated with GT-MHR design could be solved mostly effective on the basis of international cooperation.

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OVERVIEW OF HTGR UTILIZATION SYSTEM DEVELOPMENTS AT JAERI

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Abstract

Consumption of a huge amount of fossil fuels resulted from human activities since the industrial revolution has caused an enhanced global warming. In order to relax the global warming issue, that is, reduction of CO_2 emission, new energy resource/carrier and technology are required to be developed at present. Nuclear energy can satisfy a large amount of energy demands without significant CO_2 emission and its application to hydrogen production is also considered as one of the leading nuclear heat utilization because hydrogen has superior characteristics as energy carrier and its demand is expected to increase in future.

JAERI has been constructing a 30-MWt HTGR, named HTTR, to develop technology and to demonstrate effectiveness of high-temperature nuclear heat utilization. A hydrogen production system by natural gas steam reforming is to be the first heat utilization system of the HTTR since its technology matured in fossil-fired plant enables to couple with HTTR in the early 2000's and technical solutions demonstrated by the coupling will contribute to all other hydrogen production systems. The HTTR steam reforming system is designed to utilize the nuclear heat effectively and to achieve hydrogen productivity competitive to that of a fossil-fired plant with operability, controllability and safety acceptable enough to commercialization, and an arrangement of key components was already decided. Prior to coupling of the steam reforming system with the HTTR, an out-of-pile test is planned to confirm safety, controllability and performance of the steam reforming system under simulated operational conditions. The out-of-pile system is an approximately 1/20-1/30 scale system of the HTTR steam reforming system and simulates key components downstream from an IHX.

Hydrogen production using HTGR offers a quite ambitious concept for future energy systems especially when hydrogen is produced from water. JAERI has been conducting basic studies on two hydrogen production processes by water splitting, thermochemical IS and high-temperature electrolysis of steam process, as one of the future heat utilization systems of the HTTR following the steam reforming system.

Also, a design study on of three gas turbine systems, such as a direct, indirect, indirect combined cycle systems, has been performed from viewpoint of thermal efficiency and technical problems.

1. Introduction

Consumption of a huge amount of fossil fuels resulted from human activities since the industrial revolution has caused an enhanced global warming. In order to relax the global warming issue, that is, reduction of CO_2 emission, new energy resource/carrier and technology are required to be developed at present. Nuclear energy can satisfy a large amount of energy demands without significant CO_2 emission and its power generation technology by steam-turbine is said to be proven. The ratio of electricity to the secondary energy demands, however, is about 19% at present in Japan and is estimated to be only 23% in the year 2010. Therefore, positive application of nuclear energy to nonelectric field is required to relax the global warming issue. A hydrogen production system is considered as one of the leading nuclear heat utilization systems in nonelectric field because hydrogen has superior characteristics as energy carrier and its demand is expected to increase in future.

A High-Temperature Gas-cooled Reactor (HTGR) provides a high-temperature nuclear heat with high thermal efficiency being applicable to both electric and nonelectric fields. Japan Atomic Energy Research Institute (JAERI) has been constructing a 30-MWt HTGR with reactor outlet coolant temperature of 950℃, named HTTR (High-Temperature engineering Test Reactor), to develop technology and to demonstrate effectiveness of high-temperature nuclear heat utilization. The first criticality is scheduled in December 1997, and reactor performance and safety demonstration tests will be carried out for several years[1]. After that, a high-temperature nuclear heat utilization system will be coupled with the HTTR.

A hydrogen production system by natural gas steam reforming is to be the first heat utilization system of the HTTR since its technology matured in fossil-fired plant enables to couple with HTTR in the early 2000's and technical solutions demonstrated by the coupling will contribute to all other hydrogen production system. At a preliminary design conducted from 1990 through 1995, a framework and key design of the HTTR steam reforming plant have been developed. Prior to coupling of the steam reforming system with the HTTR, an out-of-pile test is planned to confirm safety, controllability and performance of the steam reforming system under simulated operational conditions. The out-of-pile system is an approximately 1/20-1/30 scale system of the HTTR steam reforming system and simulate key components downstream from an intermediate heat exchanger (IHX).

On the other hand, hydrogen production from water is considered as an ideal method for hydrogen production using the HTGR because any CO_2 emission is not expected from the system. JAERI has been conducting basic studies on two hydrogen production processes by water splitting as one of the future heat utilization systems of the HTTR following the steam reforming system. One is the iodine-sulfur (IS) process which utilizes plural chemical reactions and works like a chemical engine to produce by absorbing high temperature heat from the HTGR. Stable production of hydrogen and oxygen from water were successfully demonstrated for 8 hours in closed-cycle operation of a laboratory-scale test apparatus A study on materials suitable for corrosive process environments is under way for large-scale realization. The other is the high-temperature electrolysis of steam process which utilize the inverse reaction of a solid oxide fuel cell Electrolysis cells in the course of fabrication of electrolysis tube and planar cells have been developed, and their performance tests have been carried out.

Also, a design study on of three gas turbine systems, such as a direct, indirect, indirect combined cycle systems, has been performed from viewpoint of thermal efficiency and technical problems.

2. HTTR Hydrogen Production System by Natural Gas Steam Reforming

2.1 Design of HTTR steam reforming system

The HTTR steam reforming system is designed to utilize the nuclear heat effectively and to achieve hydrogen productivity competitive to that of a fossil-fired plant with operability, controllability and safety acceptable enough to commercialization Figure 1 shows an arrangement of main components. The HTTR reactor supplies nuclear heat of 30MW with 950℃ to parallel-loaded two heat exchangers in the reactor cooling loop, namely the IHX of 10MW at the rated heat exchanging rate and a pressurized water cooler of the remaining 20MW. The nuclear heat of 10MW transferred from the IHX to the secondary helium loop is to be utilized for 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 outlet, whereas the IHX outlet temperature is 905℃. Design specifications of the HTTR steam reforming system is shown in Table Ⅰ. The key design concept was already developed as follows[2]·

TABLE Ⅰ Design specifications of the HTTR steam reforming hydrogen production system

Pressure Process-gas/Secondary-helium	4.5/4 1 MPa
Temperature inlet at steam reformer Process-gas/Secondary-helium	450/880 ℃
Outlet at steam reformer Process-gas/Secondary-helium	600/600 ℃
Hydrogen production rate	3800 Nm^3/h
Steam-carbon ratio	3.5

TABLE II Comparison of operational conditions and performance of steam reformers.

Reformer type	Fossil-fired Plant	HTTR system
Process gas pressure	1~3MPa depending upon final products	4.5MPa at the inlet of steam reformer (>Helium pressure P_{He} of 4.1MPa)
Maximum process gas temperature	850~900℃	800℃
Maximum heat flux to catalyst zone	50~80kW/m²	40kW/m²
Thermal energy utilization of steam reformer	80~85%	78%
CO_2 emission from heat source for heat source power	3t-CO_2/h/10MW	0

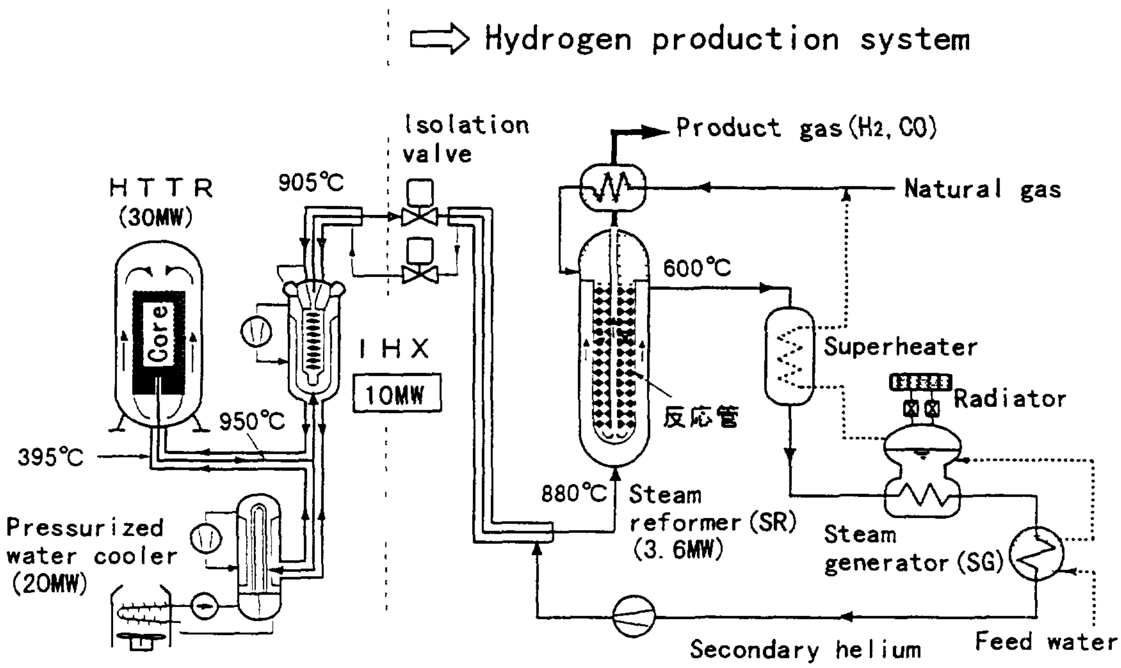

FIG. 1. Flow scheme of the HTTR steam reforming hydrogen production system.

(1) Improvement of steam reformer to enhance hydrogen productivity

Since the steam reforming conditions in the HTTR are higher pressure and lower temperature as shown in Table II, the lower hydrogen productivity is predicted compared with a fossil-fired plant. The following improvements are, therefore, introduced to the SR to increase hydrogen productivity.

1) increasing heat input to the reforming process gas which flows in catalyst tubes,

2) increasing a reaction temperature, that is a process gas temperature at the outlet of catalyst zone in the catalyst tube and

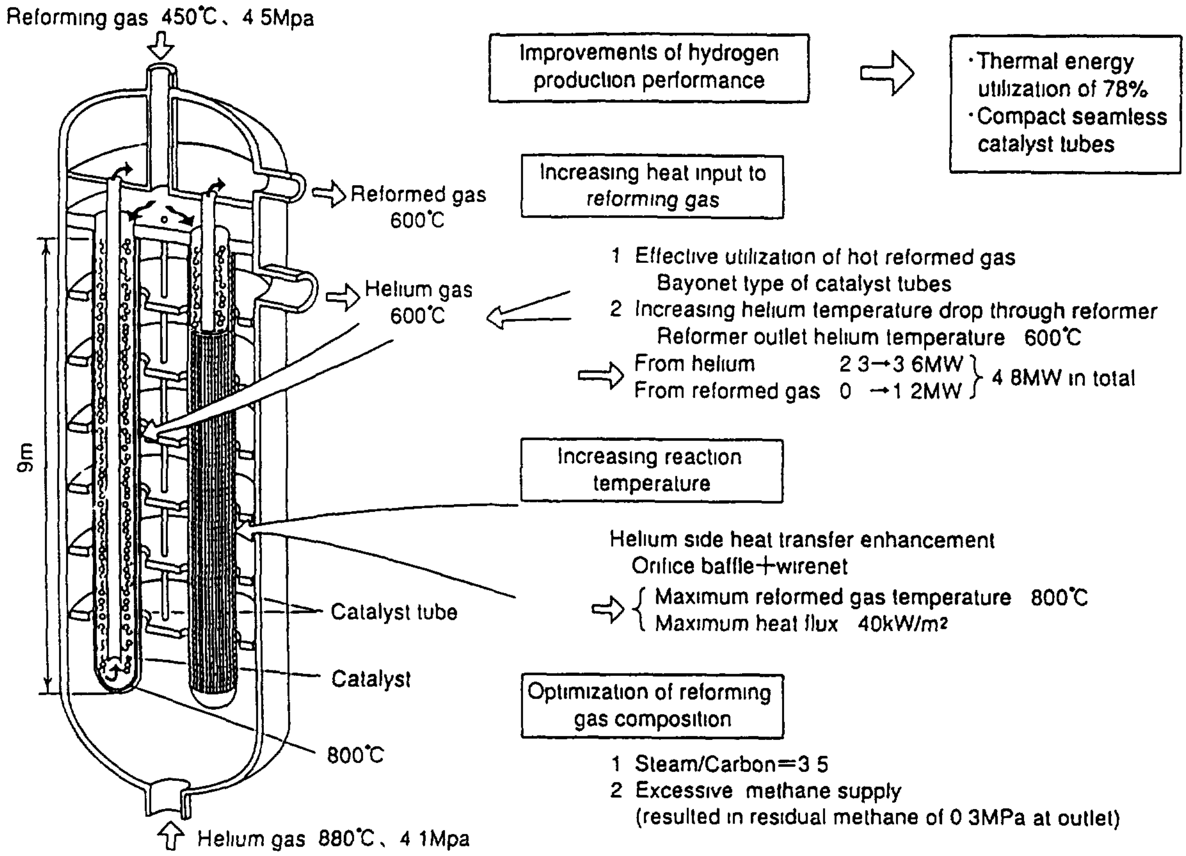

FIG. 2. Improvements of helium-heated steam reformer performance.

3) optimizing reforming gas composition so as to enhance the reforming rate

Survey of improved types of heat exchangers and an analytical study on enhancement of hydrogen production rate lead to proposals shown in FIG 2 These improvements attain a high hydrogen productivity, that is, a thermal energy utilization of 78% is competitive to that of a fissile-fired plant of 80-85% The technologies improved here are applicable also to other HTGR-hydrogen production systems because a heat exchanger type of endothermic chemical reactor is an essential technology

(2) Steam generator in the secondary helium loop for stable controllability and as a safety barrier

A steam generator (SG) installed at downstream of the SR in the secondary helium loop is found to provide the stable controllability for any disturbance at the SR due to a large capacity of heat sink In a thermal transient state that helium temperatures at the SR outlet and then at the SG inlet go up by a malfunction in the reforming process gas line, a helium temperature at the SG outlet can be kept constant at the saturation temperature of steam. Figure 3 shows analytical helium temperatures at inlet and outlet of the SR, SG and so on for the process gas feed rate of 0 to 100% of the rated flow rate. The helium temperature change is reduced from 280℃ at SR outlet to only 5℃ at SG outlet with change of the process gas feed rate. This result suggests the SG works as a thermal absorber and protects the reactor from thermal disturbance caused by the reforming system. Also, the SG is utilized to

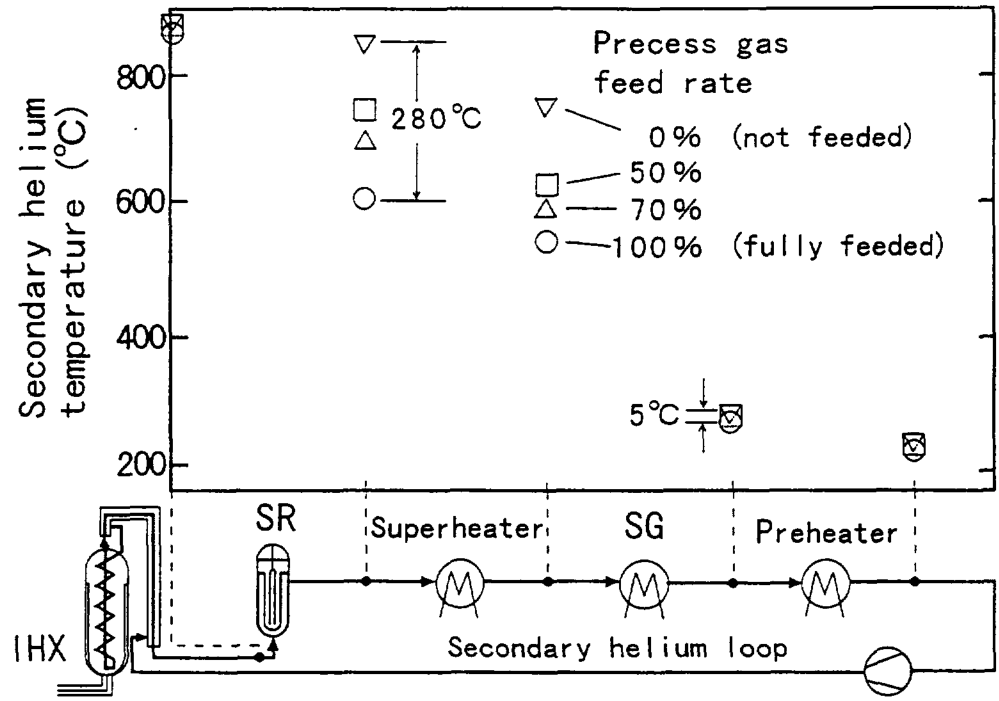

FIG. 3. Change of the secondary helium temperatures at steam reformer and steam generator for the process gas feed rate of 0 to 100%.

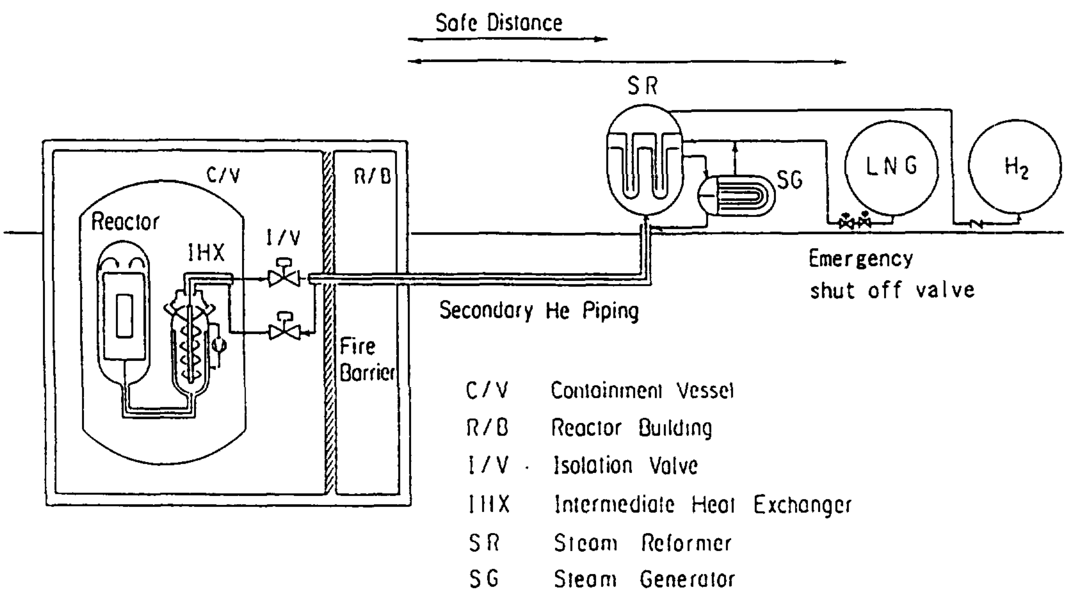

FIG. 4. Safety design concept against fire and explosion.

avoid a reactor scram due to a malfunction or accident at the reforming system. The secondary helium is designed to be passively cooled by only the SG, using natural convection of steam and condensed water between the SG and a radiator installed above the SG. This passive cooling system by the SG is now under design to meet safety requirements with reasonable configuration.

(3) Safety barrier against fire/explosion

A functional or physical barrier is required to assure the safety of a nuclear system and public. The HTTR-steam reforming system has a potential possibility of two-type fire/explosion, namely outside and inside of a reactor building (R/B). The safety design concept is schematically illustrated in FIG. 4. Against the fire/explosion outside the R/B, it is reasonable to assure safety integrity of safety-related items against a potential fire/explosion because a low possibility of fire/explosion should be assumed. Against the fire/explosion inside the R/B, however, it is the principle to take safety measures to prevent occurrence of fire/explosion. The basic design concept is to limit the ingress of amount of explosive gas from the SR through the secondary helium loop within an allowable value. A combination of isolation valves in a containment vessel and emergency shut-off valve in the process gas feed line is effective to restrict the amount of ingress gas.

2.2 Out-of-pile test

Prior to coupling of the steam reforming system with the HTTR, an out-of-pile test is required to confirm the safety, controllability and performance of the steam reforming system under the simulated operational conditions. The out-of-pile system is an approximately 1/20-1/30 scale system of the HTTR steam reforming system and simulates key components downstream from the IHX. The main objectives of the out-of-pile system are as follows:

1) design verification of performance of high temperature components, such as the SR, SG, isolation valve and so on, including hydrogen productivity,
2) investigation of transient behavior of steam reforming system, and
3) establishment of operation and control technologies so as not to give the reactor a significant disturbance at steam reforming system trouble.

The design of the out-of-pile test system has been carried out since 1995, and its construction will start from 1997.

3. Study on Hydrogen Production by Water Splitting

Hydrogen production using a HTGR offers a quite ambitious concept for future energy systems especially when hydrogen is produced from water. JAERI has been conducting basic studies on two hydrogen production processes by water splitting, thermochemical IS and high-temperature electrolysis of steam process, as one of the future heat utilization systems of the HTTR following the steam reforming system.

3.1 Thermochemical IS process

Thermochemical IS process utilizes plural chemical reactions and works like a chemical engine to produce hydrogen by absorbing high temperature heat as shown in FIG. 5. Here, the sulfuric acid (H_2SO_4) decomposition reaction is an endothermic reaction, and

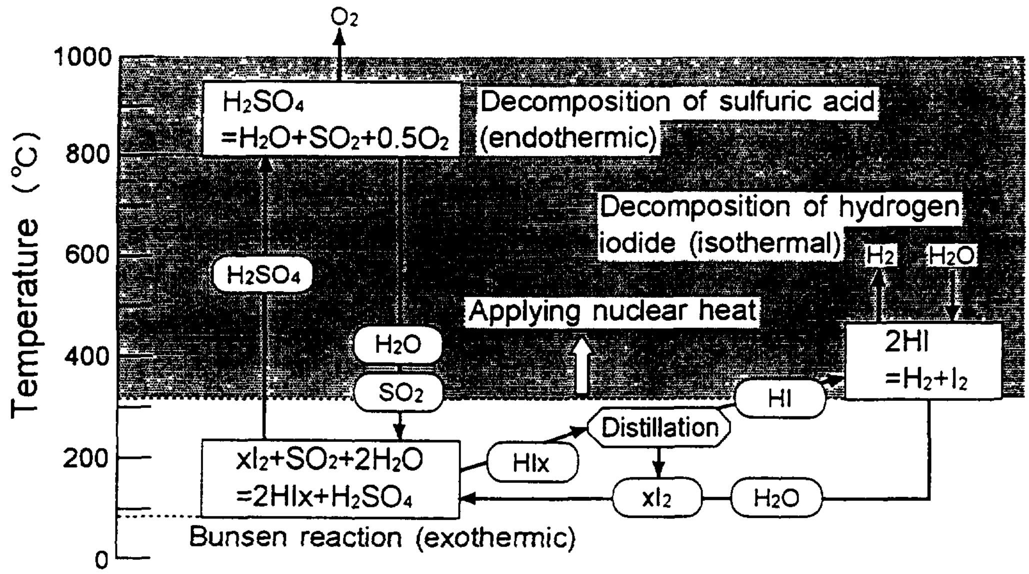

FIG. 5. Reaction scheme of thermochemical IS process.

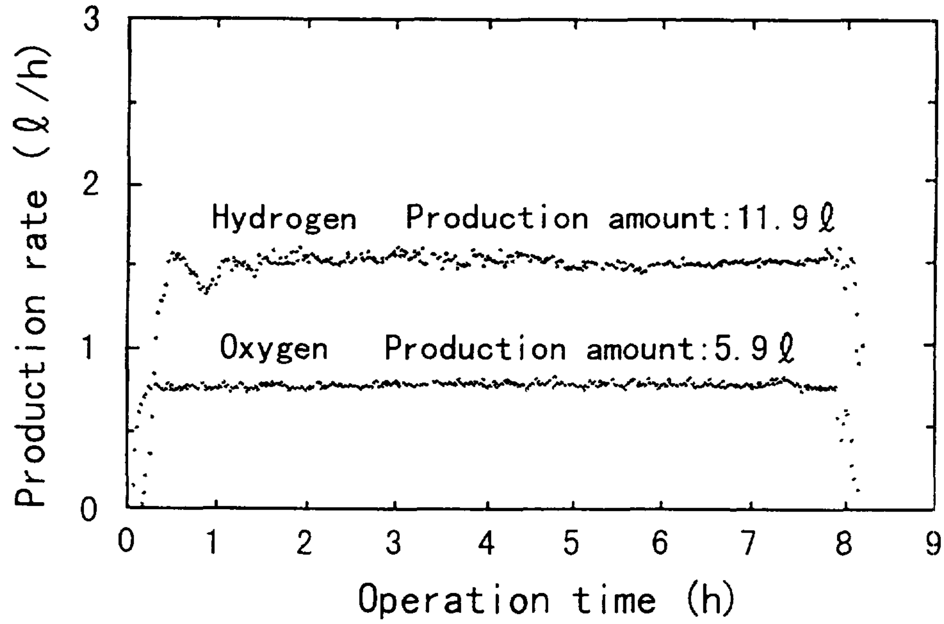

FIG. 6. Results of continuous production of hydrogen and oxygen with the laboratory-scale test.

the so-called Bunsen reaction is an exothermic reaction. The process was first proposed by General Atomic Co.[3], and has been studied at several research institutions [4-6]. The thermal efficiency of the process based on the Higher Heating Value (HHV) of hydrogen has been theoretically estimated to be 47~50% from existing thermodynamic data [3,7].

Continuous and stoichiometric production of hydrogen and oxygen for 8 hours was successfully achieved with a laboratory-scale apparatus made of glass as shown in FIG. 6. However, the obtained thermal efficiency was much less than the theoritical one.

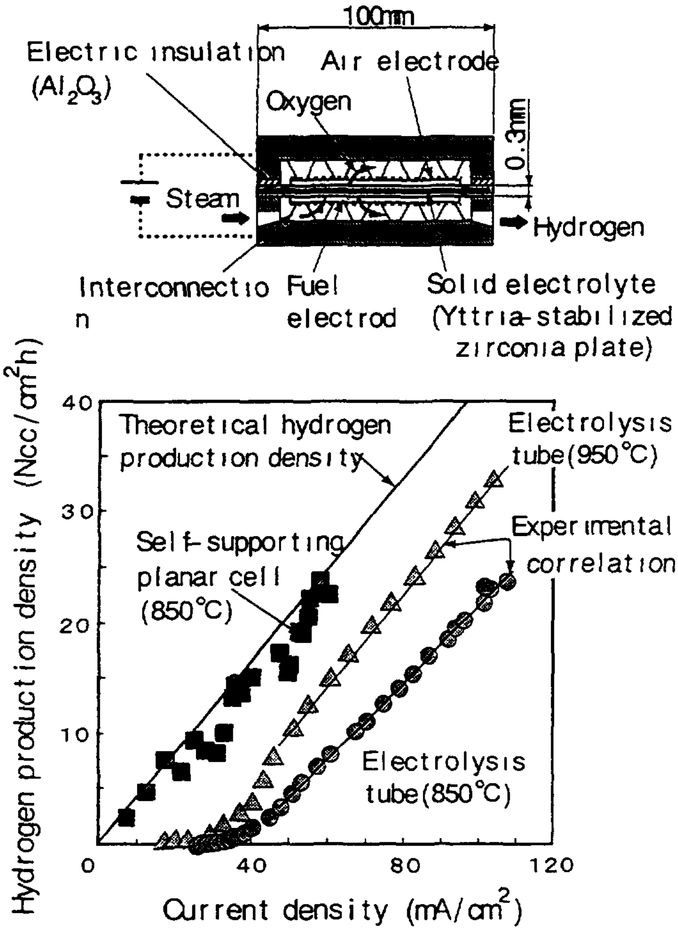

FIG. 7. Hydrogen production performance of planar and tubular electrolysis cells.

Some technologies to be developed, therefore, still remain to improve thermal efficiency. One is a rise in reaction temperature at a Bunsen reaction step, and the other is highly effective separation of hydrogen and iodine at a HI decomposition step. As for the latter, a hydrogen permselective membrane is under study to improve separation efficiency. In addition to the above issues, the development of corrosion resistance materials is necessary for commercial use Especially as for H_2SO_4 boiling environment, any material dose not exist for commercial use. Therefore, a new material based on iron-silicon alloy and an reactor vessel featuring the high corrosion resistance of silicon based ceramics are under study

3.2 High-temperature electrolysis of steam process

A hydrogen production process by an electrochemical water splitting, based on high temperature electrolysis, is considered as one of possible future processes of using a HTGR. Electrolysis cells in the course of fabrication of electrolysis tube and planar cells have been developed. A self-supporting planar cell consists of an electrolyte plate of yttria-atabilized zirconia (YSZ) and porous electrodes as shown in FIG 7. Steam is supplied to the cathode compartment with argon and hydrogen Argon is a carrier of steam, and hydrogen is a

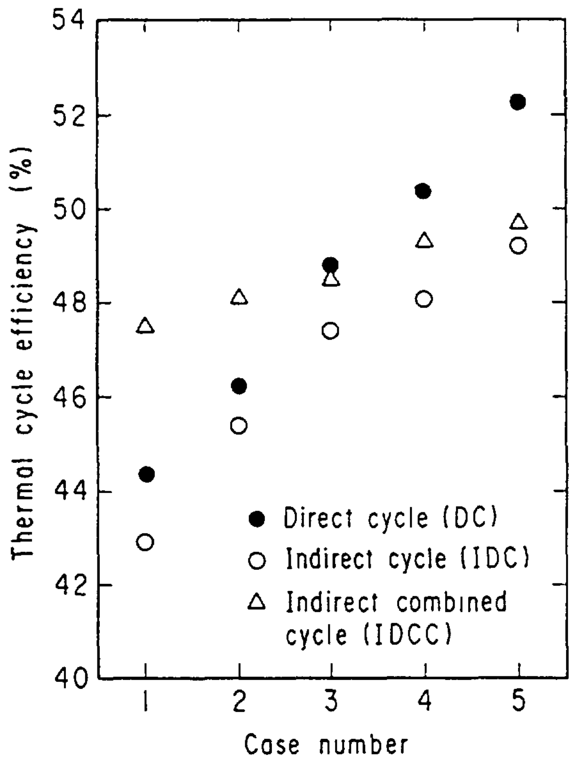

FIG. 8. Thermal cycle efficiencies of the direct, indirect, indirect combined cycle gas turbine systems.

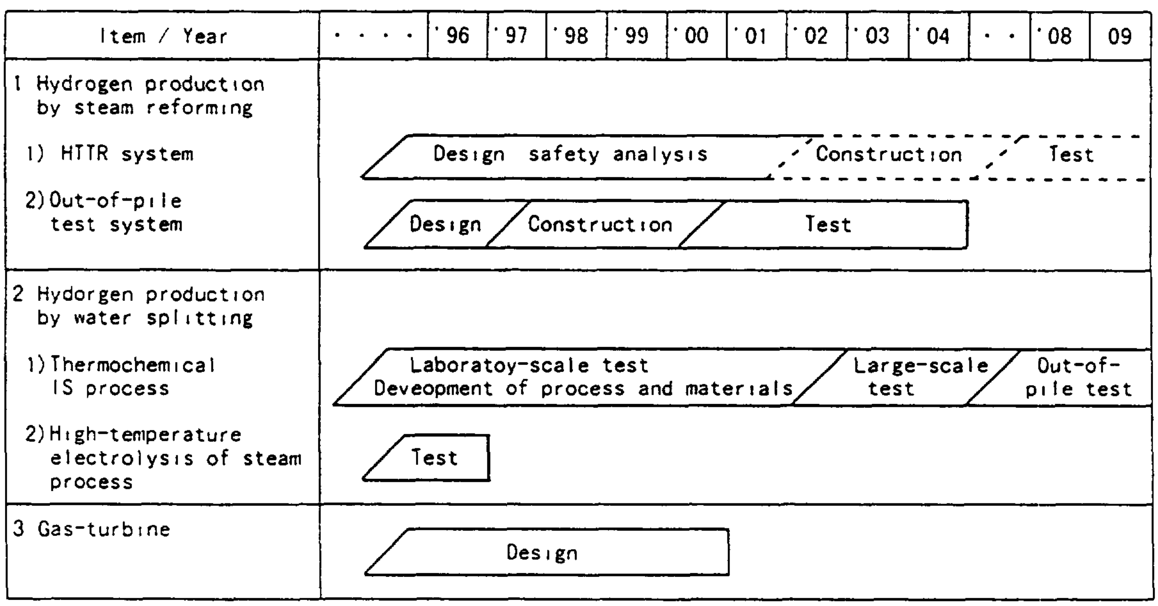

FIG. 9. Development schedule of HTGR utilization systems at JAERI.

reduced gas to keep nickel of the cathode material from oxidation. Dry air is supplied to the anode compartment. Relationship between hydrogen production rate and applied voltage for planar and tubular cells is also shown in FIG. 7. As for the planar cell, hydrogen was produced at rate of 2.3Nm^3/h at an applied voltage of 2.8V. Then hydrogen production density was 3.6Ndm^3/m^{2}h and was 1.4 times larger than that of the tubular cell at 850℃. The tests of electrolysis cells will be finished in 1996 after hydrogen productivity data is accumulated.

TABLE Ⅲ Assumed design parameters for the cycle calculation and obtained thermal cycle efficiencies.

Case		1	2	3	4	5
Design parameters						
·Turbine adiabatic efficiency	%	90	92	92	92	92
·Compressor adiabatic efficiency	%	88	90	90	90	90
·Recuperator effectiveness	%	88	88	92	95	95
·ΔTm for IHX	°C	120	120	100	100	80
Thermal cycle efficiency (DC)	%	44.4	46.2	48.7	50.4	52.3
(IDC)	%	42.9	45.4	47.4	48.1	49.2
(IDCC)	%	47.5	48.1	48.5	49.3	49.7

Note: (1) Reactor outlet temperatures are 850, 950 and 950°C for DC, IDC and IDCC, respectively.
(2) Maximum reactor inlet temperature = 500°C.

4. Design Study on Gas Turbine System

It is expected that a high thermodynamic thermal efficiency of 50% can be achieved in a HTGR gas-turbine (GT) power generation system. This means an effective reduction in the amount of waste heat, radioactive waste and nuclear fuel use. Thus the GT is considered to be one of the most useful power generation system to provide a safer environment.

A design study on thermal efficiencies for the direct and indirect gas turbine cycles has been performed. A direct cycle (DC) is the simplest system and at the same time has a great possibility to attain high thermal efficiency. However, there still remain some key technologies to be developed such as a maintenance of turbomacinery associated with a fission products plate out, a pressure seal of components in a power conversion vessel and a high temperature (500°C) pressure vessel. An indirect cycle (IDC) also has some problems, such as lower thermal efficiency compared with the DC and high temperature pressure vessel, although problems associated with fission products can be removed by an IHX. An indirect combined cycle (IDCC) is an effective candidate to circumbent these problems, since the problems regarding the reactor inlet temperature can be solved by using the heat of lower temperature in a steam cycle and thermal efficiency can be also improved compared with the IDC. Table Ⅲ and FIG. 8 show a relative comparison of thermal efficiencies for the DC, IDC and IDCC under assumed design parameters. They are assumed in accordance with the examination of technologies, which advances in the order 1 to 5. The superiority of DC in the advanced technology stage and that of IDCC in the conservative stage can be observed.

A future work in JAERI will be devoted in the field of an improvement of reactor design and safety, and studies regarding the concepts of power conversion vessel, turbomachinery, recuperator. Furthermore, a trade-off study between IDC and IDCC will be carried out. In parallel to the design, material tests of a recuperator and turbine blade are also planned.

5. Development Schedule

Figure 9 shows the development schedule of the HTGR utilization systems at JAERI. The out-of-pile test of the steam reforming system will be conducted from 2000 to 2004, and the results will be reflected to design and safety review of the HTTR steam reforming system. The construction of the HTTR system will be planned to start from 2002, and be followed by the demonstration test till 2009. The thermochemical IS process is planned as one of the future candidates of the HTTR following the steam reforming system. After a large-scale test using commercially available metal and ceramic materials, an out-of-pile test is scheduled from 2007. The design study on the gas turbine system will be carried out till 2000.

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INTERNATIONAL CO-OPERATION IN DEVELOPING THE GT-MHR

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Abstract

High Temperature Gas-Cooled Reactor (HTGR) technology and its development during the last 30 years has led to the design, construction, and operation of five graphite-moderated helium-cooled reactors with gas outlet temperatures up to 950°C and power levels up to 330 MWe. Design and licensing activities of even larger units (up to 1200 MWe) were well advanced in the U.S. when dropping electricity demand led to cancellation of these commercial units.

The experience from these early plants and related design and development activities provided a solid technology base when gas-cooled reactor development shifted towards smaller, passively safe designs. Although operating experience and past developments are still applicable, the push for safer, highly efficient and economical units, and the need for special applications, has added new requirements that demand and justify further R&D and opens the door for broad international cooperation in the further development of base technologies for HTGR applications.

In the fall of 1995, driven by budget constraints and anti-nuclear sentiments, the US government decided to discontinue financial support of the Gas Turbine-Modular Helium Reactor (GT-MHR). At that time, significant work was underway with participation of several vendors with specialized expertise in various aspects of the GT-MHR. Fortunately, the US government provided for documenting the design and development status through an orderly close-out program. Concurrent elimination of government restrictions opened the door for broader international cooperation.

Discussion between General Atomics and the Russian Ministry of Atomic Energy (MINATOM), in the summer of 1994, led to an agreement on a jointly funded design and development program for the GT-MHR. The program is initially focused on the burning of weapons plutonium that becomes available from dismantled nuclear weapons. The long term goal is to utilize the same design for commercial applications - using uranium fuel. This program took advantage of existing technologies and facilities in the US and Russia, but right from the beginning left the door open for broader international cooperation. Accordingly, in January 1996, FRAMATOME has joined the ongoing effort. Discussions are underway with other international entities to join this program.

The program is proceeding well. Several Russian laboratories/design organizations are participating with GA and FRAMATOME. Significant improvements in the power conversion system design are a clear example of the benefit of the cooperative effort.

Further work needs to be done to confirm fuel and components prior to full deployment, etc., providing ample opportunities for international cooperation in many areas.

INTRODUCTION

The Gas Turbine - Modular Helium Reactor (GT-MHR) is an advanced High Temperature Gas-cooled Reactor (HTGR) being developed in an international cooperative program involving General Atomics (GA), the Russian Federation (RF) Ministry of Atomic Energy (MINATOM), and FRAMATOME. The overall goal of the cooperative program is to first develop the GT-MHR for the disposition of surplus weapons plutonium in the RF and then to offer GT-MHR plants fueled with uranium to the international market for electricity generation.

The first step in this cooperative program was an agreement between GA and MINATOM to complete the GT-MHR conceptual design. FRAMATOME subsequently joined the conceptual design program. Other international organizations are actively being pursued for participation in the conceptual design program and are expected to join shortly.

Very satisfactory process has been made to-date. Good technical work is being performed on the conceptual design, international interest in the program is growing, and US restrictions imposed on GA regarding cooperation with RF on nuclear reactor technology have been lifted. Prospects look good that the conceptual design results will be positive and the program will move on into detail design and technology development activities leading to prototype deployment. These activities will broaden the opportunities for international participation.

Although the design work is proceeding on the basis of using weapons plutonium fuel, the project forms an excellent basis for worldwide commercial deployment of GT-MHR plants. A plutonium fueled plant will demonstrate the design and performance. A commercial plant can then be deployed based on the prototype design but fueled with uranium without need for any significant design changes.

BACKGROUND

HTGRs are characterized by a graphite moderator, helium coolant and coated particle fuel. The graphite moderator and coated particle fuel provide an all ceramic core which allows for the generation of high temperature heat energy for electricity generation at high thermal conversion efficiencies and for process heat applications. Helium coolant is the best single phase, inert fluid available for transport of high temperature heat energy from the graphite core.

The five HTGR plants built and operated to-date have validated the basic HTGR design and performance characteristics. The key characteristics which these plants have proven include:

- The ability of coated particle fuel to reliably retain fission products to very high temperatures (up to about 1600°C) and the production of coated particles in commercial quantities that meet the quality requirements necessary for high temperature retention of fission products.

- The inherent high heat capacity characteristics of HTGR graphite moderated reactor cores that provide long times for corrective measures to mitigate upset conditions without challenging safety limits.

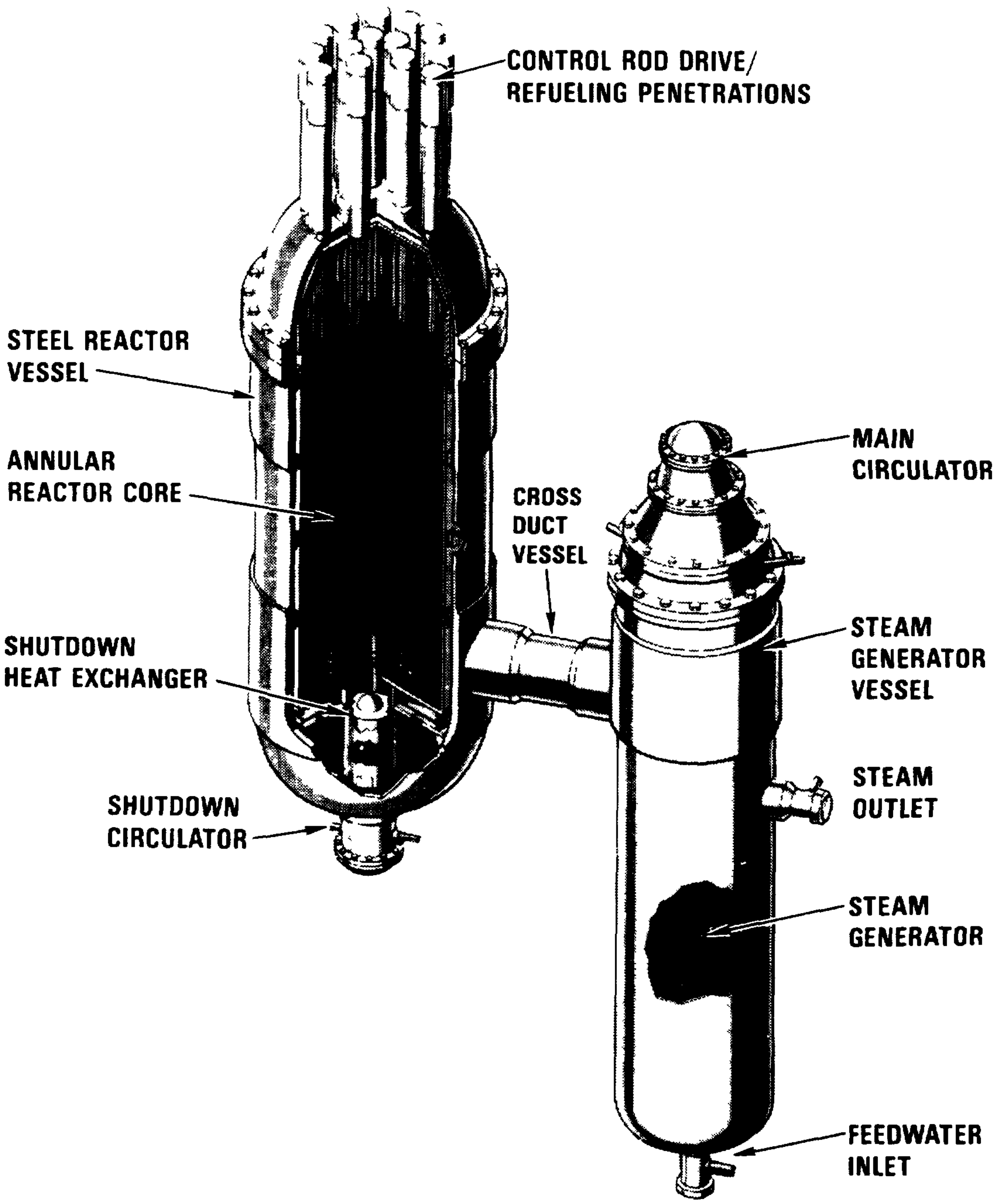

FIG. 1. 350 MWt Modular high temperature gas reactor

- The advantages of helium as a nuclear reactor coolant. Safety issues are simplified by the coolant being single phase; corrosion products are eliminated by the coolant being inert.

- The acceptability of two different fuel configurations, pebble and prismatic.

- The adequacy of the technology data base for the design, analysis, construction, and operation of HTGR reactors.

This HTGR technology data base was developed in the 1960s and 1970s in support of the deployment of large central station power plants. A construction permit had been issued in the US for the first of these plants. However, before large HTGR power plants could be deployed in the US, the oil embargo in the mid 1970s occurred and extensive energy conservation measures were undertaken reducing the electricity demand growth rate. As a consequence, the need for new large central station power plants contracted.

All of the large HTGR plants on order in the US (10 units) were canceled, along with the cancellation of more than 100 LWR plants. The LWR plants under construction were delayed causing capital costs to escalate. Then, in 1979, the Three Mile Island accident occurred which led to wide spread public fear of nuclear power and costly backfits to the existing LWR plants. The financial liabilities, the public's fear of nuclear, and the lack of need for new base load capacity combined to halt new nuclear plant orders in the US.

In the early 1980s, evaluations of the reasons for the dearth of new nuclear plant orders led to the conclusion that smaller, simpler nuclear power plants with inherent safety characteristics were needed for public acceptance. The Modular High Temperature Gas Reactor (MHTGR) was conceived to meet this need, Figure 1. The MHTGR employed a steam cycle power conversion system in common with the prior large HTGR plants.

The MHTGR design had unparalleled safety (meltdown proof), was considerably simpler because of the absence of the need for complex safety systems (completely passive decay heat removal system), and at 350 MWt/135 MWe per module was much smaller than the 1000+ MWe nuclear plant size common in that era. Furthermore, the design maintained fuel temperatures below 1600°C using completely passive means to retain fission products within the coated particle fuel. This provided substantial additional simplifications; the need for secondary containment was eliminated as were the needs for emergency plans for sheltering and evacuation of the public. There was, however, increased importance for the coated particle fuel to meet high quality standards.

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

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

- The HTGR reactor size had been reduced in developing the passively safe module design. At the same time, the size of industrial gas turbines had increased. The technology was now available for a single turbomachine to accommodate the heat energy from a single HTGR module.

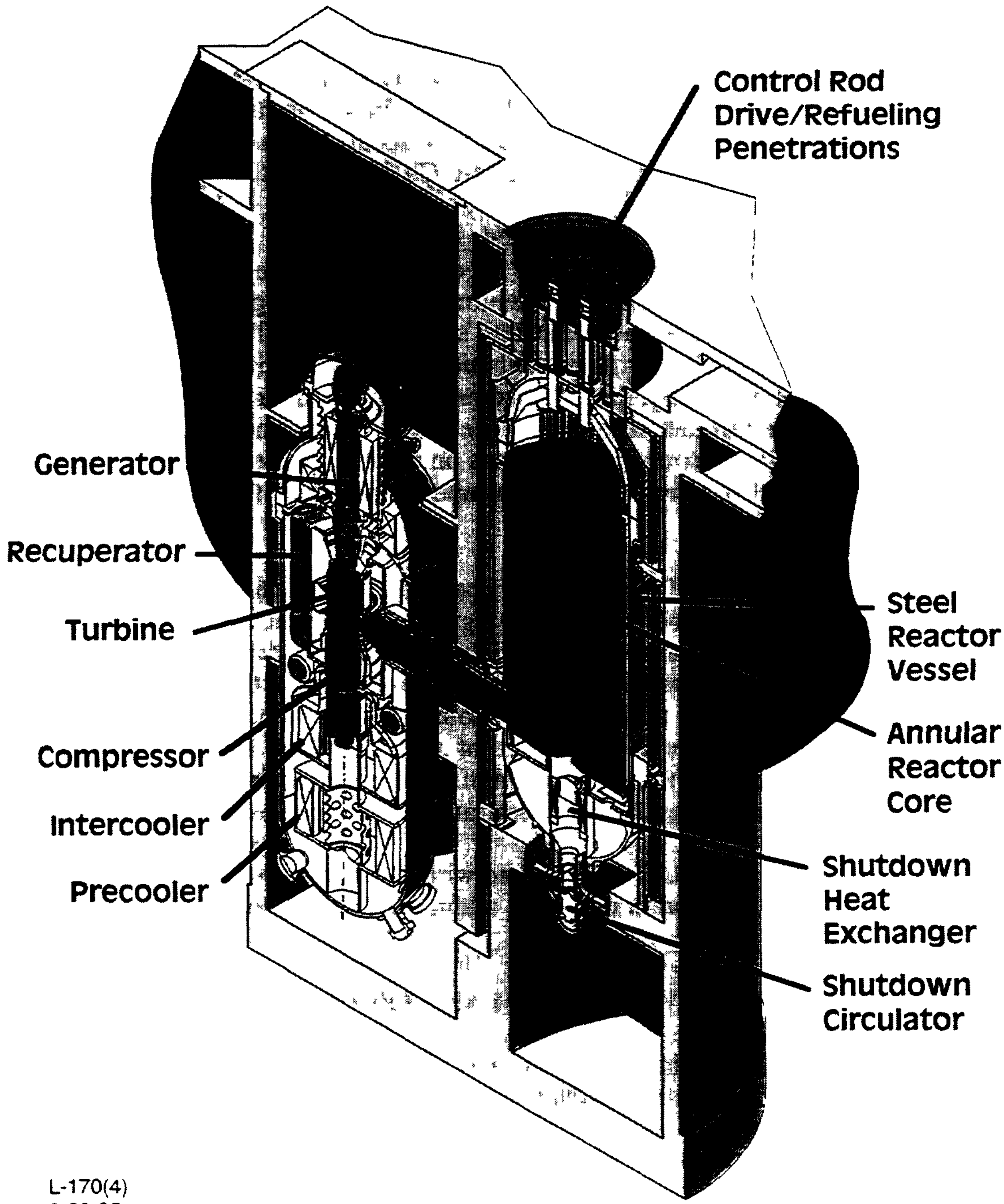

FIG. 2. 600 MWt Gas turbine modular helium reactor

- Highly effective compact plate-fin recuperators had been developed. Recuperator size and capital equipment cost are key economic considerations. Highly effective plate-fin recuperators are much smaller than equivalent tube and shell heat exchangers and provide for substantially less complexity and capital cost.

- The technology for large magnetic bearings had been developed. The use of oil lubricated bearings for the turbomachine with the reactor coolant directly driving the turbine was problematic with regard to the potential coolant contamination by the oil. The availability of magnetic bearings eliminates this potential problem.

The product of integrating these technologies into a closed Brayton cycle power conversion system coupled to a modular HTGR reactor is the Gas Turbine - Modular Helium Reactor (GT-MHR), Figure 2. The rated power of the GT-MHR was increased to 600 MWt, from 450 MWt used in the prior MHTGR, based on further evaluations which showed the higher power level was acceptable without increasing the module vessel diameter. With a net efficiency of 47.7%, a single GT-MHR module produces 286 MWe.

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

Concurrent with the 1993 selection of the GT-MHR as the reference concept for development in the US, GA and MINATOM signed a memorandum of understanding for cooperating on the development of the GT-MHR for commercial deployment. Subsequently, in early 1994, MINATOM proposed that the cooperative program focus on development of the GT-MHR for disposition of weapons plutonium. In the summer of 1994, GA and MINATOM agreed to initiate development of the GT-MHR for weapons plutonium disposition, and to jointly fund the preparation of the conceptual design, with the long term goal of utilizing the same design fueled with uranium for commercial deployment.

In January 1996 FRAMATOME joined with GA and MINATOM as a participant in the cooperative program and is providing additional funding to support the GT-MHR conceptual design work in Russia.

GA/MINATOM COOPERATIVE PROGRAM AND PROGRESS

There are many technical, political, and economic advantages of the GT-MHR cooperative program. The technical advantages include the GT-MHR system advantages (melt-down proof safety, high generation efficiency, high plutonium destruction) and the program advantages realized from reciprocal technology transfer covering a broad spectrum of scientific, engineering, materials and manufacturing know-how. The political advantages include fulfillment of joint plutonium destruction proliferation objectives, economic support for Russian institutes and industries, and development of an environmentally clean, pollution free energy source. The economic advantages include reduced development costs and a nuclear plant design for commercial deployment having low electricity generation cost.

The GT-MHR cooperative program addresses both US and Russian requirements including safety and regulatory requirements, performance and economic requirements, and operational requirements. The requirements include top level design criteria and specific design codes and standards. This approach is being used to enhance the prospect that a uranium fueled design can be readily deployed in markets throughout the world.

The GT-MHR conceptual design work is being directed by a joint steering committee composed of representatives from MINATOM, OKBM, KI, GA, and FRAMATOME. The steering committee provides oversight of project activities and establishes guidelines for conducting the work. It is responsible for establishing the overall direction and strategy of the program.

Unrestricted existing information on the US sponsored GT-MHR design has been made available to the program and this information is serving as a relatively mature starting point for the cooperative conceptual design effort. A major achievement to-date has been significantly improved power conversion component conceptual designs with major contributions from the OKBM designers, Figure 3. The current GT-MHR power conversion configuration now incorporates both US and Russian technology and is a clear demonstration of the cooperative program advantages.

A goal of the current program is to evolve it into a broad based international cooperative program and efforts are underway to foster participation of other international industrial organizations and/or governments. Several organizations in major nations with nuclear power capability have expressed interest in becoming involved with the program. Discussions are ongoing to determine appropriate terms and conditions under which they might participate. A broad based international cooperative program reduces the up-front financial burden on any one country and is considered by many international leaders to be the best way available today to develop a modern nuclear power system for commercial deployment in the world market.

Following completion of the conceptual design, scheduled for October 1997, the broad based international cooperative program would be responsible for carrying out the detailed design and development activities.

PLANNED GT-MHR DEVELOPMENT ACTIVITIES

The GT-MHR concept is based on taking advantage of existing technology to the maximum practical extent (e.g., the core is designed to use the same prismatic fuel proven in the Fort St. Vrain demonstration plant). No new technology has to be developed because the needed technologies have essentially been demonstrated elsewhere. The GT-MHR concept does, however, incorporate new features for addressing new requirements which require additional development. Principle among these features are passive decay heat removal, plutonium fuel, and use of a closed Brayton cycle power conversion system. The development activities planned for these new features are described here organized into the development tests for the reactor system and the development tests for the power conversion system.

GT-MHR Reactor System Development Tests

The development tests planned for the GT-MHR reactor system, Figure 4, include fuel tests, tests of the reactor cavity cooling system, and design verification tests of key reactor sub-systems and components. These tests are briefly described in the following.

Fuel Tests are planned to qualify TRISO-coated plutonium particle fuel for GT-MHR performance parameters and to provide data for validation of fuel performance models including fission gas release and plate-out. To supply fuel for these tests, current fuel manufacturing equipment and procedures will be adapted for processing weapons-grade plutonium into GT-MHR fuel.

Reactor Cavity Cooling System (RCCS) Tests are planned to provide design data and verify integrated performance. The tests include determining the effective conductivity of the graphite core, buoyancy-induced fluid mixing in the enclosures along the core, and emissivities of metal surfaces including the RCCS panels, reactor vessel and metallic reactor internals. Final verification of the RCCS design is planned by an integrated RCCS systems test using a scale model of the reactor metallic internals and reactor vessel.

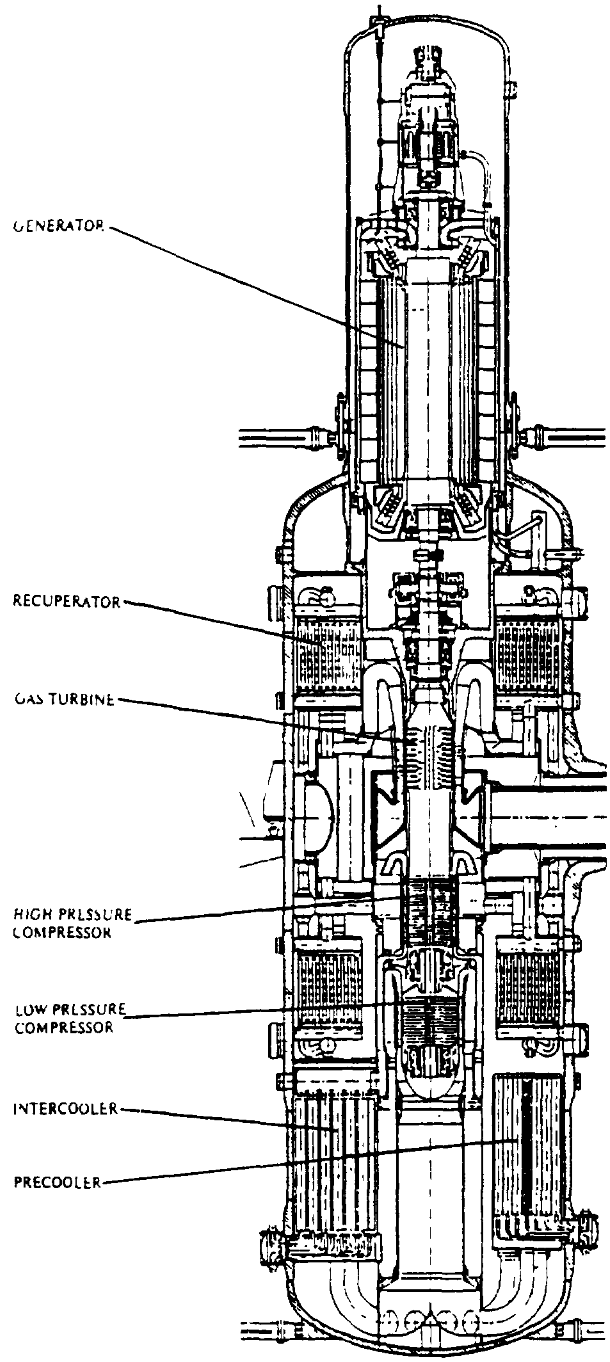

FIG. 3. GT-MHR power conversion system incorporates US and RF technology

Design Verification Tests are planned to verify the design and performance of the of the principle reactor sub-systems and components which include the following:

- The Reactor Internals and Hot Duct
- Fuel Handling System
- Neutron Control System
- Reactor Service Equipment
- Shutdown Cooling System

GT-MHR Power Conversion System Development Tests

The development tests planned for the Power Conversion System (PCS), Figure 5, include design verification tests of the key sub-systems and components and performance testing of a full-scale integrated power conversion system using a fossil-fired heat source. These tests are briefly described in the following.

Design Verification Tests are planned to verify the design and performance of the of the principle PCS sub-systems and components which include the following:

- Generator Tests
- Turbocompressor Tests
- Magnetic Bearing Tests
- Seals Tests
- Recuperator Tests
- Precooler/Intercooler Tests

PCS Integrated Tests are planned using the first GT-MHR power conversion module and will be performed in two phases using a fossil energy heat source:

- Phase I, Steady State Simulation - This phase will be done in two steps: Step 1 will reproduce full helium temperatures distribution at part pressure, and Step 2 will be full helium pressures distribution at part temperature.

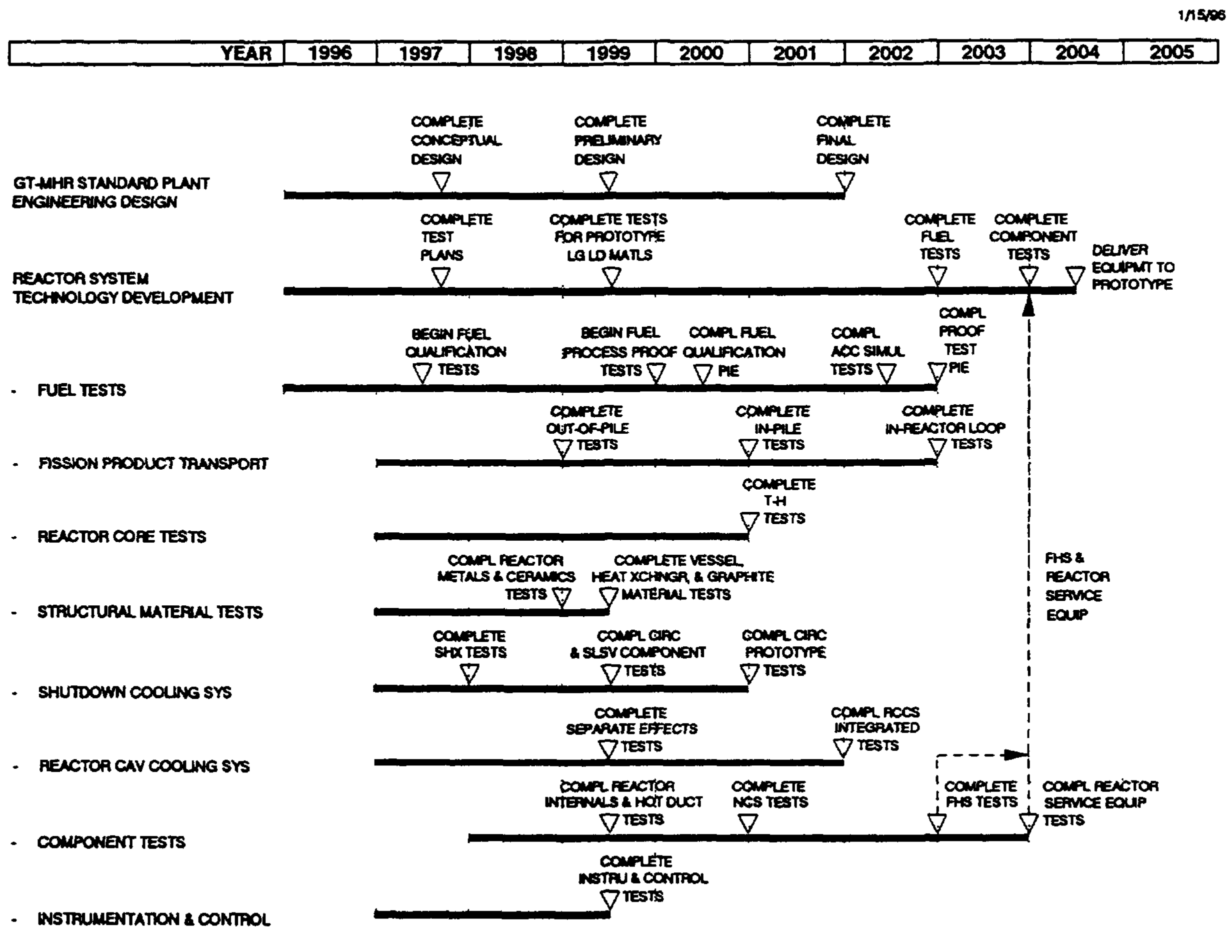

FIG. 4. GT-MHR reactor system summary development schedule

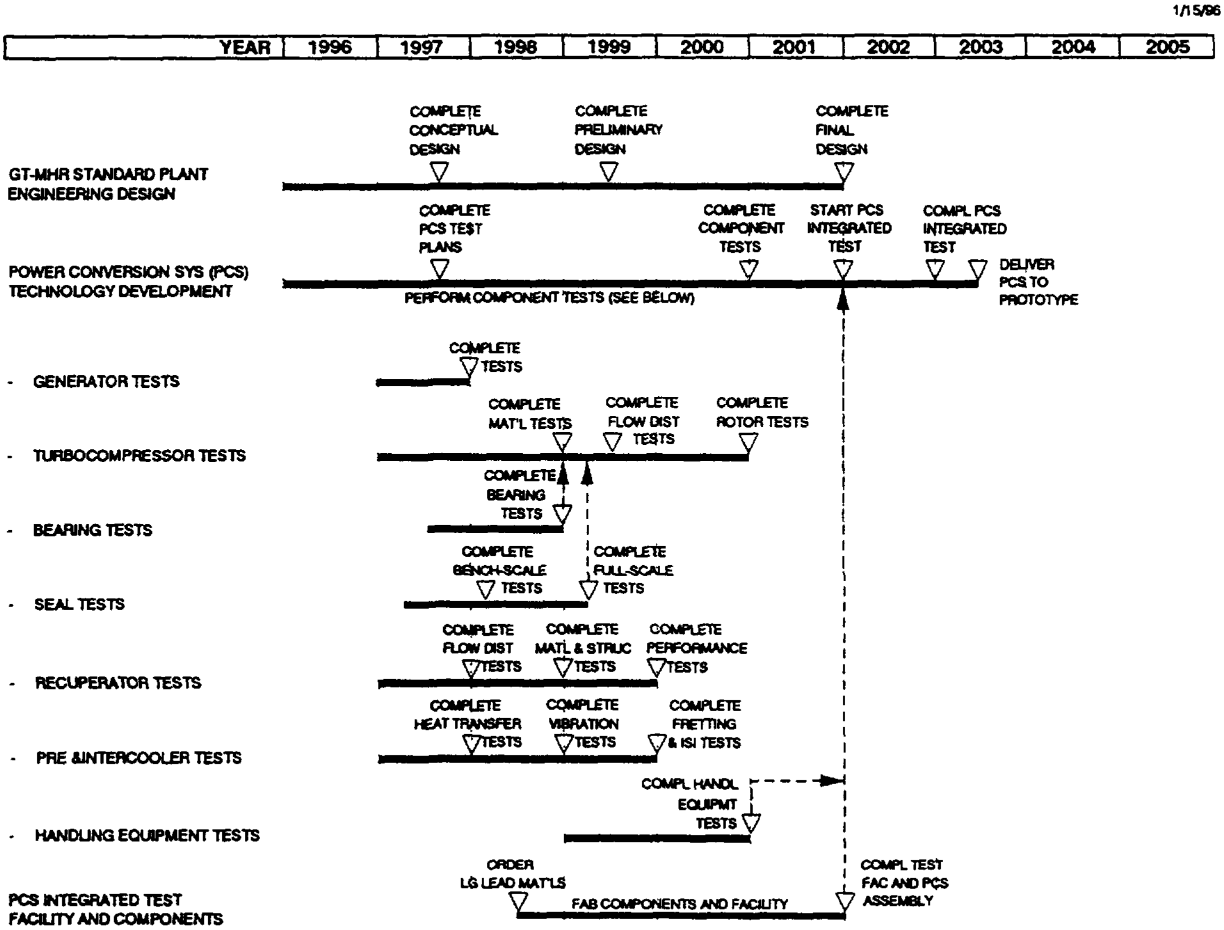

FIG. 5. GT-MHR power conversion system summary development schedule

- Phase II, Key Transient Simulation - Two transients will be run for each plant duty cycle selected according to their severity in producing thermal and pressure loads on the PCS components. One transient will start from a steady state condition in which the helium temperatures distribution has been simulated at part pressure, and the other from a steady state condition in which the helium pressure distribution has been simulated at part temperature.

COMMERCIALIZATION OF THE GT-MHR

In addition to design, development, and licensing activities, the overall program plan for development of the GT-MHR, Figure 6, includes the construction and operation of a prototype module. Performance testing of the prototype will demonstrate all of the GT-MHR systems and their operating characteristics in a fully prototypical plant.

The licensing activities will be pursued in parallel with the detail design and development activities and have the objective that the prototype plant satisfy US, RF, and other applicable international regulatory requirements. The intent is to address the licensing issues which will be directly applicable to follow-on commercial units.

Construction and operation of the prototype module, fueled with weapons plutonium, is the final preparatory step needed for commercialization. The design and development activities, the licensing activities, and the construction and performance of the prototype form the foundation for commercializing the technology. The manufacturing, construction, licensing, and operating experience obtained will provide a firm basis for the information necessary for establishing terms on the part of sellers for commercial offerings and for obtaining required financing on the part of buyers.

The participants in the international cooperative program for development of the GT-MHR will have direct access to the technology. Direct access to the technology will enable the participants, either individually or as partners, to become vendors of GT-MHR commercial plants for the world electric generation market. There is a large market projected for new electric generation capacity in the next century and the projected cost and environmental benefits of the GT-MHR show it has high potential to be highly cost effective and capture a significant share of the world market.

Advanced discussions are now being held with additional potential participants. It is expected that the participant membership will be finalized prior to the initiation of the detail design and development activities planned after completion of the conceptual design in the fall of 1997.

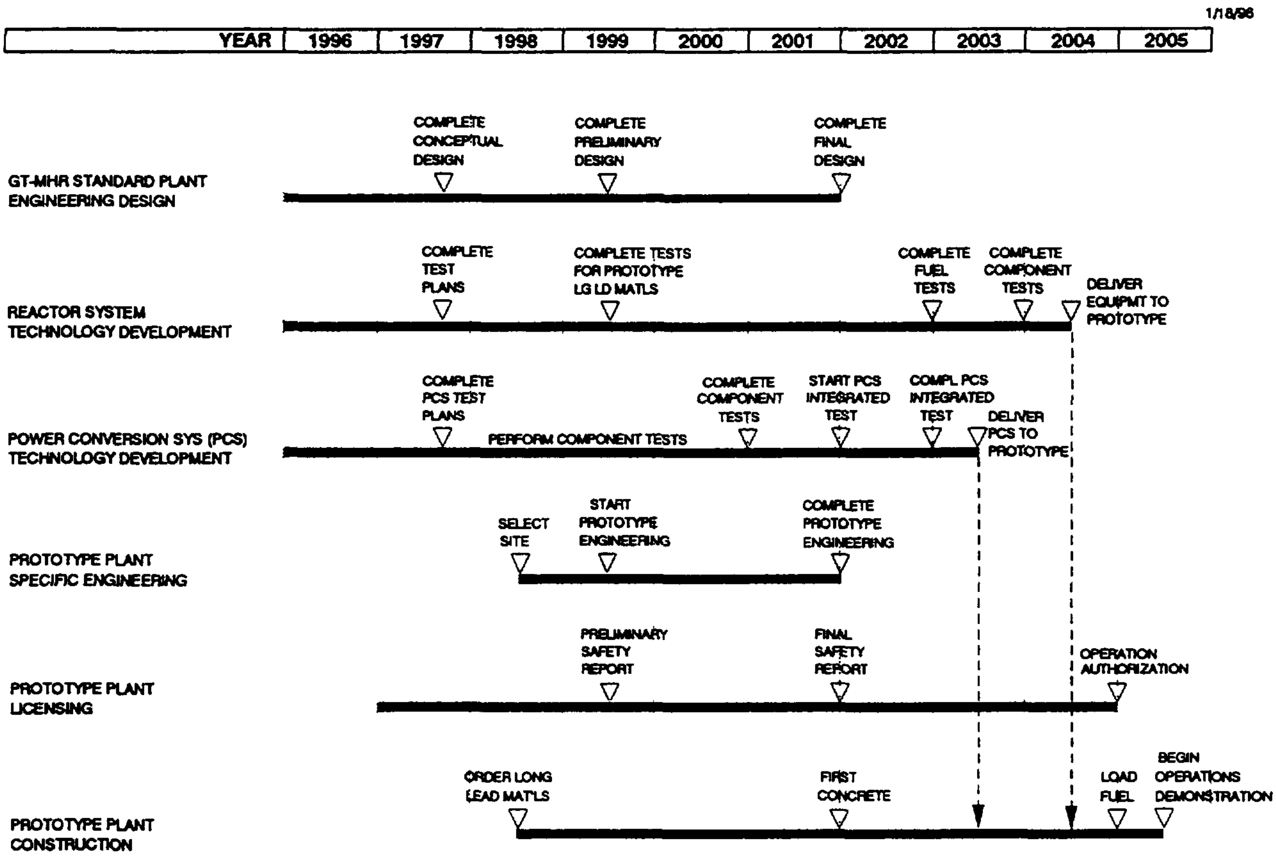

FIG. 6. GT-MHR international program summary schedule

CONCLUSIONS

The key conclusions regarding the GT-MHR background, international cooperative program, development requirements, and commercialization considerations in this paper are as follows:

- Substantial HTGR technology has been developed in support of the five HTGR plants that have been built and operated and in the programs carried out for developing the technology for large central station power plants. The key characteristics of HTGR plants, which include coated particle fuel, graphite moderator, and helium coolant, have been demonstrated.

- Repercussions of the oil embargo in 1973 followed by the Three Mile Island accident in 1979 combined to halt orders for large central station nuclear power plants in the US due to the large financial liabilities, the public's fear of nuclear, and the lack of need for new base load capacity.

- The MHTGR with a power rating of 350 MWt/135 MWe was conceived to meet the perceived need for smaller, simpler nuclear power plants with passive safety characteristics but was concluded to be non-competitive with equivalent sized coal and LWR plants. A larger MHTGR variant with a power rating of 450 MWt/175 MWe was more competitive but was still not competitive with combined cycle plants.

- To be competitive, the thermal efficiency of nuclear power has to be markedly improved to compete with modern high efficiency fossil plants. Technological developments during the last decade on industrial gas turbines, magnetic bearings and highly effective recuperators have made practical the coupling of a modular HTGR with a Brayton cycle power conversion system for the generation of electricity with net thermal efficiencies approaching 50%. The GT-MHR is the product of this coupling.

- Budget constraints and anti-nuclear sentiments have caused the US government to discontinue financial support of the GT-MHR. Fortunately, the US government provided for documenting the design and development status and eliminating restrictions on providing other countries access to the documentation.

- GA and MINATOM have initiated a cooperative program for the development of a conceptual design of the GT-MHR for weapons plutonium disposition with the long term goal of utilizing the same design for commercial deployment using uranium fuel. FRAMATOME has subsequently joined the cooperative program.

- The GT-MHR cooperative program has many technical, political, and economic advantages including the benefits realized from reciprocal technology transfer covering a broad spectrum of scientific, engineering, materials and manufacturing know-how. Design progress to-date clearly demonstrates the advantages of the reciprocal technology transfer.

- A goal of the current program is to evolve it into a broad based international cooperative program and several international organizations have expressed interest in joining. A broad based international cooperative program is considered by many international leaders to be the best way to develop a modern nuclear power system for commercial deployment.

- No new technologies need to be developed for the GT-MHR. The GT-MHR concept does, however, incorporate new features for addressing new requirements which require additional development. Principle among these features are passive decay heat removal, plutonium fuel, and use of a Brayton cycle power conversion system.

- Development tests required for the GT-MHR include fuel tests, tests of the reactor cavity cooling system, design verification tests of key systems and components and performance testing of a full-scale integrated power conversion system using a fossil-fired heat source.

- The GT-MHR deployment program includes construction and operation of a prototype module. Construction and operation of a prototype module is the final step needed for commercialization of the technology.

- The participants in the GT-MHR international cooperative program will have direct access to the technology which will enable them to become vendors of GT-MHR commercial plants for the burgeoning world electric generation market.

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RECENT ACTIVITIES ON THE HTGR FOR ITS COMMERCIALIZATION IN THE 21st CENTURY

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Abstract

Currently, the greatest concern about energy is the need to rapidly increase the energy supply, while also conserving energy reserves and protecting the worldwide environment in the coming century. Furthermore, the direct use of thermal energy from nuclear reactors is an effective way to widen the application of nuclear energy. From this standpoint, Mitsubishi Heavy Industries (MHI) has been continuing the various activities related to the High Temperature Gas Cooled Reactor (HTGR). At present, MHI is participating in the High Temperature Engineering Test Reactor (HTTR) project, which is under construction at Oarai promoted by the Japan Atomic Energy Research Institute, as the primary fabricator. Moreover MHI has been conducting research and development to investigate the feasibility of HTGR commercialization in future.

In this paper, the results of various studies are summarized to introduce our HTGR activities.

First, a basic development scenario for the commercialization of the HTGR in 21st century was formulated. In this scenario, it was clarified the development of technologies for the reactor and for the application of nuclear heat can proceed independently. This is apparently in contrast to the situation for other types of reactor. The development scenario proposed here consists mainly of a three stage process leading to the commercialization of HTGR. In the first stage, an HTGR power plant should be constructed to demonstrate the technologies and the economic characteristics of such a large-scale reactor. In the next stage, integration of the HTGR technologies and the nuclear heat utilization systems should be demonstrated by the construction of HTGRs which incorporate typical heat application systems such as a coal gasification system, an H_2 production system and a gas turbine.

In the final stage before commercialization, several types of commercial HTGR plant will be constructed and operated. Typically these plants will be modular, high performance facilities, with primary energy resource production from H_2, methanol, coal gas, and a combination of various systems.

The following were selected as the main criteria for reviewing the feasibility of the HTGR development scenario described above.

1) A design study of the HTGR plant concept and an investigation of its economic characteristics.

The conceptual design studies of HTGR rated power from 50MWe to 600MWe have been conducted. From the results of these studies, the following points have been clarified.

① The construction of a modular type HTGR is feasible using technologies proven in the operation of HTTR.

② A large-sized HTGR, such as a 600MWe plant, is feasible through the application of new technologies under development.

③ The economic characteristics were investigated. A large sized HTGR has the advantage of the cost improvement, and is expected to be competitive with the LWR.

2) A development study of the various nuclear heat application systems and the investigation of their main characteristics.

The concept of cogeneration systems using a gas turbine, a methanol production system and a coal gasification system have been developed and evaluated, from both a technical and economic perspective. The important results are following.

① The key technologies that enable a He gas turbine to achieve high efficiency have been studied and an optimized system configuration has been proposed.

② It was recognized that a system combining He gas and a steam turbine was quite promising for an advanced power plant.

③ A conceptually new coal gasification system has been proposed and developed. This system was combined with steam gasification and a partial oxidation process, which is expected to improve the coal conversion ratio.

④ The concept of a methanol production plant consisting of coal gasification and SOSE (Solid Oxide Steam Electrolysis) has been developed, and compared with other methanol production systems consisting of steam gasification and steam reforming. As a result, it was clarified that the both systems are promising from the technical and economic view points .

⑤ A conceptual design study of a new electric storage plant integrated with the HTGR, the coal gasification plant, the SOSE, the methanol production plant, and the methanol power plant has been conducted. As a result, the feasibility and cost advantage versus a pumped storage system were investigated.

1. Introduction

Currently, the greatest concern about energy is the need to rapidly increase the energy supply, while also conserving energy reserves and protecting the worldwide environment in the coming century. Furthermore, the direct use of thermal energy from nuclear reactors is an effective way to widen the application of nuclear energy.

We believe the HTGR is most promising candidate to meet these various needs because of its diverse application of nuclear energy with high temperature output. From this standpoint, MHI has been conducting the various activities related to HTGRs.

In particular, MHI is currently participating as primary fabricator in the HTTR project under construction at Oarai by the Japan Atomic Energy Research Institute.

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

In this paper, the results of various studies are summarized to introduce our HTGR activities .

2. A Scenario for HTGR Commercialization

2.1 The role of the HTGR

Based on the prospectus for energy conditions in 21st century, the following expectations of relevance to the HTGR were derived.

(1) Wider use of the nuclear energy making up for the shortage of fossil fuels

(2) Wider application of coal energy instead of natural gas and petroleum energy

(3) Higher safety potential for nuclear plants

The HTGR can supply higher temperature energy than other nuclear power systems. By exploiting this characteristic, the HTGR can play a key role in the production of hydrogen energy, which is expected as final destination of the search for a clean new energy source; this could be accomplished through technology for reforming coal in combination nuclear energy of the HTGR. Therefore, we recognized that the HTGR should be positioned as a mainstay of future energy supply systems.

2.2 The Purpose of HTGR Development

The HTGR that will emerge in the next century should have the following distinct features.

(1) It should use nuclear resources effectively in consideration of a Pu recycling plan.

(2) It should be able to produce a huge amount of hydrogen, with a view to meeting needs for into the future.

For this purpose, the HTGR should be developed with the goals of a higher output temperature and a larger sized plant.

(3) It should have cost competitiveness with other types of power supply plant.

Therefore, the cost reduction effort must always be considered in each development stage of the HTGR.

(4) It should be equipped with innovative technologies that will meet the diverse requirements of 21st century.

2.3 The HTGR Commercialization Scenario proposed by MHI.

The final target of HTGR development is focused on two systems: a hydrogen production HTGR and a coal gasification HTGR.

(1) Development Approach

At present, an HTTR is under construction and we expect to apply most of the basic technologies of the HTTR to the HTGR plant of next stage.

We believe operation of the reactor system in a demonstration plant will be necessary before construction of commercial HTGRs. This scenario differs from the approach of FBR commercialization, in which a prototype reactor should be constructed.

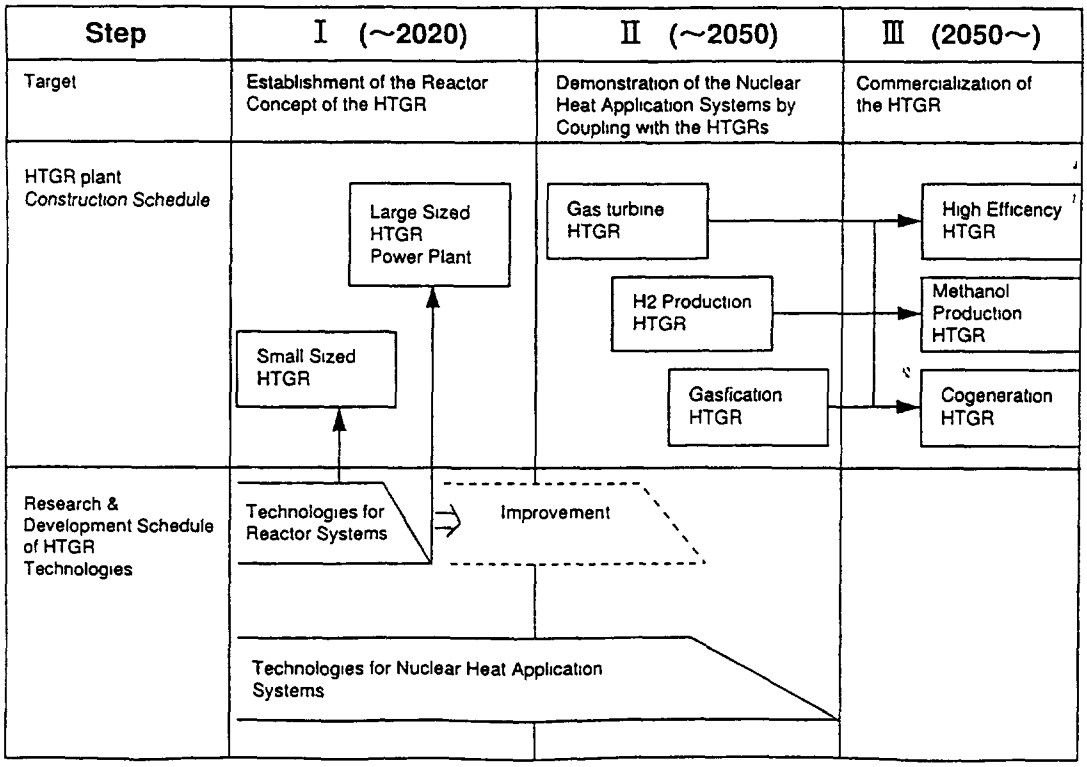

Fig.1. *Proposed Development Scenario for the HTGR Commercialization*

Generally, HTGR outlet power is limited by many restrictions, and therefore large scale explanation of technologies should not be necessary, as it is with the FBR. Fig-1 Shows the proposed approach for HTGR commercialization.

This scenario consists of three stages. In the first stage, we should demonstrate the technologies of the reactor system in which a large sized core and high performance fuels will be designed. The basic design concept of the demonstrated reactor system should be close to that of the commercial reactor system. Therefore, in the commercial stage, this design concept should prove to be cost competitive with FBRs and LWRs . With this background we propose that the HTGR demonstration plant should be an electric power plant using a conventional steam turbine system for the following reasons.

Since many of the technologies have been developed in the HTTR, the construction time should not be lengthy.

The total investment for the demonstration HTGR should be minimized and recoverable as soon as possible. It is presently expected that funding for the construction will be supported by the nuclear power utilities, therefore, the economics of the construction cost will be considered to be the most important issue.

We will improve the performance of the plant by coupling it with a gas turbine system in the next stage.

In the second stage, several HTGR plants will be constructed to demonstrate the feasibility of various nuclear heat application systems, which will be developed independently. The typical nuclear heat application systems that should be demonstrated in this stage are a hydrogen production system using SOSE (Solid Oxide Steam Electrolysis) and a coal gasification system.

In the third stage, the several types of commercial HTGR will be constructed, coupled with an optimized heat application system whose basic concepts will have been demonstrated in the previous stages. The candidates for the commercial HTGR are a coal gasification HTGR and a gas turbine electric supply HTGR. H_2 production will be a key issue having a potential to change an energy utilizing condition in human society from a fossil fuel to a H_2 fuel in far future. The use of a coal energy will widen because of a shortage of other fossil fuel like a petroleum and a LNG. However the coal fire releases a huge number of CO_2, the gasification technology will expected to convert a clean energy. The gas turbine electric supply by HTGR is also an essential technology due to it's safety characteristics, high efficiency and economics.

3. HTGR plant conceptual design study

In the HTGR development scenario, MHI proposes that an HTGR electric power supply plant will be most advantageous kind of plant to demonstrate HTGR. One of the greatest concerns about an HTGR plant is its economic characteristics; in particular, the busbar cost should be competitive with that of the LWR. From an economic perspective, it is known that a large sized plant is advantageous. To investigate the potential improvement of HTGR economics, a feasibility study was conducted for an HTGR of twice the power module of earlier systems.

To investigate the economic characteristics, a preliminary design for a very small sized HTGR was also studied.

3.1 Large sized HTGR plant design study. [1]

(1) Reactor design

To determine the maximum reactor power that, given the limits of the decay heat removal system, would result in acceptable fuel temperatures under accident conditions, we studied the feasibility of a core with the following design conditions.

1) Reactor vessel outer diameter is about 9.0 meter (maximum).

2) ZrC used as the coated fuel material with superior capacity for holding fission products.

3) Peak fuel temperatures during accidents limited to below 1800℃.

4) The control rod material uses a C/C composite.

Fig-2 shows a cross section of reactor pressure vessel, and Fig-3 shows a cross section of the core.

Table. I shows the main specifications of the large sized HTGR plant.

(2) Plant design

To improve the economic performance, we studied the feasibility of a plant having the following design conditions.

1) Coolant outlet temperature of 850℃ (maximum) to permit adoption of a gas turbine cycle in the future

2) A single cooling system per reactor

3) Steam generator and reheater co-located within the same vessel and arranged side-by-side. This means that this vessel is connected to the reactor vessel by a cross duct.

4) Steam turbine cycle for the generation of electric power.

Fig-4 shows a flow diagram of this plant.

3.2 Features of a small sized HTGR plant

Clearly, a small sized HTGR plant has an economic disadvantage; however, it has the following potential advantages as well.

- Further inherent safety characteristics
- Long term burning
- Eeay transportation with total plant system
- Reduced total investment

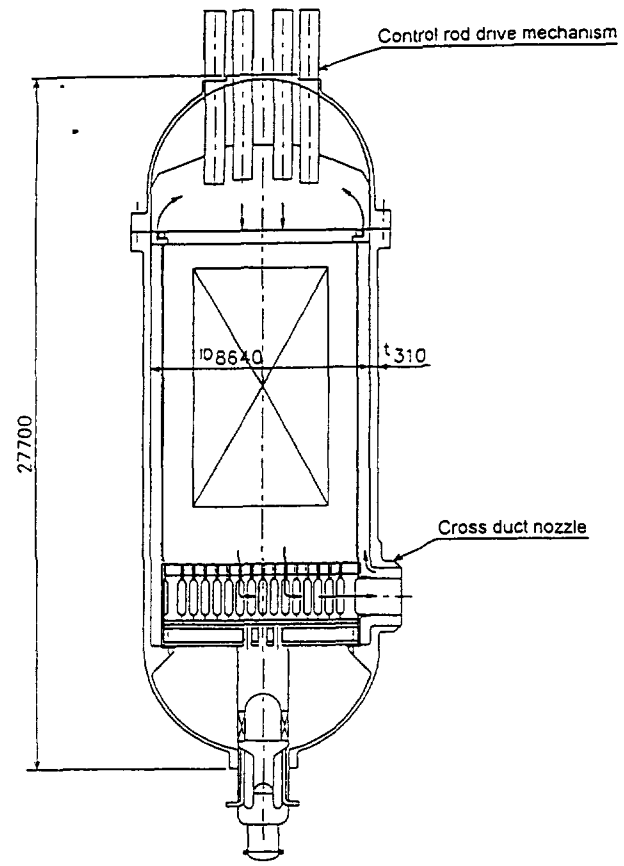

Fig.2. Reactor Pressure Vessel

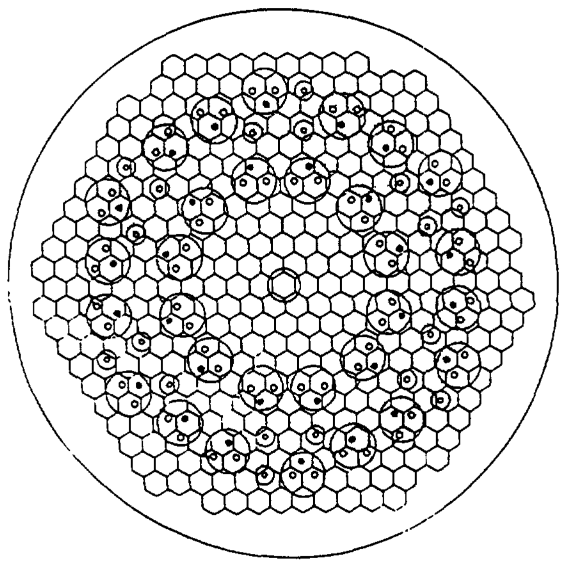

Fig.3. Core Cross Section

Table. I. Specification of Large-sized HTGR

1.	Reactor	
	Thermal power	1150 MWt
	Reactor vessel OD	9.05 m
	Active core configuration	Annular
	Effective active core diameter	3.62 m / 6.17 m
	Effective active core height	9.52 m
	Average power density	6.3 MW/m^3
2.	Primary cooling system	
	Coolant outlet/inlet temperature	850°C / 490°C
	Coolant inlet pressure	72 ata
	Coolant flow rate	2214 ton/h
3.	Fuel	
	Type	UCO
	U enrichment	19.8%
	Average fuel bumup	131000 MWD/MTU
4.	Steam generator	
	Heat-transfer pipe	Helical coil type low-fin tube
	Outside tube fluid (inlet/outlet temp.)	Primary helium gas (795°C / 490°C)
	Inside tube fluid (inlet/outlet temp.)	Pressurized water (superheat steam) (281°C / 538°C)
	Exchange calory	975 MWt
5.	Reheater	
	Heat-transfer pipe	Helical coil type low-fin tube
	Outside tube fluid (inlet/outlet temp.)	Primary helium gas (850°C / 795°C)
	Inside tube fluid (inlet/outlet temp.)	Superheat steam (338°C / 538°C)
	Exchange calory	175 MWt
6.	Steam Turbine	
	Type	Reheat extraction condensatre turbine
	Power	536.6 MWe
	Steam condition	170 ata, 538°C / 40.6 ata, 538°C

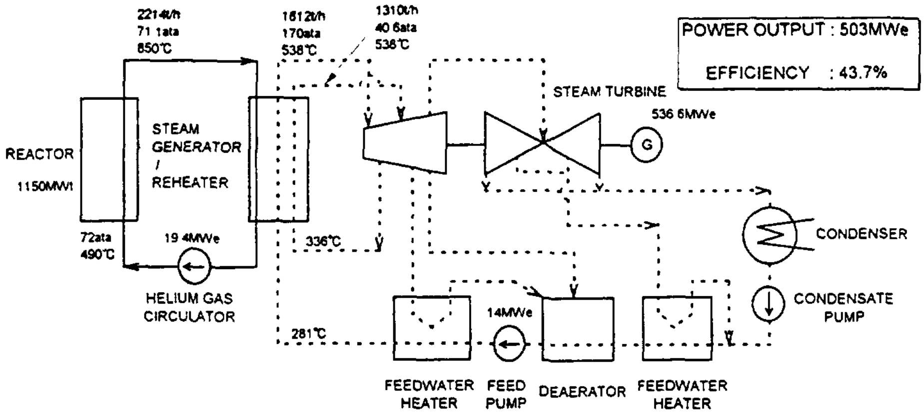

Fig.4. Flow Diagram of the Large Sized HTGR

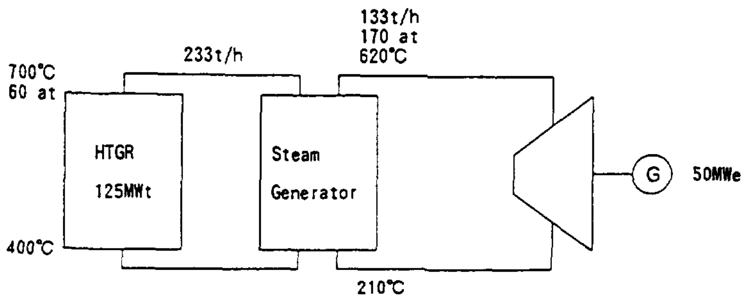

Fig.5. Flow Diagram of the Small Sized HTGR

Therefore, this type of HTGR is expected to have large marketability in the Asian area. [6] We sketched a plant concept to investigate its economics. Fig-5 shows a flow diagram.

3.3 Comparison of economic characteristics

We estimated the unit power cost for three HTGR plants having different outlet powers. Table. Ⅱ shows the results of the relative cost comparison of these plants. These results indicate that the large-sized HTGR plant has an attractive cost potential and is cost competitive with LWRs: and that the small-sized HTGR plant is expensive, but its cost is only 15% higher than that of the Modular HTGR (200MWe).

Table. Ⅱ. Cost Comparison

	Small Sized HTGR	Modular HTGR	Large Sized HTGR
(MWe) (MWt)	50 MWe 125 MWt	200 MWe 500 MWt	500 MWe 1150 MWt
Total Capital Cost	30	100	210
Unit Cost	130	100	90
Busbar Cost	116	100	91

4. Conceptual design study of nuclear plant application systems

In the commercialization scenario described in a Chapter 1, it is recognized that hydrogen is expected to be the ultimate energy source for human society. We have to develop the diverse technologies required to meet needs in far future, especially through the application of nuclear heat from the HTGR. This chapter summarizes the results of studies on the technologies of the gas turbine, coal gasification and methanol production as such application systems.

The goals of these studies were as follows.

(1) Investigation of the characteristics of the basic system

(2) Improvement of the particular characteristics of the HTGR

(3) Optimization of the nuclear heat application system

(4) Creation of an innovative concept for this system

4.1 Study of the HTGR power plant [2]

As the electric power system, we considered three different methods of generation: a gas turbine, a steam turbine and a combined system.

In this study, we surveyed plant efficiency and the principal design parameters based on the following conditions.

Reactor power: 450MWt

Outlet temperature: 850 ℃

For the steam turbine system, the maximum plant efficiency was about 42%.

For the gas turbine system, we obtained a maximum plant efficiency of 44% under the design parameters listed in TableⅢ. Fig-6 shows how the plant efficiencies change in relation to the pressure ratio between inlet pressure into turbine and outlet pressure from turbine. The combined cycle power plant was expected to produce the maximum efficiency among three systems. In this study, we optimized the power distribution between gas turbine and steam turbine, and studied the maximum plant efficiency of about 46%. Fig-7 shows the flow diagram of this combined cycle power plant.

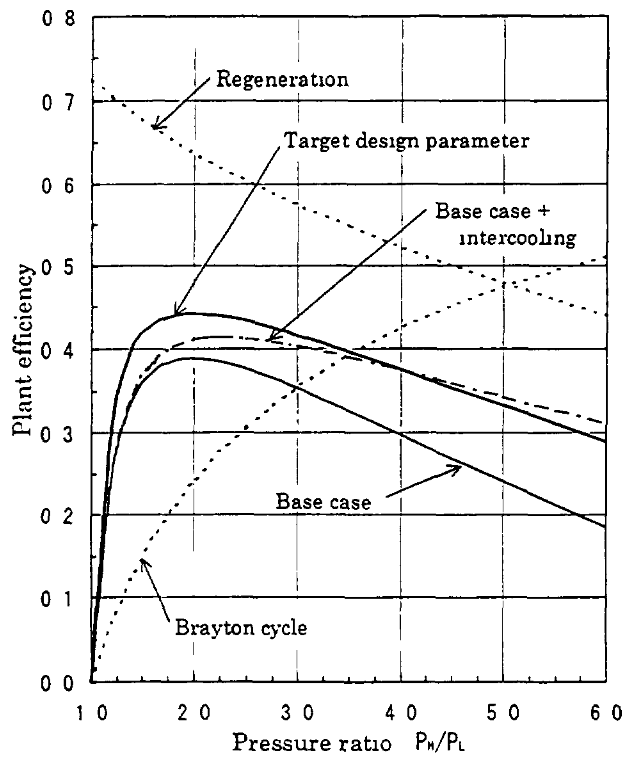

Fig.6. Plant Efficiency vs. Pressure Ratio

Table. III. Optimized Design Parameters of Gas Turbine

turbine adiabatic efficiency	92%
compressor adiabatic efficiency	89%
recuperator effectiveness	95%
generator efficiency	98%
pressure drop raio	6%

The main results of this study were summarized as followings.

(1) As a nuclear power plant system attached with the HTGR, we studied basic characteristics from the view point of improvement of the plant efficiency.

(2) The maximum plant efficiency condition and design parameters were investigated.

(3) The plant efficiency will be evaluated from 42% to 46% among three power plant and the combined cycle power plant has a highest potential to improve the plant efficiency.

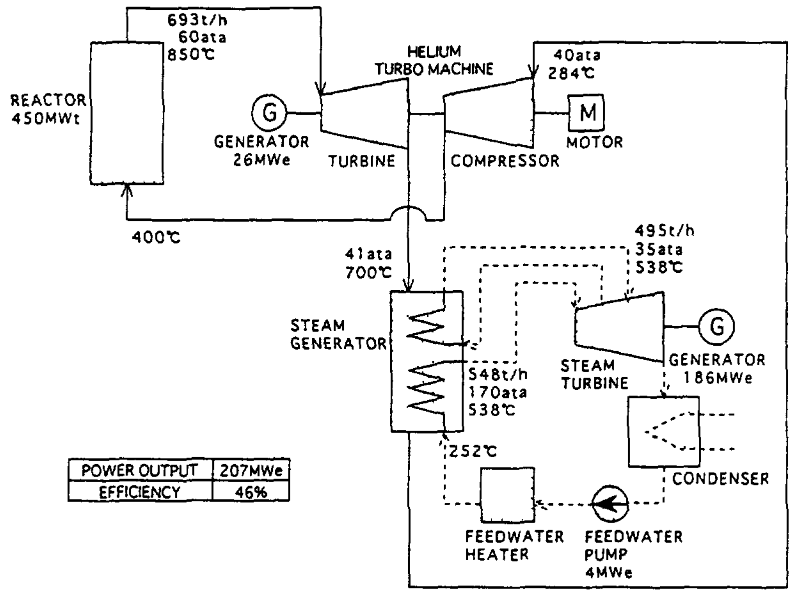

Fig.7. Flow Diagram of the Combined Cycle HTGR

4.2 Study on nuclear power cogeneration system using gas turbine system [3]

One of the most attractive point about the HTGR is considered the heat source from nuclear energy has a wide temperature range to be used compared with the LWRs.

This means the HTGR has a high potential to widen an overall thermal efficiency.

From this view point, we evaluated an overall thermal efficiency with changing a temperature levels produced with a gas turbine as the heat application system.

First, we selected four typical HTGR gas turbine systems. These were a direct and regenerative cycle; a direct, regenerative and inter cooling cycle; an indirect, regenerative and inter-cooling cycle; and an indirect, rehearing, regenerative and inter-cooling cycle. The flow diagram for each one is shown in Fig-8.

Other design parameters were set in common, as listed in Table Ⅳ.

In this study, we first sketched HTGR cogeneration systems in which a different temperature heat exchanger was equipped for each gas turbine system. We investigated in detail, the overall thermal efficiency by changing the inlet temperature of the heat exchanger from 800 ℃ to 100℃ in 100℃ steps. Fig-9 shows the typical flow diagram of HTGR cogeneration system for the direct and regenerative gas turbine system. Fig-10 shows the results of the overall efficiency for each gas turbine HTGR, with cogeneration systems adopted to the differing temperature ranges.

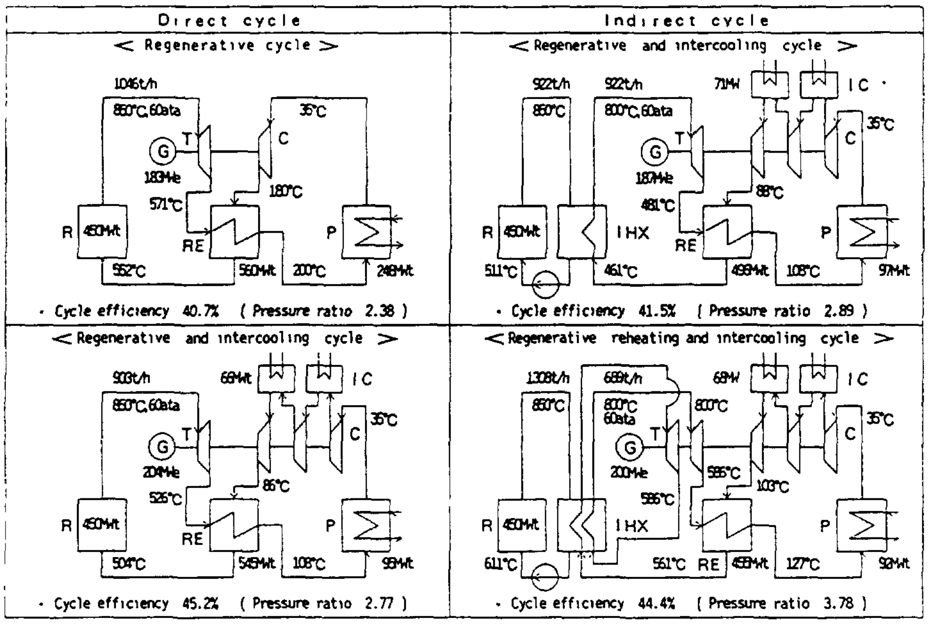

Fig.8. Heat and Mass Balance Flow of 4 Types Cogeneration System

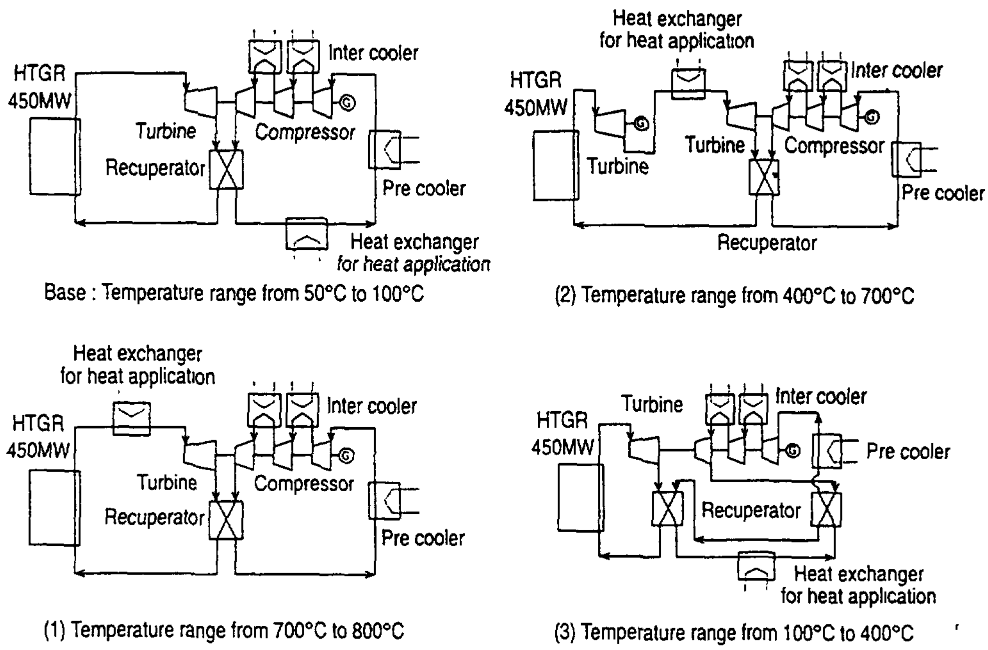

Fig.9. HTGR Cogeneration System

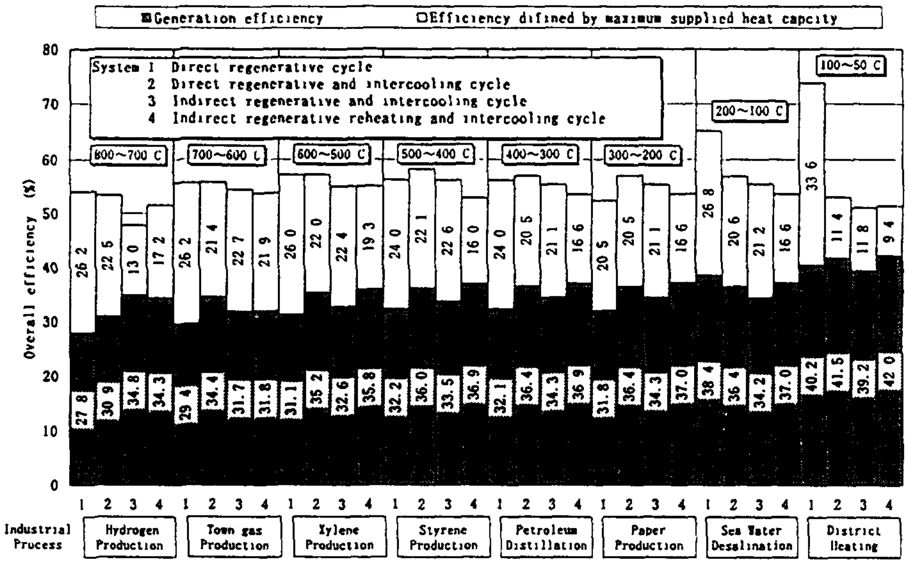

Fig.10. Overall Efficiency due to Heat Application Temperature

Table.IV. Basic Parameters of Cogeneration HTGR

Reacter power	:	450 MWt
Cove outlet temp.	:	850°C
Compressor inlet temp.	:	35°C
Turbine inlet pressure	:	60 ata
	:	92%
Compressor efficiency	:	88%
Mechanical loss	:	2.5%
Heat Loss	:	1.0%
Heat exchanger efficiency	:	95%
Generation efficiency	:	98%

From these results, we concluded the following.

The maximum overall efficiency is over 70% for the direct gas-turbine as the cogeneration system in the temperature range below 100℃.

4.3 Study on a methanol production system [4]

From the perspective of wider application of nuclear heat from the HTGR plant and conservation of fossil fuels, it is recognized that technology for reforming coal resources to a clean new energy source is essential in the future. Therefore we examined the concept of a methanol production plant and its main characteristics. Methanol is expected to provide an alternative to petroleum because of its high performance in limiting emission of CO_2.

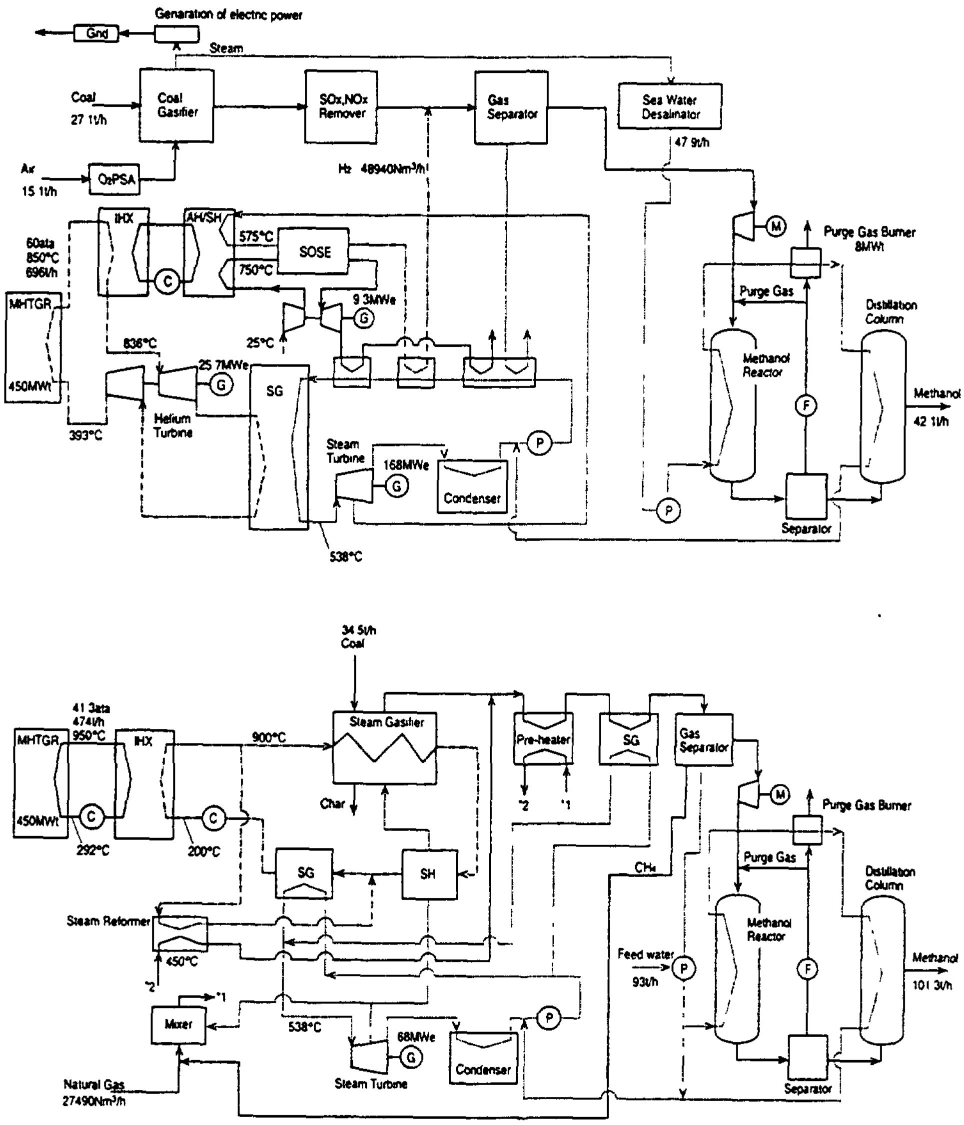

Fig. 11. Flow Diagram of Methanol Production System

Table. V. Evaluation of Design Characteristics

	System (1) Partial oxidation + SOSE	System (2) Steam gasification + Steam reforming
Economical aspect	A	B
(Methanol cost)	¥54 3/kg in 2025	¥60 8/kg in 2025
Environmental aspect	A	B
(CO_2 emission rate)	1 37kg CO_2/kg Methanol	1 42kg CO_2/kg Methanol
Utilization factor of energy	B	B
(Methanol conversion ratio)	135%	115%
(Utilization factor of nuclear heat)	17%	28%
(Utilization factor of source carbon)	95%	92%
(Utilization factor of total energy)	41%	67%
(Utilization factor of waste heat)	4%	16%
Operation flexibility	A this system can supply electricity by installing large hydrogen tank	C Capacity of electricity production is small
Relative difficulty of development	B Under the development by pilot plant	C Development of fluidized-bed reactor for steam gasification of coal is difficult
Waste heat utilization	B This system can be coupled to another process	B This system can be coupled to another process

A : Excellent   B : Good   C : Not recommend

In this study, two promising methanol production systems were proposed and their features compared. First, we reviewed the methodology of coal gasification and hydrogen production and compared several candidates. Fig-11 shows the flow diagram of above two systems in combination with the HTGR. One candidate is composed of a CH_4 reformer and the steam gasification of coal and the other is composed of an SOSE and partial oxidation coal gasification. Table. V shows the results of comparing the two systems from several points view. We also investigated methanol production systems, their characteristics and the technical solution they required.

4.4 A new electric energy storage plant using an HTGR as the heat source for methanol production [5]

This total plant concept consists of an HTGR, chemical reactors and power systems. Fig-12 shows its schematic flow diagram. The HTGR supplies high temperature heat to the SOSE, where hydrogen is produced from the steam and electric power at night. Hydrogen and coal introduced from a coal gasification system are combined in a chemical reactor to produce methanol. After the electric power is converted to methanol, the methanol is used as the fuel for the power plant. This can contribute to supplying additional electric power for to the peak load during the day.

The items studied are as follows.

1) Plant systems and component design
2) HTGR-SOSE operability, including procedures for switching between day and night operation
3) Plant layout of the systems and main components
4) Product cost evaluation

The results of this conceptual design study are as follows.

1) The total plant system was designed with the capacities listed below.
   HTGR plant rating: 450MWt
   Methanol production plant: 393tons/h
   Methanol storage capacity: 83500 ㎥
   Methanol-fired power plants: 2650MWt

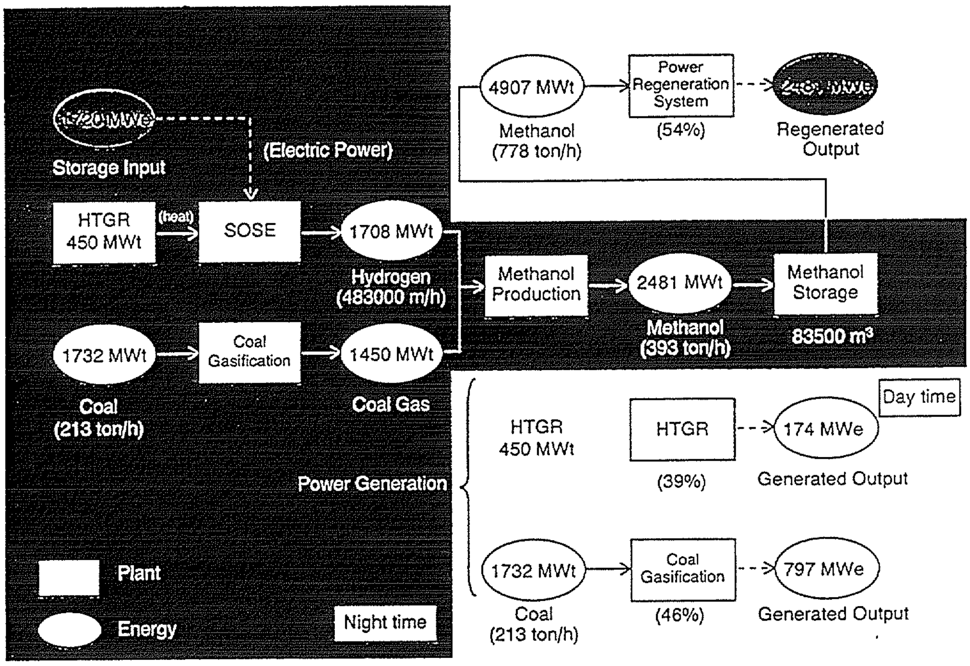

Fig.12. HTGR SOSE Electric Energy Storage System

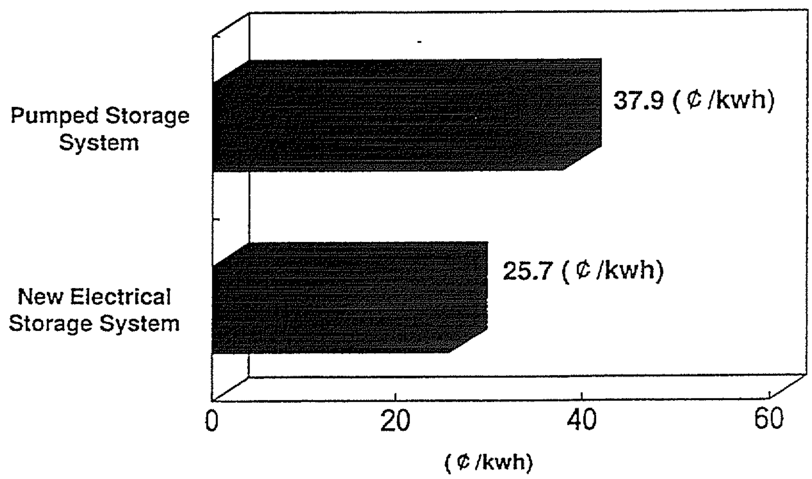

Fig.13. Comparison of Products Cost

2) HTGR plant operation was determined as follows.

Nighttime HTGR plant operation: 10 hours for H_2 production

Daytime HTGR plant operation: 9 hours for electric generation

3) The results of cost evaluation estimates that the cost of electric regeneration is approximately 20% less than that of a pumped storage system. It even has a cost advantage in hydrogen production at present marker prices due to its higher purity. Fig-12 shows a comparison of regeneration cost estimation for the regeneration system in this study and a pumped storage system.

In conclusion, a new type of electricity storage system that adapts HTGR was developed, and its feasibility and cost advantages were clarified.

5. Conclusion

In this paper, we introduced part of the results of our studies on the HTGR, which have been conducted jointly with the Japan Atomic Energy Institute, Tokyo Electric Company Ltd., and General Atomic.

We are now studying HTGR development with a long-term view toward its commercialization. The greatest concern we are now facing is establishing a framework for the development plan of a post HTTR as a Japanese national project. We must make our best efforts to realize such a plan because we believe HTGR is one of the most promising ways to solve the energy problem while protecting the worldwide environment.

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GAS COOLED REACTOR SAFETY AND MANAGEMENT

XA9846673

HTR-10 SEVERE ACCIDENT MANAGEMENT

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Abstract

The High Temperature Gas-cooled Reactor (HTR-10) is under construction at the Institute of Nuclear Energy Technology site northwest of Beijing. This 10 MW thermal plant utilizes a pebble bed high temperature gas cooled reactor for a large range of applications such as electricity generation, steam and district heat generation, gas turbine and steam turbine combined cycle and process heat for methane reforming. The HTR-10 is the first high temperature gas cooled reactor to be licensed in China. This paper describes the safety characteristics and design criteria for the HTR-10 as well as the accident management and analysis required for the licensing process.

1. INTRODUCTION

The 10MW High Temperature Gas-cooled Reactor (termed HTR-10) is a pebble bed type high temperature gas-cooled reactor with 10MW thermal power output for a wide range of possible application for e.g. electricity, steam and district heat generation, gas turbine and steam turbine combined cycle as well as process heat generation for methane reforming. It is located on the site of Institute of Nuclear Energy Technology of Tsinghua University (INET) in the northwest suburb of Beijing, about 40km away from the city. On the area of INET, there already exist a swimming pool experimental reactor and a 5MW nuclear heating reactor. The HTR-10 is under construction and it is expected to reach critical in 1999.

The main basic design features of the HTR-10 are derived from the HTR-Module, Germany. Due to the relatively small size and corresponding small power output of the HTR-10 it has own design features from safety point of view. In summarize it is much safer than the HTR-Module and therefore the auxiliary systems are more simple.

In this paper the HTR-10 safety design criteria and safety characteristics were presented. The accident analysis were described and accident management as well as corresponding measures were introduced.

2. SAFETY DESIGN CRITERIA

The following safety design criteria for the HTR-10 were set up, taking into account the inherent safety characteristics of a HTR-Modular and design requirements as an experimental reactor.

- The reactivity temperature coefficient shall be negative at any operation condition.

- Coated fuel particles (CPs) shall not fail during normal operation and anticipated operational occurrences. That is, at any accident event the maximum fuel temperature shall not exceed 1600 °C .
- The means for shutting down the reactor shall consist of two diverse systems. Main system shall perform its function assuming a single failure and shall be, on its own, capable of quickly rendering the reactor subcritical by an adequate margin from operation and accident condition. Meanwhile reserved shutdown system shall perform its function without assumption of a single failure and shall be, on its own, capable of quickly rendering the reactor subcritical by an adequate margin from operation and accident condition.
- A severe accident resulting from control rods ejection must be avoided at any condition.
- The lines connected to the reactor coolant pressure boundary shall be fitted with the isolation valve in principle. It shall be designed to ensure that the maximum rupture area at any line should not be larger than pipe area with equivalent diameter of 65 mm.
- The pressure resisting and heat resisting functions of the structure, where high pressure and high temperature coolant is contained, are separated to reduce mechanical loads on high temperature metal structure.
- The passive residual heat remove system shall be to transfer fission production decay heat and other residual heat from the reactor core at a rate such that prescribed fuel design limits and design condition of the pressure vessel are not exceeded.

3. DESIGN FEATURES

Technical design of the HTR-10 represents the features of HTR-Module HTGR design. The cross section of the HTR-10 primary circuit is shown in Fig.1.

The helium at the temperature of 250 °C enters the main circulator and is pressured, then flows into the outer coaxial pipe of the hot gas duct and enters borings in the side reflector, and flows through these from the bottom to the top. The cold gas directly enters reactor core and is heated up to a temperature of 700 °C while flowing through the pebble bed from the top to the bottom. An effective mixing of the hot helium gas is attained by the geometrical design of the bottom reflector in order to make uniform temperature distribution. Then helium flows through the center tube of the hot gas duct and enters steam generator. The heat is transferred to water in secondary circuit while the helium temperature is cooled down to 250 °C. The HTR-10 main design parameters are listed in Table 1.

The important technical features of the HTR-10 are as follow:

- Using spherical fuel elements, which are formed with coated particles.
- Reactor core design ensures that the maxim fuel element temperature of 1600 °C cannot be exceeded in any accident
- The reactor core and the steam generator are housed in two separate steel pressure vessels. They are connected by the hot gas duct pressure vessel and arranged side by side.

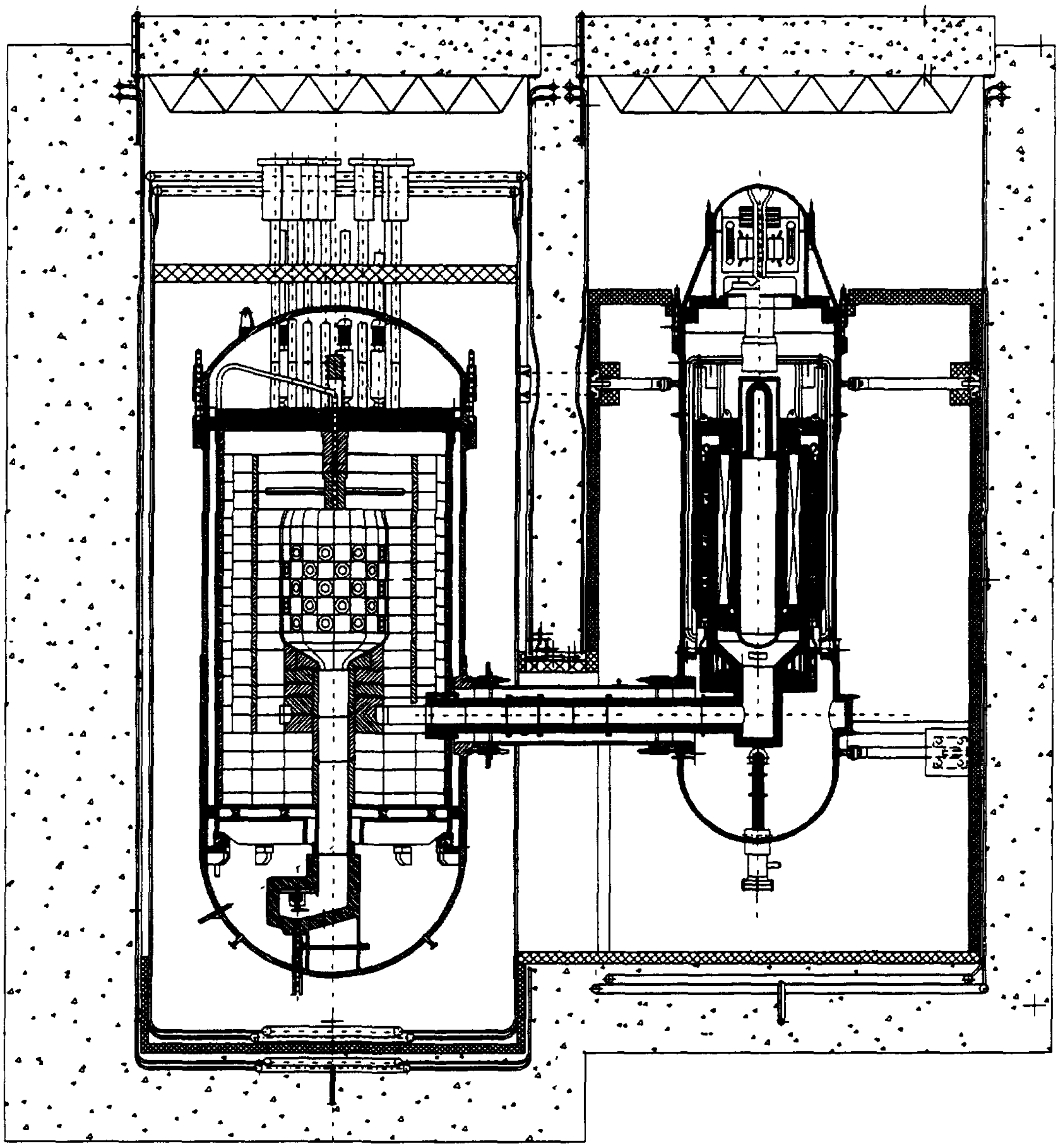

FIG 1. The Cross Section of the HTR-10 Primary Circuit

Table 1 HTR-10 Main Design Parameters

Reactor thermal power	MW	10
Active core volume	m^3	5
Average power density	MW/m^3	2
Primary helium pressure	MPa	3
Helium inlet temperature	°C	250 / 300
Helium outlet temperature	°C	700 / 900
Helium mass flow rate	kg/s	4.3 / 3.2
Fuel		UO_2
U-235 enrichment of fresh fuel elements	%	17
Diameter of spherical fuel elements	mm	60
Number of spherical fuel elements		27,000
Refueling mode		multi-pass, continuously
Average discharge burnup	MWd/t	80,000

- Active core cooling system is not required for residual heat removal in case of accident. Residual heat can be dissipated by means of passive heat transfer mechanism to surrounding atmosphere.
- The reactor core is entirely constructed by graphite materials, no metallic component are used in the region of the core.
- The two reactor shut down systems, i.e. the ten control rods and seven small absorber ball systems, are all positioned in the side reflector. They can drop into borings by means of gravity. In-core control rods are not needed.
- Spherical fuel elements go through the reactor core in a "multi-pass" pattern. Thus all fuel elements attain a relatively uniform burn-up distribution in the core. Fuel pebbles are continuously discharged via a pneumatic pulse single-exit gate (or better called "reducer") which is placed inside the reactor pressure vessel
- The integrated steam generator and intermediate heat exchanger (IHX) are designed. The steam generator is once through, modular small helical tube type. The helium circulator is installed in the steam generator pressure vessel and positioned above the steam generator and the IHX.
- A ventilated primary cavity is designed as a confinement to restrict the radioactivity release into the environment, it has not the function of gas-tight and pressure-retaining containment.
- The digital reactor protection system is used in the HTR-10.
- Use of domestic standard turbine-generator unit in secondary circuit to provide electrical power.

4. SAFETY CONSIDERATION

The basic safety features of the HTR-10 include the multi-barriers against release of radioactivity and the reactor inherent safety.

4.1. Multi-barriers against release of radioactivity

- Integrity of fuel particle coatings
  The HTR-10 uses spherical fuel elements which are formed from many small TRISO-coated fuel particles. The fuel particles are coated with two high - density layers of pyrocarbon and one layer of silicon carbide. The silicon carbide layer is very dense so that no significant quantities of radioactive gaseous or metallic fission products can be released from intact fuel particles even if their temperature is up to 1600 °C, but the maximum fuel temperature is far below 1600 °C under all normal operating and accident conditions.
- Integrity of the primary gas envelope
  The primary gas pressure envelope is the second barrier to the release of radioactive substances. The primary gas envelope of the HTR-10 is composed of reactor pressure vessel, steam generator pressure vessel and hot gas duct pressure vessel. These pressure vessels will be cooled by cold helium flow and kept in low operating temperature. The components of the pressure vessel unit are designed in such a way that the possibility of through-wall cracks can be ruled out.

- Confinement function of primary cavity:
  The reactor pressure vessel, the steam generator pressure vessel and the hot gas duct pressure vessel are positioned inside primary cavity. When the postulated event of unisolatable piping break happens, the radioactivity substances in the primary coolant and portion of the activity deposited on the surface of the primary system are released into the primary closed cavity. Afterwards, primary gas which carries radioactivity will be exhausted into the environment pass-through the vent stack. Under normal operating condition, the sub-atmospheric pressure of primary cavity can prevent leakage of radioactive gas into the reactor building.

4.2. Passive removal of reactor residual heat
Residual heat after reactor shutdown can be removed from the core to surrounding graphite structures and components and up to residual heat removal system in the primary cavity through convection and radiation. There are no any active components in this system. Heat can be dissipated through the air coolers to final heat sink-atmosphere.

4.3. Negative reactivity temperature coefficient
The HTR-10 has negative reactivity temperature coefficient for both fuel and moderator of about 8.8×10^{-5}/K and during normal operation the fuel temperature is 700 K far from allowable maximum temperature. Therefore about 5×10^{-2} reactivity compensation can be provided by negative reactivity temperature coefficient. It can offset reactivity increase at accidents of water ingress and control rods withdrawal.

5. ACCIDENT CLASSIFICATION CRITERIA

The spectrum of accident events are identified in accordance with their best-estimate frequency of occurrence and its consequence. In the HTR-10 they are divided into five categories: normal operation states (category 1), anticipated operational occurrences (or middle frequency accidents- category 2), rare accidents (category 3), limited accidents (category 4) and severe accidents (category 5).

At rare accidents and limited accidents the reactor should be maintained at subcritical condition even if the reactor could not operated in short period as well as the capacity of residual heat remove should be kept and maximum fuel temperature should not exceed 1200 °C. At rare accidents the exposure level for any individual outside the HTR-10 siting due to release of radioactivity shall be less than 1 mSv for whole body dose and 10 mSv for thyroid dose. At limited accident the exposure level for any individual outside the HTR-10 siting due to release of radioactivity shall be less than 5 mSv for whole body dose and 50 mSv for thyroid dose.

At serious accidents the exposure level for any individual outside the HTR-10 siting due to release of radioactivity shall be the same as one at limited accidents. The difference is that at serious accidents reactor mechanical integrity would not taken into consideration and only the exposure level for any individual outside the HTR-10 siting due to release of radioactivity are estimated.

6. DESIGN BASIS ACCIDENTS

The design basis accidents (DBA) are classified into several groups. There are: increase of heat removal capacity in primary circuit, decrease of heat removal capacity in primary circuit, abnormality of reactivity and power distribution, rupture of primary pressure boundary tubes and anticipated transients without scram (ATWS). The analysis results for above mentioned accidents are shown as follow:

At the accident of loss of the heat sink the maximum fuel temperature is about 868 °C and the primary pressure change is very small. Therefore it is no need to use additional measure except stopping the helium blower and inserting control rods by the means of the reactor protection initiated by signals of the ratio of primary flow to secondary flow, the helium inlet temperature raise and the rate of neutron flux decrease.

During the accident of abnormal reactivity and power distribution it is assumed that the maximum reactivity increase with one control rod withdrawal is 1.83% at the rate of 1 cm/s within the moving internal of 2.2m. The maximum power rate would be 2.7 times higher than normal power rate and the maximum fuel temperature is 925 °C. It is still no need to take additional measure.

At the accident of the rupture of primary boundary tube the tube rupture in the fuel handling system with diameter of 65mm is the most severe accident due to appropriate design. With the signals of the ratio of primary flow to secondary flow and primary pressure decrease rate the reactor should be shutdown and the helium blower should be stopped. However due to rapid release of the helium the reactor cavity pressure would be raised up 0.11MPa within few seconds and the helium would be released into the surrounding environment. Low radioactivity of the helium at normal operation will not much effect the radioactivity level at the siting surrounding. In about three minutes the primary pressure is equal to the reactor cavity pressure and the decay heat is mainly dispatched by heat conduction and radiation as the natural convection effect is very weak. Although the generated heat is higher than the heat removed capacity at first few hours, the core temperature raise is very small due to large heat capacity, that is the maximum fuel temperature is about 874 °C . It is much lower than safety limited temperature and the fuel failure would not be occurred. Meanwhile it is assumed the natural convection flow would be established and total air in the reactor cavity would be reacted with the core graphite. In this case the total amount of graphite corrosion is only 4.3 kg and the maximum corrosion rate of the spherical ball surface is about 323 mol/m^2. The coated particles are not exposed to air atmosphere and can keep their integrity.

7. BEYOND DESIGN BASIS ACCIDENTS

Beyond design basis accidents mainly include: long-term failure of the reactor cavity cooling system, simultaneous withdrawal of all control rods at power operation and at reactor start-up, failure of the helium blower shut-down, simultaneous rupture of all steam generator tubes and rupture of the hot gas duct vessel.

For the accident analysis of failure of the helium blower shut-down it is assumed that the helium blower could not be stopped due to some reason so that primary pressure and the temperature of components in the primary circle would be raised. The primary pressure would be increased to 3.5 MPa in 76 seconds and the helium would be released to the cavity, and then to surrounding environment via the filter. It has no significant influence on the surrounding. The temperature of the helium blower would be increased to about 490 °C. As there is no heavy load on the blower material it will not influence on the reactor safety. The temperature of the steam generator pressure vessel, the reactor pressure vessel and lower part of metallic internal would be raised to about 272 °C , 386 °C and 298 °C, respectively. They are lower than the material allowance operating temperature.

At the accident of simultaneous rupture of all steam generator tubes the reactor protection system is put into operation by the signal of high humidity. The initial flow rates in the broken hot and cold legs are 3.4 kg/s and 3.47 kg/s respectively. Total 102 kg water is pored into the primary circle. The oxidization corrosion rate and pressure increase is very small and can be negligible.

The accident of rupture of the hot gas duct pressure vessel is the most severe accident. The reasonable and conservative assumptions are made. That is the integrity of the reactor core is not taken into consideration at the accident analysis and the accident duration is only three days as the management measures are taken to terminate continuous air ingress into the reactor core within three days. Under above mentioned assumption at the accident of rupture of the cross duct vessel the helium would be released to the surrounding without flowing through the filter at beginning of the accident. Due to low radioactivity at normal operation it will no significant influence on the surrounding. Then the air will continually ingress into the rector core and react with the graphite. The total graphite corrosion amount is about 340kg. The graphite matrix corrosion amount is about 35% of the total graphite corrosion. Therefore about 0.75% of the coated particles are exposed and small amount radioactive material would be released to the surrounding. The exposure level for any individual outside the HTR-10 siting due to release of radioactivity is 0.36mSv for whole body dose and 1.75mSv for thyroid dose. They are much lower than specified dose limit. The fuel elements temperature is about 890 °C.

At the accident of ATWS initiated by depressurization and long-term failure of the reactor cavity cooling system the reactor can be automatically shut down by the means of the negative reactivity temperature coefficient. The temperature of fuel elements and the reactor pressure vessel are only 898 °C and 250 °C, respectively.

8. CONCLUSION

Due to the HTR-10' inherent safety features except the severe accident no management measures have to be taken. It is only in the case of hot gas duct pressure vessel failure that management measures are needed to terminate continuous air ingress into the reactor core.

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AN INTRODUCTION TO OUR ACTIVITIES SUPPORTING HTGR DEVELOPMENTS IN JAPAN

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Abstract

On the view point the most important for the HTGR development promotion now in Japan is to have people know about HTGR, the Research Association of HTGR Plants(RAHP) has paid the best efforts for making an appealing report for the past two years .

The outline of the report is described with an introduction of some basic experiments done on the passive decay heat removal as one of the activities carried out in a memmber of the association.

1. Introduction

RAHP, the only private organization studying the HTGR developments in Japan, consisting of universities, utilities and venders has been widely investigating and watching the HTGR technologies as well as the commercialization aspects in the world since 1985. THe association was reorganized recently in the framework of the comittees under the Institute of Applied Energy (IAE of Japan), but has kept same activities on the HTGR as before.

On the understsnding of the technical advantages and the importance of introducing the HTGR into the energy world through the studies, and from the view point that the most important at this stage now in Japan is to have people know the HTGR, the association has paid best efforts for the past two years for making an appealing report, especially expressed as concisely and plainly as possible for the HTGR technologies and the meanings of its introduction.

The outline and intention to be used of the report is described below , with an introduction of some basic experiments on the passive decay heat removal which is one of the key issues of the system done in Tokai University, as an example of activities of a memmber of the Association.

2. Report appealig HTGR support

2.1 Intention of the paper

The paper is made up under the subjects of ;

0 What is HTGR ?

0 Why HTGR now ?

0 How to be developed ?

In the opening page the intention of the paper is expressed. The paper is aimed for the support of the people by clearly showing our own understandings through the studies on the HTGR not only for the people in the fields concerned but also in the fields of the executives of beaurocracies, industries and universities and also for the young people who will be responsible in the next generation.

The report consists of three chapters. In the 1st chapter are explained the terms used for the HTGR and the various concepts of the HTGRs such as fuel and core or the power supply plants and heat supply plants, etc. In the 2nd chapter, the main part of the paper , is described the outline of the developments of HTGR in which the merits and the attractive featurs of the HTGR are stressed. In the 3rd chapter are described the summary and proposals.

2.2 The outline of the HTGR developments

In the 2nd chapter, the main part of the this document, the outline of the HTGR developments is described under the subtitles of ;

0 Why HTGR now ?

the reason, backgroud, stand points, direction or needs

0 What kinds of applications possible ?

0 The sizes of the plants considered.

0 What siting possible ?

0 What schedules or milestones for the developments of the HTGRs desireble ?

0 The technologies status obtained so far.

The present technology levels on the HTGRs obtained so far both in Japan and foreign countries for the further developments of the HTGR in applications.

0 What problems or subjects to be overcome ?

0 The economics of the HTGRs considered and possible ?

0 The circumstances and situations in the HTGR developments.

AS for the first theme, "Why HTGR now ?" , the paper appeals the reason why, stressing that ;

(1) HTGR can utilize the nuclear energy in much larger scopes and at much higher efficiencies. :

0 Wider utilizations of nuclear energy more than electricity.

The HTGR can expand the nuclear energy use to the other energy regions, e.g., to the transportation energy by producing hydrogen or methanole.

0 Power Generation at much higer efficiency;

o Steam turbine generation 40 %

o Gas turbine generation 50 %

o Possibility of heat utilization at much higher efficiency by cascade using

(2) HTGR can make the urban siting possible near to the place of the demands by its inherent safety characteristics of the modular HTGRs. :

0 The basic features of the modular HTGRs;

o The specific features of the coated particle fuel

FP release free even to 1600℃

Melt down free

o Helium gas cooling

No phase change

No reactibity change in consequence

o Large graphite (ceramic) core

Very slow transients in accidents

(by the huge heat capacity of the core)

No core meltdown

0 Highly achieved inherent safety features in the accidents;

o Reacter shutdown features

Not actively shutting down, but passively stopping itself

(by big negative temp. coeff. on the very large temp. rise allowance in the core)

- o Decay heat removal features;

  Not actively cooling, but passively getting cold itself

  The core temperture limitted below 1600℃ at any cases
- o Confinment of FPs;

  Through the above two inherent charactristics built in the core, the FPs are all confined in the coated particle fuels in the worst, highly hypothetical accidents.

As for the 2nd theme, **"What kinds of applications possible ?"** , the scopes to be expected or desired are illustrated as shown in the figure 1.

Besides the above, in the rest of the articles are explained the unique features of the HTGR such as ;

- o The flexibility in plant constructions owing to the multi-module plant, selecting to construct the modules by turns.
- o And thus reducing the capital burden and avoiding the capital risks.
- o Imposing less burden to the operators with little possibility to invite vital accidents by mis-handling.
- o Actual and proven technologies through the long experiences of constructions and operations to the scale as large as the demonstration reactor plants.
- o Efficient utilization of the nuclear fuel by higher burn up
- o Flexibility of the fuel cycle so as to enable to use uranium, plutonium and thorium in the same reactor core.
- o Unique capability of coated particle fuels for the efficient plutonium burning.

In the 3rd chapter, the paper concludes as the words

- o that the HTGR can greately contribute to the energy and enbironmental problems on the earth with its capability of the inherent safety, the unique ability in the thermal use of the nuclear energy by its high temperature and the flexibility of the fuel cycle, etc.,
- o and that the HTGR is of actual technologies with the long experiences in constructions and operations even to the scale of the demonstration reactor plants,

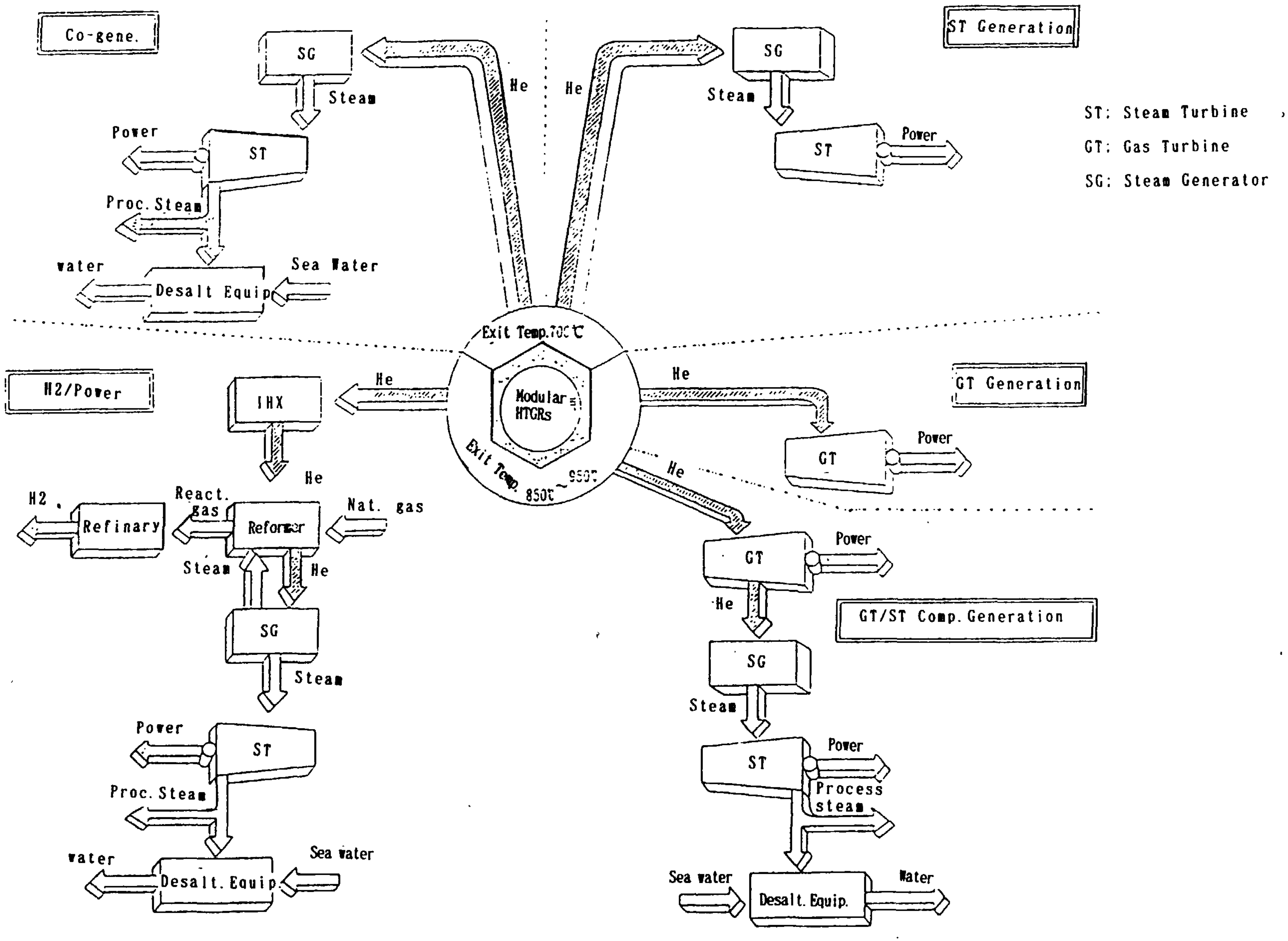


Fig. 1 What kinds of applications ?

- o and that it keeps the higher possibilities to itself by using together with gas turbine systems,
- o therefore, promoting the HTGR technology developments, making up the actual scenario to the stage of the practical use , incorporating the "HTGR development" into the national energy strategies and the actually constructiong a demonstration plants are desirable.

3. Basic experiments on the passive decay heat removal

3.1 The background

In Tokai Universiity, we have carried out basic experiments on the passive decay heat removal, one of the key issues of the modular HTGR, with subsidies from three member companies of RAHP, in the form of joint study or consigned study.

We have two themes on the experimental studies, one of which is the investigation of the air flow behaviors on the natural air draft strengthened by the the chimny effects and another is the studies on the possibilities of the attractive applications of the separated heat pipe systems both for the passive decay heat removal on the reactors.

These studies have been done together with the objects for the graduate studies of the graduate students. Therefore the studies are very basic and done by the small handmade devices mainly made by the students themselves, though, very interesting findings have been obtained so far. These results are thought to be suggesitive on desiging the actual systems or on planning larger scale experiments for the actual systems.

We would like to invite the discussions on them by showing the experimental resuls of these studies as below.

3.2 The abnormal flow phenomenon taking place on the natural air draft heat removal system experiments

Recently there has been a strong tendancy for advanced reactors to rely on the passive heat removal by means of natural air draft strengthened by chimny effects for the ultimate purpose of rejecting decay heat to the environment seen in the modular HTGR(MHTGR), modular LMR(PRISM) and advanced LWR(AP-600).

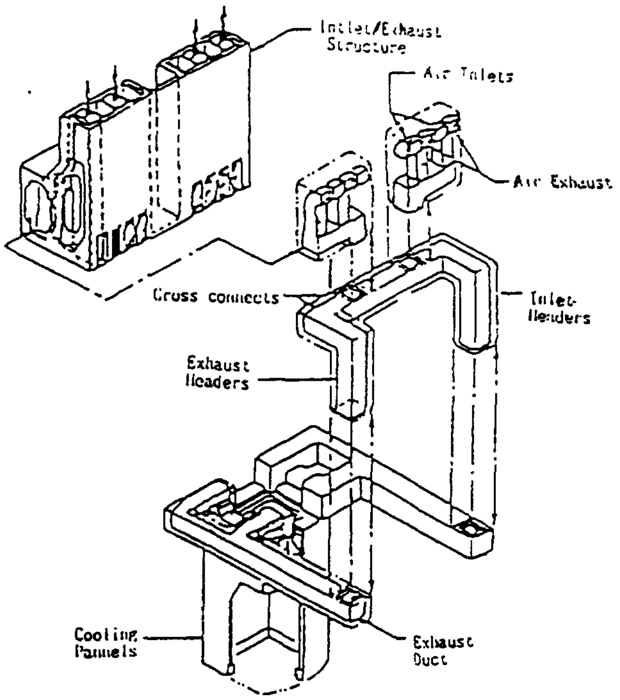


Fig.2 RCCS of MHTGR

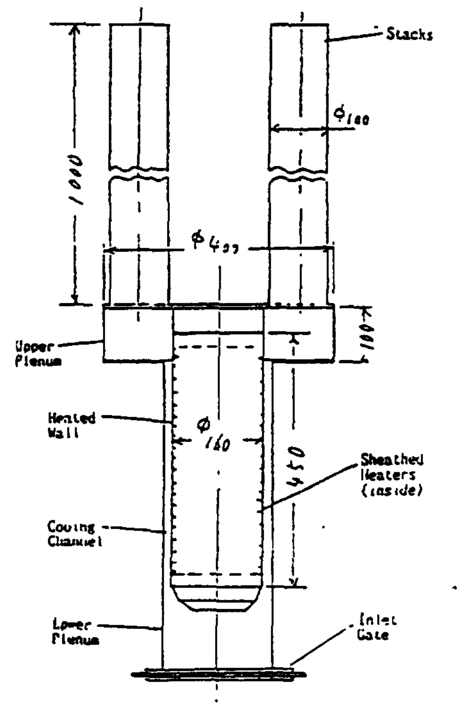


Fig.3 Experimental apparatus

Basic experiments was intended with intention of identifying the flow behaviors in the various wind conditions including typhoon and accidents such as partial blockage of the air flow pathes, etc. Abnormal flow occurence was taken place during one of the experiments.

Experimental apparatas was made with focus on the Reactor Cavity Cooling System (RCCS) of the MHTGR, the conceptial design of which is shown in Fig.2. Decay heat is transfered to the cooling pannel provided arroud the uninsulated reactor pressure vessel and then transported and rejected to atomosphere by buoyancy-driven natural circulation of outside air through the cooling pannel, plenum, stacks and finally to the ultimate heat sink of the environment. For the actual design, recutangular outlet ducts are routed inside the inlet ducts as shown in the fig. 3. The outlet ducts are insulated to reduce regenerative heating of the inlet air. Multiple stacks must be provided redundantly as the safety components by a redundancy principle in nuclear plants.

A cupper pipe 160mm in diameter and 450mm long lined with the seathed heater inside was used as the simulated heated wall. A side wall to keep air gap arround the heated wall was connected with upper plenumn upon which two pipes 100mm in diameter and 1000mm high were set, thus making a mock test device to simulate the flow functions as shown in Fig.4.

Photo. 1 Normal flow
(Both stacks discharging)

Photo. 2 Abnormal flow
(One stack discharging
／another stack inhaling)

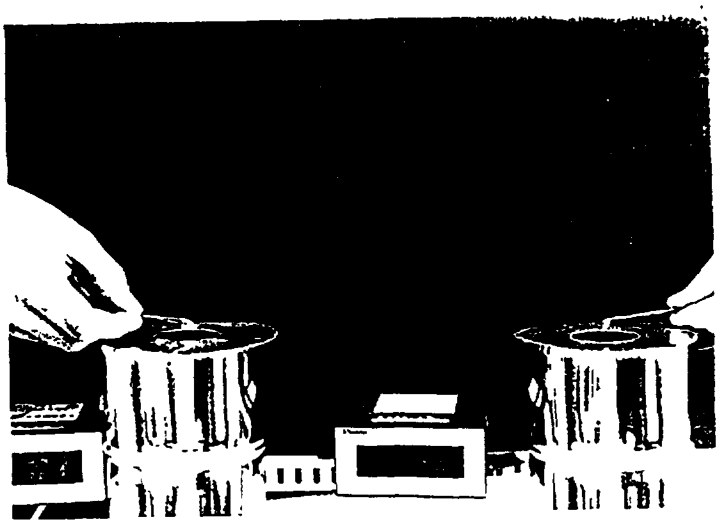


Photo. 3 Returned to the normal flow
(The flow returns to the normal flow
when the flow resistaces are added
on the extis of the both stacks)

Fig. 4 The air flow behviors at the exits of the stacks

The abnormal flow occurence was taken place when the annular gap was varied from 15mm to 10mm in the experiment demonstrating the optimum gap width for the effective cooling of the heated wall by showing agreement between the analysis and the experiments. Befor reducing the channel gap to 10mm stacks air was equally breathing out on both the stacks as shown in Photo.1 of Fig.4, where the air flow is visualized by smoke of incense sticks. However, at the moment the gap was changed to 10mm, on one side of the stacks air flow began to be reversed to suction side, as shown in Photo.2. But under this condition, if being added flow resistance on both the stacks by reducing the flow area the flow pattern retuned to the original as shown in Photo.3. This behavior was of strongly repeating characteristics.

After observing the possibility of the abnormal flow occurence of the natural air draft, the aim of experiment was shifted to clarifying the causes and identifying the conditions to avoid the abnormal phenomenon, as it was thought to be very inherent and important to these systems.

Based on the experimental results that the inlet gate flow resistance causes the same effect with the frictional resistance of the cooling channel on the abnormal flow occurence, the experimental apparatus was changed to have an

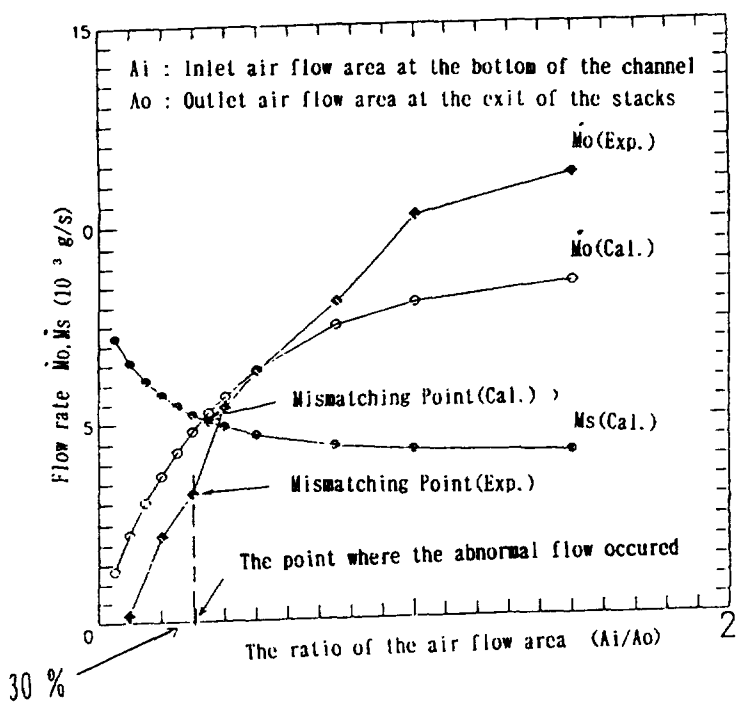


Fig.5 The comparison of the experimental results with the analysis

inlet gate to adjust the inlet flow resistance by changing the air intake area with the gap fixed at 20mm in width. The flow behaviors were examined on the various combinations of the inlet flow resistance with the exit flow resistance on the stacks. The abnormal flow occurence was taken place at the intake flow area of about 30 % of the cooling channel flow area (Fig.5), and was recovered by adding same measures of flow restriction on the stacks after the transient.

On comparisons of the experimental results with the analyses for the various combinations of the inlet and outlet flow resistances, it was concluded that the abnormal flow arised on the experimental apparatus when the conditions of the flow was as shown in Fig.5,

$$\dot{M}o < \dot{M}s$$

where $\dot{M}o$ is the mass flow rate calculated for all the path through from the inlet gate to the exit of the stacks and $\dot{M}s$ is the mass flow rate evaluated for the chimny effect caused by only the stacks.

In the subsequent experiments, however , on the experiments modified with four stacks instead of two stacks, it was proved Eq.(1) are not always met. For that reson the heat leakages along the heated air paths are doubted, being studied now with some modifications on the the experimental apparatus.

(Conclusion)

In applications of the natural air draft enhanced by chimny effect for the passive decay heat removal systems, it is advisable for avoiding the abnormal flow phenomenon that

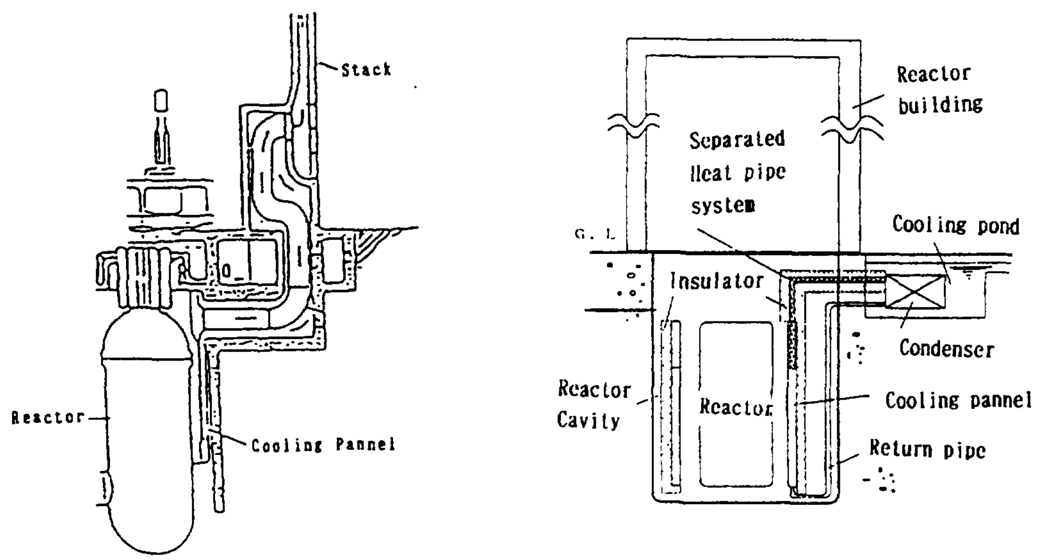
1) The air flow resistance in the inlet side (up-stream)to the heated section should be as low as possible compared with the exit side (down-stream) of the stream
2) The heat leakages along the air flow paths are douted to have big relations with the abnormal flow occurence ,and so should be carefully considered.

3.3 Studies on the possibilities of the attractive applications of the separated heat pipe system for the passive decay heat removal systems

(Experimental purpose)

Experiments on the separated heat pipe system enclosing non condensible gas, conception of which originates from so called “variable conductance heat pipe ” have been carried out with intention of using such system for the decay heat removal of the reactor systems. Basic experiments have been carried out on the experimental apparatus. The intention of our studies is to investigate the attractive applications of a separated heat pipe system in place of RCCS for the passive decay heat removal of such reactors.

The stack system in RCCS in MHTGR(Fig.6 (A))is replaced now by the separated heat pipe system in this concept as shown in (B). The decay heat in accidents is received on the cooling pannel and transmitted to the evaporator on the back of the cooling pannel, evaporating the working liquid and transporting the heat as in the form of vapor to the condenser in the cooling pond outside the reactor building. The vapor is then cooled to become liquid and returned back to the evaporator, thus recirculating in the loop and removing the decay heat without any active device. The special features of these experiments are to search for the attractive applications by introduction of the variable conductance characteristics enclosing non-condensible gas in such systems.



(A) The concept of RCCS　　(B) One concept on applications

Fig.6 Concepts on applications

(Experiments)

There is so called variable conductance heat pipe. This is a heat pipe enclosing non-condensible gas together with the working fluid in ordinary heat pipes, rendering self-control characteristics in the variable conductance as shown in Fig.7.

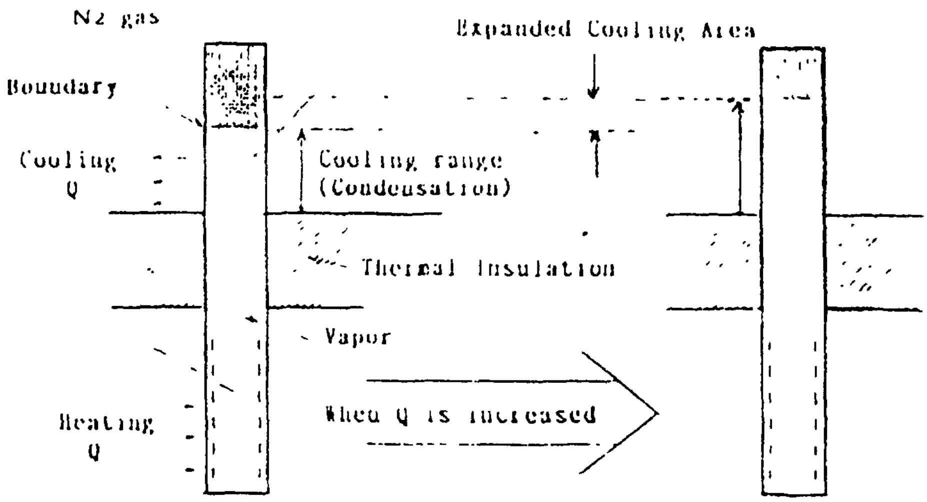


Fig. 7 The function of variable conductance heat pipe

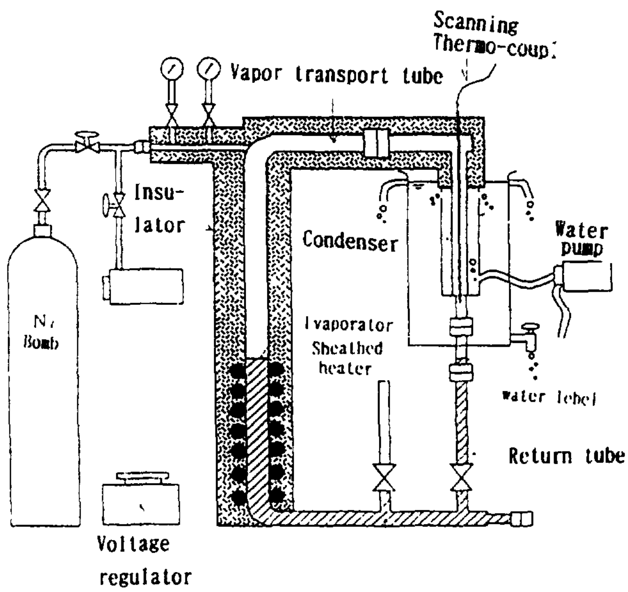


Fig. 8 Experimental apparatus

The special feature of this study is to try to introduce these functions into the separated (recirculated) type of heat pipes.

The apparatus consists of an evaporator, vapor transport tube, condenser and return tube to return the condensed liquid back to the evaporator, as shown in Fig. 8. The main parts are made of cupper tubes of about 20mm in diameter and about 1m x1m x 2m in size. Water and methanole were examined as the working fluid and nitrogen gas was enclosed as the non-condensible gas.

The behaviors of the system were examined for the parametoric changes of the heat input under the various enclosed gas pressure conditions (including vacuum). A given pressure of nitrogen gas was enclosed in the cold state before the running of each experiment.

The temperature distribution in the condenser tube on the various heat input without enclosing non-condensible gas (Po=0 MPa) examined for the base of the comparison is shown in Fig.9. In this case, it is characterized that the system temperature, i.e., vapor temperature inside the tube is rather low and the distribution is flat, and that when the heatinput is increased the temerature levels are increased corespondingly. It seems that the condensation is done for the whole region of the condenser tube.

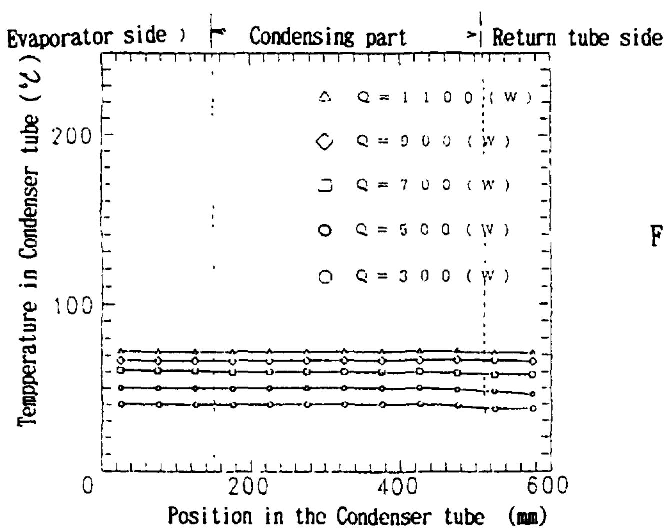


Fig.9 The temperatures in the condenser tube on the various heat input Q (Po = N_2 gas pressure enclosed = 0 MPa ; without N_2 gas)

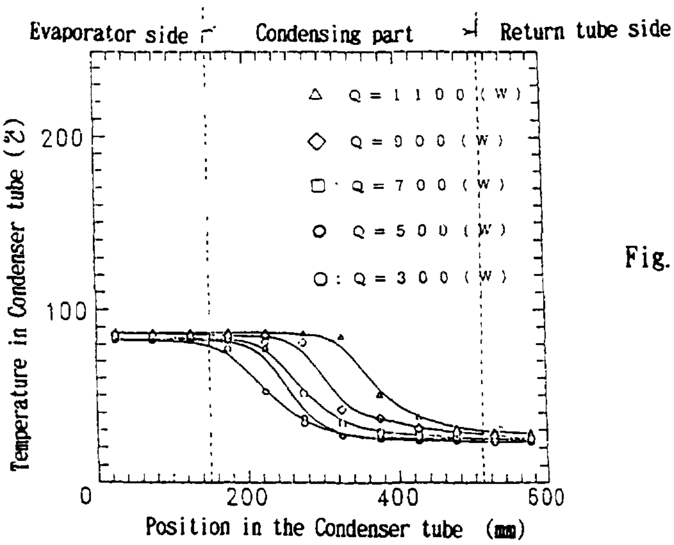


Fig.10 The temperatures in the condenser tube on the various heat input Q (Po = 0.014 MPa (0.14ata))

However, if the the non condensible gas is initially enclosed, the situation is changed drastically as seen in Fig. 10, where nitrogen gas was initially enclosed by the pressure of 0.014 MPa(about 0.14 ata). The temperature of the Evaporator side, that is the system temperature, is farely increased while in the Return tube side the temperature slops down. It seems that in the Return tube side the most part of the tube is filled with nitrogen gas and the condensation is limitted to the point of the overhanged in the tube, thus limitting the heat discharge area by condensation to give the system temperature rise. If the heat input is increased the condensation area is increased by itself, thus refraining from the temperature rise and clearly showing the self control features.

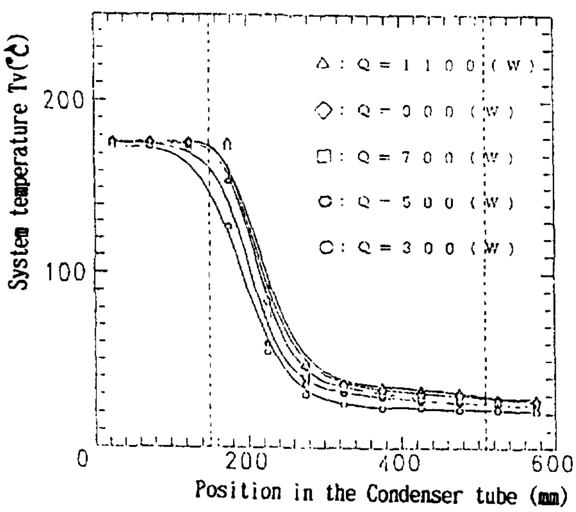


Fig. 11 The temperatures in the condenser tube on the various heat input Q (Po = 0.2 MPa (2ata))

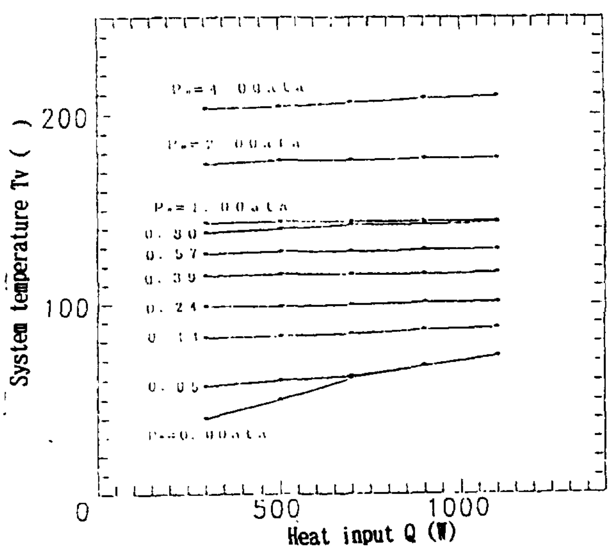


Fig. 12 The atained system temperatures on the various enclosed N2 pressure Po to the various heat input Q

Fig. 11 shows the experimental results with the nitrogen gas initially enclosed at 0.2 MPa (2ata). The system temperature grows higher in accordance with the increase of non condensible gas pressure enclosed. The data on the system temperatures in the experiments were arranged in Fig. 12 in the corelation with the heat input to the pressures of N_2 gas initially enclosed.

The comparison of the atained system temperatures with the analyses which was solved by characteristic equation taking notice of the the pressure, temperature and mass of the N_2 gas enclosd is shown in Fig. 13.

One of the results of the same experiments done on methanole as the working fluid is shown in Fig. 14 in which methanole shows the same behaviors but at somewhat lower temperature level.

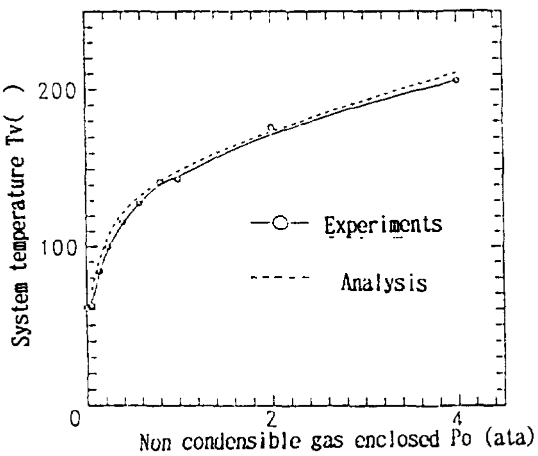


Fig. 13 The comparison of the attained system temperatures with the analyses (Q=700W, Working fluid ; Water,)

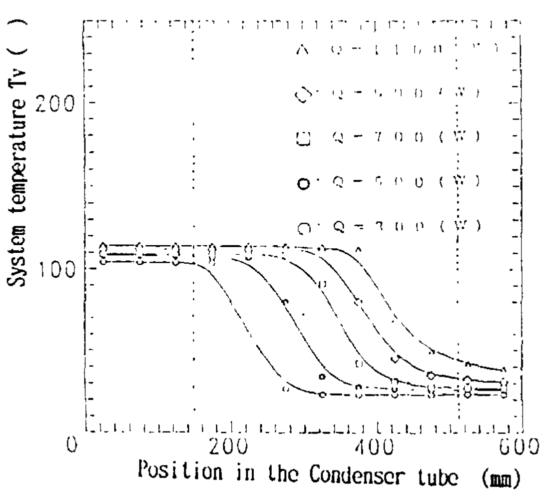


Fig. 14 The temperatures in the condenser tube on the various heat input Q when using methanole as the working fluid (Po = 0.1 MPa)

(Conclusion)

The basic experiments on the separated heat pipe system enclosing non condensible gas have been done with intention of searching for the attractive applications on the passive decay heat removal of the reactors. The results of the experiments show very clear features of self control characteristics which suggest the possibilities of the interesting applications for the passive decay heat removal in reactor system such as modular HTGRs. On both the experiments and the analysis the mechanism of the system is now clarifid and on that stand interesting applications are expected, one of which is a possibility to design a fully passive decay heat removal system which can minimize the heat loss during the normal operations with maintaining the sufficient heat removal rate during accidents.

XA9846675

MANAGEMENT SYSTEM AND POTENTIAL MARKETS FOR A HTR-GT PLANT

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Abstract

This article will discuss some aspects which could be helpful to execute a HTR-GT study successfully:

1. The prefered type of organisation for such a study; in order to achieve a maximum of support in society and industry, a minimum of through life costing and a maximum of through life support.

2. The lead time needed for such studies i.e. the design, component testing, prototype testing, the required efficiency, the type of energy in quantity and quality, financial targets, controlability, maintainability and reliability.

3. The potential markets for the nuclear gasturbine driven energy plants in the low power range. Analyses of the markets will be explained from the user's point of view on why, when and how, for what purpose, in which power range, as well as how many units per application would be required.

Introduction.

From time to time examples are published of products that are carefully designed, tested and produced with the involvement of scientists, designers and engineers and which turn out to be unmarketable and become a complete loss. One way to prevent this from happening is to invite potential users, producers and maintainers and inform them about your ideas and thoughts on design, marketing potential, production matters, financial targets and why you think it is needed. The next step is a difficult one for scientists, designers and engineers, but listen to these potential users, producers and maintainers, i.o.w listen to the market. This well-proven managementsystem is the only way to prevent disappointment and to create a maximum of support in society, a minimum of through life costing and a maximum of through life support i.o.w. the market for your product.
In the Netherlands the support for nuclear energy generation is rather slim, so to start a study for a new nuclear product is a very tense action. We followed this cautious procedure carefully, although it is certainly a new and uneasy approach for many of us.

The installation we have in mind.

What did we have in mind? In short a suitable replacement for certain diesel-driven or gasturbine-driven applications. An installation consisting of an inherently safe helium cooled pebble-bed reactor with a thermal power of less than 100 MW, directly coupled to a gasturbine in a closed cycle system, only using well-proven technologies and management systems. See Appendix 0.
Available for the replacement market around 2010.
Available at a comparable price, based on through life costing and through life support.
Suitable for the markets of :
- Combined heat and power applications
- Stand-alone heat generation
- Stand-alone electricity generation
- Ship propulsion

Driving reasons were not so much the limitation of the amounts of fossil fuels, but the CO2 problem (a major political subject in the Netherlands), the availability of well-proven technologies and management systems, the integration process of Europe with its free-market aspect and the ever increasing price of energy production. In addition the time has come to find new challenges for our youth and to set the next step in the history of energy generation.

For a market analysis it is important to consider:
1. The kind of user of your installation
2. The type of installation, i.o.w. Combined Heat and power, stand-alone electricity, stand-alone heat applications or ship propulsion
3. The required power range.

1.The kind of user is very important. The main division is:

1.a. The professional user - in other words the utilities. The characteristics of these installations are customer-built, high level power ranges stand-alone electricity generating plants with an efficiency of 35% or Steamturbine and Gasturbine combination plants (STEG) with an efficiency of 55%.

1.b. The non-professional user - this user needs energy for his industrial process like paper mills, dairies, breweries amd ship propulsion, in most cases low temperature slightly superheated steam for processes like drying, brewing and pasteurizing. This kind of process is their speciality and they hate to be bothered with the when, where and how of their power as long it is reliable, available and as cheap as possible. These installations are standardised to a very high degree and are factory designed, constructed, assembled and tested.

The installation we have in mind is ment for the second kind of user. So we started to talk to them and to listen to them to fit their usage patterns, procedures, worries, wishes, maintenance schedules, etc. into our study.

A few points of the users views:

a. They really want to stay with their core-business.
b. They like the idea of an organisation which takes care of their energy supply on the basis of through life costing and through life support.
c. The installation should be designed on the basis of the American design principle Keep It Simple and Stupid (KISS).
d. From step one onwards international cooperation should be part of the project.
e. Users accept testing and trial periods, but the first installation should be a production unit which is temporarily used as trial and test unit and not the other way around.
f. Well-proven technologies, assembling techniques and components should be used, to minimize the risks for the user.
g. The installation must be modest in size and clean. A papermaking industry wants to see steam clouds from the drying process and not a huge silo containing their energy production unit.
h. The through life costing for the new installation must be comparable to the existing plants for competitive reasons.
i. Financial targets are: a pay-back time of 10 years and an internal return rate of 15%. Considerable higher targets as the utilities demand. For the Netherlands the costs for CHP would be less than 5.5 Dutch cents per KWh (3 Dollar cents) and 8 cents per KWH (4.4 Dollar cents) in stand-alone electricity generation applications (at a crude oil price of 16 USD/barrel and an exchange rate of Dutch Guilder/USD - 1.8).

j. The energy generation process must be well controlable, depending on the demand and must be of constant quality. At the moment many users of CHP have additional firing in the exhaust boilers to control the quantity and quality of the steam for their industrial process . For this the nuclear installation is unsuitable, so a solution has to be found.

k. Prototype testing is very important as proof of safety. Its credibility increases considerably if the plant is constructed to serve as operational unit after the testing itself has been completed.

l. Although they do not refuse to believe the high availability factors you promise them, they do not like too high a price and do like to stick to their existing procedures concerning maintenance, redundancy and flexilibility.

This last paragraph is an important one: an industry using a maximum of 60 MWheat prefers 6 installations of 10 MWheat instead of 2 installations of 30 MWheat or one unit of 60 MWheat. It produces the required flexibility and redundancy. In practice the investment cost of a fossil fueled Combined Heat and Power (CHP) installation is practically independant of the installed power. (see appendix)
There seems to be no reason why this trend should not be applicable to nuclear installations of comparable power.

The procedure as described requires extra time, especially when the product, like our HTR-GT design, is a combination of well-proven technologies. This is logical, a lot of time is needed before people from different parts of the business community are comfortable with each other's opinions and capabilities. Only then they are able to get some feeling about the why and how of the other participants and to extend their support into their industry and their part of society.

The growth of this kind of public opinion has to be supported by publications in newspapers and magazines, presentations at schools, colleges and universities, by visits to potential users, suppliers and maintainers, your presence at exhibitions and by the supply of regular information to politicians and leading managers.

As was mentioned before, the support for nuclear technology is slim, the reasons why do not matter anymore. Stop being afraid of publicity and public relations. Fight back and play bluff poker. The other side is just as clever as you are.

The efficiency of the plant must at least be comparable to that of the existing plants. Not so much because of the fuel price, but because the potential owners have been fighting for high efficiency as a result of the ever increasing price of fuel and related taxes and it will frighten them to suddenly leave this path. In addition the young people in the Netherlands are well aware of the fact that the energy waste will be high on the agenda for environmental reasons after the CO_2 problem has been brought under control.

Maintainability is another aspect on which you must listen to the customer. He likes an organisation which takes care of the whole through life support, including cooperation between all users, new production, training, control, new infrastructure, licency, maintenance, documentation, and scrapping; in other words the full logistic support. This is only possible if the installation is made suitable for this design from day one onwards. There exists a management system which is able to do all this. This will be discussed in part two of this publication.

The Pooling system.

In this paragraph an aspect will be discussed, which is growing in importance; namely that of the user's ideas on the management of their Co-generation plants. The pooling-system fits with trends like "back to core business", "we will take care of you" and through life costing in combination with through life support and shared costs.

The pooling system also seems to be a set of tools to make the owner costs of an INCOGEN istallation comparable to those of the existing fossil fuel fired CHP-plants.

The history.

The system was developped at the end of World War II by the Royal Airforce of Great Britain. The aim was to improve in all aspects the world-wide exploitation of the Rolls Royce Merlin engines of the Spitfire. Since then it has further been developed and is now used by aircraft companies like KLM and British Airways. The propulsion engines are taken from the wing when needed for repair, overhaul or modification and are replaced immediately by a spare one to keep the aircraft available as much as possible. KLM manages a pool of CF-6 General Electric Engines for KLM itself and a number of other aircraft companies. British Airways does the same for the Rolls Royce aero gasturbines. Similar systems are in operation by European Airforces, for example for the Grumman F-16. A very special version is a pooling-system called "Memorandum of Understanding for the logistic support of Rolls Royce marinized gasturbines". It manages (on 01-08-'96) a pool of 109 Olympus, 97 Tynes and 46 Spey gasturbines in use with the navies of Great Britain, France, Belgium and the Netherlands. The savings are considerable. For example to keep operational all ships in the Royal Navy, using Tyne gasturbines for cruising, 22 Tynes should be available and 14 Tynes for the Royal Netherlands navy. By pooling the two navies 29 engines should suffice. A saving of 7 engines. The same saving occurs with all the spares, tooling and the workload of the repair and maintenance lines. From the military point of view one of the advantages of the pooling-system is its flexibility, which has been proven time and again in periods of war and crisis control; most lately during the Falkland crisis and the war in the Persian Gulf.

Very recently the United States Department of Justice said it approved a plan by eight nuclear utilities, to pool their equipment purchases, to share personnel and other resources in a move to cut costs. The reason this Department became involved is the Anti-Trust law in the United States.

This Pooling-organisation is called Utillities Services Alliance and manages 7 % and intends to grow to 35 % of the US nuclear power production.

The aim.

The objective of a Pooling-system is to establish arrangements for the logistic support of *common* equipment. The main advantage of a pooling-system is the *economy of scale*. In other words to increase the efficiency of the availability, the flexibility, the readiness, the new production, overhaul and repair capacity with its tooling, dedicated personnel, documentation, spares etc. for a particular type of engine or equipment. This takes place through the pooling of all available know-how, experience and investments and will lead to a reduction in exploitation costs and thus to the cost of ownership by through life support of the technical installations concerned.

How does it work?

A Pooling-system is only applicable when there is a commonality of equipment. It can comprise a whole installation like a stand-alone nuclear gasturbine installation employed to generate electricity or it can be a part of a technical installation, such as similar control equipment on different types of installations. In this article we will only consider the effect of a pooling-system on a nuclear gasturbine plant like the HTR-GT for the low power range. The owners of such a plant own, manage and share out the benefits and losses of a pool of spares, tooling, spare equipment, documentation, training facilities, test equipment en facilities, overhaul and repair shops and similar investments with subcontractors, know-how to build, maintain and scrap equipment and experience concerning the exploitation of the equipment.

The actual management is done by a dedicated team, which reports periodically to the shareholders, the owners and users of the plant.

The pool takes care of: new production in dedicated workshops, setting to work, remote control, the planned maintenance in situe and on the overhaul line, the replacement of the fuel, the handling of the nuclear waste, the analyses of defects and possible development, testing, implementation and configuration control of modifications, the training of operators, safety personnel, maintainers, the subcontractors, the planned maintenance, the investment in spares, the rest life of spares, tooling, overhaul, repair and maintenance equipment, documentation, construction, special legislation and generally applicable legislation.

Modular construction.

An important aspect in considering a pooling-system is the aspect of a modular construction. Modular built means that the total installation can be divided into units that are easy to manage and which can be replaced with spare units for repair, planned maintenance or overhaul.

So in a nuclear gasturbine installation the non-nuclear part can be and should be divided into at least the following modules: recuperator, power control, engine health monitoring system, helium cleaning system, heat exchangers, auxilliary equipment and the gasturbine itself. This will lead to a minimum of stand-down time, one of the biggest problems of the existing nuclear installations.

The maintenance time can be decreased even further by the joining together of combinations of modules in a bigger replacement unit. For the shipping applications this seems advantageous, because of scheduled docking periods.

The modules have to be interchangeable; in other words the installations will have to have a very high degree of standardization, because otherwise special procedures, tools, spares and so on would be required. This would increase the cost of ownership which is shared by all partners. So a stringent configuration control is a necessity.

This is very important. Equipment which is customer designed, built, maintained etc. is very common in the existing nuclear world (applies to most (nuclear) power plants), but increases the cost of ownership drastically. So if nuclear energy generation is to be more widely applicable, this existing way of thinking and working has to be reconsidered.

In the nuclear part of the INCOGEN-installation the replacement of the fuel is the only maintenance action which can be done in the modular way. It is advisable to place the fuel, during the usage, in some kind of open "shopping basket" which will be taken out periodically and placed in a transport container and replaced by another "shopping basket". The HTR-fuel is very suitable for such a timesaving and thus moneysaving treatment.

For the non-nuclear part the number of spare modules depends on the planned maintenance of the modules themselves and the plant they belong to, the failure rate, the accepted time to replace or to repair, etc., as discussed before. The minimum replacement time is most likely dictated by the nuclear part of the installation which has to cool properly before the helium circuit can be managed and opened for the maintenance as described.

There is nothing new in using containerizing, modular constructions and palletization of equipment. It is a well-known proven management technique. All gasturbines are modular of construction nowadays, all engineering plants have transport routes to replace and to transport heat exchangers, electromotors, generators, valves, control systems etc. for testing, maintenance, etc., especially when the stop must be as short as possible to fit in with the industrial process. Whether the unit is called a container, a module or a pallet depends mostly on the size or the professional language of the OEM and/or the owner.

Nuclear standards and norms.

Another tool to reduce the through life costing is the reconsideration of the existing nuclear norms and standards. A lot of courage is needed for an operation like this. A comparable study has been finished very recently by the Royal Netherlands Navy. The aim was to replace two very important frigates within the shrinking defense budgets. It took 5 years and they considered very carefully but seriously the existing military norms and standards, international cooperation on R & D and exploitation, modular building procedures to reduce building time, palleterizing of weapon and sensor systems, usage of aero-space technology where possible, using civil (a.o. ferries) experience and by putting this question on every aspect "what will be the penalty if we do not apply the military norm or standard?" The result is shown in the graph in the appendix. The result was a bigger ship, with less crew, a bigger flexibility for future modifications to weapon and sensor systems, less R & D costs, no loss of employment, same speed and fuel costs, same through life costing and same through life support. See Appendix 1.

The market analysis for the Netherlands.

The first step was to study the existing market of fossil-fueled, gasturbine driven plants in the Netherlands.

A prefered power range was established by analysing the total existing population of gasturbine driven power plants for the markets as mentioned before. See appendix page 2-A.

The next step was to take out the gasturbines used by electricity generating companies, de Gasunie and oil companies. See appendix page 2-B.

Thereafter we had to establish the age of the existing plants.

See appendix page 2-C.

The last step was to establish the shipping market.

See appendix page 2-D.

During each step the philosophies of experienced users were taken into consideration.

The results of this market analysis are:

1. The 20 MWth installation is dominant as well for Co-generation as for ship propulsion.
2. The total number for Co-generation is 75 units.
3. There seems to be no market in this power range for stand-alone electricity generation or heat generation in the Netherlands.
4. The market for shipping (Dutch and Dutch Antillian flag) is 205 units.
5. The replacement market will start to come into effect in about 2010.

Western Europe.

In Europe Co-generation, and thus nuclear CHP, has real possibilities for export, mainly because of the positive effect on the production price in an industrial process. This is of special interest to companies in countries with comparable salaries, because this gives the owner of a Co-generation plant an advantage over his competitor. After all, separate generation of heat and electricity is more expensive. In the open market this effect will be the main driving force behind the export possibilities of Co-generation. Another positive force will be the lower burden on the environment as far as emissions of combustion gas and waste heat are concerned, as well as the higher total efficiency of the Co-generation installations, although we are of the opinion that this issue plays a lesser role outside the Netherlands. The promotion efforts have not been very impressive so far, but the Advisory Committee of the Ministry of Economic Affairs will start coordinating the promotion efforts in 1997. When the realisation of any export could actually start, is difficult to estimate. There is, of course, a negative force in this process as well. In many countries the national power generating boards have and will try by all means to maintain their monopoly. This monopoly can only be broken by pressure from governments as part of the open market policy.

The project bureau "Warmte/Kracht Koppeling" did a study on the export possibilities of CHP (Combined Heat and Power) in Europe. An extract was published in the magazine "IPG - International Power Generation", issue september, page 38:

"The following are some recommendations for European and national government actions to unlock the potential for CHP, and given in descending order of importance:

- Political will is needed to remove barriers and to provide a framework, giving clear opportunities to those wishing to benefit from the development of high efficiency CHP systems.
- The efforts of the energy industries, in particular the electricity industry, to move from suppliers of energy to providers of energy services, including CHP, should be reinforced.
- Liberalisation in electricity markets can bring major opportunities for the development of CHP through, for example, access to grid networks and the introduction of greater competition in power generation.
- Under circumstances where the economical conditions are similar, priority should be given to CHP investment over a conventional plant.
- The internationalisation of environmental costs in energy prices should be supported as a mechanism for promoting CHP, while ensuring that industry and other energy users derive real economic benefits from it.
- Member state governments and the EU as a whole should set ambitious but achievable targets for CHP for 2005 and 2010.
- Wider efforts should be made to reinforce and develop CHP/DH networks at the urban level. Such systems can be multi-fuel, offering significant economic flexibility."

The conclusions are mentioned here with the permission of the "Project bureau Warmte/Kracht Koppeling".

Stand-alone electricity or heat production with an installation of 20MWth seems to be a small market in Western Europe.

The possibilities in the market for shipping are comparable to the Dutch shipping market.

The market worldwide.

The possibilities for the export of a Nuclear Gasturbine Installation of 20 MWth around the world do not differ widely from the possibilities in Europe, although the requirement for stand-alone electricity generation is much bigger. Diesel generators in the power range of 5 to 15 MW are in use on most islands around the world (such as Indonesia, the Phillipines, the Dutch Antillies and the Seychelles) or in places were the investment in a distribution system is not very cost effective or not feasable.

The market for the small HTR-GT is not only the replacement market for these diesel engines but also occurs in areas where the supply of fossil fuel and/or the investment and maintenance costs are a major factor in the cost of power generation and distribution. In other words, areas with:

1. a low population density, difficulty to maintain distribution systems and remote from a fossil fuel supply.
2. a high density of population and industry and remote from a fossil fuel supply or from the sea or main rivers.

For the development of these markets priority should be given to areas where industry is very much needed to create employment. In other words energy means employment, which leads to economic growth.

Which application (stand-alone power generation, heat generation or co-generation) will be chosen, depends on the climate and/or the industrial process for which the installation is intended.

As mentioned before the number of diesel engines used for electricity generation in remote areas and on islands is enormous. This is another huge potential market which should be developed.

The market for shipping.

The market for shipping has a very conservative and peculiar character. This has always been the case and history teaches us that changes in the propulsion of ships have always been emotional and have taken place in a relatively short time. This last aspect has mainly been caused by economic (smaller crews or better delivery schedules) or military reasons. So it is most likely that history will repeat itself in this case. An additional negative force in this market is the reluctancy to accept nuclear driven ships in many harbours.

Conclusions.

Due to trends like "back to core business", future owners of an energy plant will increasingly look for means to reduce costs of ownership in combination with guaranteed through life support.

The pooling-system satisfies these aspects to a great extent. For the design the tool of reconsidering the existing nuclear norms and standards is absolutely necessary.

It should be pointed out that all the markets mentioned above, can best be opened by building a Nuclear Gasturbine Installation. This installation can prove and show time and time again the inherent safety of the design to anybody. In this installation all the modern philosophies like modular construction, containerizing where practical, remote control, easy to maintain, pooling-management system etc., are implemented, leading to an acceptable and understandable price.

The modular construction makes it possible to build and use the described HTR-GT installations in even the most remote areas.

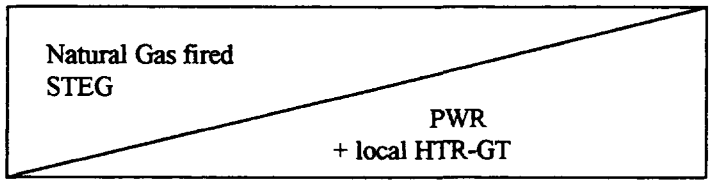
The HTR-GT fully satisfies the intentions to stop further climatic changes due to the CO^2 emissions around the world.

Appendix 0

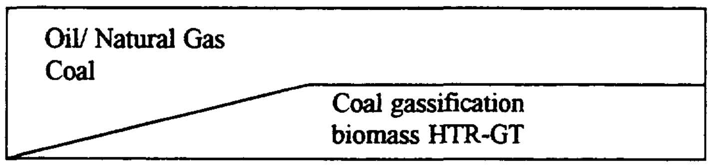
Energy Density

Estimated changes in future Energy supply per type users

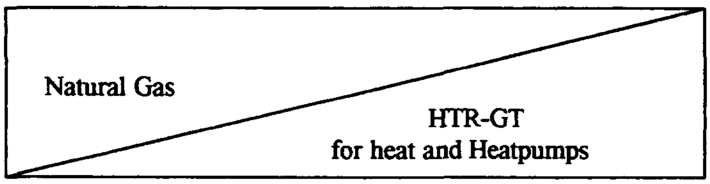
E. Utilities, see A.



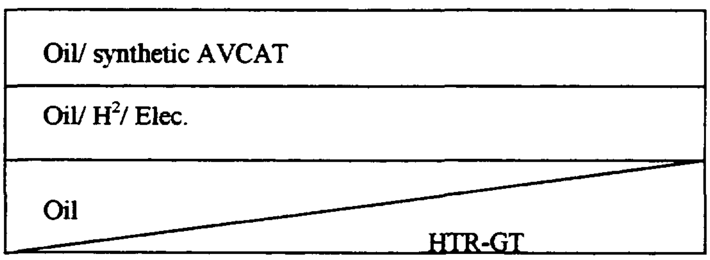
D. High Temperature heat, industry



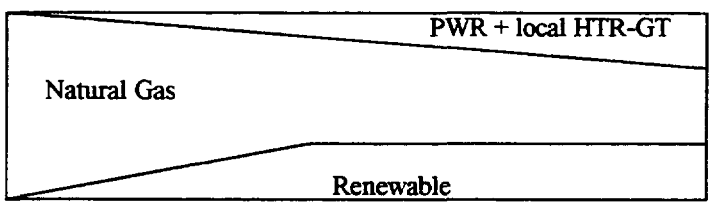
C. Low temperature heat
beer/paper/dairies



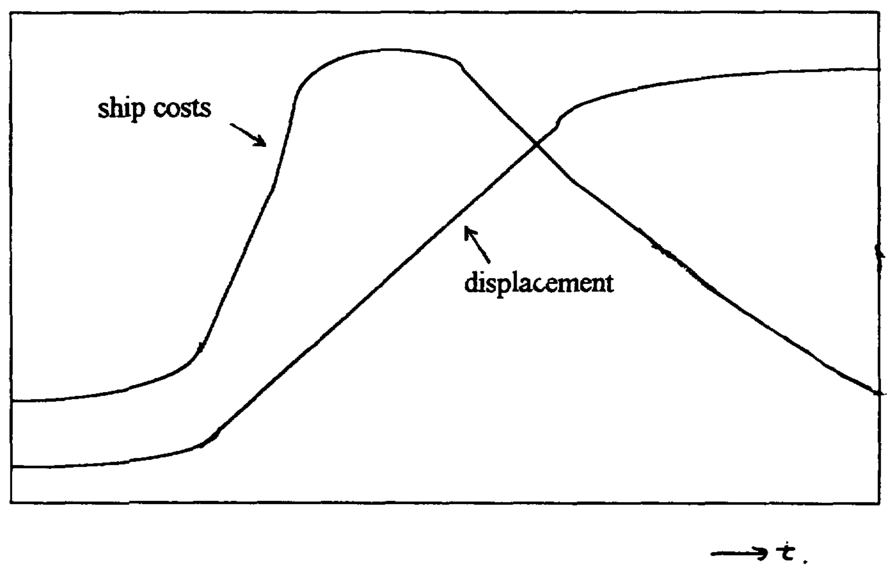
B. Transport Air
1000 x A Road
Sea



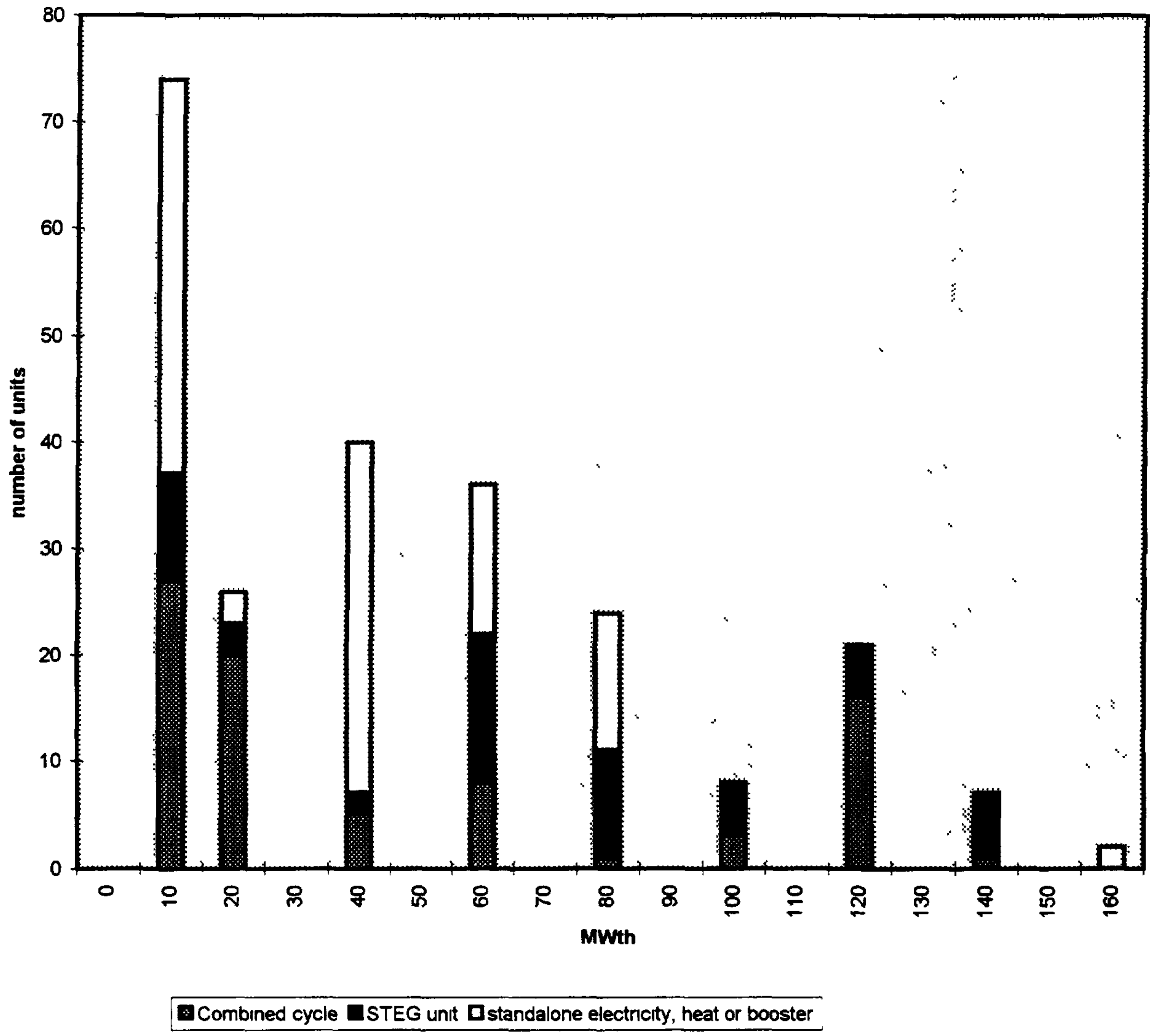
A. Households
Offices
see abv. E



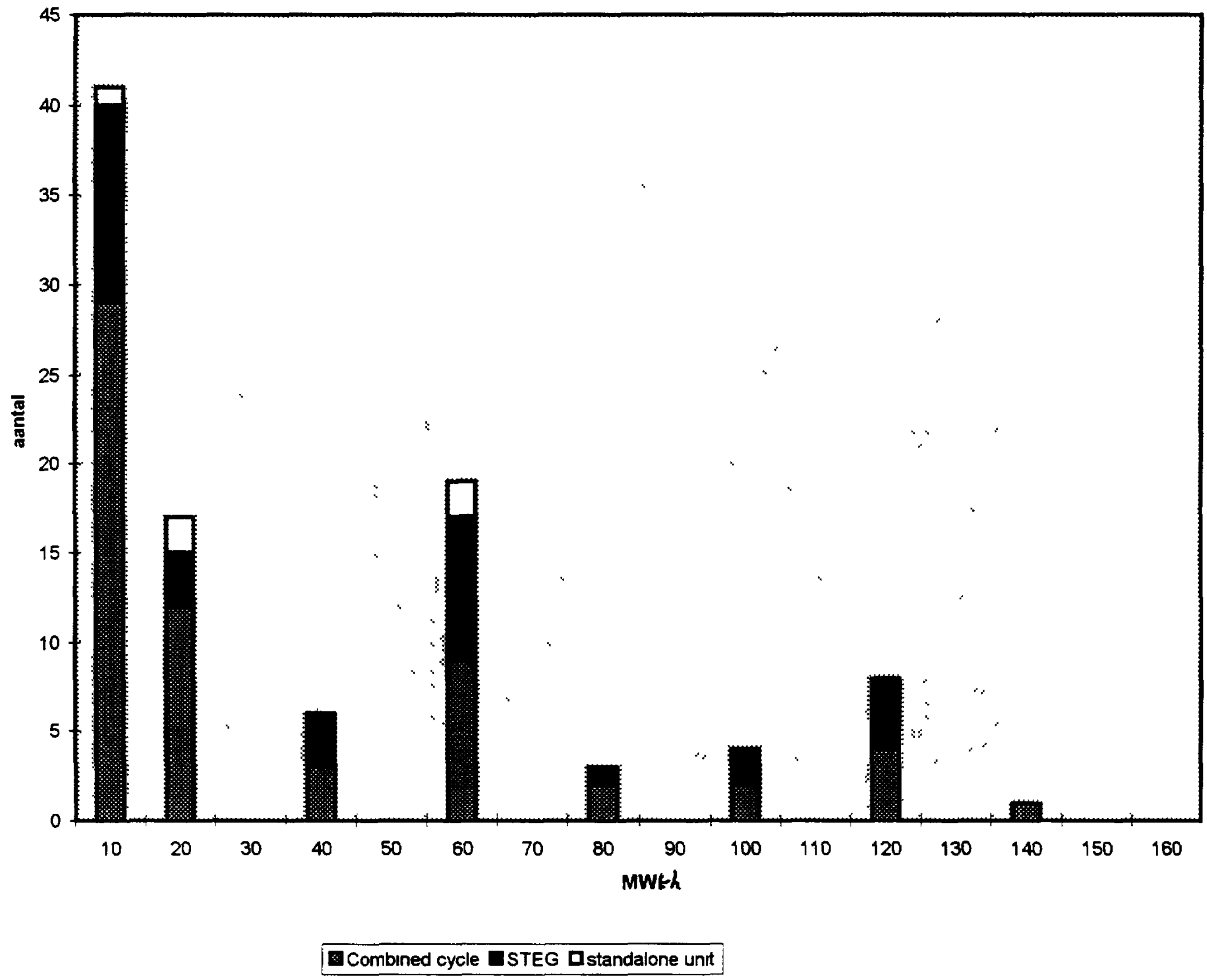
Decrease of military norms and standards



Gasturbine population in the Netherlands

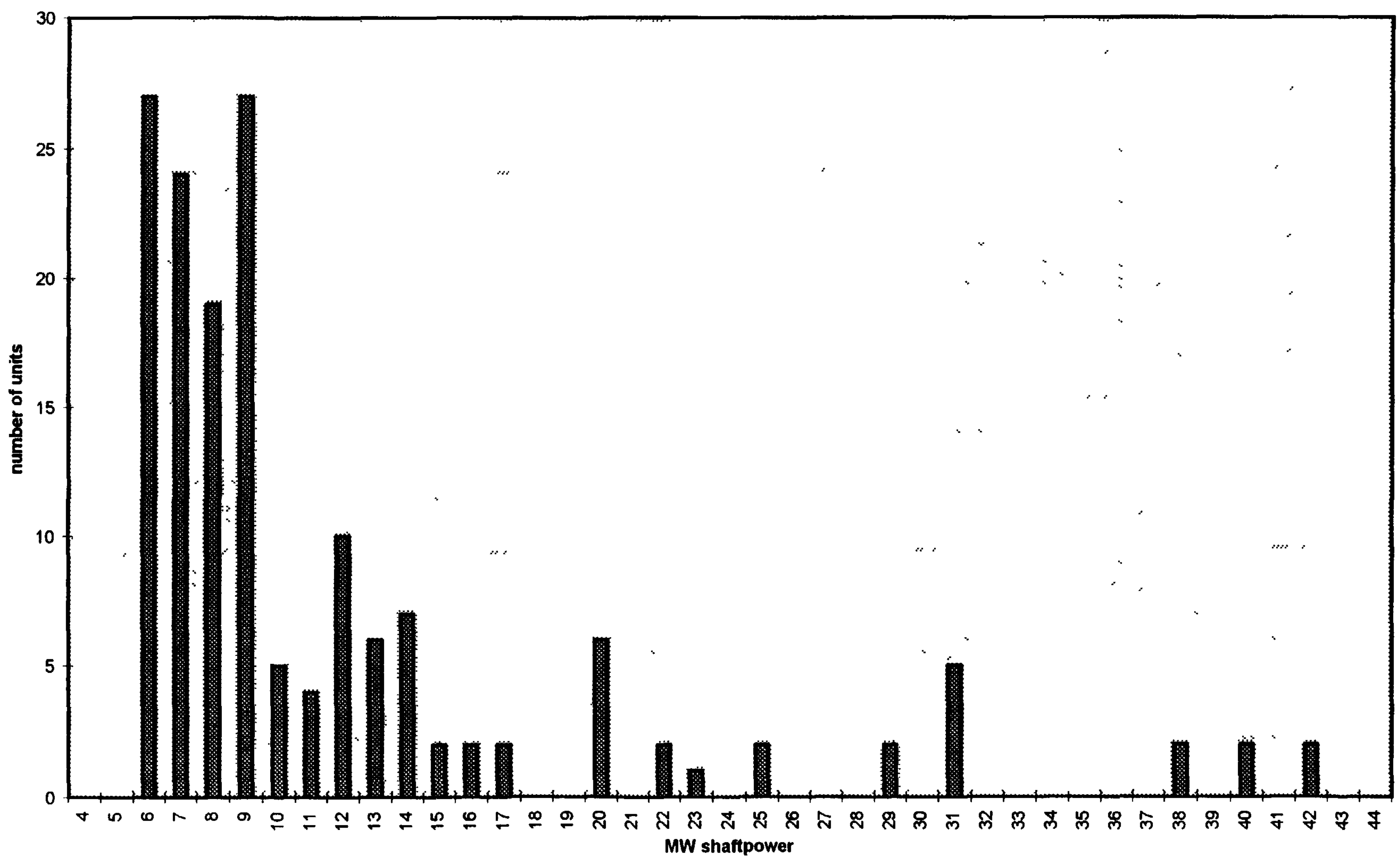


Gasturbine Population privately owned



Gasturbine population privately owned. Replacement market 40 years later.

Propulsion of the Dutch and Antillian merchantfleet



XA9846676

APPROACH ON A GLOBAL HTGR R&D NETWORK

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Abstract

The present situation of nuclear power in general and of the innovative nuclear reactor systems in particular requires more comprehensive, coordinated R&D efforts on a broad international level to respond to today's requirements with respect to public and economic acceptance as well as to globalization trends and global environmental problems.

HTGR technology development has already reached a high degree of maturity that will be complemented by the operation of the two new test reactors in Japan and China, representing technological milestones for the demonstration of HTGR safety characteristics and Nuclear Process Heat generation capabilities.

It is proposed by the IAEA 'International Working Group on Gas-Cooled Reactors' to establish a 'Global HTGR R&D Network' on basic HTGR technology for the stable, long-term advancement of the specific HTGR features and as a basis for the future market introduction of this innovative reactor system.

The background and the motivation for this approach are illustrated, as well as first proposals on the main objectives, the structure and the further procedures for the implementation of such a multinational working sharing R&D network. Modern telecooperation methods are foreseen as an interactive tool for effective communication and collaboration on a global scale.

Introduction

Several industrial, national and international approaches to the market introduction of High-Temperature Gas-Cooled Reactors (HTGRs) have been undertaken in the past and are still going on in different parts of the world with various time horizons and market sector targets. Up to now, none of these stand-alone attempts have led to a commercial follow-up project, although the maturity of the HTGR technology has reached a considerable level and in spite of the important know-how that could be gained from operation of the test reactors (Peach Bottom, DRAGON, AVR) as well as from the demonstration reactors (Fort Saint Vrain, THTR) complemented by comprehensive operation experience from other Gas-Cooled Reactors (GCRs).

This situation seems to be symptomatic of new nuclear reactor lines in particular as no innovative nuclear reactor system at all has overcome the threshold to commercial application in the last few decades anywhere on the globe. In the meantime it appears that also the introduced reactors have to face stagnation, moratoriums and problems of economic and public acceptance in most industrialized countries, a far cry from the promising expectations of 'pioneering times'.

Radical changes have been experienced in the structure of the nuclear industries in recent years, too. Former competing suppliers for the different national and international markets for nuclear power plants have merged into international consortiums as 'global players'. And even the situation of the utilities is going to change drastically mainly due to privatization and competition in the enlarged and interconnected electricity distribution networks.

Extremely low prices for fossil energy carriers combined with remarkable enhancements in the efficiencies of fossil-fired power plants generate additional economic pressure on existing and new nuclear power plants Together with time ranges being continuously extended for the reserves of fossil energies, the 'classical' pro-nuclear argumentation based on anticipated price escalations and shortages of fossil energy carriers are being increasingly eroded in spite of growing awareness of their environmental impacts and their influence on climate change

The case for nuclear power has to be redefined within this 'changed world' in a realistic way, generally and also separately for the different nuclear technologies An active response to the current situation and expected evolution is necessary and more appropriate than - passively - waiting for unfavourable developments in the energy market with, e.g , price jumps, shortages and the need for drastic environmental regulations restricting the use of fossil energy carriers

Most of the important boundary conditions for the future of nuclear power are of a global nature or at least valid for many areas and countries in the world, because they are embedded in a more general 'globalization' of problems as well as of economic relations and dependencies The latter will be fostered by global information and (interdisciplinary) collaboration networks setting new historical milestones and chances for mankind

Against this more general background, today's obstacles and problems for nuclear power have to be understood as challenges for further research and development (R&D) using all the potentials of modern technology and the sound basis from R&D programmes and operational / manufacturing experience which is globally available for regaining economic and public acceptance

Globalization / internationalization of pre-competitive medium-/long-term R&D efforts, e g for innovative nuclear reactor systems may also be an adequate response to the shrinking industrial involvement beyond short-term commercial projects and to the cuts in the national nuclear programmes in many countries Besides that, such an approach will broaden the potential market from the very beginning and stabilize the technology base for later use

The 'rules of the game' for real globalization of R&D in a work-sharing and complementary manner have not yet been established in general But it should be possible to create new collaboration structures on a fair and collegial base not only for distributing the efforts and development risks across several 'shoulders', but also offering adequate benefits from the collaboration for all participants according to their contributions

The pooling of know-how and scientific / industrial capabilities on a broader international level has become easier than ever before by the effective use of recent telecooperation methods. It seems to be essential to make extensive use of these new communication techniques in an interactive way to achieve 'living structures' that have the ability to grow into a pragmatic collaboration tool

Lessons from the Past

Especially in HTGR history there has been a confusing multiplicity of different technical concepts, projects and applications. This underlines, on the one hand, the impressive potential and flexibility of this technology, but, on the other hand, valuable human and financial resources have been wasted on this internal struggle for the best technical solution and thus also created a burden for the reputation of HTGR technology

Passionate discussions on the differences and anticipated benefits of favoured technical or conceptional characteristics often concealed the overriding common elements, the common aims and the common responsibility for the success of the development efforts.

It would be better for the future, first to reveal the common technical sources and the coherences of the different approaches as a common, interconnecting technology base. Future projects can make use of this and add the incremental, specific R&D and design work needed for a definite concept.

Sometimes HTGR projects have been propagated as alternatives to the existing nuclear reactor types causing a lot of polarization and rejection within the nuclear community. It is more appropriate to define HTGRs as complementary to the existing Nuclear Power Plants (NPPs) and their evolutionary or more innovative derivatives with which HTGRs might establish future symbioses.

It should not be forgotten that the general safety philosophy of the HTGRs - rejected for a long time by many proponents of active safety systems - has partially found its renaissance in many advanced NPP designs. And this is a very important contribution to the future of nuclear power representing a real success for HTGR development up to now. The former conflict between passive and active safety systems has dissolved into a pragmatic coexistence of both safety approaches.

Focusing R&D only on the requirements of a specific project often turned out to be a qualification of 'frozen' technologies impeding the development of innovative solutions due to fixed and ambitiously short timescales. Instead of this attitude there should be a balanced mix of on-going innovations and of short-term qualifications of practicable solutions to maintain enough flexibility to react with innovative technologies to more stringent requirements concerning safety and economy as well as to actual chances for the market introduction. This view takes into account the fact that a nuclear reactor represents a combination of different technology elements/disciplines each having its own individual development potential following timescales that need not always fit the time limits of specific projects although they might open up promising options for continuous innovation of the technology.

Bad experience was made in the total focusing of R&D and commercial efforts only on one project thereby coupling the fate of the whole technology to the success of a stand-alone project. The establishment in parallel of a more generic technology base programme with a long-term orientation covering the essentials of the HTGR innovation potentials would help not only to conserve valuable know-how but also to supply necessary improvements for keeping pace with other technologies. Such a technology base programme for HTGRs could have the same justification as e.g. R&D on fusion reactors, both offering safe, clean and reliable energy for different purposes as a long-term option. But this approach needs a broader collaboration beyond purely national or commercial programmes including all the partners within the HTGR community.

The aspects of 'spin-off' and cross-fertilization might in future be even more important than in the past, either to transfer HTGR technologies to other conventional and nuclear applications or to profit from other developments outside the HTGR field. Why not use, for example, coated particle technology for other reactor types as well or for the final disposal of problematic waste, taking all the benefits from an applied technique afterwards or in parallel for the HTGR? This would provide an industrial background instead of an exotic atmosphere!

Possibly the most important lesson from the past is how to cope successfully with the problem of first-of-a-kind (FOIK) cost. Neither in the frame of national programmes nor in purely commercial attempts or in combinations of both, even with some bi- or international extensions, was it possible to proceed beyond the operation of test and demonstration reactors. The latter even being such a financial burden for their utilities that both demonstration reactors (FSV, THTR) were shut down before a long-term technology demonstration could be established for a commercial follow-up project. And this fate could not be excluded even for the smaller test reactors due to still very significant annual subsidies.

Broader structures are needed to divert the FOIK and the cost of pre-running R&D. Broader scopes for the markets and economic incentives are needed to generate enough motivation for commercial involvement. More stability is needed to provide more confidence for investors and public funds. Convincing R&D results are needed for the innovative systems complementing the use of proven technology. Technological perspectives are needed to assure steady response to more stringent safety and economic requirements in the future and to avoid technological dead-locks.

The Case for the HTGR

In the course of time, the main motivation arguments for HTGR development have shown some significant changes depending on the prevailing technical /political priorities, opinions, problems, economic assessments and perspectives of energy supply or availability of resources either fossil or nuclear.

Due to the flexibility of HTGR technology it was always possible to modify the HTGR concepts corresponding to the shifts in the overall requirements, but this was often out of phase, reaching the aim at the very moment when it disappeared, just as in the fable of the tortoise and the hare.

Today's requirements may be mainly governed by the aspects of

- public acceptance / coping with severe accidents
- economics
- waste handling
- environmental protection / climate change, but
- no anticipated energy resource availability problem for the time being.

The response to these aspects may be the key to the future of nuclear power and give additional drive, especially for the further development of HTGR technology.

The largest market for nuclear power is to be seen in those areas of the world that refused to build NPPs in the last few decades either due to acceptability problems or due to high investments and lack of competitiveness with cheap fossil-fired plants. So, regaining public acceptance, while retaining the economic goals, is an essentially commercial task opening a huge window outside the saturated markets where nuclear power has already gained significant shares in energy generation.

The most convincing way to cope with severe accident scenarios might be to avoid them by physical laws rather than to mitigate the consequences. It also should not be forgotten that the application of Nuclear Process Heat (NPH) in an industrial environment might only be acceptable from an industrial point of view if neither the high commercial investments nor the production lines are endangered by accidents of the nuclear energy supply source. Industries and insurances might even ask for more stringent safety requirements than licensing authorities controlling compliance with legal standards to avoid catastrophic releases of radiotoxicity and relocations, also in case of hypothetical accidents.

Broadening the field of application for power plants by including heat utilization is in accordance with the actual trend towards Combined Heat and Power (CHP) plants in the conventional energy supply area and is - by good reasons - very fashionable amongst environmentalists CHP may significantly enlarge the market for NPPs as well as their contribution to reduced noxious impacts on the environment combined with improved economics. A specific 'claim' for HTGRs can be assumed especially for high temperature ranges, gas turbine and process heat applications

Enhancing the thermal effectiveness and using CHP processes not only reduces power generation cost but also the relative waste production per unit of energy used This provides another argument in favour of the HTGR, which might be a good candidate for burning of long-lived radiotoxic wastes and fissile materials, too

The concept of coated particles opens new options for an extremely high burn-up of such wastes avoiding further reprocessing in between The encapsulation of the fuel kernel enables the burned fuel or wastes to be directly disposed, because this already represents an excellent barrier for final disposal beside its importance for the safety characteristics of the plant itself Thus, the technology of coated particles could represent a future alternative or complement to vitrification as well as to the burning of MOX fuel in the present configuration, possibly having economic advantages over multiple reprocessing and conditioning strategies

The market approach of standardizing modular reactors to power sizes that represent even a down-scaling compared to THTR and FSV keeps the technology within the frame of experience But this experience must be kept viable for commercial confidence in the new concepts[1]

Standardized modules also open up the potential of a quick learning curve and lower cost by the use of proven conventional equipment and by series production that has not really been introduced in the nuclear field up to now on a broader scale comparable to the competing conventional power plants It is not only important to focus on 'revolutionary' technical concepts but also on realizing new structures following general industrial trends

- A broadening of the market,
- a consensual set of user/licensing requirements,
- a fair and cost-effective sharing of the manufacturing efforts and
- a broader collaboration in the R&D field

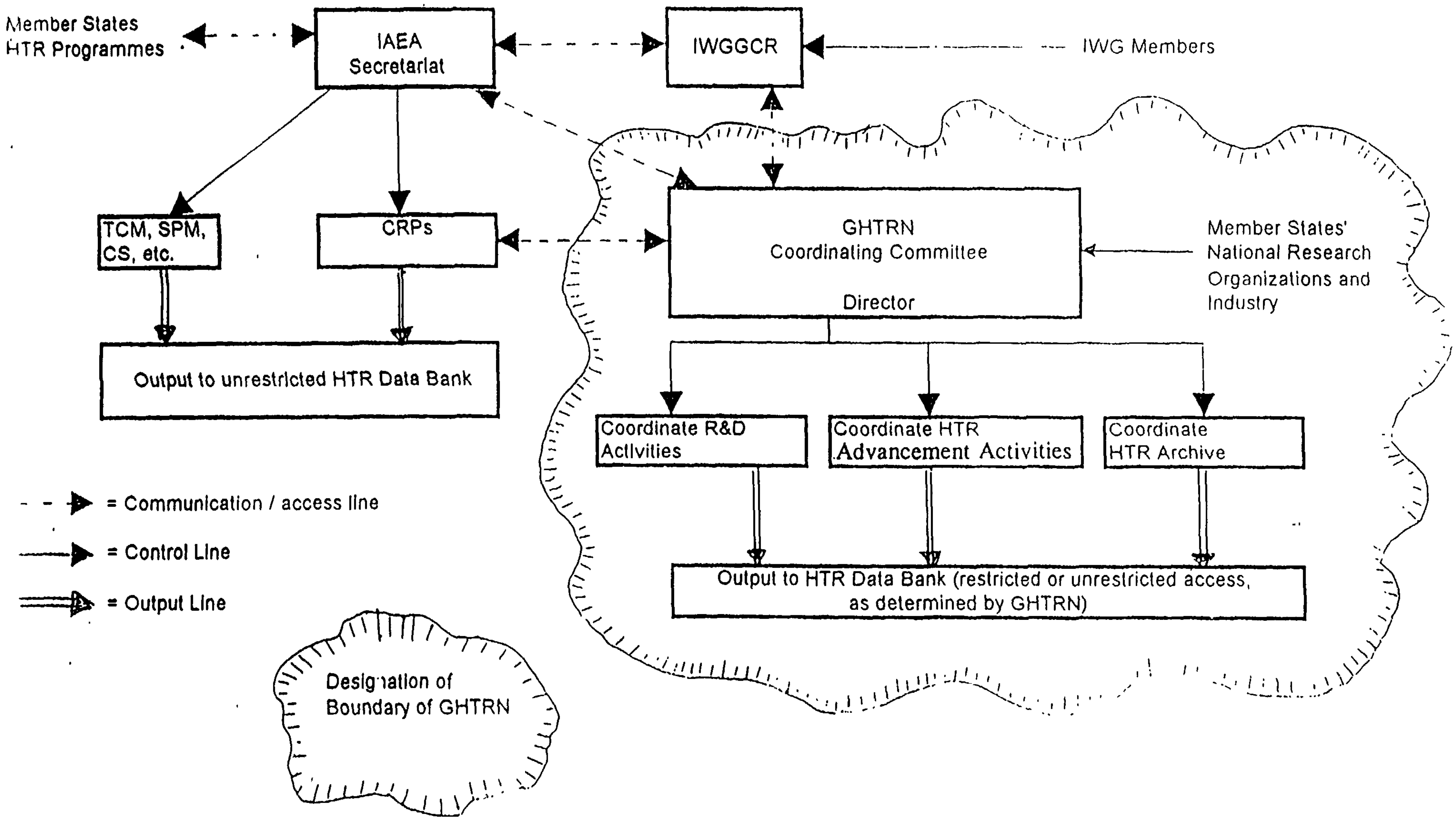
on an international or global basis could represent important preconditions for the introduction of the HTGR as an innovative nuclear reactor system

Aspects of nuclear safety, environmental protection, long-term safety of waste disposal and affordable, reliable energy supply are really global aspects and challenges The HTGR has the ability to respond to these requirements and it is an urgent task to find an appropriate and effective collaborative strategy

The Need for a Global HTGR R&D Network (GHTRN)

In actual fact, there is a broad HTGR and GCR know-how basis available in different countries in the world comparable or even superior to other innovative or evolutionary reactor developments, if it is be merged in a synergistic way on a global level Not only one new HTGR test reactor will start operation within the next few years but also a second, both together offering an exceptional R&D platform for the demonstration of HTGR characteristics under today's requirements of public acceptance and market conditions.

IAEA / GHTRN Interface Structure



Despite the small power sizes, these two reactors will provide a nearly full-scale demonstration of the passive safety philosophy of future HTGRs and for the first time high-temperature process heat will really be coupled out of a nuclear plant and will be combined with heat utilization systems representing new milestones in the on-going history of nuclear technology!

But only the symbiosis of existing and coming know-how represents a convincing basis for future commercial applications of this technology. Reduced financial/political support for nuclear development programmes enforces a tighter and more effective collaboration of the reduced crews who are still 'on board' globally sharing their know-how, infrastructures and facilities to achieve a common aim.

First discussions - on a brainstorming basis - have been recently held as an outcome of the 13th IAEA International Working Group on Gas-Cooled Reactors (IWGGCR) and this discussion will be continued to define a consensual approach to the general rules, purpose, scope, structures, contents and collaborative tools.

The first recommendations were to limit the scope to pre-competitive, joint R&D and project - independent basic technology aspects under the working title 'Global HTGR R&D Network (GHTRN)' under the auspices of the IAEA for the peaceful uses of nuclear energy. Present and future commercial projects can make use of the accumulated know-how and infrastructure within the GHTRN under conditions still to be defined, but supportive of the market introduction of HTGR technology.

It seemed to be appropriate to orientate the approach to the general structures and contracts of the 'International Thermonuclear Experimental Reactor (ITER)' Programme as it has a broad international partnership and also claims a long-term potential as an 'environmentally acceptable and economically competitive source of energy to be developed for the benefit of all humankind'.

On this basis, drafts for the General Charter of the GHTRN have already been formulated and the purpose and overall programatic objectives defined:

- to establish a coherent global R&D approach on a work-sharing basis to make best use of existing know-how, manpower, test facilities, test reactors, test object manufacturing capabilities etc.

- to provide a global information exchange, documentation and communication network on HTGR-related R&D

- to jointly review technological progress and to propose priorities as well as new R&D activites on a common basis

- to acquire funding for joint R&D programmes by national / international governmental bodies and private industries

- to serve as a stable, long-term R&D basis, independent of, and complementary to short-term HTGR projects and

- to generally promote the specific features and applications of HTGR technology worldwide

The IAEA is asked to play a strong role also in supporting the process of forming the GHTRN and to offer advice and provide infrastructures for different activities. Figure 1 illustrates preliminary ideas on interactions between the IAEA and GHTRN.

First steps under the scope of the GHTRN could be, for example:

- to establish a worldwide overview of existing and on-going HTGR-related R&D and to define common elements and work-sharing structures

- to perform joint test programmes and code validations using the existing test facilities and especially the test reactors (HTTR and HTR 10) to complement operating experience with HTGRs

- to establish a systematic documentation of existing and future know-how also introducing modern information techniques for ease of communication

- to create an information stock exchange as a pragmatic start-up and a know-how pool on general HTGR technologies as well as proposals for communication with future commercial projects

It seems to be essential to apply recent telecooperation and archiving methods from the very beginning as these innovative tools are the driving force for globalization trends in general and capable of handling more complex collaboration structures. They also enable an interactive growing process instead of a 'classical' centralized approach no one can afford any more. The HTGR should also be present on the Internet to provide up-to-date information on progress to the HTGR community as well as to decision makers in politics and industries for generating additional support and trust in the further development, benefits and chances of HTGR technology.

But the most important message of a 'Global HTGR R&D Network' on the common base technology would be a signal to the world that the HTGR community is united in support of the future of this reactor type.

The report represents the author's personal opinion and need not necessarily coincide with the official governmental / institutional view in all aspects.

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AN OVERVIEW OF THE LICENSING APPROACH OF THE SOUTH AFRICAN NUCLEAR REGULATORY AUTHORITY

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Abstract

This paper describes the approach adopted by the South African Nuclear Regulatory Authority , the Council for Nuclear Safety (CNS) in licensing nuclear installations in South Africa.

An introduction to the current South African legislation and the CNS philosophy pertaining to the licensing of nuclear installations is discussed.

A typical process for granting a nuclear licence is then presented.

The risk assessment process, which is used to verify compliance with the fundamental safety standards and to establish licensing requirements for a specific nuclear installation, is discussed.

Based on the outcome of this assessment process, conditions of licence are set down. The generic content of a nuclear licence and mechanisms to ensure ongoing compliance with the risk criteria are presented.

The regulatory process discussed in this paper, based on such a fundamental approach, may be adapted to any type of nuclear installation taking into account plant specific designs and characteristics.

1.0 CNS licensing philosophy

In terms of current national legislation, the Nuclear Energy Act No 131 of 1993, nuclear installations must be licensed by the national nuclear regulatory authority, the South African Council for Nuclear Safety (CNS). The legislation is broad empowering legislation, requiring licences to be subject to any conditions deemed necessary to ensure that the associated risk of nuclear damage remains consistent with the criteria laid down by the CNS.

In the early 1970's when the Koeberg project was in the planning stage, the forerunner to the Council for Nuclear Safety established fundamental safety standards [1] for the purpose of licensing Koeberg. The standards laid down were risk based and the licensing approach adopted was to require that:

- the design basis of the plant should respect prevailing international norms and practices and,

- that a quantitative risk assessment should demonstrate compliance with the CNS safety criteria.

As an example the design basis for the Koeberg nuclear power plant, was derived from the international norms and practices applicable to a standard 900Mwe Pressurised Light Water Reactor e.g US 10 CFR50 complemented by additional French rules prevailing at the time.

The risks to the operators and members of the public posed by the installation, related to a range of identified initiating events and accident scenarios, were required to be assessed by way of a probabilistic risk assessment and to respect the laid down criteria. The criteria were expressed in terms of individual risk and population risk with a bias against more severe accidents. The assessment was also required to demonstrate on a deterministic basis that public and occupational exposures arising from normal operations would be compliant with a dose limitation system employing dose limits of 250 μSv and 50 mSv respectively and a requirement to maintain doses as low as reasonably achievable. In addition to the requirement to demonstrate compliance with the laid down safety criteria, the operators are also obliged to establish and maintain a comprehensive emergency plan.

A similar approach has been used in the licensing of the facilities of the Atomic Energy Corporation.

The current applicable CNS nuclear safety criteria are documented in the CNS Licensing Document LD-1091 , which forms an integral part of the nuclear licence requirements. A summary of the current CNS nuclear safety criteria is presented in Table 1.

2.0 Typical CNS licensing process

The typical licensing process adopted by the CNS is indicated in Fig.1.

The first step triggering the licensing process is a formal application by a potential licensee, to the CNS, in accordance with the Nuclear Energy act.

Following the formal application by a potential licensee, a series of discussions take place between all the relevant parties and the CNS to identify the range and type of information necessary to ascertain the licensability of the activity. These discussions typically take place at the conceptual/design phase to identify the design, construction and operational requirements in order to satisfy the CNS safety criteria.

During this initial phase the potential licensee will be required to provide details in support of the proposed nuclear activity to be undertaken. Likewise the CNS will provide some appropriate licensing guidelines as indicated by A and B on Fig. 1. This will inter alia include the following:

1. the fundamental safety criteria

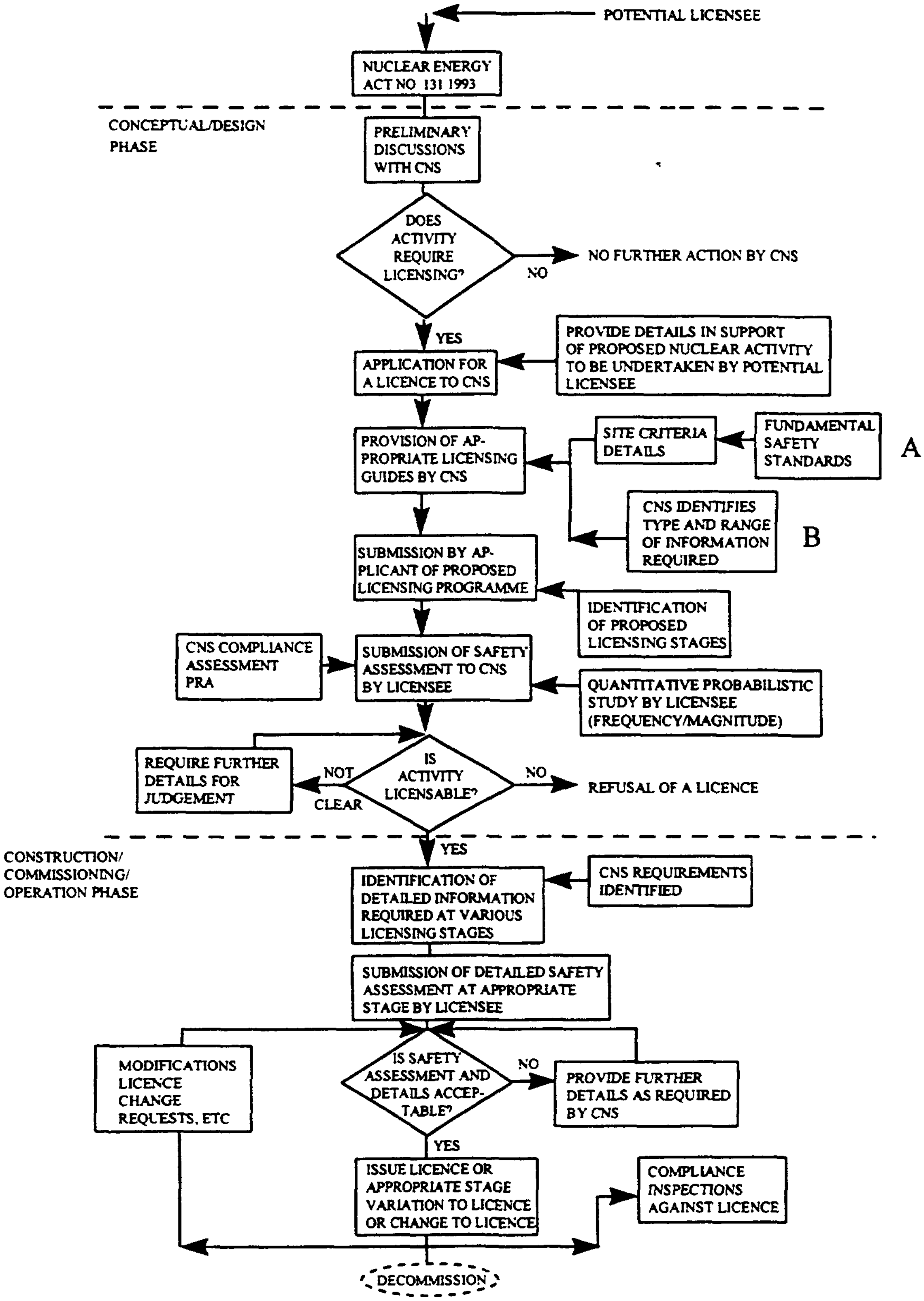


Figure 1 CNS licensing process

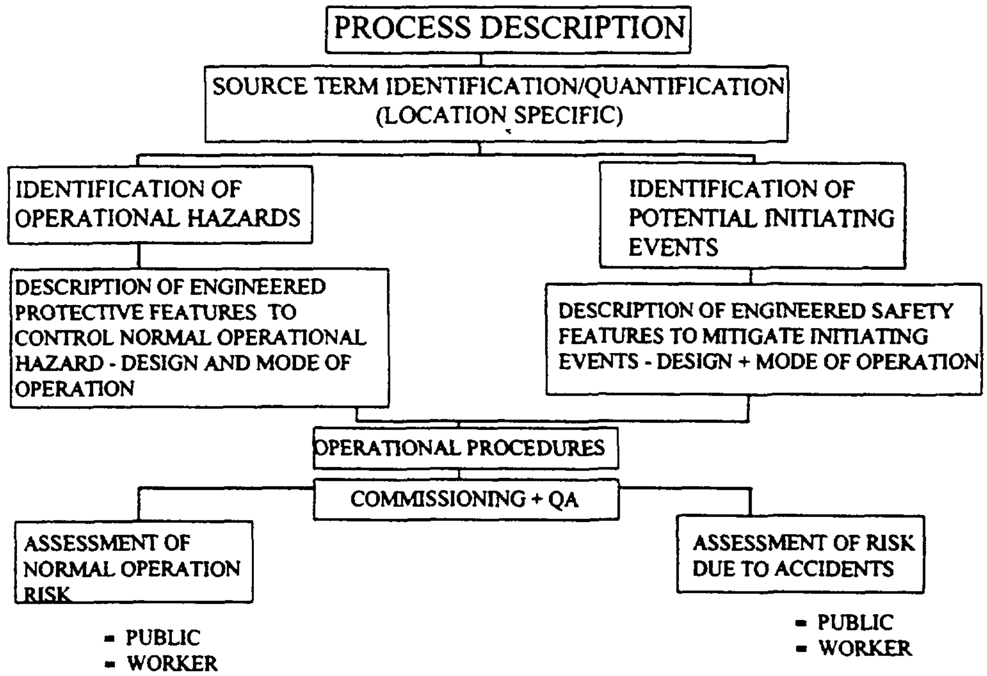


Figure 2 Risk Assessment Method

2. the identification of the type and range of information required to assess the risk to operators and members of the public under both accident conditions and normal operations. Fig. 2 illustrates the method used to assess the risk.

Following the submission by the applicant of a proposed licensing programme, which identifies the various licensing stages, an intent document in the form of a Preliminary Safety Assessment Report (PSAR) will be submitted to the CNS. This will, inter alia, include a quantitative probabilistic study.

The review of the PSAR and other supporting information will be used to ascertain whether the activity is licensable. Typically this will mark the end of the conceptual phase and the start of the remaining licensing stages leading to eventual operation.

For complex projects such as the licensing of the Koeberg Nuclear Power Plant, the licensing process used by the CNS is a multi stage process. Typical primary licensing stages are:

1. Nuclear installation siting
2. Design of the installation
3. Construction and commissioning
4. Radioactive material/ nuclear fuel on site
5. Start up/criticality (where applicable)
6. Power raising (where applicable)
7. Plants operations
8. Decommissioning

Each stage of the process is supported by a safety assessment, including a quantitative risk assessment to demonstrate to the satisfaction of the CNS that the relevant CNS safety criteria are met.

3.0 Safety assessment philosophy

As indicated above a safety assessment is required for the approval of each stage of the agreed licensing programme. A typical safety assessment is based on three major elements which are closely interdependent:

1. Scientific and engineering principles
2. Radiological protection principles
3. A quantitative risk assessment

The type of information required at the various stages of the safety assessment covers a wide range of topics which inter alia include the following :

- Site criteria details
- Seismic properties, weather statistics, demographic predictions
- Design drawings, specifications and supporting calculations
- Manufacturing/construction programmes and procedures
- Quality assurance and control programmes
- Commissioning plan
- Operating and maintenance procedure including accident management
- Testing regime
- Organisational structures including staffing and training
- Radiological protection programme including waste management
- Emergency planning programme
- Physical security

3.1 Scientific and engineering principles

The licensing requirements, related to the engineering principles, are that the design basis and operating rules of the plant respect prevailing international norms and practices.

The basis for the engineering assessment is to respect the principle of defence-in-depth. Defence-in-depth is a concept which applies at all stages in the life of the facility from design to decommissioning/dismantling. This principle consists of recognising technical, human or organisational failures and to guard against them by successive lines of defence.

There are several categories of lines of defence which can typically be divided as follow:

- **Prevention** of failure
- **Monitoring or detection** to anticipate/detect failure
- **Means of action or mitigation** to mitigate consequences of failure

Likewise there are several lines of defence which are normally provided for each category. Some typical examples are given below:

- Built in multiple barriers to limit the release of radioactive material into the environment
- Adequate safety margins in the design basis
- Selection of adequate protection and safeguards systems
- Diversity and redundancy of equipment
- Choice of adequate materials and equipment
- Adequate manufacturing requirements
- Adequate rules for monitoring and detection
- Adequate operating principles
- Adequate organisational structures etc..

The relevant applicable measures, ensuring defence in depth, are defined and implemented respecting acceptable rules, codes and standards based on international experience.

The number and properties of the barriers or lines of defence is commensurate with the risk posed by the installation and also with the uncertainties associated with the determination of the risk.

3.2 Radiological protection principles

The safety assessment associated with any facility or practice is required to address four areas of risk which pertain to both the operator and members of the public under conditions of both normal operation and accidents. The principles of radiation protection should be considered to be related to that part of the assessment which addresses normal operation. The two main principles of radiation protection apply here having assumed that the practice has been justified.

- All exposures must remain below the annual dose limit
- All exposures must be kept As Low As is Reasonably Achievable (ALARA)

Notice should be taken of the reference to dose rather than risk in the assessment for normal operation where dose is regarded as a surrogate for risk. Furthermore, the circumstances under which this dose must be determined, must be limiting so that operational flexibility is allowed and that it would be unlikely that the relevant limits would be transgressed. In this way, the assumptions of the safety assessment for normal operation are deterministic as distinct from the probabilistic approach adopted for the assessment of risk due to accidents.

3.2.1 Dose Limitation To The Operator

In respect of dose limitation to the operator, it must be demonstrated that the annual individual dose limit for the operator of 50 mSv y^{-1} will not be exceeded. No assessment can really predict what the annual exposure to any given operator will be due to the uncertainty in how much time will be spent in each area and the attendant uncertainty in doserate at any given time. Therefore, the assessment must demonstrate that a certain strategy of radiation protection provisions is in place which encompasses both design and operation.

In terms of design, the facility is required to be segregated into zones of defined ambient doserate, airborne and surface contamination on the basis of the location of radioactive equipment, and limiting leakrates and source terms. Shielding and ventilation assessments are conducted to identify the classification required for any given zone. Areas where the radiological hazard is subject to change are provided with installed radiation monitoring to warn operators, and additional design features for contamination control are implemented mainly on the basis of the need for maintenance and In Service Inspection.

The strategy of provisions which must be implemented operationally is known as the operational radiation protection programme. This programme, must, in essence, be able to achieve certain fundamentals;

- The recognition and quantification of all potential and actual hazards during normal operation
- The prospective implementation of control measures to ensure compliance with the dose limitation system
- The recognition and quantification of radiological hazards associated with abnormal situations and the implementation of appropriate radiological controls

The strategy of provisions which satisfies these principles will therefore identify and review the magnitude of operator exposures on a continuous basis such as to ensure compliance with the annual dose limit.

3.2.2 Dose Limitation To Members Of The Public

The limitation of dose to members of the public requires that the highest exposed member of the public would not incur an annual dose in excess of 250 $\mu Sv\ y^{-1}$. The two stage process of design assessment and operational control is implemented to ensure compliance with the dose limit to members of the public.

In terms of the design stage, the annual liquid and gaseous effluent discharges are predicted based upon the process of radioactive effluent and waste treatment intended to be followed. From such an assessment, which traces the migration of radioactivity from its point of origin, to point at which it will be discharged, all radionuclides and their associated discharge quantities can be identified using limiting assumptions which are chosen judiciously in order not to compromise operational flexibility. Dose conversion factors for all identified radionuclides discharged into both transport pathways can then be calculated by use of liquid and atmospheric dispersion codes, and the use of site-specific input data. The impact of the discharges can then be assessed at the design stage and compared to the annual dose limit to determine whether they are acceptable or whether modification of the waste treatment systems or processes is required.

Once consolidated into an agreed figure for all radionuclides of concern, the control of annual discharges of radionuclides themselves provide a surrogate for the control of dose. The operational programme which is implemented in this regard is known as the effluent management programme and is closely linked with the design assessment. Radiological effluent which originates from systems which are potentially highly contaminated are discharged in batches only after radionuclide-specific analysis. Where the potential is lower, discharges are allowed on a continuous basis and sampled, for confirmation, as a composite. In this way, dose to members of the public is also assessed and reviewed on a continuous basis such as to ensure compliance with the annual dose limit.

3.3 Quantitative risk assessment principles

The acceptability of the design basis and general operating rules is evaluated by mean of a quantitative risk assessment, performed by the licensee, to verify that the proposed installation complies with the fundamental safety criteria of the CNS.

The philosophy adopted by the CNS is that a risk assessment is necessary to maintain a clear perspective on the fundamental objectives of nuclear regulation, namely to protect the public and plant personnel from the risk of nuclear accidents, and a means of assessing the extent to which these objectives are being achieved.

TABLE 1. Fundamental safety standards of the CNS

	ACCIDENTS	**NORMAL OPERATION**
ASSESSMENT TYPE	**PROBABILISTIC**	**DETERMINISTIC**
POPULATION		
AVERAGE:	10^{-8} fatalities/person/ annum (per site)	Controlled by limitation of individual risk
INDIVIDUAL:	5×10^{-6} fatalities/annum	250 μSv/annum
PLANT PERSONNEL		
AVERAGE:	10^{-5} fatalities/annum	Risk controlled by the operational radiation protection programme
INDIVIDUAL:	5×10^{-5} fatalities/annum	

For this reason the licensee of a nuclear installation is required to perform and maintain a quantitative risk assessment, approved by the CNS, to demonstrate compliance with the fundamental risk criteria laid down by the CNS. These include the probabilistic criteria given in table 1, and a bias against larger accidents applicable to risk to the public. These criteria are applicable to the licensed site, including all activities on the site and interactions between them.

Regulatory control on the development, updating and application of the PRA is exercised by means of the licensing document LD-1091 to ensure that the PRA is maintained valid, reflects the current status of the plant, includes local and international experience feedback, and is used for decision-making on safety related matters including plant safety improvements, risk management, limitations on urban development and emergency planning.

In addition to the use of risk criteria, safety indicators at different levels (such as core damage frequency, containment failure probability etc) are used for decision making in support of the principles of defence-in-depth and ALARA.
As regards risk to the public, the principle stages of a risk assessment for a nuclear power plant are illustrated in figure 3.

As the risk assessment is formally approved by the CNS and is effectively controlled by the nuclear licence, it serves as a valuable framework for focused technical discussions with the licensee on safety related matters.

Historically, numerous requirements have been imposed on licensees based on the findings of the various plant-specific risk assessments. These include hardware modifications, procedural and management improvements. In addition, specific licence conditions have been established or relaxed on the basis of a risk assessment. The risk approach to nuclear regulations in South Africa is presented in more detail in [2].

4.0 The Nuclear Licence

Based on the outcome of the safety assessment, a nuclear licence is granted. Conditions of licence are generated to ensure compliance with the laid down safety criteria.

According to the multi stage process, described in fig.1, a variation of a specific site licence will be issued at each stage following the review of the specific stage safety assessment. Conditions of licence are then amended accordingly.

A typical licence for a fully operational nuclear power plant will include the following conditions:

- a valid plant description and configuration to be maintained together with a modification control procedure;

- the maintenance of a valid and updated risk assessment;

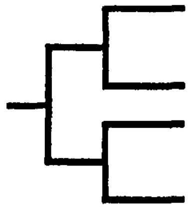
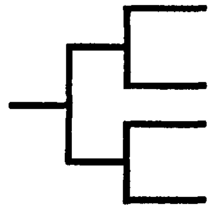
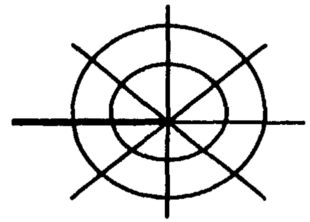
	PRE-CURSORS	INIT EVENTS	LEVEL 1 EVENT TREE	PDS	PDS FREQU.	CONTAINMENT EVENT TREE	BIN	BIN FREQU.	
		~ 50 IEs		72 PDSs			18 BINS		
Safety Indicators					CMF			CCFP	IND. RISK OPER. RISK 
Criteria/ Goals					Derived Goals			Derived Goals	CNS Risk Criteria
INPUT	Occurrence Data	Fragility Data Component Reliability Human Factors External Data Transient analysis		Acc. Procs. OTS Maint. Sched. Outage Planning		Severe Accident Management Guides Human Factors Severe Acc. Issues			Emergency Planning Urban Development Siting of Facility

FIGURE 3. PRA stages, criteria and data

- establishment and compliance with operating technical specifications;
- normal operations, incident and accident procedures;
- an in-service inspection programme;
- a maintenance programme;
- an operational radiation protection programme;
- a waste management programme;
- an emergency plan;
- a routine and occurrence reporting programme;
- a quality management programme;
- reports, approvals

Changes to the issued and approved licence, or addition to an existing installation, must be requested by the licensee, for the approval of the CNS. This request must be supported by a safety assessment including a risk assessment, which is based on the principles detailed above.

4.1 Compliance with risk criteria

Once the licence is issued it is necessary to ensure that the conditions of licence are complied with. Although it is the responsibility of the licensee to ensure that this is accomplished, the CNS generates an independent compliance inspection programme which covers most of the licence conditions.

4.1.1 Licensees responsibilities

In conformance with the conditions of licence, a monitoring programme is implemented by the licensee, and regularly assessed by the CNS, to verify compliance with the risk criteria on an-on going basis. The major elements of this programme are described below:

- ♦ Plant Operational Feedback
    - In Service Inspection Programme (ISIP)
    - Maintenance and Testing Programme
    - Occurrence/Incident Reporting and Analysis including trending

- Hardware and Procedural Modifications
- Emergency Planning Exercises
- Quality Assurance Audits
- International Experience Feedback Analysis

4.1.2 CNS surveillance and compliance inspection programme:

Complementary to the licensee's monitoring programme, a comprehensive independent surveillance and compliance inspection programme is developed by the CNS to verify compliance with the licence requirements and to identify any potential safety concerns. Most licence conditions are subjected to the inspection programme, which is implemented by the CNS staff. Some of the major topics covered by this programme are :

- occurrences/incident assessment and trending
- monitoring of selected safety related systems e.g ventilation, fuel handling equipment, electrical power supplies, effluent discharge control systems etc
- compliance with the installation operating technical specifications
- operational radiological protection e.g occupational and public exposures, environmental monitoring
- emergency plans
- maintenance and ISI activities
- physical security
- quality assurance audits
- staff training
- reactor plant operator examination and licensing

Complementary to this on going safety review, the CNS also requested the Koeberg nuclear power station to carry out a periodic safety review. The rationale, objectives and methodology of this periodic safety review are presented in detail in [3].

5.0 Conclusion

This overall licensing process has been successfully applied to the licensing of complex projects such as the Koeberg nuclear power station, and also to other nuclear installations in South Africa e.g the Atomic Energy Corporation (AEC).

The CNS view is that this approach, which maintains a clear logical link between the fundamental safety criteria and the plant specific design and characteristics, may be applied consistently to reactor designs other than light water reactors.

REFERENCES

[1] SIMPSON, D.M., et al., Population Dose Evaluation and Standards for man and his Environment (Proc. Symposium Vienna, 1974) IAEA-SM-184/31

[2] HILL, T.F , Risk Approach to Nuclear Regulations in South Africa presented at the "Executive Meeting on Risk Based Regulations and Inspections", Stockholm, Sweden , 12-14 August 1996.

[3] CLAPISSON, G.A., et al., Activities of the South African Regulatory Authority in the Safety Review of Nuclear Power Plants presented at the International Symposium on Reviewing the Safety of Existing Nuclear Power Plant (Vienna 1996) IAEA-SM-342/68

DEVELOPMENT OF THE PEBBLE BED MODULAR REACTOR PLANT IN SOUTH AFRICA

XA9846678

TECHNICAL STATUS OF THE PEBBLE BED MODULAR REACTOR (PBMR-SA) CONCEPTUAL DESIGN

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Abstract

The reactor study is well underway seen from a broad spectrum of disciplines and technology. The objective power output with a high efficiency direct cycle power conversion unit remains promising after compiling the first critical analysis of the core and the power conversion unit. The stability and controlability of the system are demonstrated by the engineering simulator.

The main system and components are basically specified for costing purposes. A first plant layout has been completed demonstrating the positions of main components, personel movement, installation methods for large components etc.

A cryptic report style presentation includes study opjectives, indicating quiding documents, giving an overwiew of design and analyses work done as well as a few sketches and diagram are included in this paper. Most of these sketches and diagrams are small replicas of large drawings and are therefore not readable but can be used as references.

1. OBJECTIVE OF STUDY (CONCEPTUAL DESIGN)

The following technical and economical questions have to be answered before any recommendations can be made.

Technical Feasibility:

- Can we achieve a sufficiently high reactor power output within the required safety criteria?
- Is the Brayton Cycle with three separate shafts stable? (See attached .diagram (1))
- Can we fullfill the clients operational requirements (operating profiles)?
- Can we prove high system efficiency claimed for HTGRS worldwide?
- Is this reactor as safe as internationally claimed (Is it inherent-, passive- and catastrophy free/safe, is it licensible)?
- Is the technology basis, internationally seen, to such a standard that basic research and development is not required and that a first-off full scale plant can be constructed directly as reference plant?
- Is modularization possible?

Economic Feasibility:

- Can we achieve the capital cost objectives for *first-off, pre-production* and *production* units?
- Is operational cost within the owners requirement specification?
- Is the owners investment protected by the said safety characteristics?

2. WORK PLANNING

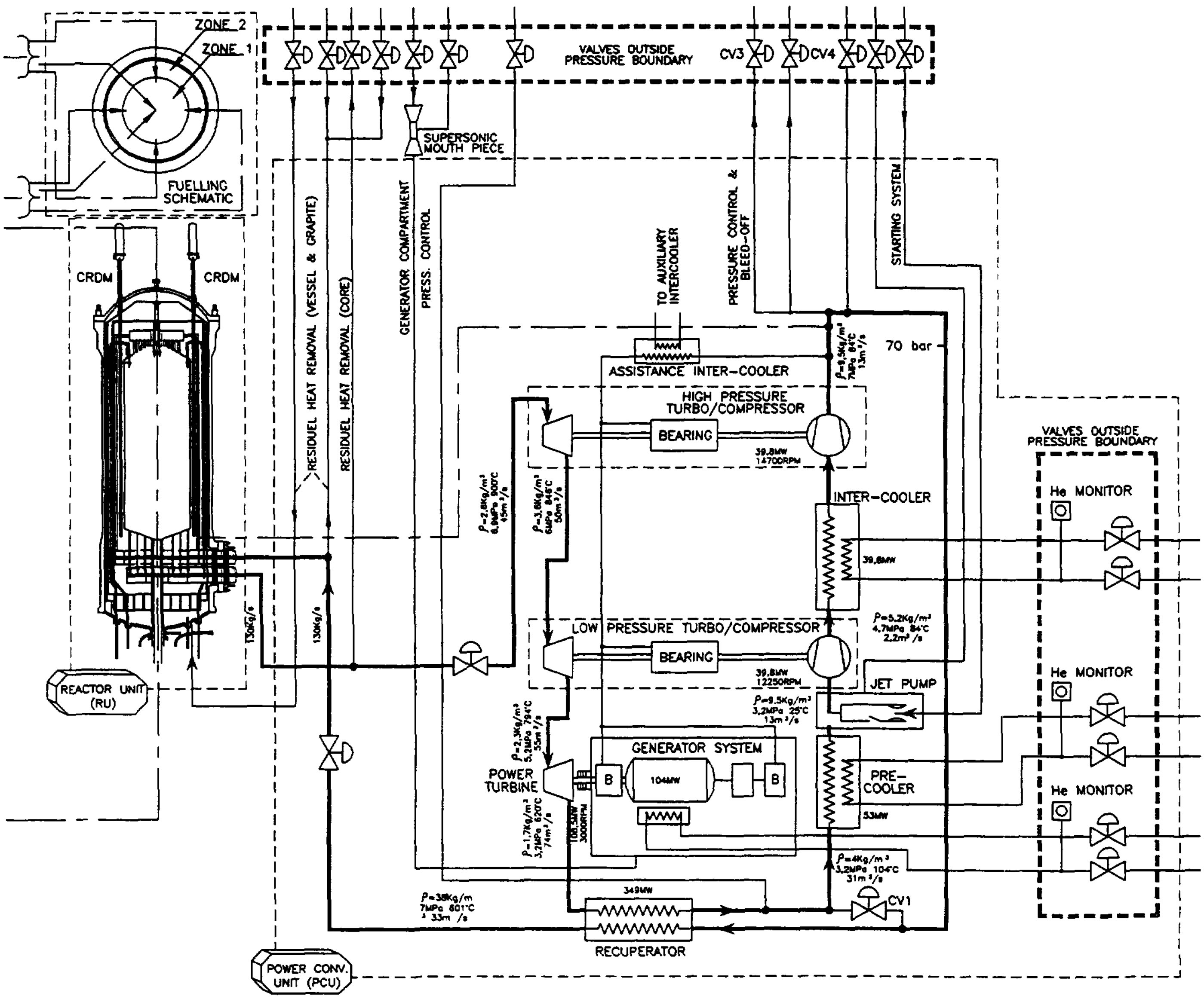
The execution of work to meet the study objectives is basically captured in three diagrams, i.e.

***The Acquisition Plan** indicating:* (See attached Figure 2)

◊ The global plan (long term planning)

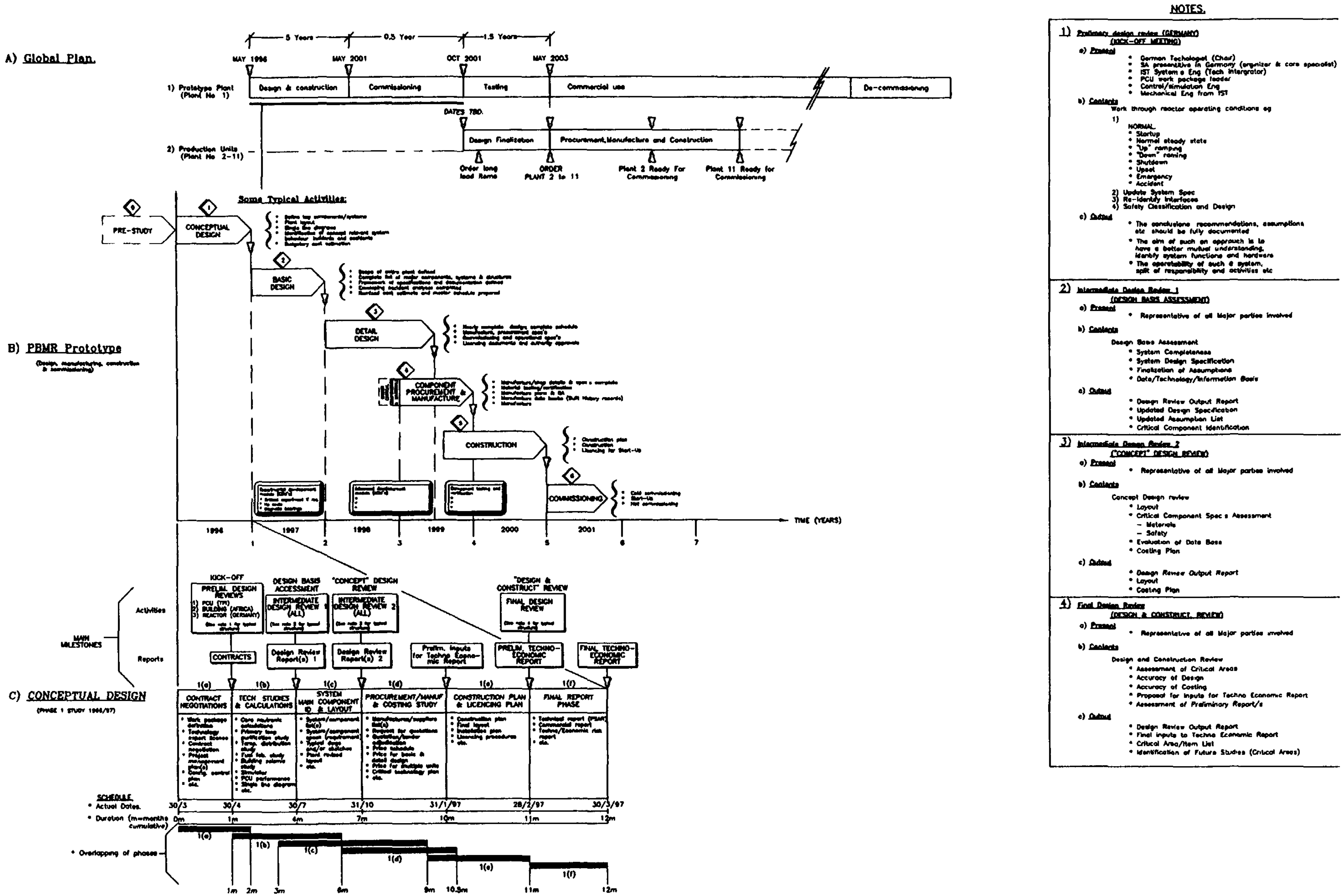
◊ The prototype plant design and construction plan.

◊ The detailed conceptual design (1996 study) activities including milestones, milestone documentation and design reviews. (See attached Figure 3)



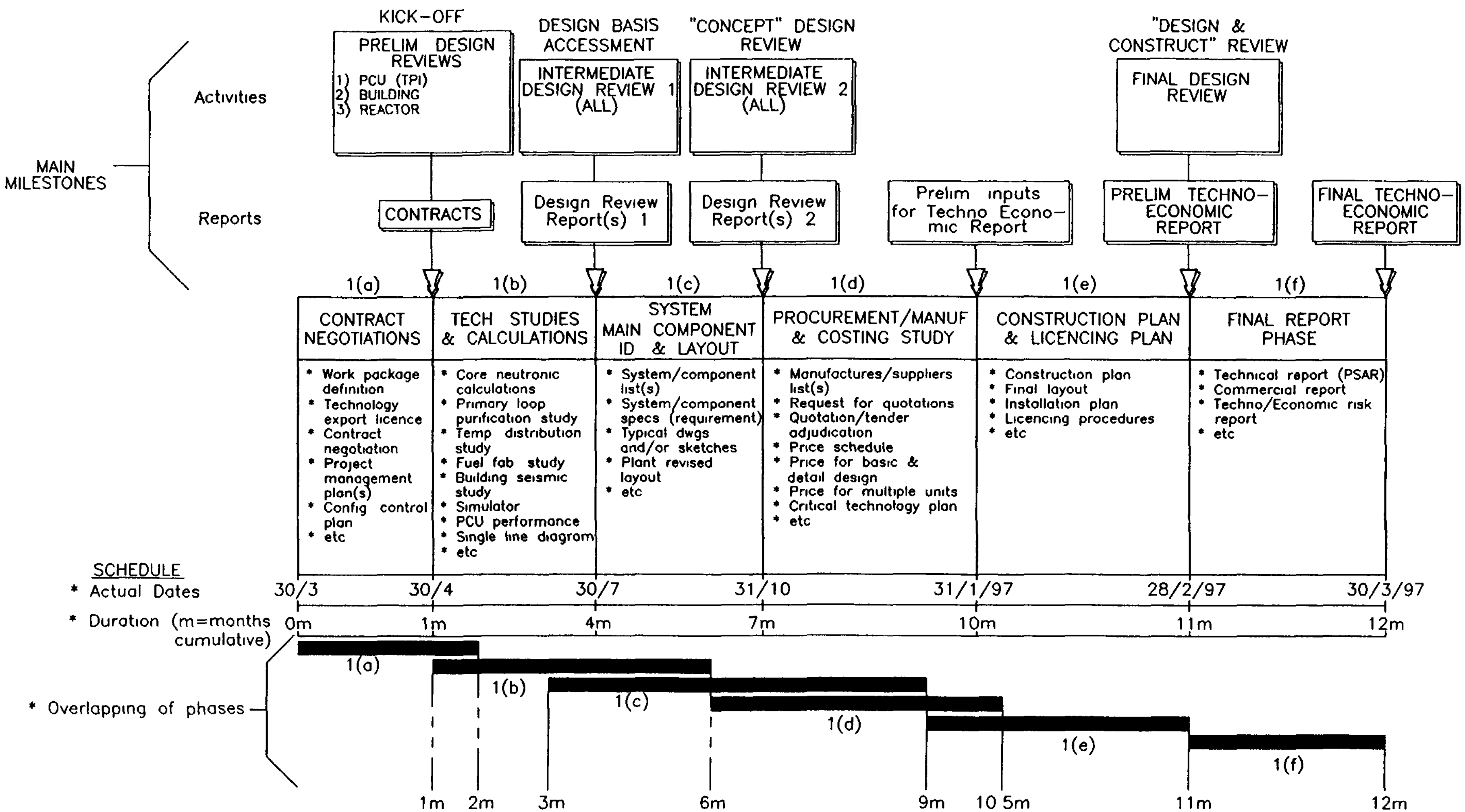
MAIN POWER SYSTEM SCHEMATIC

FIG. 1



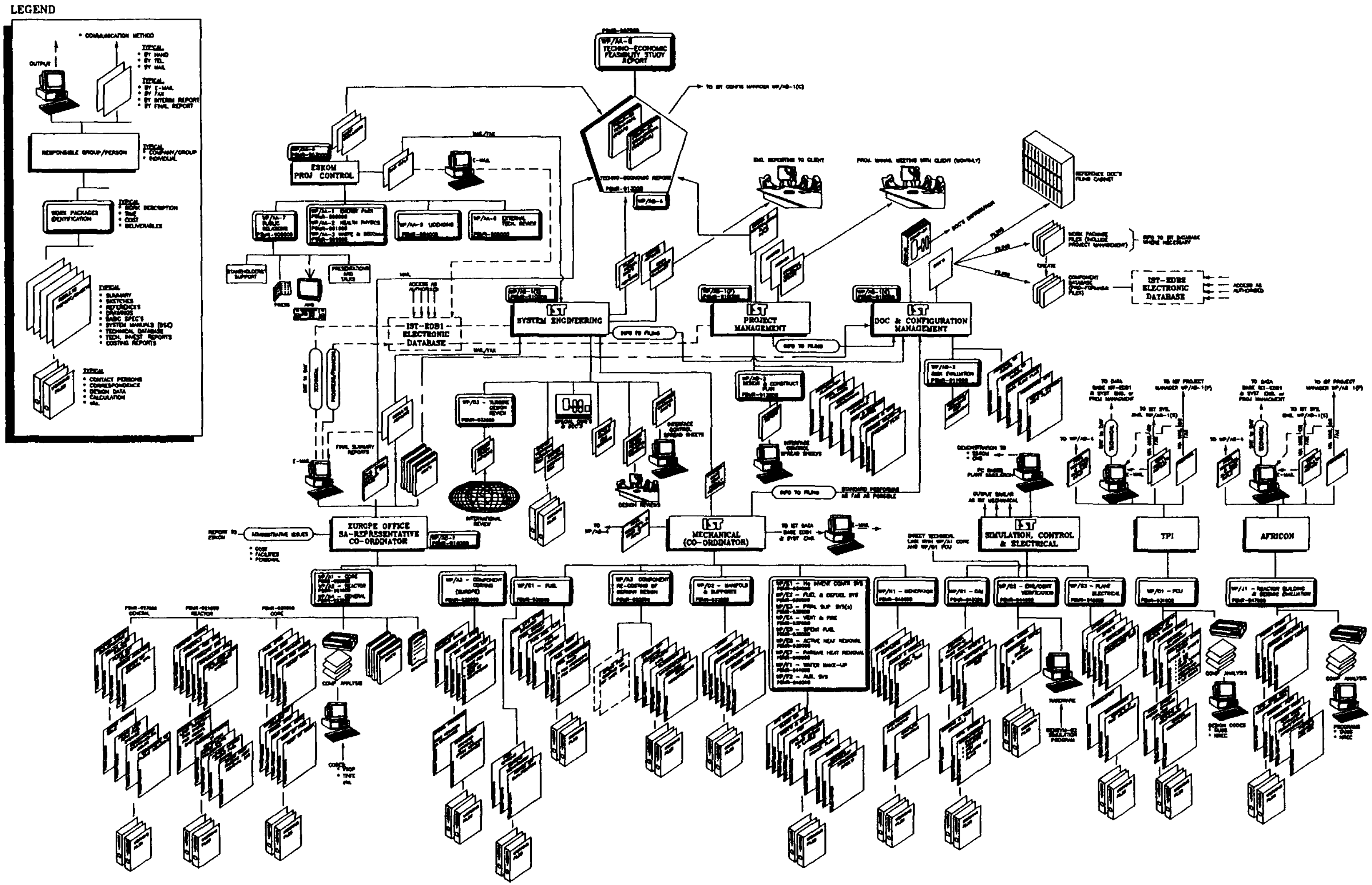
PBMR ACQUISITION PLAN

FIG. 2



PBMR CONCEPTUAL DESIGN PLANNING

FIG. 3



INFORMATION AND DATA FLOW PROCESS PLAN for PBMR PHASE 1 STUDY

FIG. 4

- ***The Information and Data Flow Process Plan** indicating:*
  (See attached Figure 4)

  ◊ Flow of specialist data through the Systems Engineering and configuration management to the Final Report.

  ◊ The use of communication methods e.g. paper copies, E-Mail, and Internet files.

  ◊ Day-to-day database and system design data (System Manuals).

  ◊ Licensing and international design reviews, etc.

- ***Work Breakdown Structure** indicating:*

  ◊ The Architect Engineering responsibilities.
  ◊ Other work packages and responsible parties.

3. THE WORKING TEAM

Presently a wide range of engineering disciplines are involved (*excluding* major manufacturing companies contribution):

Discipline	LOCAL		OVERSEAS CONSULTANTS
	Full time	Part time	
• Management, Promotion and Marketing	1	2	2
• Reactor Physics includes core design, shielding etc.	1		5
• Mechanical Systems & components	4	3	4
• Electrical and Control	2	2	
• Fuel		3	2
• Simulation		2	
• Special Materials Studies	2	2	2
• Reactor Building		3	
• Safety and Licensing	1	2	
• Documentation/Data Control	1	1	
TOTAL	12	20	15

A total of 47 technical and management, plus 3 administrative people involved, of which all technical people have vast nuclear experience.

4. PLANT IMPORTANT FEATURES AND SPECIFICATION; AN ABSTRACT FROM OVERALL DESIGN BASIS (See attached Figure 5)

CHARACTERISTIC (OBJECTIVE)	STATUS/REMARKS
• 220 MW_{TH}	Achieved
• Pebble Bed Core	Accepted
• Fuelling scheme ◊ on-line recycle ◊ on line or off-lineOTTO-P, use burnable poisons	• Recycle scheme accepted as reference • OTTO-P has good potential
• Modular design to suite most sites	A compromised design approach is followed. eg. depth below ground level, seismic spectra etc.
• Net power range 0 to 100%	Full range load following possible (specified power step and ramping under discussion) .
• Spent fuel remains inside reactor building.	• Sufficient space for 20 full power years. • Passive, natural air cooling to be analysed still.
• Passive residual heat removal.	• Possible by natural air circulation. • Possible by water/air system. • Reactor vessel maximum temperature, and alllowable temperature gradients through the wall under consideration.
• No active safety related systems required. (Therefore a new definition for Reactor Protection System).	Believing the strong negative temperature coefficient of reactivity and believe the core neutronic design codes, e.g. the temperature of the fuel can never exceed the maximum safe allowable temperature under most extreme postulated accidents.
• Core outlet temperature - 900°C.	Limit by turbo-machinery maximum allowable temperatures (specific technology limitation).
• Core inlet temperature ≈ 560°C for optimized cycle efficiency.	This temperature exceeds allowable temperature for RPV materials (creep). The reactor coolant flow path design allows for this

	high temperature return gas.
• Reactor vessel operates at low temperature. (120°C)	Introduce recuperator bypass for cooling purposes. This results in poor efficiency at low inventory operation.
• System maximum pressure = 70 bar	Followed world trends. Cost vs. pressure optimization is not done.
• Minimum absolute pressure variation in the Alternator compartment.	Achieved by orientation of power turbine to ensure minimum abs. pressure at shaft Labyrinth seal. This principle may change because of other considerations.
• All magnetic bearings.	• He cooled • No oil. • Low maintenance • Behaviour under environmental conditions to be investigated.
• Water ingress into reactor impossible.	• Cooling water pressure always lower than primary loop helium pressure under operational conditions. • Total water inventory can be collected in the heat exchanger pressure housings.
• Easy access to first turbine and generator.	Housed in pressure containing housings (bells) which can be removed fairly easy. Couplings to pipes under investigation, e.g. for ease of removal.
• Design approach based on "many-off" and continuous production.	
• Short construction time.	Equipment grouped as modules, mounted on ski's and factory tested. Civil construction after installation of equipment should be avoided.
• Installation and maintenance equipment to be shared between units.	Capital cost for these equipment shared between units.
• Reactor building to be a low pressure confinement	• A proper containment building is not required.

	• Break disks with large flow filters provided (Provide for blowdown accident) • Building act as energy absorber for aeroplane crash. Penetration of larger/faster aeroplane components possible. The reactor pit is the ultimate barrier.
• A gaseous waste system not required	This topic is under discussion.

5. MAIN GUIDANCE DOCUMENTS

Apart from the Work breakdown structures, Hardware breakdown structures, time schedules etc., the following technical documents were compiled:

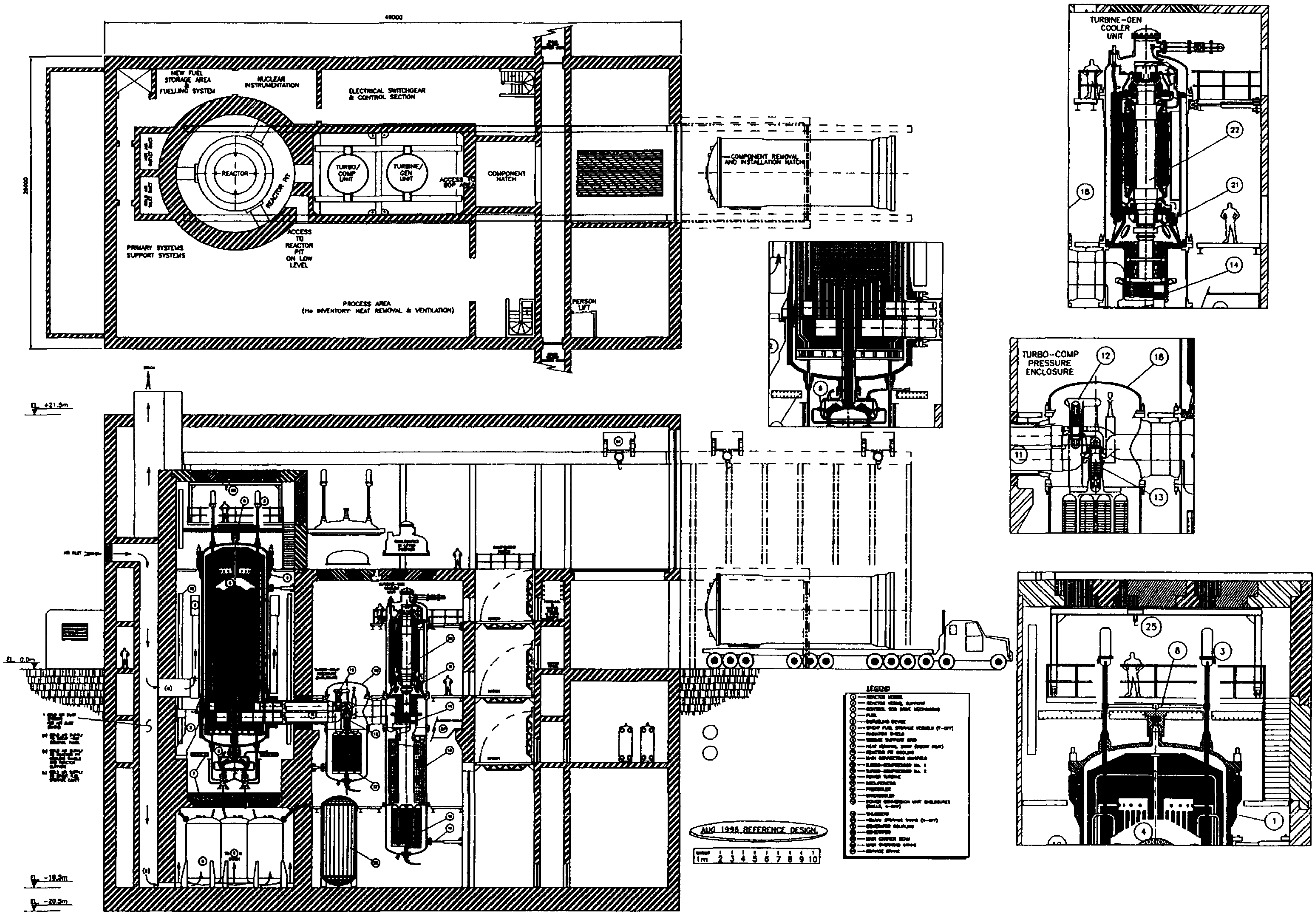
TOPIC	PURPOSE	STATUS
Technical Description.	Consolidation of 1995 discussions.	Outdated.
Quality Assurance Plan for the PBMR-SA.	Acquisition, information and data flow plans for the conceptual design study.	Active.
Overall plant Specification and Interface Control Document.	Quick reference of the plant characteristics and change control thereoff.	Active.
Index of Techno-Economic Feasibility Study Report. Volume 1 - Commercial Volume 2 - Technical Description		Done
Design bases for most of the systems and main components.	Input for conceptual design of systems and components.	Done

6. PLANT OVERALL PROCESS AND LAYOUT.

- Overall process diagram has been developed. (See attached Figure 6)

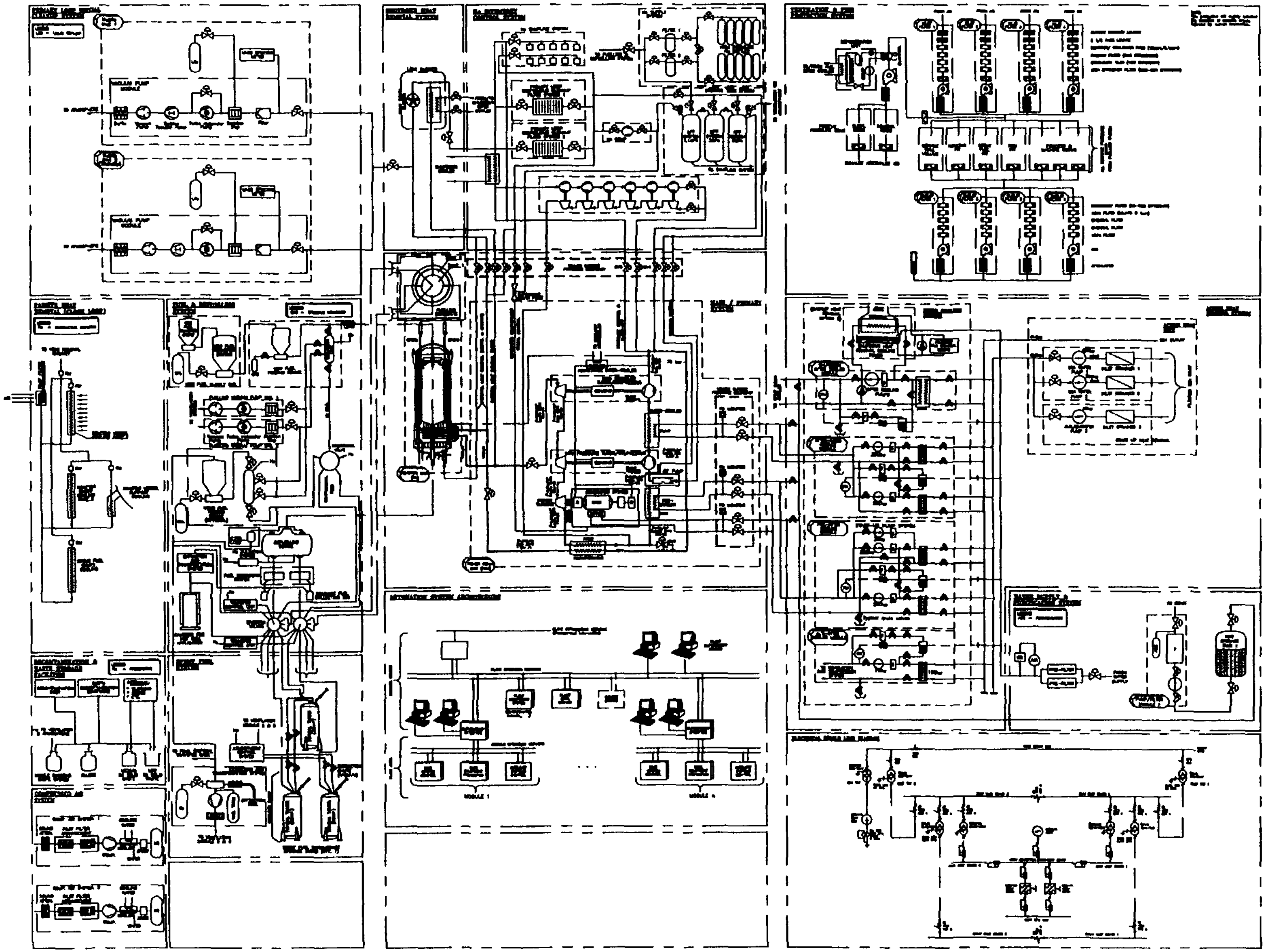
- The systems have been functionally grouped for example

  a) ***Active Heat Removal*** includes all active heat removal functions.
  b) ***Main Power System*** includes all primary components needed for prime power production (this includes the alternator).
  c) ***Helium Inventory Control System*** includes helium make-up, primary loop cleanup and inventory control etc.).



PEBBLE BED MODULAR REACTOR CONCEPT (100MWe)

FIG. 5



REVISION "B" NOTES:

THIS DIAGRAM IS BASED ON DISCUSSIONS DURING THE PRELIMINARY DESIGN JUNE 1998

CHANGES TO THE PREVIOUS ISSUE:–

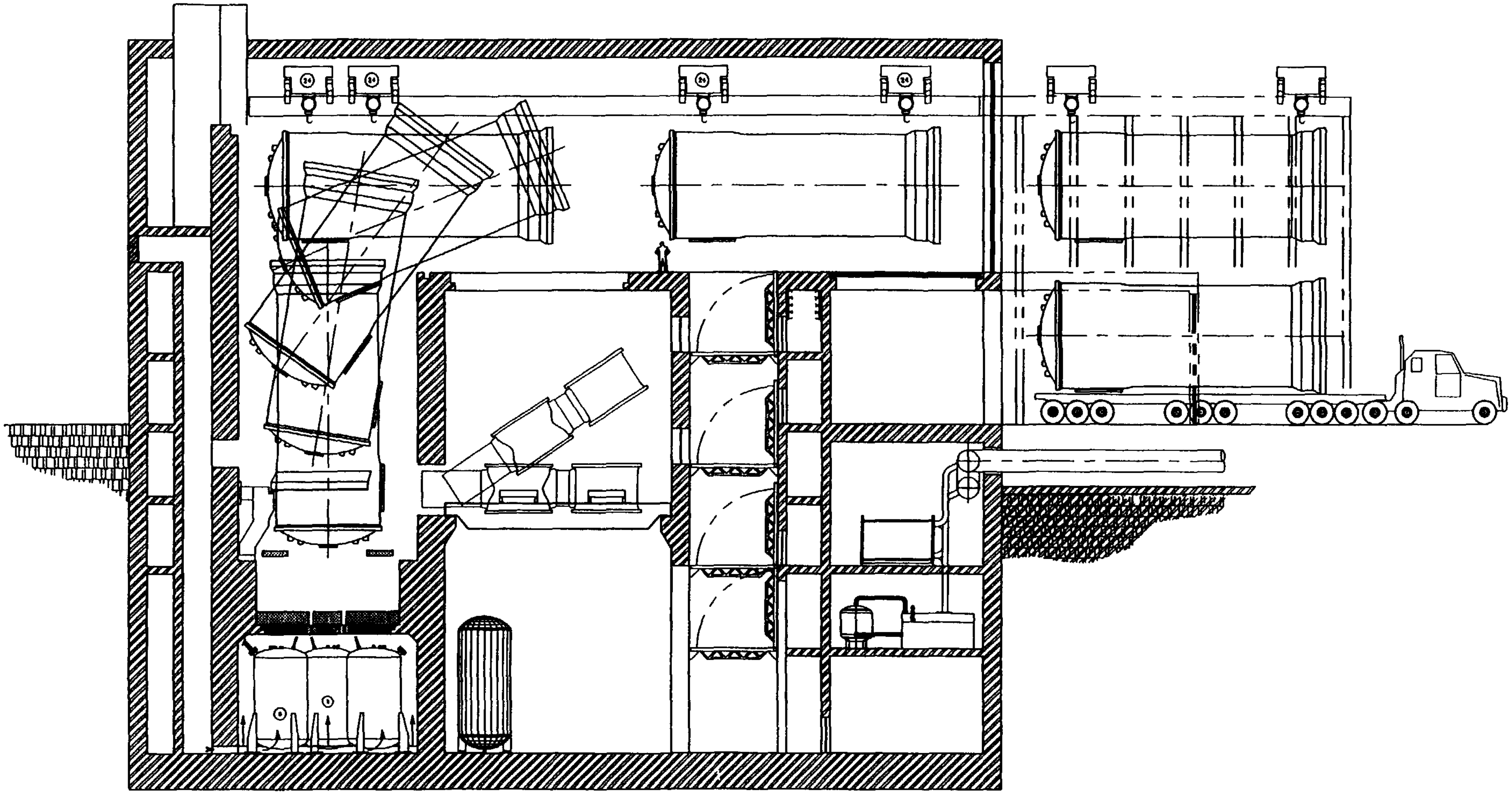
- RECYCLING OF FUEL (SEE ZONE 2 FUELLING SCHEME)
- MULTIPLE SMALLER SPENT FUEL TANKS
- JET PUMP IN LOW PRESSURE POSITION
- He INJECTION IN GENERATOR COMPARTMENT
- REACTOR PRESSURE ZONE SEPARATION

REVISION "C" NOTES:

UPDATED He INVENTORY CONTROL SYSTEM

PBMR – OVERALL SCHEMATIC DIAGRAM

FIG. 6



PBMR-SA PEBBLE BED REACTOR VESSEL INSTALLATION

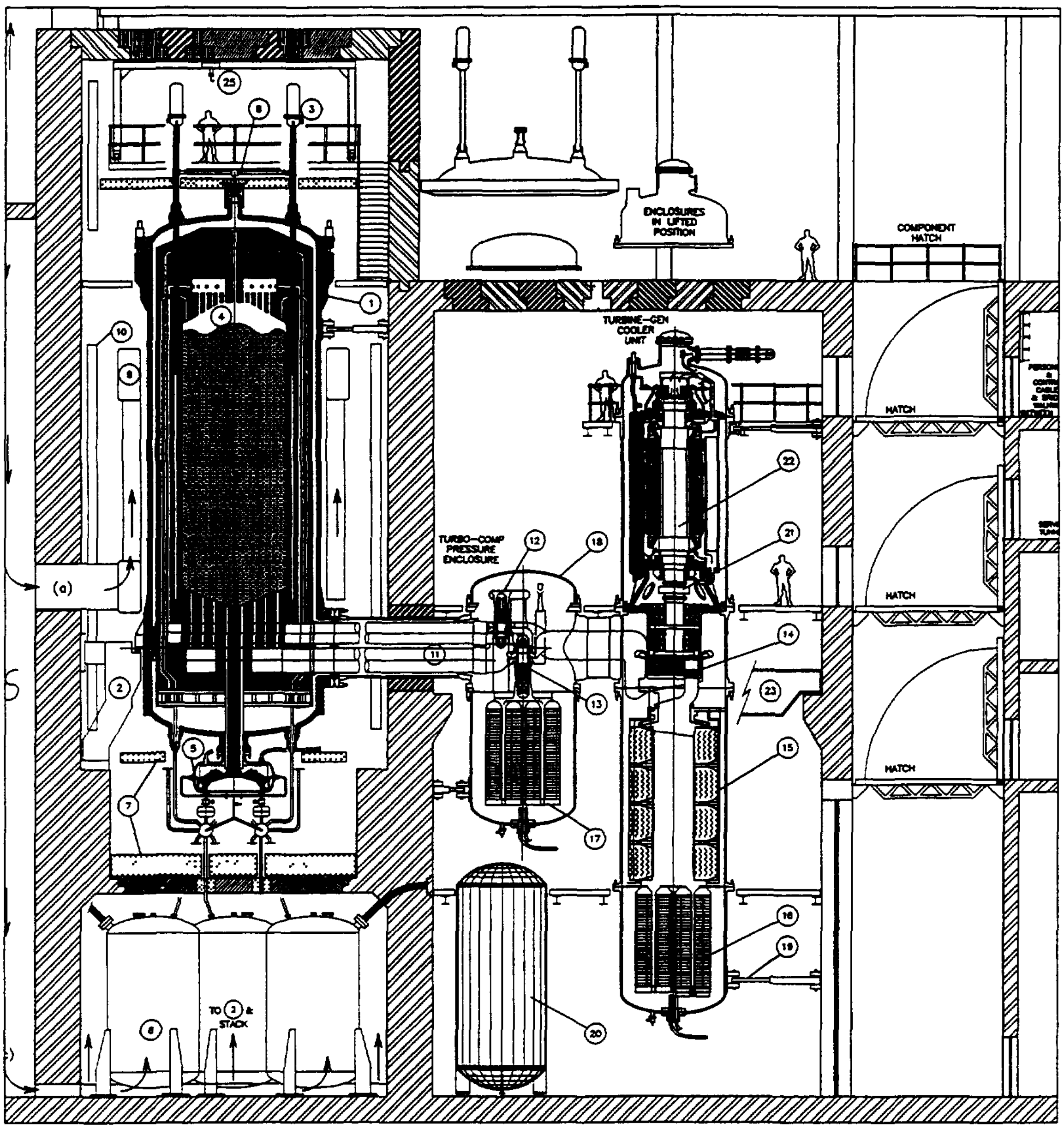
FIG. 7

- The plant is designed in order to make extension possible (addition of a new modules) without hampering the operation of the other units
- Space in the building is not yet optimized. Allocation of physical space for functional systems is to a large extend done.

7. SPECIAL STUDIES AND ANALYSIS

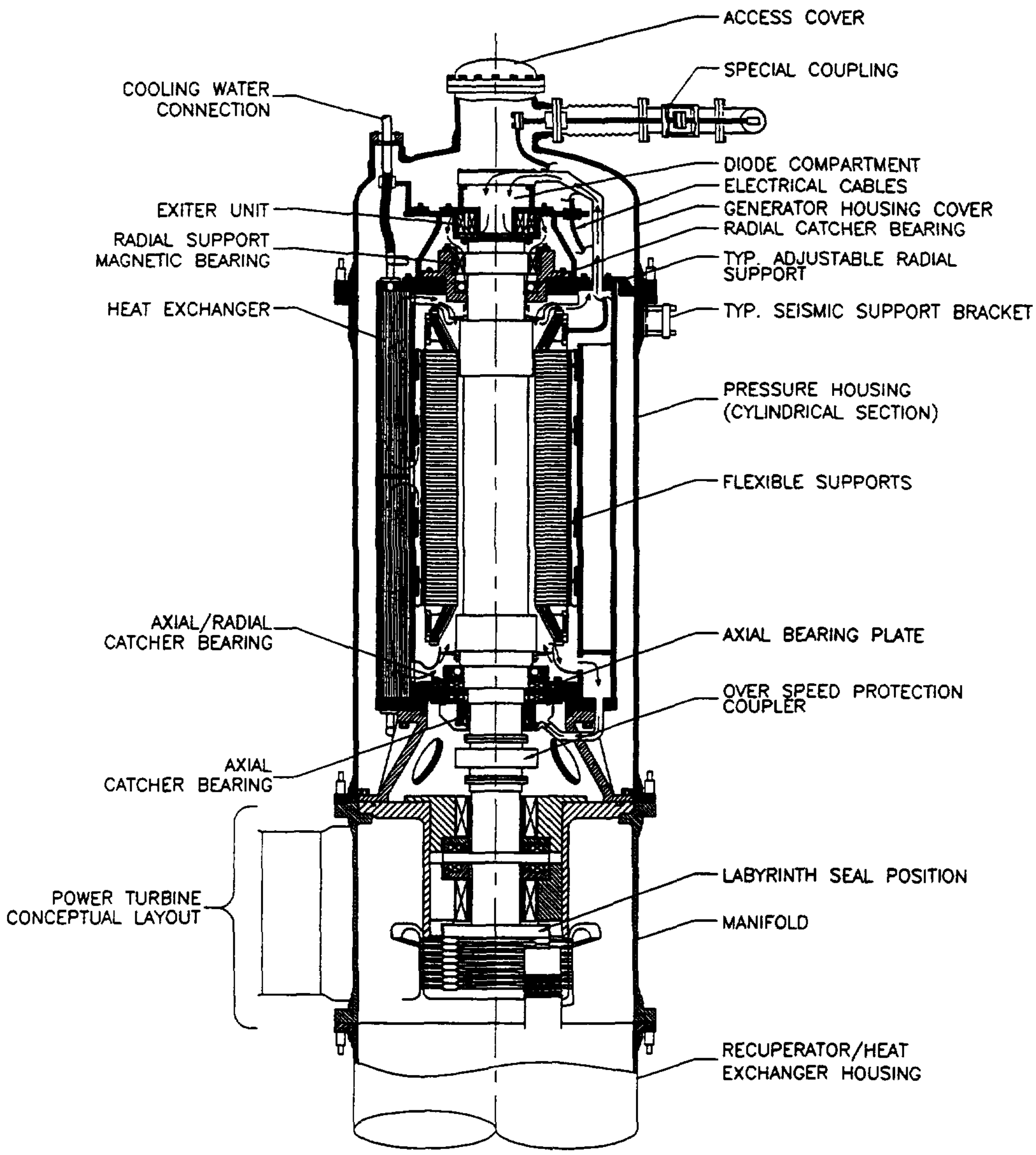
The following special studies have been done or approach completion.
(See attached Figure 7)

- Installation of the reactor vessel. .
- Reactor layout (See attached Figure 8)
- PCU layout.



MAIN POWER SYSTEM

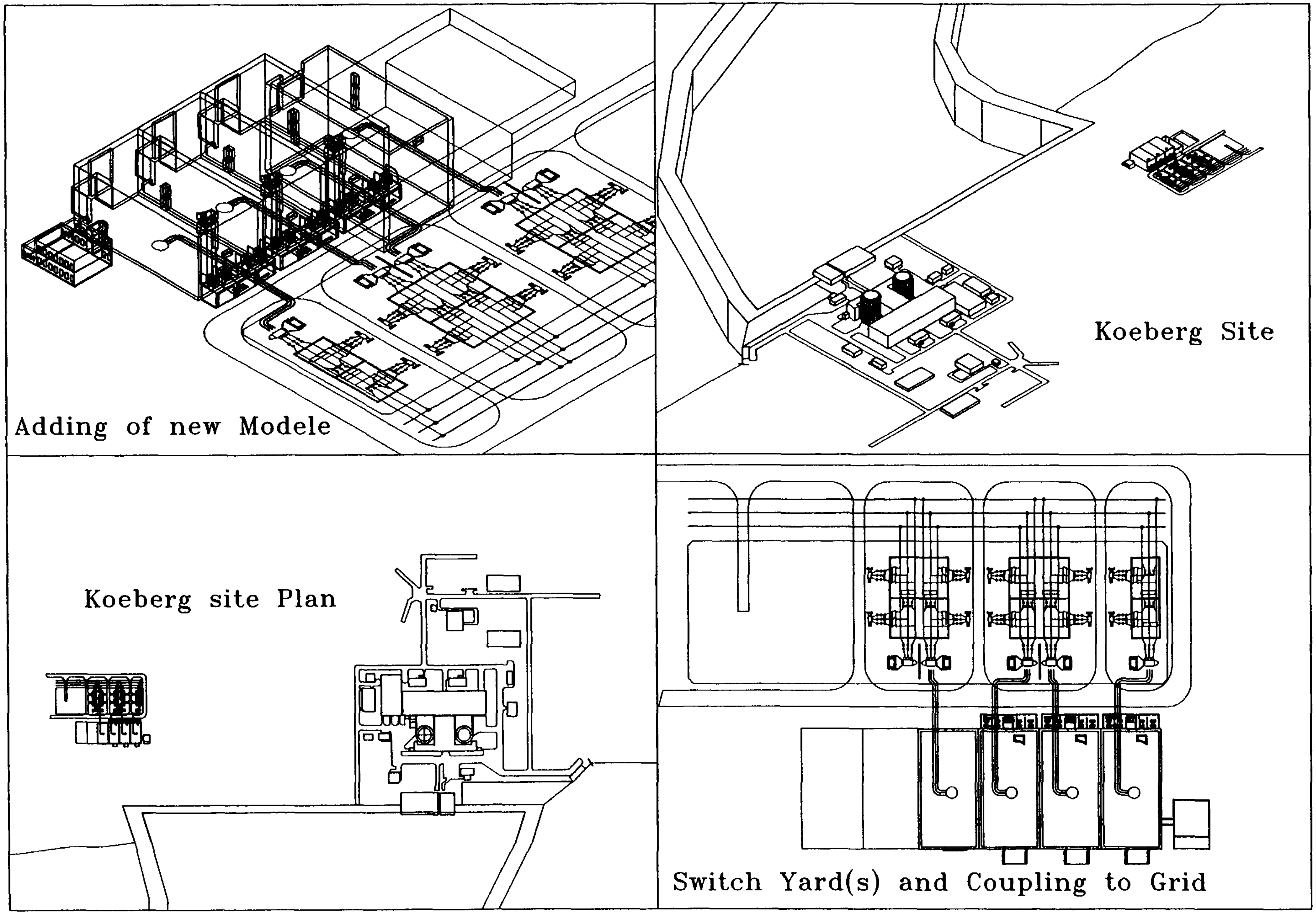
FIG. 8



GENERATOR CONCEPTUAL LAYOUT

FIG. 9

- Connecting of PCU to reactor.
- Different helium inventory control system proposals.
- PCU system and component performances.
- Reactor core performance.
- Pressure and temperature equalization following a reactor trip.
- Electrical *house* power demand study.
- Passive heat removal analyses (including the chimney).
- Reactor vessel thermal stress sensitivity (sensitivity study)
- Reactor building and main power system seismic behaviour.



KOEBERG SITE LAYOUT CONCEPT

FIG. 10

- Fuel supply and manufacturing options study.
- Alternator layout, supporting, coupling and operational modes. (Requirement study). (See attached Figure 9)
- KOEBERG site layout (See attached Figure 10)

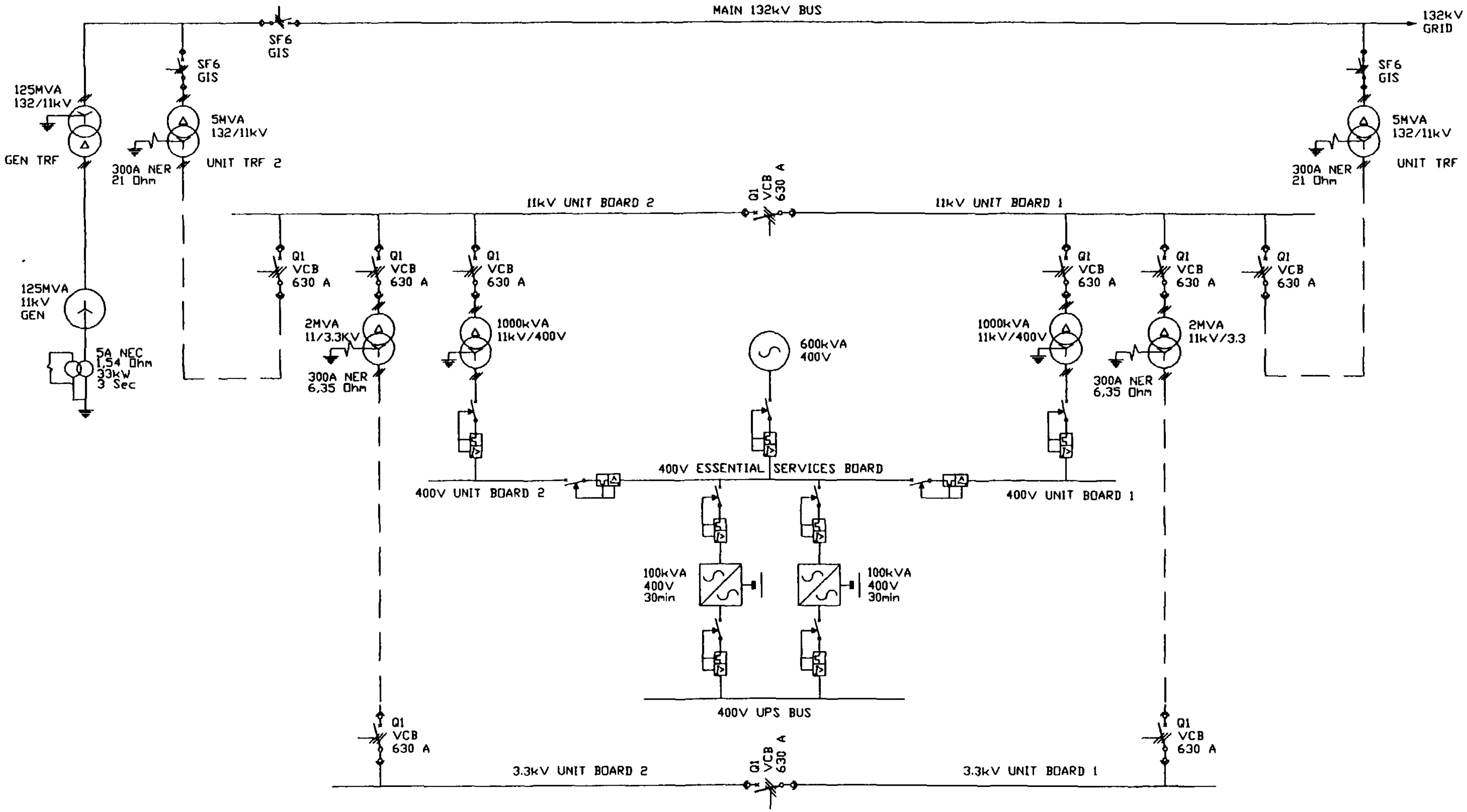
8. BRIEF OVERVIEW OF SOME OF THE INDIVIDUAL SYSTEMS

- Systems covered by other presentations during this TCM:
  - The Reactor (core design)
  - The Power Conversion Unit (PCU)
  - Helium Inventory and control system
  - Passive heat removal.
- Some comments on Systems of interest:
  - ***Active heat removal system***.
    * The sea as ultimate heat sink
    * Cooling tower for shutdown, startup and maintenance.
  - ***The shutdown heat removal***
    * basically keeps the reactor vessel cool when Brayton Cycle not functioning
    * Flow will be through the graphite reflector.
    * Core will remain hot during operation of this system.
  - ***The fuel and defueling systems***
    * An on-line operable system.
    * This system is 100% redundant
    * 2 zone core loading capability.
  - ***Electrical system*** (See attached Figure 11)
    * A 132 kV main bus with step down transformators to 11 KV, 3,3 KV and 400V busses and switchgear.
    * Electric demand is split into startup, shutdown and run categories.
    * A 600 kVA diesel-generator will serve as startup and stand-by electric supply source if grid is not available.
  - ***Automation System Architecture*** (See attached Figure 12)

    Minimum scope safety-related Reactor Protection System to ensure that the initial conditions preceding a possible accident are safe (e.g. reactor temperature and integral energy - which has to be removed as decay heat).

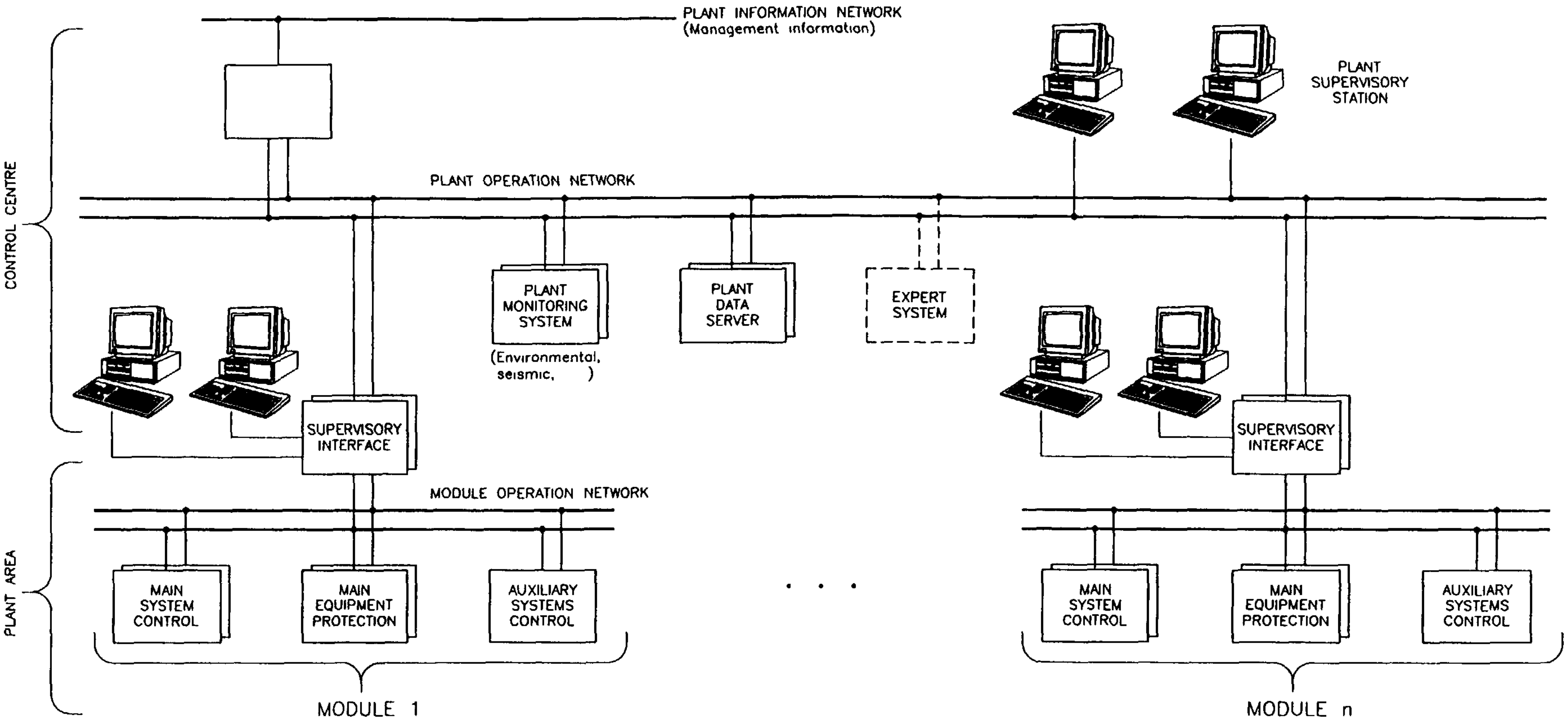
    The rest of the system is normal industrial automation equipment. Use is made of:

    * programmable logic controllers
    * computer-based man-machine interface devices with an absolute minimum of discrete control buttons / indications)



PBMR POWER SYSTEM SINGLE LINE DIAGRAMME

FIG. 11



AUTOMATION SYSTEM ARCHITECTURE

FIG. 12

- Networking between controllers
- redundancy of processors and networks as and where appropriate

Each reactor module has an independent automation system, situated in the plant area. Operator stations for all modules are in one common control centre. Provision is made for one plant supervisory station (but no technical support centre, no remote shutdown facility).

Plant common automation equipment (with a degree of redundancy as and where appropriate) may be:

- plant monitoring systems (environmental, seismic, ...)
- plant data server

9. COMPONENT SPECIFICATION AND COSTING

The following main components costing is at present under discussion. (tender inquiries out)

- The reactor vessel.
- The graphite.
- The fuel.
- The alternator.
- The turbo machinery and heat exchangers
- The building
- The electrical equipment include the main transformers.
- The active heat removal pumps and heat exchangers.
- The ventilation system.

10. THE ENGINEERING SIMULATOR.

- Acceptable level of stability of the Brayton Cycle is demonstrated.

- Operating at different levels of power showing correlation with other methods of predictions.

- The behaviour of the turbo machinery is demonstrated during ***step*** and ***ramping*** of power and helium inventory changes.

- The simulator already plays an important role in identification and simplification of a control system.

(Detail of the simulator is presented in a separate paper).

11. CONCLUSION

- The simplicity of the proposed reactor still prevails.

- The inherent safety is captured in the fuel and plant configuration.

- A new licensing framework is possible.

- The direct cycle is a non-omissible option for future power generation.

- The technology base internationally available makes it possible to design and build a fullscale prototype plant directly without extended research and development required.

- We foresee no technical and financial reason why the public cannot accept this power plant as an electric power production unit.
- Variations of the proposed concept are possible to suit potential clients requirements.
- It is one of the world's most fascinating new developments.
- Our study is basically on time as scheduled, and will be finished with great enthusiasm.

XA9846679

LICENSING OF THE PROPOSED PBMR-SA

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Abstract

This paper describes the preliminary criteria, which are intended to be used by the South African regulatory authority (CNS), for licensing of the ESKOM proposed - South African high temperature gas-cooled Pebble Bed Modular Reactor (PBMR-SA).
The CNS intends to apply the existing CNS licensing approach together with some international design criteria used for advanced reactors as well as international experience gained from the safety evaluation of the MHTGR, THTR, etc. A major requirement to this type of reactor is that it should comply with the current CNS risk criteria and provide, as a minimum, the same degree of protection to the operator, public and environment that is required for the current generation of nuclear reactors.

I. INTRODUCTION

The South African high temperature gas-cooled Pebble Bed Modular Reactor (PBMR-SA) proposed by Eskom[1] is now at a preliminary design stage. In this paper the CNS approach to the establishment of a licensing basis for this reactor design is discussed against the background of the regulatory approach applied in South Africa and fundamental safety standards laid down by the CNS[2] as well as an international approach developed for the MHTGR[3] and other advanced reactors[4].
The approach followed by the CNS as regards reactor regulation is that the licensee must present a safety case for the plant which must include a credible design basis respecting international norms and practices, and that the safety case includes a quantitative risk assessment demonstrating compliance with the fundamental safety criteria laid down by the CNS [5], which include:

- The CNS risk criteria for risk to the public and plant personnel due to accident conditions and normal operating events
- Defence-in-depth
- The ALARA principle

In view of the fact that the PBMR-SA is to a large extent a unique design, the design rules and design basis have to be largely developed by the South African licensee in consultation with the CNS. This must be performed in accordance with the fundamental safety standards of the CNS and prevailing norms and standards.

II. EXPECTED DESIGN FEATURES

International experience indicates that high temperature gas-cooled modular reactors have unique safety characteristics [6]. The most important being:

- slow response to core heat-up events, because of the large heat capacity and lower power density of the core;
- the very high temperature that the fuel can sustain before the initiation of fission-product release;
- passive reactor shutdown with a modest temperature rise;
- passive decay heat removal features;
- high-integrity coated fuel particles, which function as the initial fission-product barrier and primary reactor containment system;
- minimum requirements for operator actions and the low sensitivity of the design to operator error.

All these contribute to the long time intervals necessary for the implementation of corrective actions.

Therefore, the PBMR-SA design has the potential to eventually demonstrate a number of favourable characteristics, in particular:

- the potential for only minor core damage and fission-product release over a wide range of severe challenges to the plant;
- the objective of reduced dependence on human actions and reduced vulnerability to human error;
- a long response time of the reactor under accident conditions, which provides sufficient time for evaluation and corrective actions;
- the capability to demonstrate by test, the significant safety features and safe performance of the plant over a wide range of events;
- the potential for the development and successful performance of high quality fuel and passive safety characteristics.

As a result of these expected safety features, the following important differences between the existing LWR and the PBMR-SA design could be expected in the PBMR-SA design:

- a shift in emphasis from mitigation features to highly reliable protection features;
- reduced number of systems, components, and structures needed to be classified as safety related;
- Possible relaxation in the number of lines of defence or barriers.

III. DEVELOPMENT OF A LICENSING BASIS

The basic requirements introduced in chapter I will have to be applied to the PBMR concept design to derive plant-specific safety requirements for the purposes of establishing an appropriate licensing basis. The basic objective is to underpin the risk requirements with “deterministic” safety principles. This must be performed by the licensee in consultation with the CNS in the following stages:

1. Identification of accident and operational events.

All scenarios which could result in exposure of *the public or plant personnel* to risk during normal operation of the plant or as a result of an accident, either due to direct exposure or release of radioactive material from the facility must be identified. In this regard all sources of radioactive material, release mechanisms, release pathways, components, systems, plant processes, activities and events which could impact significantly on risk must be identified. The significance of such events will ultimately be determined by comparison with the safety criteria of the CNS.

2. Establishment of a classification scheme for accidents and operational events

It is proposed that a classification scheme for accident and operational events should be developed to facilitate the development of deterministic design rules. This scheme would effectively serve as a surrogate for the CNS risk criteria, but would facilitate the development of design rules by adherence to a number of principles outlined below.

Criteria for public and operator exposure should be developed for each event category along with appropriate frequency ranges with the following objectives:

- Adequate safety margins against the CNS criteria for public and operator risk must be assured, taking into consideration that the risk to the public from the presence of the PBMR must be added to that of any other existing facilities at the site (e.g. the Koeberg PWR reactors at the Koeberg site).

- Although the proposed design should provide, as a minimum, the same degree of protection for the public and environment as required of the current generation of nuclear reactors, an enhanced level of safety is expected.

- The classification scheme must be made as stable as possible (i.e. not subject to the variabilities typical of quantitative risk assessments) by giving adequate attention to prevailing uncertainties. In view of the limited experience feedback from reactors of this type, consideration of uncertainties are expected to play a mayor role in establishing the licensing basis for the PBMR.

- The consequence analyses must be based on conservative assumptions (at least for event categories I and II) with due consideration to uncertainties as discussed above.

- In view of the advanced nature of the proposed design as a potential source of energy for the next 40-50 years, it is expected that the safety level for these reactors must be shown to be better than the current generation of LWRS, particularly as regards accident prevention. The emphasis on accident prevention as opposed to accident mitigation could provide a basis for decisions regarding less defence-in-depth requirements than for the current generation of LWR's.

The classification scheme for a set of event categories proposed by ESKOM [1] and used for MHTGR[3] with some modifications may be considered at this stage. The list of events is given in tables 1-4.

Event Category I. Events in category I (EC-I) (table 1) are equivalent to the current anticipated operational occurrence (AOO) class of events considered for LWRS. The frequency range for these events goes from a number per year down to approximately 10^{-2} per reactor-year, which corresponds to the frequency of events that may be expected to occur several times during the life of the plant.
Events with frequencies exceeding 10^{-2} per annum may not lead to any significant dose to the public or plant personnel. This would encompass events in categories 1 and 2 for LWRS, category 1(for LWRS) referring to normal operating events (i.e. frequency > 1), while category 2(for LWRS) refers to transient conditions which are expected to occur during the operating life of the plant (frequency between 1 and 10^{-2}).

Table 1. Event Category 1

Number	Event
EC-I-1	Main-loop transient with forced core cooling
EC-I-2	Loss of main and shutdown cooling loops
EC-I-3	Control-rod-group withdrawal with control rod trip
EC-I-4	Small heat-exchanger leak
EC-I-5	Small primary-coolant leak

Table 2. Event Category II

Number	Event
EC-II-1	Loss of heat transport system (HTS) and shutdown cooling system (SCS) cooling
EC-II-2	HTS transient without control rod trip
EC-II-3	Control-rod withdrawal without HTS cooling
EC-II-4	Control-rod withdrawal without HTS and SCS cooling
EC-II-5	Earthquake
EC-II-6	Moisture inleakage
EC-II-7	Moisture inleakage without SCS cooling
EC-II-8	Moisture inleakage with moisture-monitor failure
EC-II-9	Moisture inleakage with heat-exchanger dump failure
EC-II-10	Primary-coolant leak
EC-II-11	Primary-coolant leak without HTS and SCS cooling

Event Category II. Events in category II (EC-II) (table 2) are equivalent to the current Design Basis Accident (DBA) categories 3 and 4 for LWRS and would be selected in a manner consistent with that for selection of an LWR DBA envelope. Specifically, events in EC-II would:

⇒ Be identified, using in-depth engineering assessment and judgement, complemented by PRA methods, to include internal events in the frequency range of 10^{-2} to 10^{-4} per year, with the aim of ensuring that any event expected to occur over the lifetime of a population of reactors is included.

⇒ Include external events relevant to the reactor site.

⇒ Be subject to single-failure criteria and other traditional conservatism, depending on the assessed accident consequences, with no credit for non-safety-grade equipment, etc.

Events in this category would require conservative analysis, as is currently done for LWRS.

Table 3. Event Category III

Number	Event
EC-III-1	Moisture inleakage with delayed heat-exchanger isolation and without forced cooling
EC-III-2	Moisture inleakage with delayed isolation
EC-III-3	Primary-coolant leak with neither forced cooling nor helium purification system pumpdown
EC-III-4	Inadvertent withdrawal of all control rods, without reactor trip for 36 hours: (1) reactor system pressurised, with forced cooling available (2) reactor system pressurised, with reactor cavity cooling system (RCCS) cooling only (3) reactor system depressurised, with RCCS cooling only
EC-III-5	Station blackout for 36 hours: (1) reactor system pressurised (2) reactor system depressurised
EC-III-6	Loss of forced cooling plus RCCS cooling for 36 hours: (1) reactor system pressurised, RCCS 25 percent unblocked after 36 hours (2) reactor system depressurised, RCCS 25 percent unblocked after 36 hours
EC-III-7	Rupture of justifiable number of heat exchanger tubes with failure to isolate or dump heat exchanger: (1) reactor system depressurised, with forced-circulation cooling maintained (2) reactor system depressurised, without forced-circulation cooling
EC-III-8	Rapid depressurisation. Double-ended guillotine break of crossduct with failure to trip (assume RCCS failed for 36 hours and 25 percent unblocked thereafter). Partial control-rod insertion after 36 hours.
EC-III-9	Severe external events consistent with those imposed on light-water reactors.

Event Category III. Events in category III (EC-III) would correspond to those severe events beyond the traditional DBA envelope that should be used by the designer in establishing the design bases. The events in this category would be identified using engineering assessment and judgement in conjunction with probabilistic techniques and expert opinion.
Specifically, events in EC-III would:

⇒ Include internal events in the frequency range from 10^{-4} to 10^{-7} per year, including external events not covered in EC-II, consistent with the approach taken for future LWRS.

The events proposed by ESKOM for emergency planning, together with additional bounding events to account for plant-specific uncertainties (as for the MHTGR) as listed in Table 3 are to be considered in this category.

In selecting the events to be included in EC-III, the design would be specifically reviewed to identify those events with the potential of a large release, core melt (or equivalent), or reactivity excursion to ensure that adequate prevention or protection is provided before these events could be excluded from this category. EC-III events could be analysed on a best-estimate basis.

Event category IV. Events in category IV (EC-IV) (table 4) would be used in assessing the extent for offsite emergency planning necessary and evaluation of the inherent extended safety features of the PBMR-SA. These events would be analysed in terms of risk and the events proposed by ESKOM for Beyond Design Basis Events as well as vessel failure and graphite burning events are to be considered in the EC-IV category at this stage.

Table 4. Event Category IV

Number	Event
EC-IV- 1	Reactivity events including spurious reactivity insertion and indefinite failure to trip when required: (1) pressurised conduction cooldown wfthout reactor trip; (2) control rod withdrawal (all rods); (3) hypothetical control rod ejection; (4) large moisture ingress.
EC-IV- 2	depressurised conduction cooldown with immediate and indefinite loss of the HTS, SCS. and RCCS.
EC-IV- 3	large moisture ingress events with and without heat-exchanger isolation: (1) large moisture ingress with isolation (2) moisture ingress without isolation.
EC-IV- 4	rapid vessel depressurisation due to catastrophic interconnecting manifold failure.
EC-IV- 5	catastrophic failure of core support.
EC-IV- 6	hypothetical vessel failure
EC-IV- 7	hypothetical vessel failure with graphite totally burnt out

Event Frequencies

Large uncertainties may exist in PRA results, especially in the lower frequency ranges. Therefore, in selecting and analysing the events, consideration must be given to the treatment of uncertainties. Accordingly, where the event categories include in their definition a frequency value, this frequency value is intended to be a guideline only and is not to be considered a rigid limit for which compliance must be rigorously demonstrated.

If it is demonstrated that the PBMR-SA design posses the safety features anticipated above [6] and a negligible risk of the significant fission products release can be justified, the classification scheme for a set of event categories could be changed to only three categories, in particular:

- Events in category I - anticipated operational occurrence;
- Events in category II - accidents with no release of fission products;
- Events in category III - accidents with release of fission products.

3. Establishment of basic safety rules for design and operations

Basic safety rules for the design and operation of the plant should be established with the objective of meeting the requirements arising from the classification of accident and normal operating conditions described in previous section. These rules should form a link between the requirements arising from the above risk-based classification and the detailed plant design basis. These rules should form the basis for licensing and deal with issues such as:

◊ Requirements on the need for and integrity of the various barriers, e.g. fuel, reactor vessel and containment structure based on the event categories defined above, covering inter alia:
  - Mitigation of internal events (coolant leaks/core damage accidents)
  - Protection against external events (fire, flooding, air crash)

◊ Requirements on protection systems, i.e.:
  - Need for and classification of diverse engineered safeguard systems
  - Levels of redundancy of the above systems
  - Availability requirements

◊ Design requirements relating to radiological protection (e.g. waste treatment, shielding, ventilation, installed radiation monitoring)

◊ Plant siting requirements.

It can be expected that the licensing requirements for the first unit will include an extended more comprehensive commissioning and testing programme, requiring possible additional hardware, to verify and validate the design safety parameters.

As the first unit will most probably be situated on the Koeberg site, the interaction with KNPS will therefore need to be addressed.

4. Establishment of a detailed design basis and general operating rules

Once the basic safety rules have been established in step 3, the detailed design basis must be established to ensure compliance with these rules as well as the CNS safety criteria.

The design basis will have to be demonstrated to be credible and developed respecting acceptable international norms and practices.

Protection features identified thus far which must be accounted for in the design basis must include the following specific concerns:

- Protection of metal components from continued exposure at elevated temperatures and from hot helium during postulated transients

- Prevention of uncontrolled access of air or moisture to hot graphite and fuel particles.

In parallel with the development of the design basis, operating rules need to be established including inter alia:

- Operating technical specifications
- In-service inspection programme
- Maintenance and testing programmes
- Radiation protection programme
- Waste management programme
- Occurrence reporting programme
- Quality management programme
- Emergency plan.

IV. CONCLUSION

The final determination of whether the plant is licensable is contingent on the following:

1) Satisfactory completion of the preliminary design review;
2) Satisfactory resolution of all safety issues, identified during the review process and possible additional safety issues that may be raised at later stages during the review process;
3) Satisfactory completion of the final design review by a competent independent reviewer;
4) Conformance with applicable CNS rules, LDs and LGs current at the time of any future licensing action;
5) Satisfactory completion of the final design and licensing review by the CNS;
6) Satisfactory completion of research and development activities, including successful design, construction, testing, and operation of the first unit before design certification.

Acceptability of the PBMR-SA design by the CNS will be determined on the evaluation of both deterministic safety and PRA analysis.

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

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PBMR PHASE 1 STUDY: SEISMIC AND STRUCTURAL DESIGN CONSIDERATIONS — AN OVERVIEW OF PRINCIPLES

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Abstract

This paper briefly reviews the principles involved in the planning and design of the proposed facility to cater for seismic and structural loads. The conceptual layout is discussed, as well as the different load characteristics and scenarios. An outline is given of a model used to estimate the seismic loads, whereafter the different analytical models are discussed.

1. Background

This paper briefly reviews the structural and seismic design of the proposed facility. The structure serves the main role in supporting the sensitive equipment, protecting it from the external environment, internal and external missiles, and providing some rudimentary protection against the sudden uncontrolled release of radio-active material. The building will need to be designed to provide the necessary protection, but also be adaptable to different environments and circumstances. The current prototype is being planned for the Koeberg site in the Western Cape (close to Cape Town), but the ultimate goal is to be able to place such facilities at virtually any location in the world. In designing this prototype, the objective has therefore been set to ensure that it can be 'transported' to a wide range of sites: of particular interest is the seismic input parameters.

The paper reviews the forces and conditions to which the building will be subjected. The principles involved in planning the layout of the building are reviewed. Next the fundamental issues involved in estimating the seismic input parameters are reviewed, as well as the principles to be used in performing the seismic analyses.

2. Design considerations

The concrete structure that will house the proposed system is being planned to fulfil a multiple range of functions. The different input requirements are outlined below.

2.1 *Structural Requirements*

2.1.1 *Strength*

The structure will be expected to fulfil the normal operating loads, as well as a number of critical (accident) conditions. Normal loads to be considered would include the following:

- the support of heavy equipment
- slight under or over pressures during operational conditions

- construction loads (e.g. installation of the process equipment, etc)
- normal heat loads during operational conditions

During accident conditions, it is anticipated that the following conditions would have to be catered for:

- protecting the process equipment from missiles, both internally generated (e.g. turbine blades, broken pipes, etc) and external missiles (e.g. aircraft)
- the protection against slight over-pressures during a potential accident, and
- provide sufficient strength against seismic loads during earthquakes.

2.1.2 *Stiffness*

The sensitive nature of the equipment calls for a relatively stiff supporting structure to prevent relative movements between support points of the equipment. A key design parameter in planning the layout of the building has therefore been to ensure that the building does not exhibit excessive relative displacements during dynamic loads, and that piping and equipment are not subjected to large differential movements of its supports. Research has shown that such relative displacements could seriously jeopardise the structural integrity of the process equipment. Furthermore, resonance between the different modes of the equipment and the building structure should be prevented, as this could seriously jeopardise the safety of these critical equipment.

However, the supporting soil of the site could furthermore impact on the dynamic loads experienced by the building. Experience and extensive research have shown that substantial amplification of the seismic motions can be expected in the case where the facility is situated on soft soil deposits. Care should therefore be taken to ensure that such amplification does not occur. Special care would be required where such situations do occur: it is usually advisable to prevent resonance between building and the supporting soils.

2.2 *Integrity*

The second key objective of the building will be to ensure sufficient protection of the process equipment from the natural external environment. This entails the protections against the external corrosive sea air, as well as the potential damaging effect of the ground water.

The prior matter will be addressed easily through high quality concrete construction. Similarly, the structure will be designed to withstand at least the minimum over pressure that could be expected during the initial phases of a possible blow-down of the system: special pressure release gates will be provided to release any potential over-pressure as soon as possible. It is therefore not anticipated that the structure will tested for leak tightness, and not special effort will be put into creating an absolutely leak-tight environment.

The prototype plant will be situated on the Koeberg site, where the sand cover over the underlying rock is approximately 20m thick. The water table is also quite high, being right next to the sea. It can therefore be expected that substantial water proofing will have to be provided. A cavity wall system is currently being investigated for this purpose.

2.3 *Structural Layout*

The primary structure is planned to consist of a number of heavy concrete walls and floor required to support the process equipment (see Figure 1). The large latter force will be resisted by the internal floors, while a vertical load carrying system is being devised to provide for the support of the vessel, the heavy overhead crane, the numerous large pressurised gas containers, and the roof itself. Special care will be given to finding a solution which takes into account the dynamic response of the roof during seismic events.

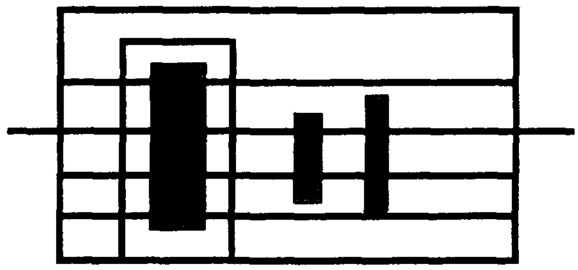

Figure 1: Conceptual Layout of Building

A key role to be played by the structure is the protection against external missiles. It will be designed to withstand the following two cases:

- Cessna 172 (which would not be allowed to penetrate the external wall), and
- Phantom F1, (which would not be allowed to penetrate the protective boundary surrounding the process equipment.

This protective boundary is illustrated in Figure 2.

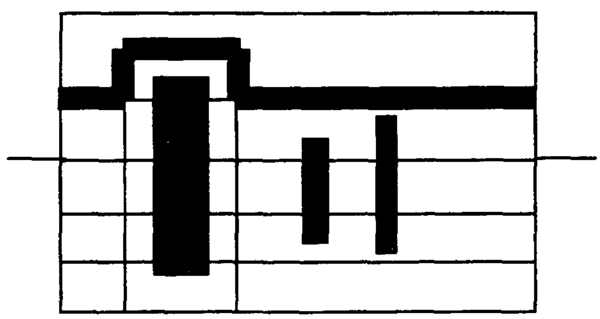

Figure 2: Protective Boundary Above Process Equipment

3. Seismic Design

3.1 Seismic Input Parameters

3.1.1 Properties of earthquake

Effects of Earthquakes

Earthquakes represent some of nature's most catastrophic events which could wreak havoc in any community. Fortunately, the South African region is known for its seismic stability, and only a relatively small number of earthquakes occur. However, earthquakes do induce significant forces in buildings and other structures, and extensive damage could occur in buildings if the necessary precautions are not taken. It is therefore necessary to take cognisance of these effects during the design stage.

An earthquake is manifested by a sudden, random, rapid movement of the ground which could have a duration of between 2 to 20 seconds. A typical recording of the ground motion in the three directions are uncorrelated. The maximum horizontal acceleration measured during an event can reach 1 or 2g (where g is the gravitational acceleration of 9.81 m/s^2) but in most cases it is less than 0.5g. In frequency terms, the dominant frequency of the motion generally lies in the range of 2 to 20 Hz (i.e., the ground vibrates at frequencies of between 2 and 20 Hz).

Classification of Earthquakes

Various scales are used to measure the degree of intensity of an earthquake. The two more commonly used scales are the Richer magnitude scale, and the Modified Mercalli Intensity scale. The Richter scale is used most commonly where sufficient instrumentation is available to measure the motion of the earth. This scale is an open-ended scale, and larger events can reach 8.5 on this scale. The largest event recorded to date in Southern African is the main shock of 29 September 1969 at Ceres, which registered 6.3 on the Richter scale. The Modified Mercalli Intensity scale is generally used in cases where no instrumentation recorded the event, and where local evidence needs to be used to estimate the severity of the event. The earthquake at Ceres was assigned an MMI value of 9, being the largest recorded in this region.

Earthquakes are induced by a range of different mechanisms in the upper layers of the earth crust. The major cause of earthquakes is the sudden release of energy when two adjacent sections of the crust move with respect to each other. Such movement typically occurs at the interplate boundaries of the earth. This results in a concentration of earthquake epicentres along the American coastal belts, in the Philippines, and in some parts of the Mediterranean. Other causes include volcanism, and other effects which could cause stress concentrations in the rock layers, such as an increase in load on the earth crust due to the construction of dam, or the seasonal accumulation of water in some swamp areas. However, these situations generally result in relatively small seismic events.

Effect of Distance

The epicentres of larger natural earthquakes are located at depths of between 25 and 40 km below the surface. From there the energy is transferred as P (compressional), S (shear) and Raleigh waves through the earth to areas further away. The intensity of the observed ground motion (and therefore the MMI observed at a particular position) attenuates with distance from the epicentre. Although not strictly correct, this attenuation of intensity can be represented by a logarithmic function expressed as:

$$\log I = \log I_o + a \log R + c$$

where I_o is the intensity at the epicentre,

I is the intensity at distance R, and

a and c are constants.

Therefore, sites further afield from seismically active zones are affected less significantly than those close by.

3.1.2 *Seismicity in Southern Africa*

Earthquakes have been recorded in the Southern African region since as early as 1620, but a scientific record of events has only been kept for the past 20 or 30 years (Fernandez 1970 - 1986). Largely due to a lack of data, seismologists have not been able to construct a clear model of seismic activity in this country, and quite large error bounds are therefore by necessity included in any codified guidelines. It can be expected that this may also be the case in a number of other countries where these plants may in future be situated. A brief outline of the methodologies used here is given to illustrate that the lack of seismic information should not preclude the development of some model of seismic behaviour of the region.

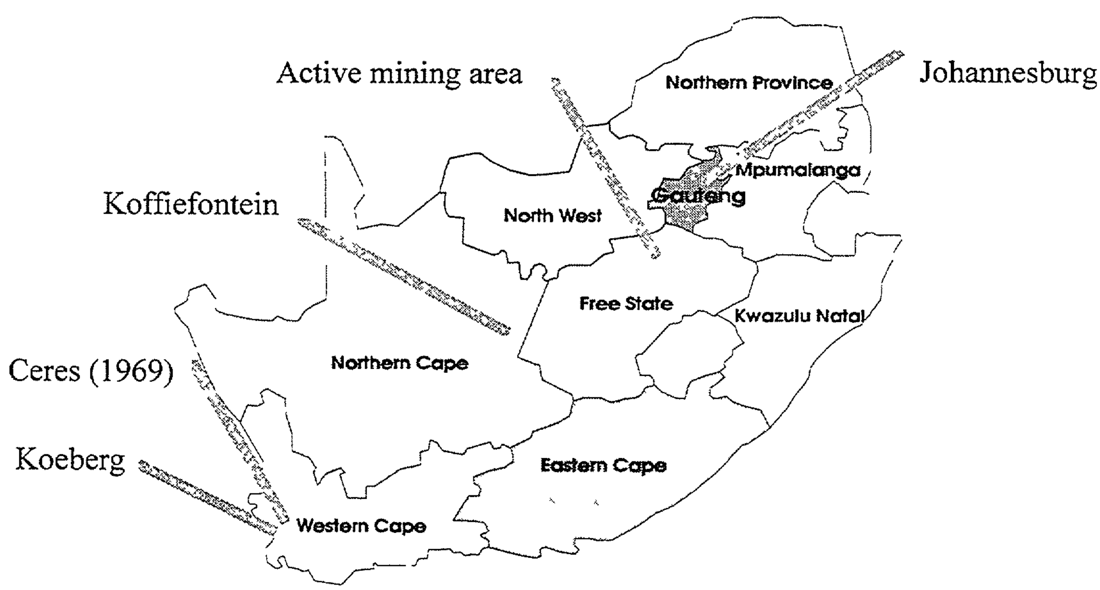

Figure 3: Points of Interest and Seismically Active Areas

From the available data it is clear that there seems to be higher concentration of earthquakes in the Cape Town - Ceres area, where some of the larger events took place (see Figure 3). A number of larger earthquakes have also occurred in the Zululand area, as well as in Lesotho and in Koffiefontein. A larger number of earthquakes (albeit small) occur in the northern region of Botswana, and in the vicinity of new large dams, such as the Kariba dam. This data was used to develop the seismic hazard map used in the design codes. Alternatively a probabilistic approach could be used to develop design spectra (Wium and Opperman, 1986).

A large concentration of earthquakes are also found in the major active gold mining areas of Transvaal and the Orange Free State. The epicentres of these events are typically located at depths of between 2 and 3 km, and they generally have quite short duration. Due to their very shallow epicentres, they tend to attenuate very rapidly, resulting in only limited impact on constructed facilities.

3.2 Seismic Design Parameters

This section briefly summarises the methods used to develop the seismic risk model. The methodology used in this work is based on the work of Cornell (1968), and on later extensions implemented by Anderson and Trifunac (1978) and Lee and Trifunac (1985). Two numerical models were developed to represent the unique properties of the Southern African region. The various aspects involved in this procedure are briefly discussed below.

3.2.1 Seismicity

The occurrence of earthquakes in a given region can be represented by a model of the seismicity which describes the location of seismic events in the region, the frequency of these events, and their magnitude or intensity at the epicentre. Typically, the region can be divided into a number of different seismic zones or features with which earthquakes can be associated. The most obvious features are faults and point sources, but in many cases, earthquakes cannot be associated with any particular feature, and a region therefore has to be treated as an area of distributed seismic activity. It is assumed that throughout each of these zones, there is an equal probability of an earthquake of a specific intensity or magnitude occurring anywhere in such a zone.

This model is expressed in terms of the Modified Mercalli Intensity (MMI) at the epicentre. Although a model of magnitudes would be more apt, little or incomplete magnitude data exists on earthquakes in Southern Africa prior to the 1970's. Seismologists therefore have to rely on records of MMI values.

The Seismological Data Bank (consisting of a comprehensive list of all recorded earthquakes in Southern Africa from 1620 up to 1983) was obtained from the Geological Survey of the Department of Mineral and Energy Affairs (Fernandez, 1985). All events after 1900 were selectively grouped into seven seismic zones, and only those events subsequent to the turn of the century were retained. Fore- and after-shocks were judiciously excluded from the list or earthquake.

However, a close inspection of the cumulative energy of the earthquakes shows that the available data on the seismic history of Southern Africa is not complete. Had this been the case, the average trend of the cumulative energy would have been constant for the full period from 1900. This certainly is not the case. Instead, in many regions this data is only complete from the early to late nineteen fifties. This data can hardly be used to correctly estimate the seismicity in a region.

On the other hand, it would seem as though the record of seismic events is not complete for the smaller events that occurred prior to the fifties, and that the list becomes progressively more complete in the smaller events as the time goes on. For instance, it seems that in some areas all large events (those with an intensity greater than or equal to 8) may be complete from as early as 1652, being the period for which written records are available. These events are so large that they would have been recorded by the public had they occurred. On the other hand, smaller events (with intensities between 5 and 7) may be complete from approximately 1900, while smaller events only seem to be complete from 1950 or even later.

It was therefore decided that all available data be used to compile a more representative model of the seismicity of each region. This can be done by establishing the average rate of events of a particular intensity during that period for which the events of that intensity is complete, and by extrapolating these events over a longer period. It is suggested that the longest available period be used as the basis for calculating the seismicity: in this case, it would be for the period from 1652 to 1983, being the period during which the Southern Cape has been populated on a continuous basis. The number of events that occurred during a shorter period should therefore be extrapolated to cover this full period.

All that remains to be done is to select that period for which the data of each intensity is complete. It was therefore decided that this can be regarded as that period during which the moving average of the annual number of events remain constant, or approximately so. These through I were used to estimate the applicable periods of complete data for each seismic region. Three alternative methods were used to establish the list of earthquakes.

3.2.2 *Attenuation*

The seismicity model discussed above expresses the probability distribution of events at the epicentre. However, the effect of such an earthquake rapidly diminishes with distance. This phenomenon is represented by an attenuation model. Various expressions have been suggested to represent the attenuation of earthquake waves due to the distance between the epicentre and the site (Howell, B.F. (Jr.) and Schultz, T.R., 1975). Isoseismal maps can be used to derive a suitable expression for Southern Africa. Typically, the intensity at a site (I) is expressed as a function of the intensity at the epicentre, the epicentral distance (R) and the focal depth (h).

3.2.3 *Engineering manifestations of earthquakes*

In the design of a structure, it is necessary to predict its response by using engineering input parameters, such as the peak ground acceleration, or the response spectrum for the design earthquake. MMI values cannot be used for this purpose.

No strong motion records are available of any significant natural earthquakes in Southern Africa. Functionals expressing the peak acceleration, response and Fourier amplitude spectra as functions of the site intensity have been proposed, and are based on a large number of world-wide earthquakes (Anderson and Trifunac, 1979; Cornell, Banon and Shakal, 1975; Trifunac and Brady, 1975). These correlations represent the relationship between two site parameters (e.g. MMI and peak acceleration), and are not significantly influenced by the geological characteristics of the region. This data can therefore be transferred from one seismic area to the next, and can be used to develop seismic input parameters for sites in Southern Africa.

3.2.4 *Final model*

The above expressions and functionals were obtained from observed data, and inherently exhibit a certain degree of scatter. Therefore, they need to be treated as probability distributions. Using this data, it is possible to derive a probabilistic model of the response at a given site by considering the seismicity of all seismic sources in the vicinity, by attenuating the effect to the site, by converting the MMI value to the response variable of interest, and then by integrating these functions for all seismic sources to obtain, for example, the response spectrum values at a range of frequencies. The response spectra obtained from these models can easily be used to generate the design time histories by using other methods for generating artificial earthquakes that are compatible with the design response spectra (Gasparini and Vanmarcke, 1976).

It is beyond the scope of this paper to further discuss the details, but the methods are more fully explained in the work of Cornell (1968), and Anderson and Trifuncac. Models for the seismicity and attenuating effects have been developed for the Southern African region, and are discussed in the following sections.

3.3 *Seismic Design Principles*

It was pointed out above that the dynamic interaction between the structure and the supporting soil needs to be taken into account. A complex model is being developed for this purpose: in this particular site, it can be expected that the dynamic interaction between the horizontally layered sand and the structure could be quite substantial. Care will be taken to ensure that a resonant situation is not created. At other sites, special care will be needed to prevent such interaction. Where possible, a more rigid site should be selected, as that would obviate the use of special soil improvement techniques (e.g. piling, large foundation mats, etc)

Should the site characteristics require it, it may be necessary adapt the structural layout in one of the following ways:

- modify the structural layout to change its dynamic behaviour
- consider alternative support positions for equipment (which should at all costs not be done)
- use more flexible support elements

This may require some additional attention when designing the plant for a subsequent project.

3.4 Design Philosophy

The general philosophy being followed in this project is to provide a design that would withstand a certain level of seismic excitation for at least 0.5g: the current design level is 0.3g, but it is anticipated that the target will be achieved without serious design repercussions. This will allow the design to be used on a wide range of sites throughout the world, without having to be requalified.

3.5 Seismic Analyses

A dynamic analysis model of the system and the structure will be compiled, which will represent the dynamic properties of the various components. However, in order to optimise the structural layout, and to provide meaningful input to the designers of the process equipment, a phased approach is being followed in modelling the systems. In the development, four levels of models are will be developed. These are the following:

- 2 Planning models (100 - 200 elements), one consisting of individual models for the building and process equipment, and one consisting of a combined model for both systems
- Design model (500 - 3000 elements)
- Licensing model (3000 - 10 000 elements)

The different types of analyses to be performed are outlined below.

3.5.1 *Response Spectrum Analysis*

A decoupled 3-D dynamic analysis of the primary loop will be performed. In this analysis, a floor response spectrum will be generated for the operating level of the concrete structure using the dynamic properties of the concrete structure. In this decoupled analysis of the concrete structure, the primary loop will be modelled as lumped masses to reflect the interaction effect between the primary loop and the internal concrete structure.

In the second analysis, the complete piping system, and the concrete structure will be considered. The primary loop will be once again connected to the internal structure as discussed above. The basemat will be represented by a six degree of freedom model which will allow complete freedom to rotate and displace, but which assumed that the basemat itself will be rigid, thereby representing the soil-structure interaction

A computer program (SSICAL) has been developed in-house for the purpose of calculating the dynamic soil-structure interaction. It is based on the Substructure Theorem, and encompasses a model synthesis in the frequency domain. This technique recognises the fact that the natural frequencies and model shapes of a structure with a fixed base are easily obtainable, while the soil is characterised by frequency-dependent stiffness functions.

An equivalent (frequency dependent) dynamic stiffness matrix can be compiled which represents the dynamic properties of the structures and the soil. For this, it is necessary to calculate the mode shapes of the structure on fixed base (i.e., ignoring all soil-structure interaction). The study will consist of the following steps:

- develop models for the structures, and calculate the modes and frequencies on a rigid base;
- generate the time histories, and calculate the Fourier transfers of each;
- calculate the dynamic stiffness of the combined model, and then calculate the response at each frequency;
- calculate the inverse Fourier transfers for each record to obtain time histories for the displacement and rotation of the base; and
- evaluate the response of the structures and pipes, using the time histories and prescribed displacements.

3.5.2 Combined Model

Since the concrete structure supports the primary loop, and as the primary loop will be quite heavy compared with the internal concrete structure itself, the primary loop model will be incorporated in the internal concrete structure model to form a combined model so that the interaction of these two substructures could be accounted for.

The coupling of the primary loop and the concrete structure will be achieved by means of sub-structuring and the use of generalised constraints equations for constraining the supports of the primary loop to their respective support locations in the internal structure. In this analysis, the primary loop will be considered as the higher level substructure and the internal concrete structure as the lower. The support locations were selected nodes in the internal structure, and were constrained to the six degrees-of-freedom of the beam element modelled in the centre of the structure. Since the structure will be supported on a rigid base, no interaction could occur between the internal and containment structures. The latter structures were therefore excluded from this analysis.

3.5.3 Licensing Model

This model will consist of a very detailed set of components in which the different key elements will be assessed. Detailed stress calculation will be performed, as will the accelerations be calculated for sensitive equipment.

4. Summary

This paper briefly reviews the scope of work being undertaken to address the seismic analysis and structural design of the proposed new facility. It is pointed out that methods are available for compiling seismic histories for the facility, even in the absence of comprehensive seismic histories.

The importance of the seismic and structural inputs have been highlighted, and the importance is stressed regarding the role of the structure in providing appropriate support, stiffness and protection to the equipment. It is clear that these issues will be of significant importance during the licensing applications, and appropriate attention needs to be given to the matters.

HELIUM STORAGE AND CONTROL SYSTEM FOR THE PBMR

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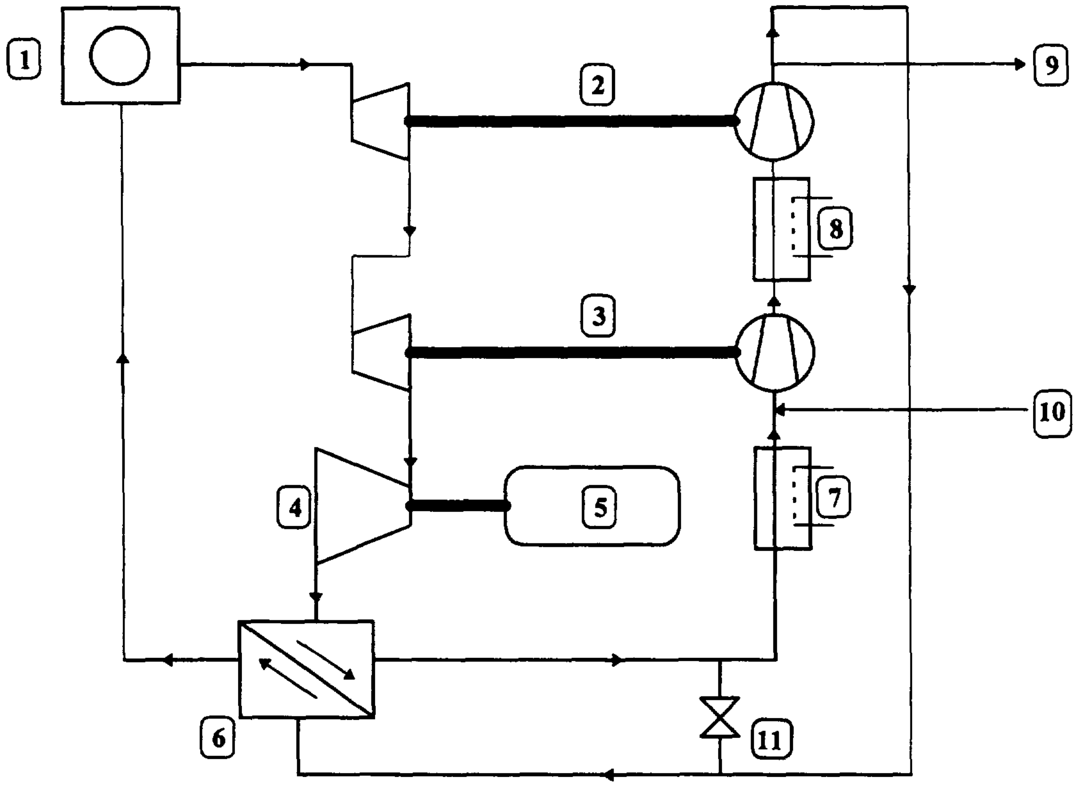
Abstract

The power conversion unit will convert the heat energy in the reactor core to electrical power. The direct-closed cycle recuperated Brayton Cycle employed for this concept consists of a primary helium cycle with helium powered turbo compressors and a power turbine. The helium is actively cooled with water before the compression stages. A recuperator is used to preheat the helium before entering the core. The start of the direct cycle is initiated by a mass flow from the helium inventory and control system via a jet pump. When the PBMR is connected to the grid, changes in power demand can be followed by changing the helium flow and pressure inside the primary loop. Small rapid adjustments can be performed without changing the helium inventory of the primary loop. The stator blade settings on the turbines and compressors are adjustable and it is possible to bypass reactor and turbine. This temporarily reduces the efficiency at which the power conversion unit is operating. Larger or long term adjustments require storage or addition of helium in order to maintain a sufficient level of efficiency in the power conversion unit. The helium will be temporarily stored in high pressure tanks. After a rise in power demand it will be injected back into the system. Some possibilities how to store the helium are presented in this paper. The change of helium inventory will cause transients in the primary helium loop in order to acquire the desired power level. At this stage, it seems that the change of helium inventory does not strongly effect the stability of the power conversion unit.

1. Introduction.

The primary cycle used for the heat-transfer process is shown in figure 1. It is the Rankine cycle, but when used with a single-phase, gaseous working fluid instead of a condensing working fluid, it is termed the Brayton cycle. It is called the direct Brayton cycle if the working fluid does not transfer its heat to a secondary (steam)cycle, but instead is used to power the turbine of the generator system directly. Two helium powered turbo compressors are used to maintain the required pressure in the system. In order to improve efficiency a counterflow heat exchanger, known as a regenerator or recuperator, is introduced which transfers heat from the gas flow into the first compressor towards the gas flow into the reactor. A further increase in efficiency is obtained by active cooling of the gas with water before each of the two compressors with a pre- and intercooler. A more detailed analysis of the power conversion unit is given by Liebenberg [1]. The initial filling of the primary cycle is performed by injecting clean helium from high pressure supply bottles into the system through a jet pump. This will start the circulation of the helium in the primary cycle and will cause the turbines of the compressors to start rotating and to build up the required pressure ratio. When the system is filled up to the point at which the pressure becomes sufficient to sustain the operation of the power turbine of the generator system, the plant is said to be operating at base load. In order to meet an increasing demand in power output, helium is further injected from the storage tank system until the required or nominal power level is reached. The variations in demanded power level between base load and nominal operation can be followed by changing the helium flow in the primary cycle. Rapid control



1 reactor unit
2 high pressure turbo/compressor
3 low pressure turbo/compressor
4 power turbine
5 generator
6 recuperator
7 precooler
8 intercooler
9 helium inventory outlet point
10 helium inventory inlet point (jet pump)
11 bypass valve

FIG. 1. The primary cycle of the power conversion unit.

in a small range up and down (≈ ±10%) is provided by adjustable stator blade settings on the turbines of generator and compressors. A reactor and turbine bypass system provides a large control range downward in power level (up to 100% load rejection instantaneous). Opening the bypass valve reduces the helium mass flow through the reactor core and subsequently less heat is transferred to the power conversion unit. The mass flow through the compressors is not effected by opening the bypass valve. In order to keep a constant pressure ratio over the compressors their turbines must extract the same power from a reduced mass flow. This is achieved by adjusting the stator blade settings on the compressor turbines. As a result the generator turbine can extract less power from the mass flow, and the efficiency of the power conversion unit decreases. In order to reduce the compression work the mass flow through the compressors is reduced by removing helium from the primary cycle. At the same time the mass flow through the reactor core is kept constant (that is the required power level), so the bypass valve gradually will close as the helium inventory decreases. When the bypass valve is closed, the removal of helium should stop and the efficiency close to nominal operation will be restored. The helium will temporarily be stored in pressure tanks. The removal or injection of helium from or to the primary cycle is a slow process (tens of minutes) compared to the nearly instantaneous blade settings on the power turbine or the operation of the bypass valve. As such, it will only be used for longer term adjustments. In the following section the application of inventory compressors (powered by helium turbine or electrically) will be considered in order to reduce storage tank size and number, and a simple model will be discussed for the calculations. In section 3 four different configurations will be presented, section 4 will discuss the time behavior, and in section 5 a preliminary conclusion will be drawn as to which design is most appropriate.

2. Modeling the helium inventory system.

It is possible to design a storage system without extra inventory compressors. The system compressors will perform the work required to store the helium in pressure tanks. At the high pressure point in the primary cycle, i.e., after the second compressor, helium will flow out of the system into the storage tanks due to the pressure difference between system and tank. The helium will flow back into the system at the lowest pressure point in the primary cycle (before the first compressor) due to the pressure difference between tank and system. For such a storage system the following relations can be derived:

$$f = d \cdot g \quad , \tag{1}$$

where f is the factor by which the pressure of the helium in the system increases after outflow of helium from a storage tank, and d·g the factor by which the pressure of the helium in the system decreases when helium flows out of the system into the storage tank. In order to prevent waiting for the pressures in tank and system to settle down to an equilibrium (the last part takes the longest time) and in order to prevent minor oscillations in mass flow direction that probably will occur around equilibrium, the mass flow from system to tank will be stopped when the ratio of tank pressure divided by system pressure equals g. The value of g will be close to 1, for instance 0.97. For the pressure ratio after injection back into the system a similar factor g' is defined, the ratio of system pressure divided by tank pressure. This factor g' is only important for the volume of the storage tank. If K_s is the pressure ratio maintained by the system compressors, and d (f) the factor by which the pressure in the storage tanks (the system) increases, then can be written:

$$K_s = f \cdot d \tag{2}$$

The number of stages n (= number of pressure tanks or pressure zones) required to increase the system pressure from a pressure at base load p_0 to a final pressure at nominal power p_f (p_0 and p_f measured in same pressure point) is given by:

$$f^n = \left(\frac{p_f}{p_0}\right) \tag{3}$$

From equations (1), (2) and (3) follows n according to K_s, g, and p_0 , which are design parameters for the system. When n is determined, f can be chosen from equation (3). This factor influences the storage tank size according to the helium mass in the system before ($m_{s,1}$) and after ($m_{s,2}$) injection from the storage tank:

$$\frac{p_{s,1}}{p_{s,2}} = f^{-1} = \frac{m_{s,1}}{m_{s,2}} \qquad \Rightarrow \Delta m_s = m_{s,2} - m_{s,1} = m_{s,1}(f-1) \tag{4}$$

with $p_{s,1}$ and $p_{s,2}$ the pressures related to $m_{s,1}$ and $m_{s,2}$. A similar relation holds for the storage tank:

$$\frac{p_{v,1}}{p_{v,2}} = f \cdot g' = \frac{m_{v,1}}{m_{v,2}} = \frac{\rho_{v,1}}{\rho_{v,2}} \qquad \Rightarrow \Delta\rho_v = \rho_{v,2} - \rho_{v,1} = \rho_{v,1}(\frac{1}{f \cdot g'} - 1) \qquad (5)$$

with $\rho_{v,1}$ ($\rho_{v,2}$) the helium density in the tank before (after) injection into the system. With $\Delta m_s = \Delta\rho_v \cdot V_v$ follows for the volume of 1 tank:

$$V_v = \frac{m_{s,1}}{\rho_{v,1}} \cdot f^2 \cdot g' \cdot \frac{(1-1/f)}{(f \cdot g'-1)} = \frac{m_{s,2}}{\rho_{v,2}} \cdot \frac{(1-1/f)}{(f \cdot g'-1)} \qquad (6)$$

The model can describe an isothermal or an adiabatic process, the difference will be expressed in the densities $\rho_{v,1}$ and $\rho_{v,2}$.

It can be shown that the volumes of all n tanks will be equal by substituting $\rho_{v,1} = f^2 \rho_{s,1}$ (with $\rho_{s,1}$ the density in the primary loop at injection point before injection) in equation (6) and by defining $V_s' = m_{s,1}/\rho_{s,1}$, with V_s' the effective system volume that can be calculated from the average system pressure and temperature at given inventory mass:

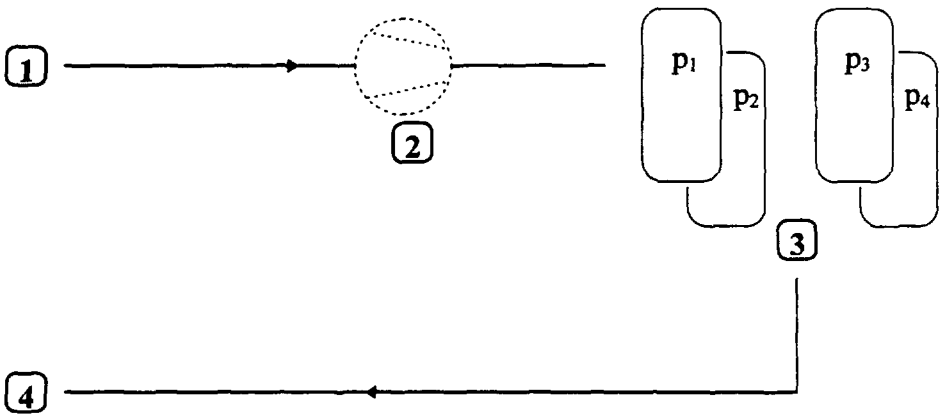
$$V_v = V_s' \cdot g' \cdot \frac{(1-1/f)}{(f \cdot g'-1)} \qquad (7)$$

If inventory turbo compressors are added to this system (figure 2) with a combined pressure ratio of K_i (depending on mass flow) then K_s has to be substituted by $K_s \cdot K_i$ in the previous equations and the storage tank size becomes V_v/K_i .

3. Configurations of storage system.

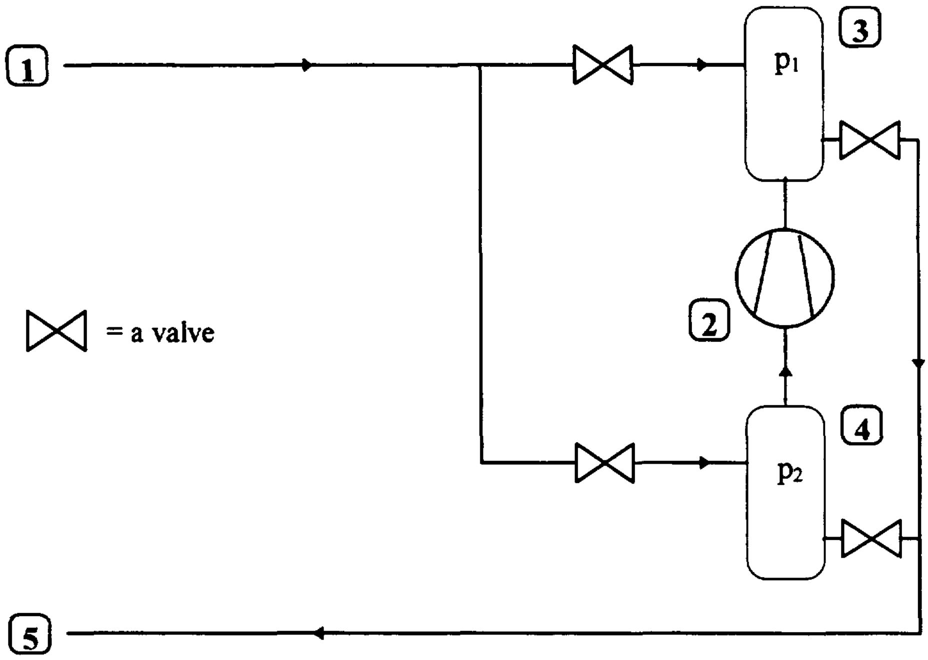
Although it is an advantage to control the helium inventory without using extra inventory compressors for storage, it requires very large tank sizes. For the PBMR-SA four tanks, each at different pressure and around 200 m^3, are required to store the helium when reducing power output down to base load. This operation is reasonably fast: within 15 minutes all tanks are filled. Emptying the tanks might take longer due to a temperature drop in the tank because of the adiabatic expansion of the helium into the system. The pressure will drop with the temperature and the tank can not be emptied to the extent needed. Injecting warm helium from the primary cycle into the tank, or actively heating of the tank might be a solution. Introducing extra inventory compressors to fill the tanks reduces the number of pressure tanks and their size, and influences the speed at which the storage system operates. Figure 2 shows these options. Turbo compressors will hardly effect the speed, being capable of handling large volume flows, but are limited to a certain pressure ratio they can achieve. Positive displacement (reciprocating) compressors are able to pressurize the tanks up to larger pressure differences, but will slow down the storage process.

In order to combine the advantage of not using extra inventory compressors in the storage system and reducing the tank size, a compromise can be found in installing two tanks: one tank is the original highest pressure tank operated without additional



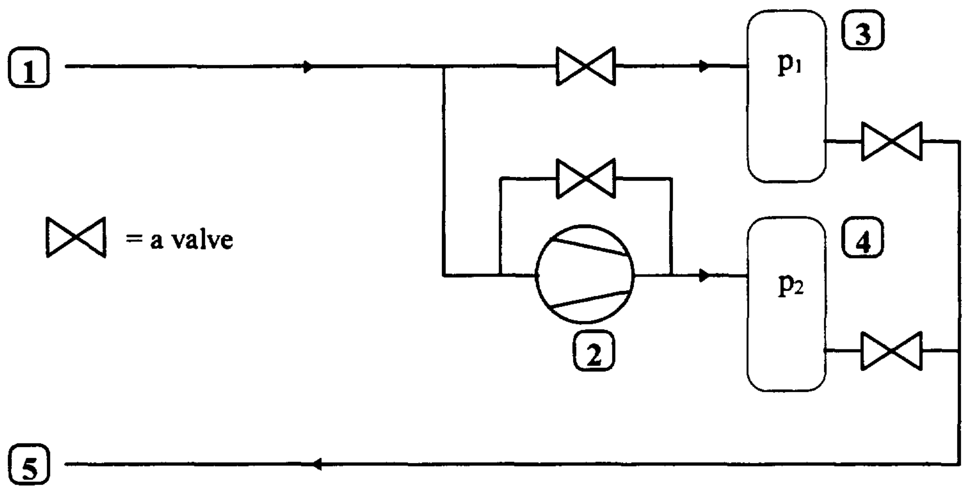
1 helium inlet flow from high pressure point in primary cycle
2 optional (turbo)compressor
3 storage tanks at different pressures
4 helium outlet flow to low pressure point in primary cycle

FIG. 2. The inventory storage tank system with four pressure tanks. Number and size of tanks can be reduced by applying (turbo)compressors.



1 helium inlet flow from high pressure point in primary cycle
2 positive displacement compressor
3 storage tank at high pressure
4 intermediate or control tank at lower pressure
5 helium outlet flow to low pressure point in primary cycle

FIG. 3. The inventory storage tank system with one high pressure tank and one intermediate or control tank.



1 helium inlet flow from high pressure point in primary cycle
2 positive displacement compressor
3 storage tank at high pressure (same as in figure 2)
4 storage tank at lower pressure
5 helium outlet flow to low pressure point in primary cycle

FIG. 4. The inventory storage tank system with a positive displacement compressor which compresses directly from the system.

compressors, and the other tank is used as an intermediate or control tank from which the helium is compressed to the high pressure tank. This configuration is shown in figure 3. A slightly different configuration is presented in figure 4, using again the highest pressure tank without additional compressors, but now the rest of the helium is compressed into the second tank with a positive displacement compressor.

If both the tanks are 200 m^3, the PBMR-SA is capable of load following without inventory compressors in the range of 100%-45% for both configurations. For further reduction in case of the control tank the inlet valve would close and the control tank would be emptied. If after some time a further power reduction is requested, the inlet valve of the control tank will be opened and even in the range below 45% a fast inventory reduction is realized. However, there are several drawbacks to this system. Firstly, if there is no further reduction in inventory required after some time, but an increase then the compression work 'in advance' has been wasted. Secondly, it is psychologically wrong to allow the original system pressure (before opening the control tank) to drop by filling the control tank and then to start compressing from this lower pressure level. The configuration of figure 4 does not have these drawbacks: firstly, the compression only takes place when needed, and secondly the compression is started from the highest possible point in the primary cycle towards a pressure point in the storage system that is lower than in the other configuration. The next section will discuss the time behavior of the tanks when operated without compressors.

4. The time behavior of the inventory storage tank system.

In this section the duration of filling and emptying the storage tanks or system will be considered when operated without compressors. Clearly, when compressors are used, their specifications in terms of power and volume capacity will define the time necessary for storage. When the pressure difference between system and tank is large enough to occasion a 10% or greater decrease in the helium density, the flow is no longer considered to be incompressible. The helium flow from the system at pressure $p_s(t)$ into a storage tank at $p_v(t)$ through an isentropic nozzle can then be described by solving the energy balance for an adiabatic, horizontal flow process:

$$\frac{\dot{m}(t)}{A} = p_s(t)\sqrt{\frac{kM(t)}{RT_s}\left(1+\frac{k-1}{2}\cdot M(t)^2\right)^{\frac{(k-1)}{(k+1)}}}\ , \tag{8}$$

with $\dot{m}(t)$ the mass flow through the nozzle with area A, R the gas constant for helium, T_s the system temperature. $M(t) = v(t)/c$ is the Mach number, with v(t) the flow velocity and c the velocity of sound in the gas. The ratio of heat capacities at constant pressure and volume is defined by $k=C_p/C_v$.

An analytical description for this time-dependent flow process can be found as long as the pressure ratio of system and tank remains larger than the critical pressure ratio:

$$\frac{p_s(t)}{p_v(t)} > \left(\frac{2}{k+1}\right)^{k/(k-1)} \tag{9}$$

When equation (9) is valid, the flow velocity will be constant and equal to the velocity of sound in the gas and the Mach number equals unity. The mass velocity is then proportional to $p_s(t)$ (equation (8)) and independent of the downstream pressure. The flow is said to be choked. Defining :

$$C = \sqrt{\frac{kM(t)}{RT_s}\left(1+\frac{k-1}{2}\cdot M(t)^2\right)^{\frac{(k-1)}{(k+1)}}}\ , \tag{10}$$

and solving the differential equations after substituting equation (9) for the choked process (M=1):

$$\frac{dp_s(t)}{dt} = -\dot{m}(t)\cdot\frac{RT_s}{V_s'}$$

$$\frac{dp_v(t)}{dt} = \dot{m}(t)\cdot\frac{RT_v}{V_v} \tag{11}$$

gives the time t_m for which the flow will be choked:

$$t_m = \frac{1}{P'\cdot A\cdot C}\ln\left\{\frac{p_s(t=0)}{p_v(t=0)}\cdot\left(\frac{2}{k+1}\right)^{\frac{k}{k-1}}\right\} = \frac{1}{P'\cdot A\cdot C}\ln\left\{K_s\cdot g\cdot\left(\frac{2}{k+1}\right)^{\frac{k}{k-1}}\right\} \tag{12}$$

with $P' = R\left(\frac{T_s}{V_s'}+\frac{T_v}{V_v}\right)$.

The temperature of the system at inlet and outlet point is kept constant by the circulation through the reactor core, but the temperature in the tank is not constant. In case of flow into the tank it is reasonable to assume that the temperature in the tank is approximately constant. Firstly there exists a fairly large heat buffer in the form of the helium present. Secondly the hot system gas is cooled close to the temperature in the tank by adiabatic expansion. In case of flow out of the tank a satisfactory description for the temperature in the tank still has to be formulated.

After substituting the values for the PBMR-SA it follows that the storage flow is no longer choked after 44 seconds (initial storage flow at 2% of the main flow, and gradually decreasing). The pressure in the system has dropped about 20%. When the flow is no longer choked, the velocity of flow also becomes time dependent and a numerical approach of the problem seems more appropriate.

Analytically also interesting is the time tail of the flow process when the densities in system and tank differ about 10%. Then the gas can be considered as incompressible and an average constant density $<\rho>$ between tank and system can be assumed. If at that point $\Delta p(t_0)$ is the difference in pressure between tank and system, then

$$\frac{d\Delta p(t)}{dt} = -P' \cdot \dot{m}(t) \quad ,$$

and $\dot{m}(t) = A \cdot \sqrt{2 <\rho> \Delta p(t)}$. (13)

After solving for $\dot{m}(t)$ follows:

$$\dot{m}(t) = -P' A^2 <\rho> t + A\sqrt{2 <\rho> \Delta p(t_0)} \quad (14)$$

This last part of the flow takes just over 2 minutes (starting with storage flow at 2% of main flow and gradually decreasing). In reality the diameter of the pipe will be adjustable and the mass flow out or in will be kept at a constant percentage of the mass flow in the primary cycle in order to maintain stability. Preliminary results from the simulator [2] indicate that storage at a rate of at least 2 % of the primary mass flow does not create stability problems. Higher rates are probably possible and have to be tested.

5. Conclusions.

At this point there is an inclination towards the storage system with four large tanks without compressors because of its simplicity from control point of view. Also from an economical point of view it seems the most advantageous system, provided that the building size has not to be altered significantly due to the large volumes of the tanks.

The system is also fast in terms of in and out flow, but thoughts must still be given to the extra pressure drop in the tanks due to adiabatic expansion. An adequate description must be developed to predict this behavior. With the help of the simulator the upper limit for the storage flow rate will be studied, and what transients are to be expected in the power conversion unit.

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POWER CONVERSION UNIT FOR THE SOUTH AFRICAN DIRECT CYCLE HTGR

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Abstract

The system parameters chosen to optimise the thermal efficiency of the Eskom PBMR whilst maintaining component simplicity is discussed Power Conversion Unit components, which are now at a preliminary design stage, comprise a precooler, two turbo units consisting of a turbine driven compressor recuperator and a power turbine, driving an alternator Design aspects of every component is mentioned and the inventory method of power control is explained with reference to start-up and shut-down events making the system an effective load following device, down to 4% of full power Application of the same design principles for HTGRs smaller than 25MWe is discussed

1 SYSTEM DESIGN PHILOSOPHY

A two stage intercooled compression, recuperated Brayton cycle was chosen for the Power Conversion Unit to operate in direct mode with the HTGR With a reactor outlet temperature of 900°C, the overall total pressure ratio for maximum cycle efficiency, based on anticipated component efficiencies, is 2.5 but in order to lower the reactor inlet temperature and reducing turbomachine size, an optimised pressure ratio of 2,7 was used Other parameters resulting from such a choice and a reactor thermal rating of 220 MW, is given in Table 1 A system electrical output of 100 MW could be expected under Koeberg sea water temperature conditions of 18°C

The thermodynamic cycle and flow diagram of the system, operating under full power conditions, is shown in Figure 1, coinciding with the process properties of the various conditions given in Table 1

Contrary to the traditional, an inventory method of power control is utilised This causes the reactor inlet and outlet temperatures to be constant, minimising thermal stress cycling, whilst the speed of the turbo machines, its' blade settings, volume flow rates and their pressure ratios are all constant over the entire operating range, from full to 4% power A requirement that the system must respond to up to a 10% of load, step function in load, necessitates variable stator blade geometry on the turbo machines because inventory transfer to and from storage is relatively slow Other special measures required to assist inventory control under transient conditions, is discussed later

2 SYSTEM COMPONENTS

Design detail of PCU components, as at this preliminary stage, is given below The design requirement for the present first phase of the contract is to generate such detail, that reliable cost and performance figures would be obtained Off-design and detail performance of the turbo machines is therefore not available yet The PCU general assembly is shown in Figure 2

2 1 LP TURBO

The LP Turbo consists of a 9 stage axial flow compressor mounted on the same shaft as its driving single stage, parallel flow axial turbine The rotor is suspended by electromagnetic bearings Rotor speed is chosen such that an optimised specific speed is found without compromising turbine blade stresses Elementary turbine rotor blade cooling will be utilised to increase blade creep life and mtbo Stator blades have a single cooling channel which will also scavenge hot gas ingress to the disc front and the inter-disc space

Table 1. PMBR Process Parameters

Position	Parameter	Unit	Value
Cycle	Core mass-flow	kg/sec	117.8
	Reactor thermal power	MW	220
	Reactor outlet temperature	°C	900
	Alternator electrical output	MW	101.6
	Overall pressure ratio	—	2.7
	Cooling water temperature	°C	18
	Reactor cooling massflow	kg/sec	4,4
	Alternator efficiency	%	97,5
	Cycle efficiency	%	47.9
	System efficiency	%	46.2

Turbo		Unit	HP comp	LP comp	HP turb	LP turb	Power turb
	Shaft speed	r/min	18 000	15 000	18 000	15 000	3 000
	Isentropic effy	%	87	87	89	89	90
	Inlet diameter	mm	476	586	573	661	1487
	Hub-tip ratio	—	0,7	0.664	0,89	0,90	0.928
	Blade tip speed	m/sec	449	460	540	520	234
	Load coefficient	—	0,306	0,306	1,6	1,7	2,2
	Flow coefficient	—	0,51	0,51	0,77	0,75	1,02
	No stages	—	9	9	1+1	1+1	8
	Power	MW	48,5	48,5	49.1	49,2	105,3
	Cooling flow	kg/sec			0.13	0.12	12.22

System	Condition	Pressure MPa abs	Temperature °C
	1	2,593	27
	2A	4,277	105
	2a	4.271	26
	2	7,0	103
	3	6,972	558
	4	6,825	900
	5A	5,581	820
	5a	4,498	740
	5	2,65	570
	6	2,594	141

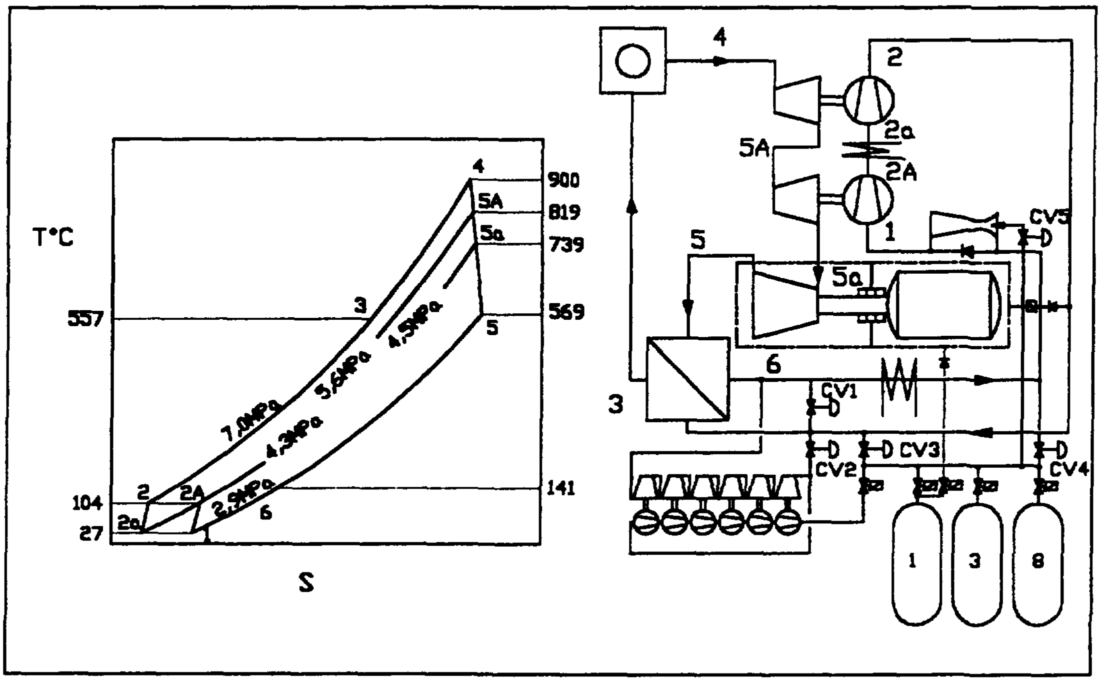

FIG. 1. Thermodynamic cycle and flow diagram.

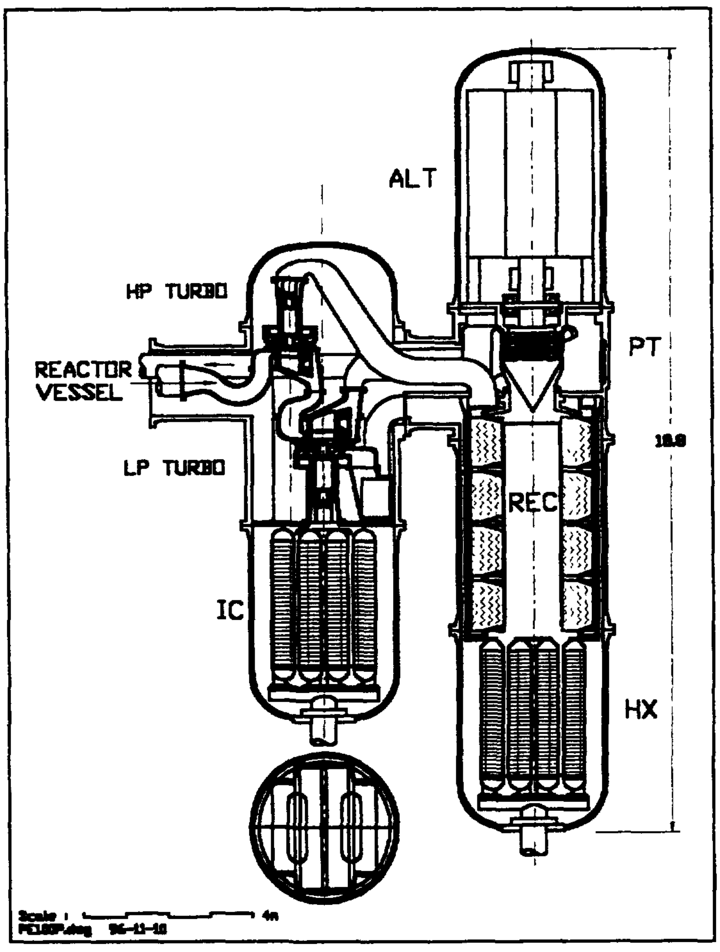

FIG. 2. Eskom PBMR PCU lay-out.

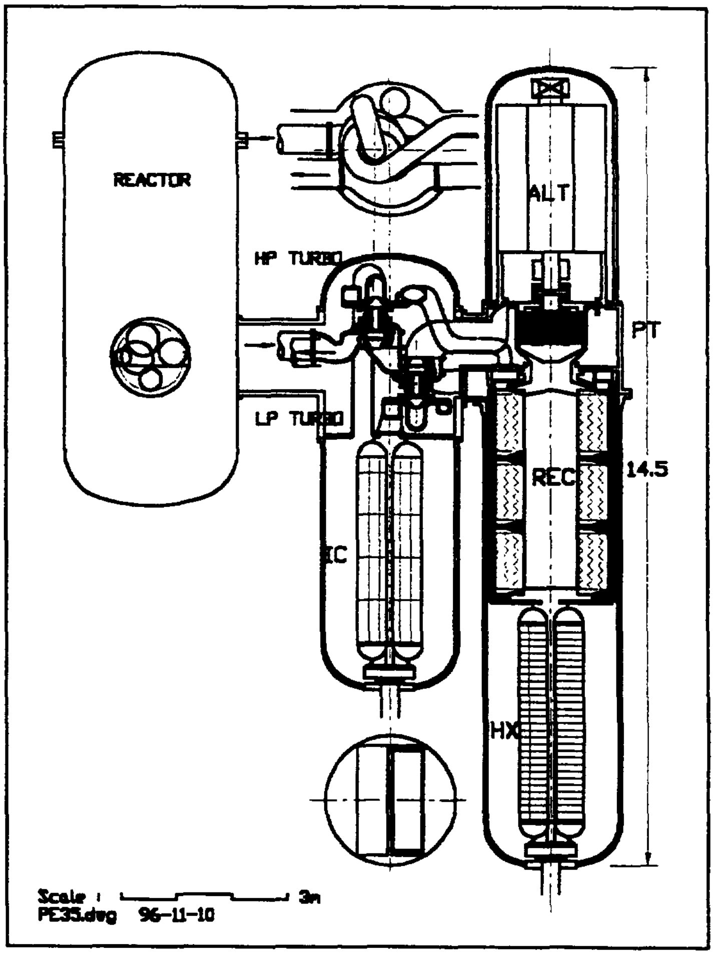

FIG. 3. 35MWe PCU lay-out.

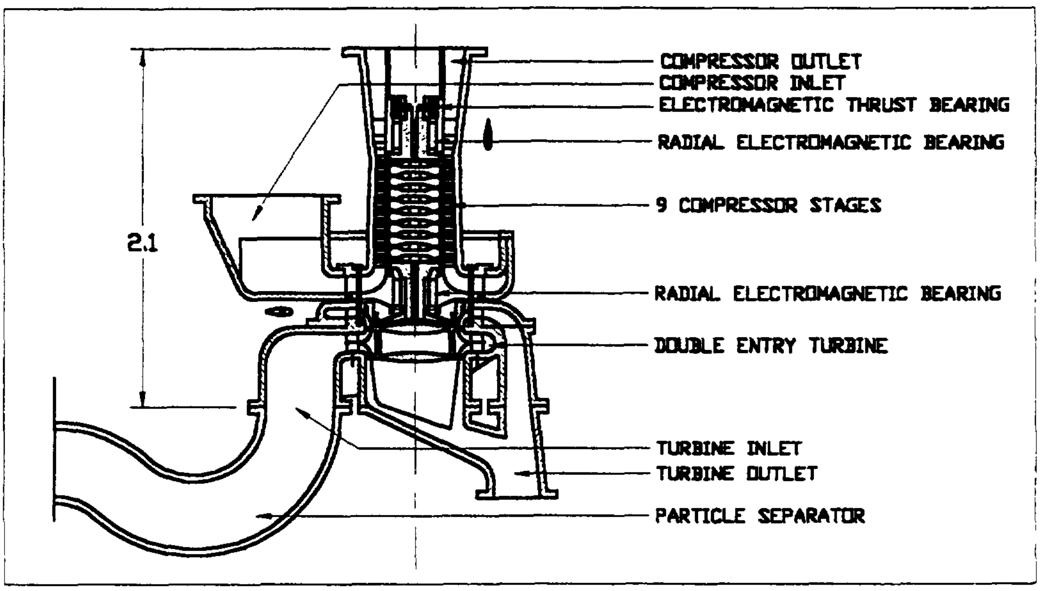

FIG. 4. HP Turbo lay-out.

Mechanical lay-out is shown in Figure 4 The turbine end overhangs the radial bearing, so that the bearing compartment could be kept at a low temperature It also obviates the need to cross the hot turbine stream for bearing services Compressor and turbine is mounted such, that minimum thrust force is generated The turbo compartment is kept at system high pressure (condition 3) from where cooling and sealing flow could be drawn

Apart from improved maintenance resulting from the electromagnetic bearing having no friction, and therefore no wear, the absence of vibration transmitted to the base structure is also an advantage Rotor dynamics is designed such that the bending frequency is at least 30% higher than the maximum over-speed rotational frequency This simplifies the control algorithm, making the use of identical bearing components for the two turbos possible and improves bearing reliability

Aerodynamic design is biased towards maximum efficiency with compressor and turbine characteristics shown in Table 1 Secondary losses on the compressor blades will be counteracted with a small blade tip clearance due to the electromagnetic bearing and casing treatment over the rotor swept area

Compressor and turbine is fitted with variable geometry stator blades Turbine mass flow could thereby be changed from 120% down to 5% of nominal The compressor could only handle +25% down to -22% of design volume flow rate at the design pressure ratio The 10° blade angle adjustment needed for this range can be accomplished with servo-actuators within half a second

2 2 HP TURBO

The HP Turbo is a scaled-down version of the LP Turbo, with practically the same design parameters The higher turbine inlet temperature makes rotor blade cooling more critical and it requires a larger cooling gas flow rate Comparative dimensions of the two turbos are given in Table 1

2 3 POWER TURBINE

The mechanical design of this machine is stretched towards a very high hub-tip ratio of 0,928, caused by the requirement that turbine speed has to be 3000 r/min, driving a 50 Hz alternator directly This mechanical complication more than compensate for the gearbox or frequency converter which would otherwise had to be used The power turbine and alternator is protected from an overspeed condition if the normal control malfunctions, by a turbine by-pass valve

Under full power the turbine has a load coefficient of 2.2 and a flow coefficient of 1 02 at a blade tip speed of 234 m/sec, which are all in line for a high efficiency The stator blades are adjustable over a range of 80% to 150% of nominal mass flow

The rotor shaft is also of overhung construction, mounted directly on the alternator shaft the rotor is suspended on electromagnetic bearings, which are located in the alternator compartment This volume is continuously purged through filters from condition 2a, escaping via a labyrinth to condition 6, which anchors the pressure in the compartment

2 4 RECUPERATOR

A compact design utilising a perforated fin plate has been used, because of the very large heat transfer which has to be accomplished by this unit The flow channels are wedged in order to fit into cylindrical format, as shown by the lay-out drawing, Figure 5 The 1300 sectors are joined together by a top and bottom mounting ring to form a donut shaped cassette The pressure difference between the two sides is 4MPa and only a small defect will cause a major mass-flow by-pass It is therefore important to finally inspect and locally repair the construction, which is possible with this lay-out A further advantage is in the system integration, where no complicated ducting is required

Four cassettes are stacked on top of each other to finally compose a 3 76 m length of finned counter flow channel Recuperator performance is listed in Table 1

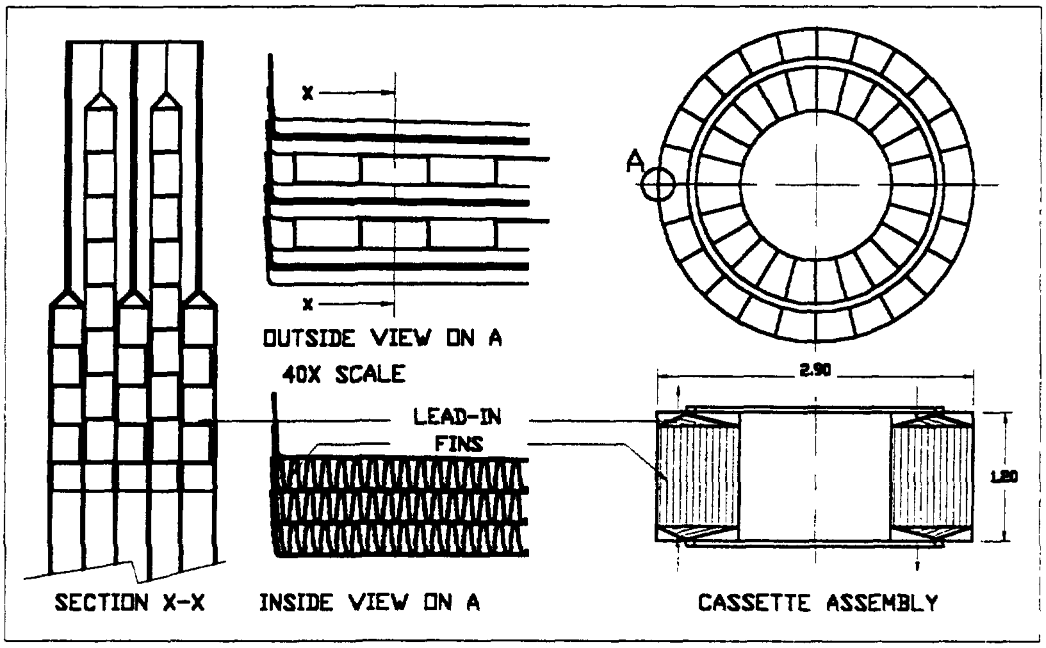

FIG. 5. Recuperator construction.

2.4. HEAT EXCHANGERS

The thermodynamic loading of system cooler and intercooler is so close that identical units could be used for both. As shown in Figure 2, a unit consists of 4 tube bundles supported by a frame. Every bundle has a 2-pass group of 8 externally finned tubes, such that the entire flow path consists of 16 rows arranged in triangular pitch. The 15,75 mm inner diameter tubes, finned to a diameter of 34 mm is used on 38 mm pitch with a length of 3,4 m. The 8 different U-tubes are rolled and welded into the tube sheet, onto which the water manifold is welded. The tubes are mid-way supported to avoid vibration.

Water supply to the coolers is from a plate water-to-water heat exchanger to isolate the water side from sea water and to have a controllable volume where gas leakage could be detected. The supply from these isolation heat exchangers could be done at such a low pressure that leakage into the helium system is impossible.

2.5. ALTERNATOR

The construction of this machine is not resolved yet. Integration with the turbine on the same shaft with electromagnetic bearings is envisaged.

3. CONTROL

Power control is achieved by changing system inventory. This is done by transferring helium to or from a system of storage vessels and process. Ramping up takes place by letting helium from storage to system low pressure line (condition 1) within pressure differential which will ensure a rate of 1MW/min. Ramping down is done from system high pressure (condition 2) to storage, but under special conditions has to be augmented by a set of inventory compressors. System load may drop faster than manageable by this equipment, when a by-pass valve from condition 2 line to condition 6 line will cause the necessary load rejection.

The system will cope with power changes in the order of ± 10% of load per second by changing the stator blade setting simultaneously on the two compressors and three turbines.

Table 2. 35MW System Process Parameters

Position	Parameter	Unit	Value
Cycle	Core mass-flow	kg/sec	58.7
	Reactor thermal power	MW	84
	Reactor outlet temperature	°C	900
	Alternator electrical output	MW	36.4
	Overall pressure ratio	—	1.9
	Cooling water temperature	°C	28
	Reactor cooling massflow	kg/sec	2.4
	Alternator efficiency	%	96
	Cycle efficiency	%	43.8
	System efficiency	%	41.6

Turbo	Parameter	Unit	HP comp	LP comp	HP turb	LP turb	Power turb
	Shaft speed	r/min	24 700	22 100	24 700	22 100	3 000
	Isentropic effy	%	81	81	89	89	90
	Tip diameter	mm	467	523	368	408	1300
	Hub-tip ratio	—	—	—	0.84	0.84	0.94
	Blade tip speed	m/sec	603	606	476	473	205
	Load coefficient	—	0,75	0,75	1.5	1.5	1.65
	Flow coefficient	—			0,72	0.65	0,79
	No stages	—	1	1	1+1	1+1	8
	Power	MW	17.0	17.0	17.2	17.2	38.3
	Cooling flow	kg/sec			0.1	0.09	1.72

System	Condition	Pressure MPa abs	Temperature °C
	1	3.684	40
	2A	5,078	93
	2a	5,078	37
	2	7,0	89
	3	6,988	647
	4	6,974	900
	5A	6.099	845
	5a	5.29	789
	5	3.712	666
	6	3.684	148

Table 3. PCU Component cost, US $'000

Component	Eskom PBMR	35 MWe Unit
HP Turbo	1 800	900
LP Turbo	2 000	1 000
Power turbine	2 300	1 100
Recuperator	3 500	1 800
Cooler	1 200	600
Intercooler	1 200	600
Pressure vessels	3 200	1 900
Pipework	800	400
Alternator	5 000	1 900
Total	21 000	10 200

3 3 START-UP

The system will be brought into power ready state by the following steps

Fill system with helium at 150kPa abs

Start low volume cooling water circulation

Start the reactor and ramp the temperature to 400C

Energise the jet pump from the low pressure storage vessel This will cause a helium circulation in the system and if the turbo's blade angles are set correctly, they will start to turn and accelerate

Ramp reactor temperature to 900°C and increase inventory to 20% The by-pass valve must be used to control the speed of the turbos This is necessary to ensure an alternator enclosure pressure of at least 500kPa abs, in order to avoid flash-over

The power turbine can now be brought to synchronous speed by means of it's stator blades and switched to take over the house load Full cooling water circulation could then be switched

When switched to the grid, the system can start at any power level lower than 20% with the closure of the by-pass valve and ramp from there at a rate of 1MW/min

4 SMALLER SYSTEMS

By sacrificing power capacity, a reactor system could be built which will never need a re-fuelling activity Such a system, using a 20 year PAP reactor with centrifugal compressors instead of axial machines, is configured to be manufactured with the present SA capability The penalty paid by such a system is

Power capacity 35MWe vs 100MWe

System efficiency 43% vs 47%

Although the fuel recycling system is absent, the reactor vessel is marginally smaller in diameter, but longer to hold the 20 years' supply The reactor cost per MW could then be more than double that of the Eskom PBMR version

A typical lay-out is shown in Figure 3, with coinciding process parameters in Table 2 The turbine arrangement of the turbo units is shown as two stages in series, whereas a parallel arrangement will be used, as with the Eskom PBMR machines

5 PCU COST

No cost figures are available at this stage Cost objectives of the PCU components are given in Table 3, quoted in US $

XA9846683

SIMULATION OF THE PEBBLE BED MODULAR REACTOR NATURAL AIR CONVECTION PASSIVE HEAT REMOVAL SYSTEM

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Abstract

Cooling of the Pebble Bed Nuclear Reactor under evaluation in South Africa is primarily effected by the flow of helium through the cavity which contains the nuclear fuel. However, apart from this, a certain amount of heat flows from the reactor cavity, through the graphite barrel and reactor vessel to the containment building and ultimately to the environment During normal operation this passive heat loss represents approximately 1MW for a 100MW reactor and constitutes an undesirable loss of power. In the event of a shutdown or loss of main coolant, however, this passive heat removal is relied upon to remove the decay heat from the core. A study was initiated to simulate the process of this heat removal to provide an indication of the maximum vessel temperature and power transfers after shutdown. However, there is a lack of precise data indicating values for thermal conductivity, heat transfer coefficients, heat capacities or even densities. This paper describes the assumptions made and the manner in which these data were estimated so as to provide what is hoped to be a reasonably accurate estimate of the behaviour of the passive heat removal process.

1. INTRODUCTION

The Pebble Bed Nuclear Reactor being studied in South Africa is a high temperature helium cooled nuclear reactor with a Brayton cycle for the power conversion unit. The cooling of the reactor is mainly effected by the flow of helium through the cavity which contains the nuclear fuel. Apart from that, however, a certain amount of heat flows from the reactor cavity, through the graphite barrel and reactor vessel to the containment building and ultimately to the environment. This is what is termed the passive heat loss from the core. During normal operation this heat loss amounts to approximately 1MW for a 100MW reactor and constitutes an undesirable loss of power. In the event of a shutdown or loss of main coolant, however, this passive heat removal is relied upon to remove the decay heat from the core.

The purpose of this study was to simulate this process of heat removal to give an indication of the maximum vessel temperature and power transfers after a shutdown. This study formed part of the work involved in producing an engineering simulator for the pebble bed reactor.

The main problem encountered during this study was the lack of precise data indicating values for thermal conductivity, heat transfer coefficients, heat capacities or even densities. This paper describes the assumptions made and the way in which these data were estimated so as to provide what is hoped to be a reasonably accurate estimate of the behaviour of the passive heat removal process.

2. DESIGN BASIS

Figure 1 shows the civil structures that were taken into account in the study.

Discussing these briefly, we see the reactor cavity (labeled "gas" in the figure), surrounded by a graphite barrel. The incoming helium flows through channels inside this barrel before entering the cavity. Between the barrel and the reactor vessel there is a narrow space that is intended to carry a small "leak flow" of cold helium taken from the highest pressure point of the main coolant loop. This leak flow has a temperature of approximately 90 degrees Celsius and keeps the reactor vessel cool. The leak flow enters the cavity after cooling the vessel.

For power calculations, the graphite barrel is in fact considered to be made up of three parts. An inner graphite barrel, a middle graphite barrel through which the inlet channels pass and an outer carbon insulating barrel.

The vessel exterior is in contact with the air in the containment building. A passive cooling structure consisting of hollow panels assists in cooling the vessel. This is called the reactor cavity cooling system (rccs). Air flows from the environment into the lower part of the panels, up through panels and out back to the environment through a chimney. The vessel transfers heat to the rccs via conduction and radiation. The main purpose of the rccs is to keep the vessel cool in the case of failure of the "leak flow" cooling. A similar vent cooling system exists with the purpose of cooling the containment pit.

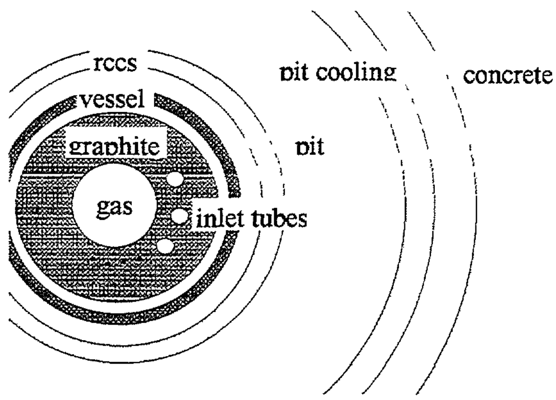

Figure 1 : Passive Heat Removal Structures - Plan

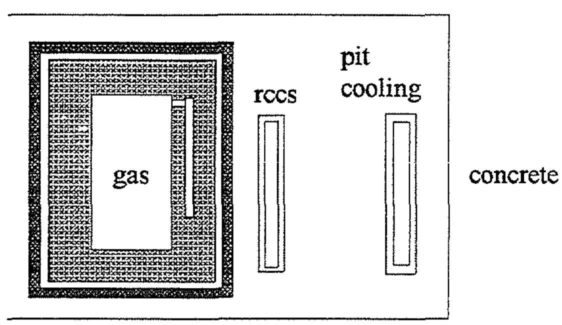

Figure 2 : Passive Heat Removal Structures - elevation

The civil structures, and the spaces between them have the following dimensions (SI units). These dimensions were obtained from civil drawings.

Table 1 : Passive Heat Removal Structure Dimensions

structure	inner diam	outer diam	Hght	Thick ness	inner area	outer area	vol
core gas	-	3.6	10	-	-	158	57
graphite	3.6	4.7	14	0.5	158	206	113
carbon (steel liner)	4.7	5.8	14	0.5	206	255	113
HE inlets	-	0.1	11	-	-	221	5.5
He leak flow	5.8	6.0	14	0.1	255	263	26
Vessel	6.0	6.3	16	0.15	300	316	46
vessel-rccs air	6.3	6.5	9	0.1	178	183	18
RCCS steel	6.5	7.1	9	0.3	183	200	30
RCCS air	6.5	7.1	9	0.3	183	200	57
pit air	7.1	9	20	0.95	446	565	480
pit vent steel	9	9.3	14	0.15	395	409	61
pit vent air	9	9.3	14	0.15	395	409	60
concrete	9.6	12.4	22	1.4	663	857	1064

The structures are assumed to have the following characteristics (SI units). The density of gas is calculated using the ideal gas law. (R for helium = 2078 J/KgK. R for air = 287 J/KgK)

Table 2 : Passive Heat Removal Structures Characteristics

structure	density (Kg/m^3)	thermal conductivity (W/mK)	specific heat capacity (J/KgK)	emissivity (W/mK^4)
core gas	P/RT	see note	5193	-
graphite	2240	168	710	-
carbon (steel barrel)	3510	8.9	854	0.4e-8
HE inlets	P/RT	see note	5193	-
He leak flow	P/RT	see note	5193	-
Vessel	7850	55	500	2.4e-8
vessel-rccs air	P/RT	see note	1005	-
RCCS steel	7850	55	500	0.4e-8
RCCS air	P/RT	see note	1005	-
pit air	P/RT	see note	1005	-
pit vent steel	7850	55	500	0.4e-8
pit vent air	P/RT	see note	1005	-
concrete	2400	1.5	880	5.3e-8

NOTE

The transfer of heat from a solid wall to a gas (helium or air) and vice versa is calculated using the Grigull equation which produces a heat transfer coefficient dependent on various parameters of the gas and the wall and which takes into account natural convection.

3. MODEL OVERVIEW

The model is broken up into the following sections

- Power transfers
- Temperature evolutions

4. POWER CALCULATIONS

Power flow due to conduction through a solid body is calculated using the following form of equation

$$P = \lambda \frac{A}{l}(T_{hot} - T_{cold}) \tag{1}$$

In the case of power flow from a solid to a gas, natural convection of the gas is assumed and the following formulae are used

$$P_{wall-gas} = \lambda_{conv} A(T_{wall} - T_{gas}) \tag{2}$$

where

$$\lambda_{conv} = \frac{\mu \lambda_{gas}}{H_{wall}} \tag{3}$$

and

$$\mu = 0.55(G_r P_r)^{1/4} \qquad \text{for } 1700 < GrPr < 1e8 \tag{4}$$

$$\mu = 0.13(G_r P_r)^{1/3} \qquad \text{for } GrPr > 1e8 \tag{5}$$

Pr is the prandtl number for the gas given by

$$P_r = \frac{\eta c_p}{\lambda_{gas}} \qquad \text{where } \eta \text{ is the gas dynamic viscosity} \tag{6}$$

$$G_r = \frac{g\beta \Delta T \rho^2 H^3}{\eta^2} \tag{7}$$

where
g is gravitational constant 9.8
β is the volume coefficient of expansion (1/Tgas for a gas)
ΔT is abs(Twall - Tgas)
ρ is the gas density (calculated using ideal gas law PV=mRT)
H is the height of the wall
η is the gas dynamic viscosity

Power flow due to radiation from one solid body to another is calculated as follows

$$P = \frac{A(T_{hot}^4 - T_{cold}^4)}{(\frac{1}{\varepsilon_{hot}} + \frac{1}{\varepsilon_{cold}} - \frac{1}{\varepsilon_{bbody}})} \tag{8}$$

where
ε is the emissivity of the body concerned
A is the exchange area between the two bodies.

Table 3 : Decay Heat Power Profile

time after shutdown (seconds)	power (MW)
2 s	75
200	40
400	7.4
600	4.4
1000	3.5
2000	3
3000	2.7
5000	2.2
7500	1.8
11500	1.5
15000	1.3
25000	1.15
30000	1

5. ASSUMPTIONS

All metal structures are assumed to have an emissivity corresponding to unpolished aluminium except the reactor pressure vessel which is assumed to be treated so as to have an emissivity corresponding to something between unpolished brass and oxidized copper.

Heat transfer from the rccs and pit cooling metal structures to the vent air is assumed to be according to natural convection.

Heat transfer from the rccs and pit cooling vent air to external air is calculated on the basis of a certain flow rate. Two base cases are considered - 0.1m/s and 10m/s. This can be maintained either by natural convection or forced convection. The way in which the flow is guaranteed is irrelevant to this study. It is considered, however, that these flow rates are not unrealistic for natural convection.

The profile of the power generated in the core after a shutdown is shown below. This profile is not significantly affected by whether the shutdown is due to rod insertion or loss of coolant (negative temperature reactivity). The temperature calculations always assume that flow of the main coolant is reduced to zero.

6. POWER FLOW PATHS

The following power flows are considered

Conduction with natural convection coefficient
- core gas to graphite inner barrel
- graphite mid barrel to inlet gas

Conduction
- graphite inner to graphite mid barrel
- graphite mid barrel to carbon outer barrel

Conduction with natural convection coefficient
- carbon outer barrel to helium leak flow path
- helium leak flow path to vessel

Radiation
- carbon outer barrel to vessel
- vessel to rccs
- vessel to concrete

Conduction with natural convection coefficient
- vessel to air between vessel and rccs
- air between vessel and rccs to rccs
- air between vessel and rccs to pit air
- rccs to pit air
- pit air to pit cooling stack
- pit air to concrete
- rccs to vent air
- pit cooling stack to vent air

Radiation
- rccs to pit cooling stack
- pit cooling stack to concrete

Power flow due to mass flow
- Helium flow through core
- Helium flow through leak flow path
- Air flow in rccs vent
- Air flow in pit cooling vent

Optionally, power by conduction from the vessel directly to the rccs is considered.

7. TEMPERATURE EVOLUTIONS

Temperature in a particular structure is calculated as follows

$$\frac{dT}{dt} = \frac{(\Sigma Pin - \Sigma Pout)}{mc_p} \qquad (9)$$

where
m is the mass of the structure
Cp is the specific heat capacity of the structure

8. TEST SCENARIOS

The following test cases were simulated. Temperatures are shown in Kelvin.

8.1 LOFC (with leakflow)

Main helium flow is reduced to zero but the leak flow is maintained.
The rccs is assumed to be separated from the vessel.

Results are shown in figure 3.

The fuel and gas temperatures rise sharply at first and then start to fall off as the decay power reduces and the heat is slowly evacuated to the graphite barrel.
At the same time the graphite temperature rises slowly.
The vessel temperature stays constant due to the cooling effect of the helium leak flow.

8.2 LOFC (no leakflow) - vent air flows about 0.1 m/s

Main helium flow is reduced to zero and helium leak flow is lost.
The rccs is assumed to be separated from the vessel.
An air flow of 0.1m/s is assumed in the cooling stacks which seems conservative when thinking of natural convection at a temperature of about 320K.

Results are shown in figure 4.

The vessel temperature rises due to the loss of leakflow cooling. A peak of about 650K seems likely.

8.3 LOFC (no leakflow) - vent air flows about 10 m/s

This is the same test as above except that the vent air flow is assumed to be 10m/s.

Results are shown in figure 5.

The difference in vessel temperature is very small. A peak of about 630K seems likely.
The effect on pit air temperature is more pronounced.

8.4 LOFC (no leakflow) - vent air flow 10m/s - vessel joined to rccs

This test is the same as above but the rccs is assumed to be physically attached to the vessel. The thermal conductivity is assumed to be that of steel divided by ten. This is to allow for perhaps only a 10% effective join between rccs and vessel (to allow inspection channels or air flow paths for example)

Results are shown in figure 6.

The result is that the graphite barrel temperature is marginally reduced and that the vessel temperature is significantly reduced (without apparent effect on pit air temperature however).
A peak vessel temperature of about 550K seems likely.

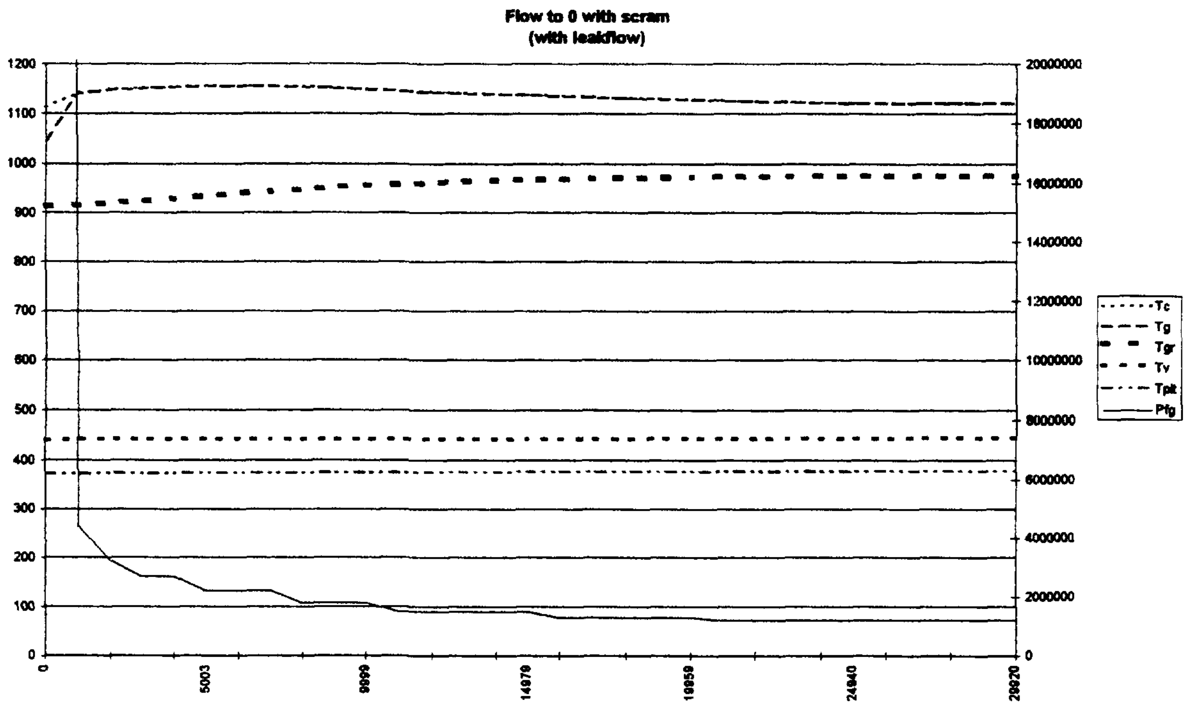

Figure 3 : LOFC with leakflow

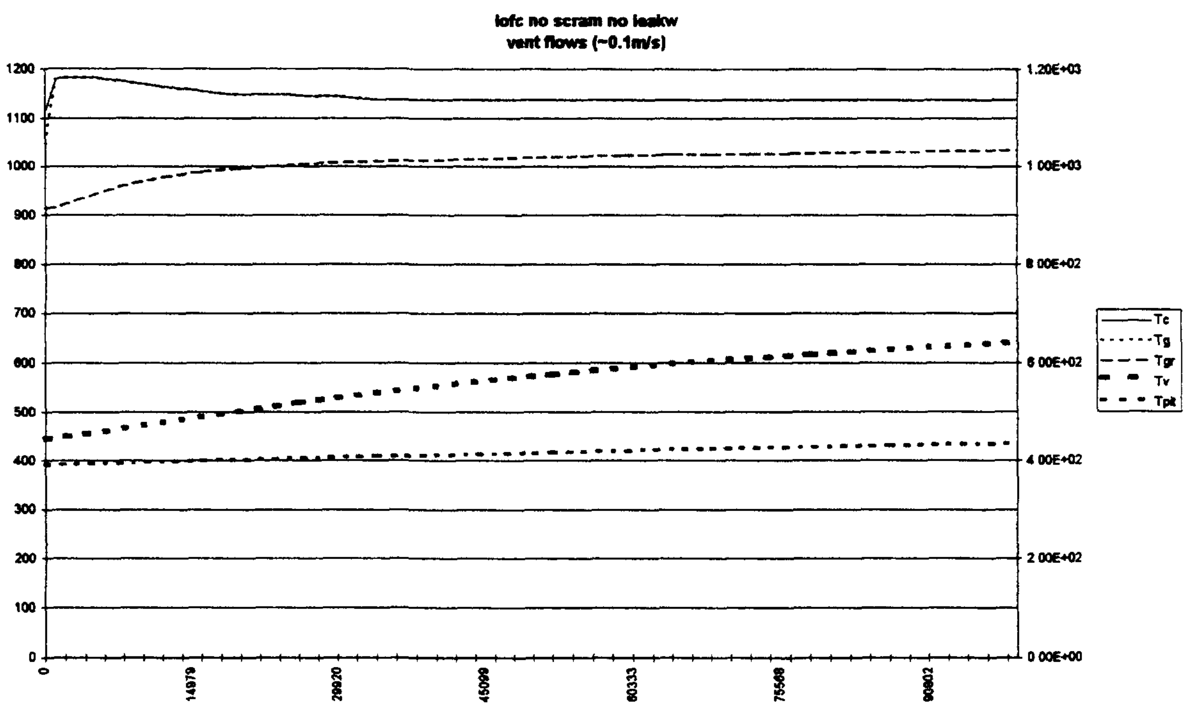

Figure 4: LOFC - no leakflow - 0.1m/s vent air flow

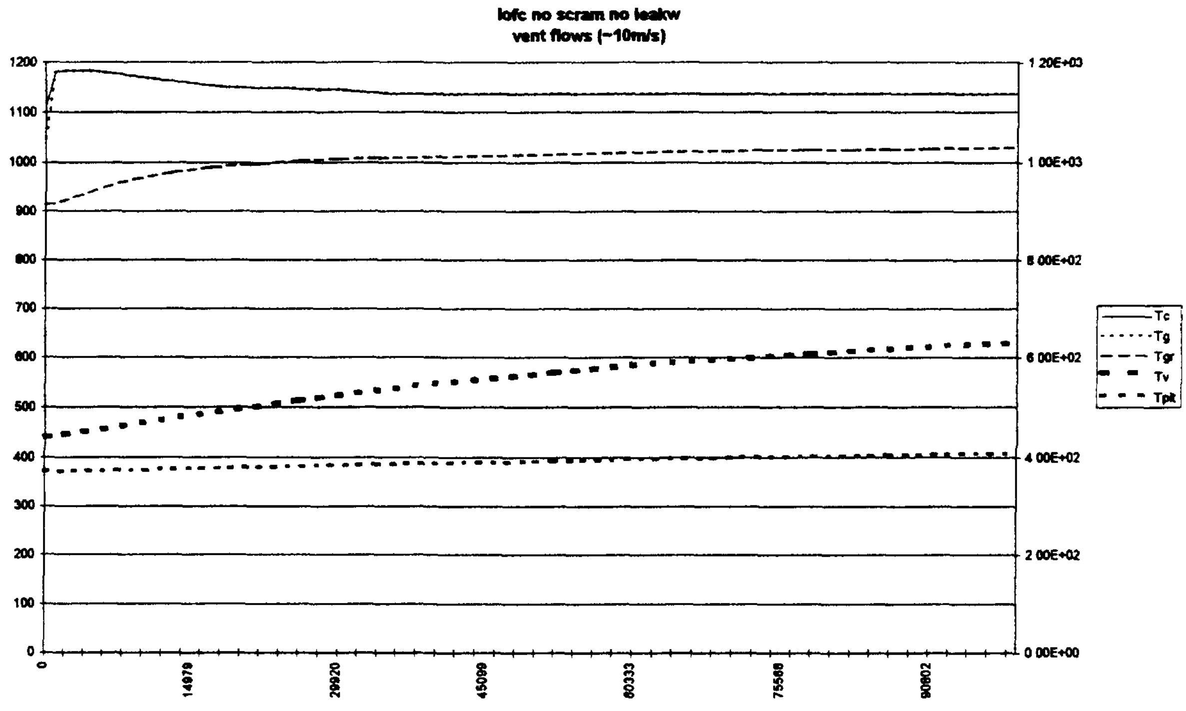

Figure 5 : LOFC - no leakflow - 10m/s vent air flow

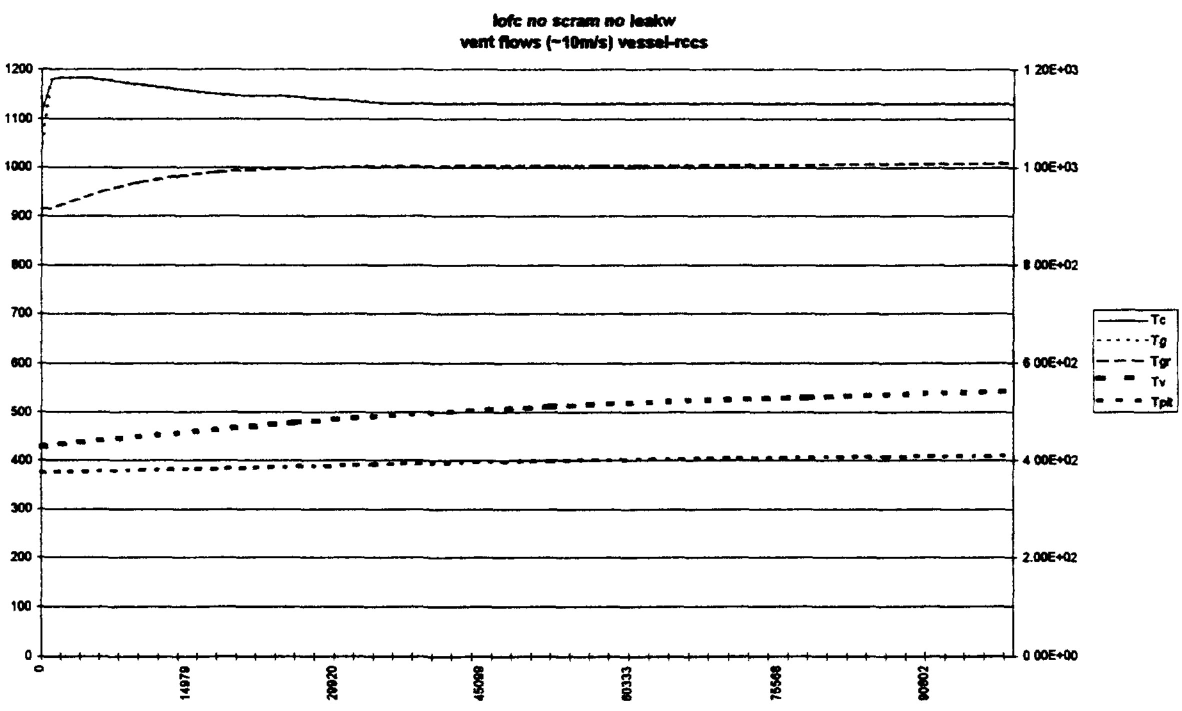

Figure 6 : LOFC - no leakflow - 10m/s vent flow - vessel touching RCCS

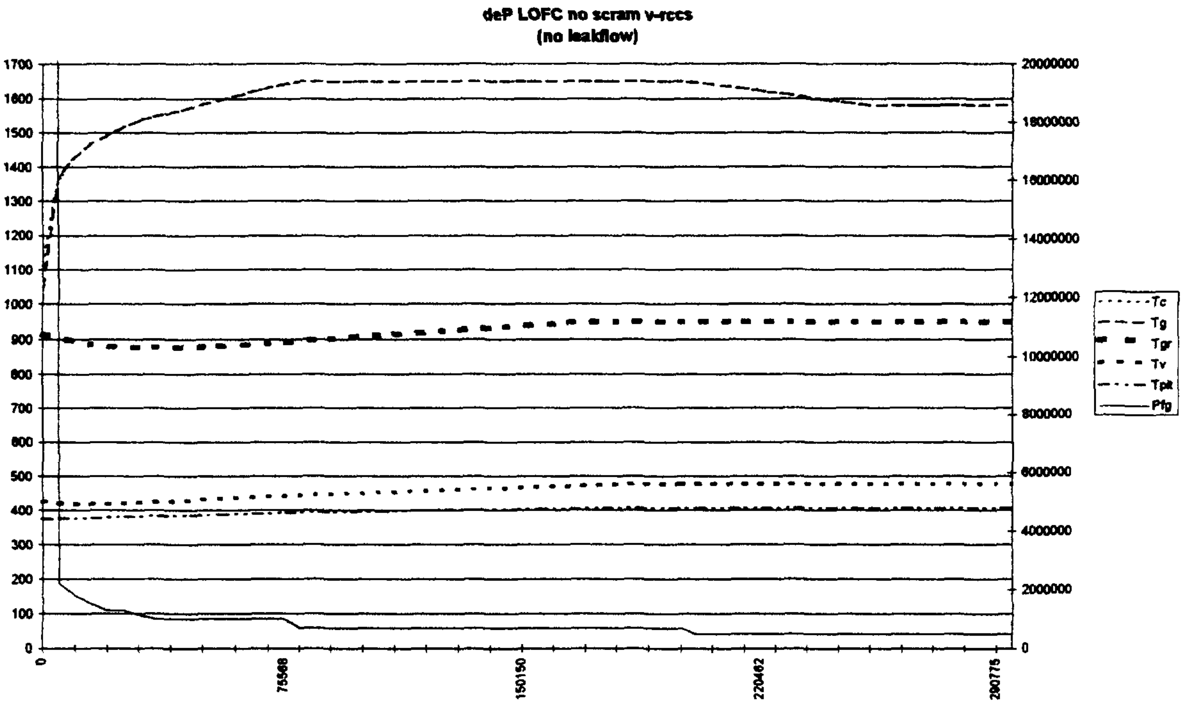

Figure 7: Depressurised LOFC - no leakflow - vessel touching RCCS

8.5 Depressurised LOFC (no leakflow) - vessel joined to rccs

The last test shows the temperature evolutions following a depressurised LOFC accident.

The results are shown in figure 7.

In this accident the pressure in the core, core inlet tubes and leak flow channel is reduced to atmospheric pressure. The characteristics of the gas are assumed to remain those of Helium (i.e. no air ingress).

A surprising result is that the temperature of the graphite barrel and the vessel decrease to begin with. This can be explained by virtue of the fact that the conductivity (including contribution due to convection) of the depressurised helium is considerably less effective than at high pressure. This means that the heat flow from core gas to core barrel has been abruptly reduced following depressurisation. But radiation from the barrel to the vessel continues. Therefore the barrel temperature will reduce.

Similarly with the vessel. On depressurisation, conductive heat flow from the barrel via the leak flow channel to the vessel is reduced. But heat removal via the pit and rccs is unchanged. Therefore the vessel temperature will reduce.

Because heat removal has been made less effective, the core gas and fuel temperature will rise considerably until the elevated temperatures cause increased heat power flow from the core.

9. CONCLUSION

The conclusions that can be drawn from the above study are the following.

Firstly, the assumptions made for characteristic data and coefficients produced realistic results without having had the need to tune them to expected results.

Natural air convection alone appears capable of limiting the vessel temperature to below 650K (380 degrees C). The cooling of the reactor vessel can be dramatically improved by having conductive contact between the vessel and the cooling stack.

The effect of reduced air flow in the cooling stacks has a minimal effect on the vessel and core temperatures in the time scales considered in this study. The consequence of this is that a failure or blockage of the stacks need not be considered important since one has about 10 hours available for corrective action.

FOLLOW UP ON AIR COOLED PASSIVE HEAT REMOVAL

During the presentation on the air cooled passive heat removal system it was observed that vent flows in the rccs and pitvent stacks of 0.1m/s could not possible evacuate the decay heat of the core and that vent flows of 10m/s were unrealistic.

The pages below indicate power flows and heat xfer coefficient in the rccs at beginning and end of a typical transient. On studying the data the following observations can be made.

It is important to avoid drawing false conclusions from the passive heat removal paper. Remember that the initial conditions for all the transients studied were a steady state full power core that had been cooled by the helium bypass flow so that the vessel and graphite barrel were cool. This means that during the shutdown transient in which helium bypass flow is absent a lot of the energy that is produced becomes stored in the graphite barrel and vessel and goes into heating these structures up. This energy is later drawn out slowly on the conntainment side to be evacuated by the cooling stacks at greatly differing rates depending on the vent flow of 0.1 or 10.0 m/s. Studying the data below we see that at 0.1 m/s the power flow out of the stacks after 3 hours is about 25KW each - virtually nothing. At 10 m/s the power flow after 3 hours is about 100KW in the rccs and about 176KW in the pit cooling stack - just over 1/4MW together.

This means that the air cooling system is able to maintain the vessel temperatures at levels shown in the paper only if the initial condition is as stated above and presuming that the shutdown occurs in a normal way. A sustained ATWS with longterm elevated temperatures - for example - would most likely result in elevated temperatures of varying degrees in different parts of the system. At best, therefore, the air cooled system can provide a support service together with other cooling structures to provide vessel cooling and decay heat evacuation.

A quick aside to look at the approximate amount of energy that is stored in the graphite and vessel during the transients described in the paper.

The relevant data are

Mgraphite = 508480Kg	Mvessel = 361100Kg
Cp graphite = 710	Cp vessel = 588

Consider a case where in 30 hours we observe the following temperature rises

delta T graphite = 110	delta T vessel = 200

The total energy stored in the system is given by Q = M.Cp.dT

Qgraphite = 39 7e9 J	Qvessel = 42.4e9 J

The total energy evacuated by the stacks is very approximately estimated by a power flow of 27.6KW to be 27.6e9 J

The above three terms total to 109.7e9 J

Compare this to the decay heat produced after 30hrs estimated at about 117e9 J.

The missing energy can be easily accounted for as being stored in various other structures such as the cooling stack metal parts and the concrete.

Vent flows at 10m/s

t	Pv-airin	Rv-rccs	Prccs-air	Prccsflow	Ppvent-air	Ppventflow
Wrccs	Wpvent	HXcoeff-rccsair	Wvolrccs	Wvolpvent		
0 00	0 00	0 00	0 00	506115 09	0 00	542266 19
71 94	53 96	0 00	10 00	10 00		
100 00	98865 28	64079 92	99192 20	99192 61	189069 86	189069 98
71 94	53 96	5 57	10 00	10 00		
200 00	99532 82	64277 24	99192 20	99192 61	188912 25	188922 70
71 94	53 96	5 57	10 00	10 00		
300 01	99956 82	64511 93	99192 20	99192 61	188757 03	188767 14
71 94	53 96	5 57	10 00	10 00		
400 02	100323 80	64747 02	99192 20	99192 61	188601 80	188611 59
71 94	53 96	5 57	10 00	10 00		
500 02	100690 98	64982 49	99192 20	99192 61	188446 59	188456 03
71 94	53 96	5 57	10 00	10 00		
600 00	101058 42	65218 35	99192 20	99192 61	188291 45	188300 47
71 94	53 96	5 57	10 00	10 00		
9400 90	136032 08	89316 53	101976 77	101968 37	176198 77	176208 38
71 94	53 96	5 61	10 00	10 00		
9500 51	136418 97	89612 26	102057 05	102047 80	176198 77	176208 38
71 94	53 96	5 61	10 00	10 00		
9600 12	136806 13	89908 52	102137 34	102127 23	176198 77	176208 38
71 94	53 96	5 61	10 00	10 00		
9699 73	137193 63	90205 33	102217 56	102208 87	176198 77	176208 38
71 94	53 96	5 61	10 00	10 00		
9799 34	137581 36	90502 69	102297 90	102288 31	176198 77	176208 38
71 94	53 96	5 62	10 00	10 00		
9898 95	137969 36	90800 59	102378 13	102369 95	176198 77	176208 38
71 94	53 96	5 62	10 00	10 00		
9998 56	138357 59	91099 05	102458 49	102449 38	176198 77	176208 38
71 94	53 96	5 62	10 00	10 00		

Vent flows at 0 1 m/s

t	Pv-airin	Rv-rccs	Prccs-air	Prccsflow	Ppvent-air	Ppventflow	Wrccs	Wpvent	alpharccsair	Wvolrccs	Wvolpvent
0 00	0 00	0 00	0 00	5061 15	0 00	5422 66	0 72	0 54	0 00	0 10	0 10
100 00	98822 87	64054 63	24106 43	23317 93	21260 21	21053 44	0 72	0 54	3 70	0 10	0 10
200 00	99412 72	64222 54	23590 46	23552 75	21112 03	21093 82	0 72	0 54	3 67	0 10	0 10
300 01	99755 88	64428 05	23613 84	23586 42	21118 46	21109 00	0 72	0 54	3 68	0 10	0 10
400 02	100059 33	64633 92	23644 63	23617 20	21132 33	21122 88	0 72	0 54	3 68	0 10	0 10
500 02	100347 30	64840 16	23675 41	23647 98	21146 19	21136 75	0 72	0 54	3 68	0 10	0 10
600 00	100632 19	65046 80	23706 15	23678 76	21160 16	21150 63	0 72	0 54	3 68	0 10	0 10
9600 12	129685 16	87407 83	26912 55	26867 46	22410 04	22400 65	0 72	0 54	3 77	0 10	0 10
9699 73	130017 10	87673 48	26958 94	26913 87	22424 00	22414 54	0 72	0 54	3 77	0 10	0 10
9799 34	130349 30	87939 66	27005 33	26960 27	22437 87	22428 44	0 72	0 54	3 77	0 10	0 10
9898 95	130681 67	88206 34	27051 72	27006 67	22451 74	22442 34	0 72	0 54	3 77	0 10	0 10
9998 56	131014 31	88473 55	27098 14	27053 07	22465 61	22456 22	0 72	0 54	3 77	0 10	0 10

t	Pvairin	Rvrccs	Prccsair	Prccsw	Ppventair	Ppventw	Wrccs	Wpvent	alpharccsair	Wrccsvol	Wpventvol
0 00	0 00	0 00	0 00	5061 15	0 00	5422 66	0 72	0 54	0 00	0 10	0 10
100 00	98822 87	64054 63	24106 43	23317 93	21260 21	21053 44	0 72	0 54	3 70	0 10	0 10
200 00	99412 72	64222 54	23590 46	23552 75	21112 03	21093 82	0 72	0 54	3 67	0 10	0 10
300 01	99755 88	64428 05	23613 84	23586 42	21118 46	21109 00	0 72	0 54	3 68	0 10	0 10
400 02	100059 33	64633 92	23644 63	23617 20	21132 33	21122 88	0 72	0 54	3 68	0 10	0 10
500 02	100347 30	64840 16	23675 41	23647 98	21146 19	21136 75	0 72	0 54	3 68	0 10	0 10
600 00	100632 19	65046 80	23706 15	23678 76	21160 16	21150 63	0 72	0 54	3 68	0 10	0 10
9600 12	129685 16	87407 83	26912 55	26867 46	22410 04	22400 65	0 72	0 54	3 77	0 10	0 10
9699 73	130017 10	87673 48	26958 94	26913 87	22424 00	22414 54	0 72	0 54	3 77	0 10	0 10
9799 34	130349 30	87939 66	27005 33	26960 27	22437 87	22428 44	0 72	0 54	3 77	0 10	0 10
9898 95	130681 67	88206 34	27051 72	27006 67	22451 74	22442 34	0 72	0 54	3 77	0 10	0 10
9998 56	131014 31	88473 55	27098 14	27053 07	22465 61	22456 22	0 72	0 54	3 77	0 10	0 10

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POTENTIAL AND LIMITATIONS IN MAXIMIZING THE POWER OUTPUT OF AN INHERENT SAFE MODULAR PEBBLE BED HTGR

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Abstract

The past development of modular pebble-bed HTGRs in Germany led to two well-defined reactor designs, namely the 200 MWth HTR-MODUL and the 250 MWth HTR-100 by SIEMENS/Interatom and ABB/HRB, respectively. Recently the South African utility, ESKOM, decided to include the pebble-bed HTGR design as a future supply option.

In contrast to the German designs, ESKOM prefers a direct cycle helium turbine system on the power conversion side. This imposes certain modified boundary conditions on the reactor design and enables a higher plant efficiency.

Nuclear and thermal-hydraulic investigations have been performed at KFA-ISR to determine the potential and limitations of increasing the unit thermal power output of the reactor compared to the former german designs. In doing so an upper limit for the maximum fuel element temperature of 1600°C was observed. The impact of all modifications in view onto the efficiency of the nuclear control and shut-down systems was also considered.

The results obtained so far demonstrate the well-adapted and conservative design of the SIEMENS HTR-MODUL within a 10% safety margin to the higher region. The introduction of graphite noses has a remarkably positive influence on the shut-down and control systems, while the positive effect on the maximum accident temperature depends strongly on the fast neutron dose-related thermal conductivity of the nose graphite. Considering the fact that effective conductivity of the pebble-bed core is maintained at high temperatures, the temperature effect due to the noses are of secondary influence at this point.

1. Introduction

The development of modular pebble-bed HTGRs in Germany during the last 20 years has led to two well-defined reactor designs, namely the 200 MWth HTR-MODUL and the 250 MWth HTR-100 by SIEMENS/Interatom and ABB/HRB, respectively. Although negotiations especially for the HTR-MODUL e.g. with the former Sowjet Union were promising to a certain extent, neither of these two systems has been commercially realised up until now. Nevertheless primarily because of the outstanding inherent safety properties of that reactor concept the world-wide interest in pebble-bed high temperature reactors is not only still remaining but seems to extent further. At the Institute for Nuclear Energy Technology (INET) of the Tsinghua University in Beijing, China, a 10 MW pebble-bed HTR is being built today and recently the South African utility, ESKOM, decided to include the pebble-bed HTGR design as a future supply option into the considerations. Especially this activity may be looked at as a new starting point for the commercialization of this reactor type.

Against this background the question of finding the limits for the unit power output of such reactors was raised again in order to increase the overall system economy.

In contrast to the German designs, ESKOM prefers a direct cycle helium turbine system on the power conversion side. This imposes certain modified boundary conditions on the reactor design, such as a higher gas inlet temperature. On the other hand, the direct cycle enables a higher plant efficiency.

Starting with the well-known 200 MW_{th} HTR-MODUL nuclear and thermal-hydraulic investigations have been performed at KFA-ISR to determine the potential and limitations of increasing the unit thermal power output of the reactor. This paper will first give an overview of some basic considerations to the possibilities and limits for the maximum power output.

Following the consequences of these considerations a more detailed core-design will be presented. Finally the results will be discussed and upper limits for the power output on the basis of today's knowledge will be indicated.

2. General Aspects and boundary conditions for the maximum power output

Modular pebble-bed reactor designs reflect the philosophy of a new quality of safety, the so-called 'catastrophe-free' nuclear reactor technology. Thereby they are governed by three main boundary conditions:

1. Self-acting stabilization of the nuclear power and the fuel temperatures at permissible values in all operational and accident situations
2. Self-acting removal of the nuclear decay heat in all possible accident situations
3. Control-and Shut-down equipment placed outside the pebble-bed.

Item 2 and 3 impose limitations to the reactor geometry, while item 1 restricts the operational excess reactivity bound in movable control elements to certain maximum values. Therefore the reactor designs are characterized by tall and small cylindrical core shapes because of reaching a large surface-to-volume ratio and by a restricted region of part-load operation because of the xenon override problem. The introduction of item 3 into the design conditions was a direct consequence of the german THTR-300 experience, where control- and shut-down rods directly inserted into the pebble-bed led to a significant amount of fuel element damage.

The condition, to control and shut-down the reactor from outside the pebble-bed leads to a very strong limitation of the core diameter for purely cylindrical configurations as can easily be shown by some calculations for the dependence of the maximum effectivity of control systems as a function of the core diameter. The results of such calculations are shown in figure 1. Using the nuclear data for the 200 MW_{th} SIEMENS HTR-MODUL it is found, that the core diameter may not be increased significantly over 3m, when a cold shut-down reactivity efficiency of about 14% nominal is required. An increase of the core diameter to 3.5 m will reduce the reactivity efficiency to less than 75% if a shut-down system like in the HTR-MODUL is used. Even if the density of the shut-down elements is further increased to the theoretical maximum the reactivity efficiency will be less than 85 %, which is too low.

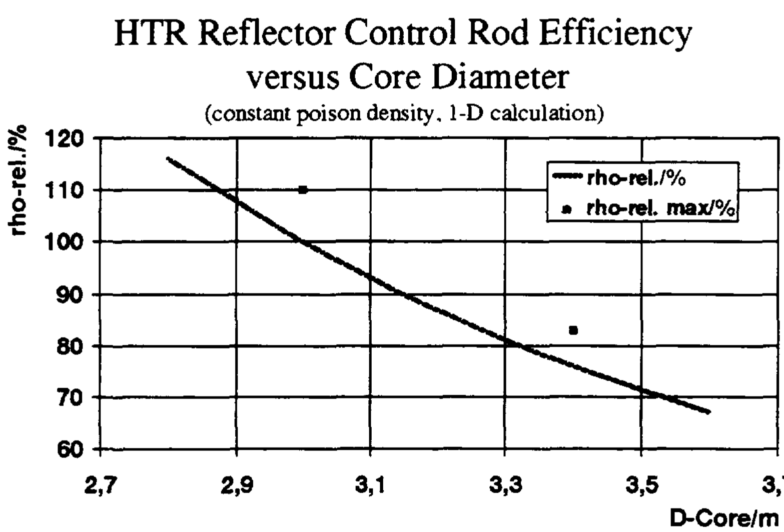

FIG. 1. Reflector Rod Efficiency

As a consequence of this the diameter of a purely cylindrical core is limited to 3 m, if control and shut-down is provided by reflector absorber systems. This situation can only be changed in either going from the pure cylindrical core shape to other geometries or in falling back to a shut-down philosophy with control-rods directly moving into the pebble-bed. A design concept with a different core-geometry will be discussed later on.

Concerning item 2 of the list of conditions, the safe decay heat removal, the situation seems to be a little more flexible. Increasing the core diameter in this case has two basic effects, namely the increase of the path-length for the heat transport inside the core, which will increase the temperature difference over the core, and the increase of the reflector surface, which will decrease the temperature difference over the reflector and outer components up to the surface cooler. A rough estimation already shows, that the second effect is larger than the first, which gives some potential for enlarging the power output with increased core-diameter. A more detailed dynamic calculation of a depressurisation transient shows an even greater potential as is shown in fig. 2.

Maximizing the Power-Output
of an inherent safe MODUL-HTR

Maximum Core-Power as a Function of Core-Radius

Lambda-eff like V S O P

			SC (250°C - 700°C)		GT (600°C - 900°C)	
R_{Core} / m	$D_{Refl.}$ / m	Power / MW	T_{max} / C	$t(T_{max})$ / h	T_{max} / C	$t(T_{max})$ / h
1 50	0 80	200	1476	32	1608	27
1 75	0 55	250	1602	50	1708	38
2 00	0 30	300	1693	64	1776	50
1 50	0 80	230	1588	32	*	*
1 75	0.55	215	*	*	1592	38
2 00	0 30	240	*	*	1598	50
2 00	0 30	270	1601	64	*	*

FIG. 2. Maximum Power output of a Cylindrical Modular Pebble-Bed Reactor

The figure shows the maximum fuel temperature after a depressurisation accident as a function of the core diameter for a typical steam-cycle system and for a gas-turbine cycle. Caused by the much larger average temperature during normal operation the 1600 degree limit will be reached at lower power outputs for the gas-turbine cycle. It should also be kept in mind, that the results shown here don't take into account, that there are additional uncertainties to be accounted for, which finally will reduce the maximum allowable power again to some extent. In any case the results indicate, that concerning the 1600 degree limit there is an almost linear dependence of the achievable power from the core diameter.

Because of the limitation in control- and shut-down capability an increase of the core diameter has to be accompanied by some modifications in geometry, if the reflector-based shut-down philosophy is maintained. The easiest way to do this was already exercised in the german AVR-reactor as well as in the HTR-100 design. By introducing graphite constructions, the so-called 'noses', which protrude from the side reflector region into the core a possibility is opened to extend the reflector control to a region much closer to the core centre and thereby to a region of much larger neutron importance. In doing so, the effectivity of shut-down absorbers can be dramatically increased, as is shown in figs. 3 and 4. In these figures the results of preliminary 2-D neutronic reactor calculations are shown using previously generated rough nuclear data. The important result is, that the achievable reactivity effect is about 14% to 16 %

for cold shut-down and about another 6 % to 8% for control purposes. This is in the order of the required figures and indicates the principal possibility of such a concept. By variation of the geometric layout of the absorbers additional potential can be gained.

The introduction of noses into the core design thus allows to increase the power output without running into the trouble of lack in control- and shut-down reactivity capabilities from outside the pebble bed positioned absorber systems.

HTR Core Model with Noses

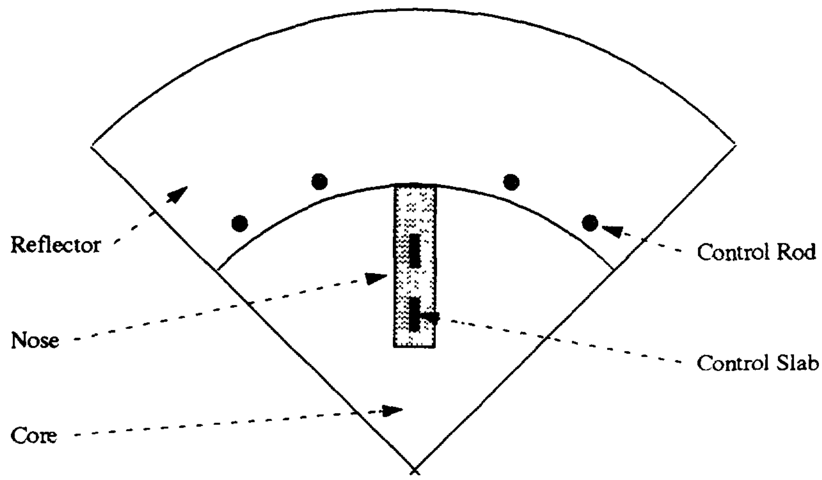

FIG. 3. HTR Core Modell with 'Noses'

Control System Efficiency in HTR Core with Noses

Results of 2-D Diffusion Calculations

Case	k-eff (2-D r-φ)	Reactivity / %
no Control System	1.11801	0.00
Control rods	1.05779	*5.1*
Front Control Slabs	1.01288	9.3
Rear Control Slabs	1.06611	4.3
All Control Slabs	0.96690	*14.0*
Total Control Sytem	0.90714	<u>20.8</u>

FIG. 4. Efficiency of Control Systems

The next question is related to the effect of 'noses' on the maximum accident fuel temperature. Here the comparison of the effective heat conductivity inside the pebble-bed with that of the 'nose'-graphite is helpful. In fig. 5 this is shown for typical data. As can be seen from the figure, depending on the type of graphite and the fast dose, there is equilibrium between the effective pebble-bed heat conductivity and the graphite conductivity in the temperature range between 1000 °C and 1500 °C. This corresponds fairly well to the situation in at least some parts of the core at accident temperatures. It may be therefrom concluded, that no strong influence of the 'noses' on the maximum accident fuel temperature can be expected.

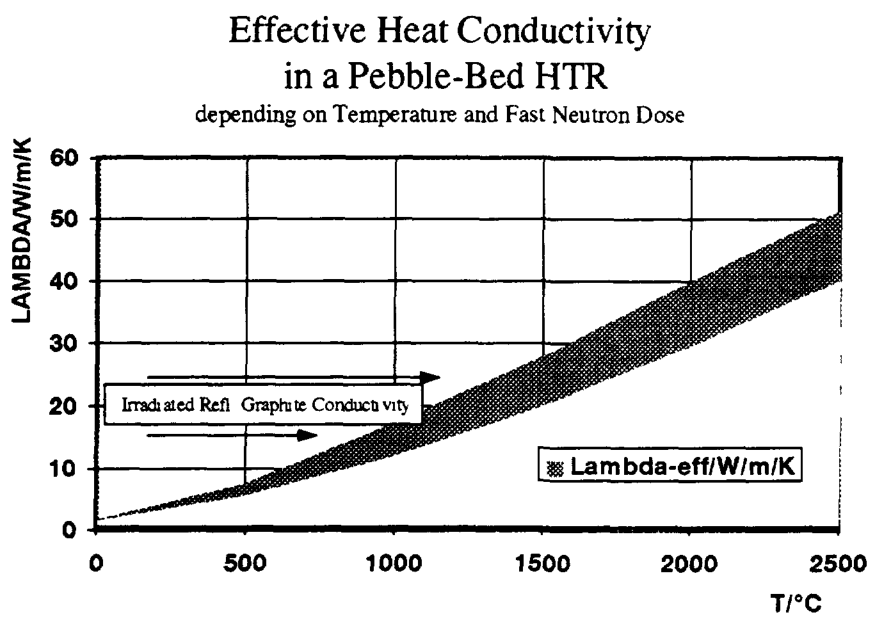

FIG. 5. Heat Conductivities in a Pebble Bed HTR

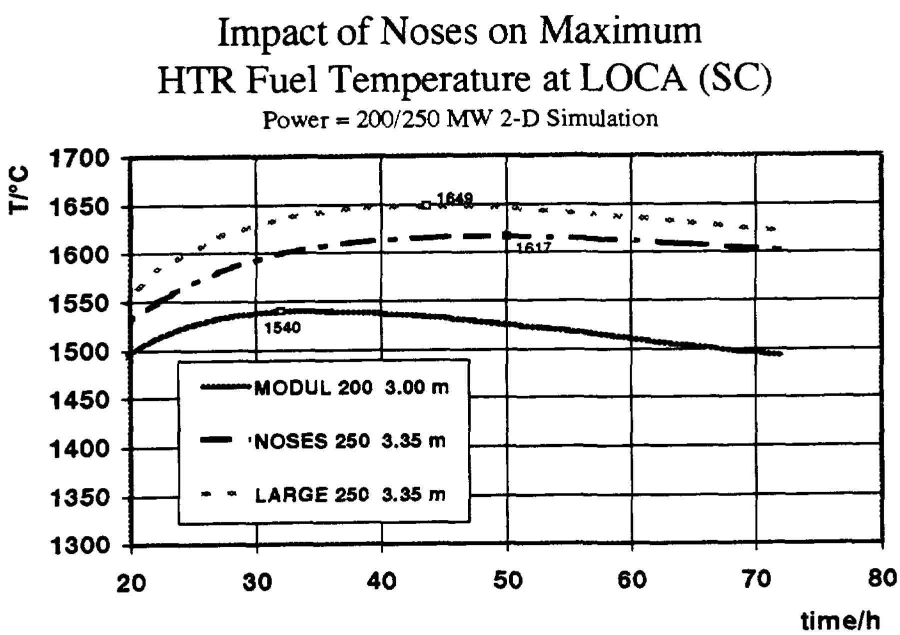

FIG. 6. Nose Influence on LOCA Temperature for Steam Cycle

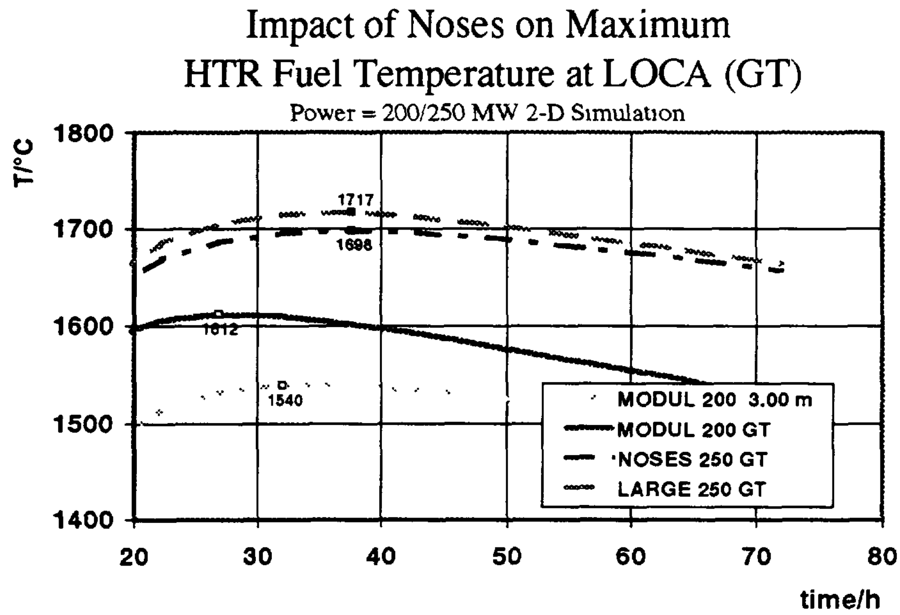

FIG. 7. Nose Influence on LOCA Temperature for Gas Turbine Cycle

To validate this statement, 2-D finite element calculations have been performed, comparing cores without and with 'noses' under otherwise identical boundary conditions. The results are summarized in figures 6 and 7.

Whereas fig. 6 shows results for steam-cycle conditions in fig. 7 data obtained for a gas-turbine system are given. In both cases as a reference the temperature evolution for the 200 MW_{th} HTR-MODUL is shown, too.

In both cases, the core with noses gives slightly lower maximum fuel temperatures than the purely cylindrical core. This is due to the fact, that relatively optimistic data for the graphite conductivity in the noses have been used ($\lambda_c = 25$ W/m/K). Nevertheless the differences are not very significant. The results show, that a power output of 250 MW cannot be expected for the cores under consideration. ($D_c = 3.35$m). Fig. 7 also shows the fact, that there is a remarkable increase in the maximum fuel temperature for the gas-turbine cycle compared to the steam cycle, which additionally decreases the chance of enlarging the power output.

In summarizing all these results the introduction of noses into the core design enables the control and shut-down from outside the pebble-bed even for core diameters in excess of 3 meters. On the other hand the benefits related to the maximum fuel temperatures in case of a depressurization accident are only marginal. Because of the reactivity effect, which is seen to be most essential, a core-design with noses seems to be the most promising way to slightly increase the power output over the up to now discussed 200 MW_{th}.

It should be mentioned however, that there are further options in the core design, which have not been discussed here. Introducing other fuel-loading and managing schemes but the standard multipath option opens an additional field of optimization. A more detailed presentation of this will be given in another paper of this TCM (E. Mulder, E. Teuchert).

3. A core-design including noses

3.1. Modelling the reactor

In case of the 200 MW HTR-Modul reactor a 2-dimensional treatment was sufficient for the evaluation of the burnup history of the fuel elements and of the coupled spatial distribution of the neutron flux and of the thermal power. Introducing graphite noses into the core region, however , disturbs the azimuthal symmetry of the core. Thus, in view of the strong coupling between the flux distribution, the neutron spectrum and the burnup of the fuel, a 3-dimensional evaluation of these items becomes necessary. For this purpose, the 3-dimensional version of the V.S.O.P. (96)-code was used in order to investigate the neutronics of the reactor. V.S.O.P. (96) /1/ is the successor of V.S.O.P(94) /2 /, now solving the burnup equations for 28 heavy metal isotopes from Th-232 through Cm-244, and using burnup-dependent resonance integrals for the lumped U-238 and (if present) for Th-232.

The continuous fuelling of the reactor provides a multipassing of the fuel elements (10 times), i.e. 1/10 of the loaded fuel elements are fresh ones, 9/10 have already passed from the top to the bottom of the core 1 through 9 times. 1/10 of the elements, which have reached their final burnup (80 MWd/kg $_{HM}$) after their tenth passage, are finally disloaded.

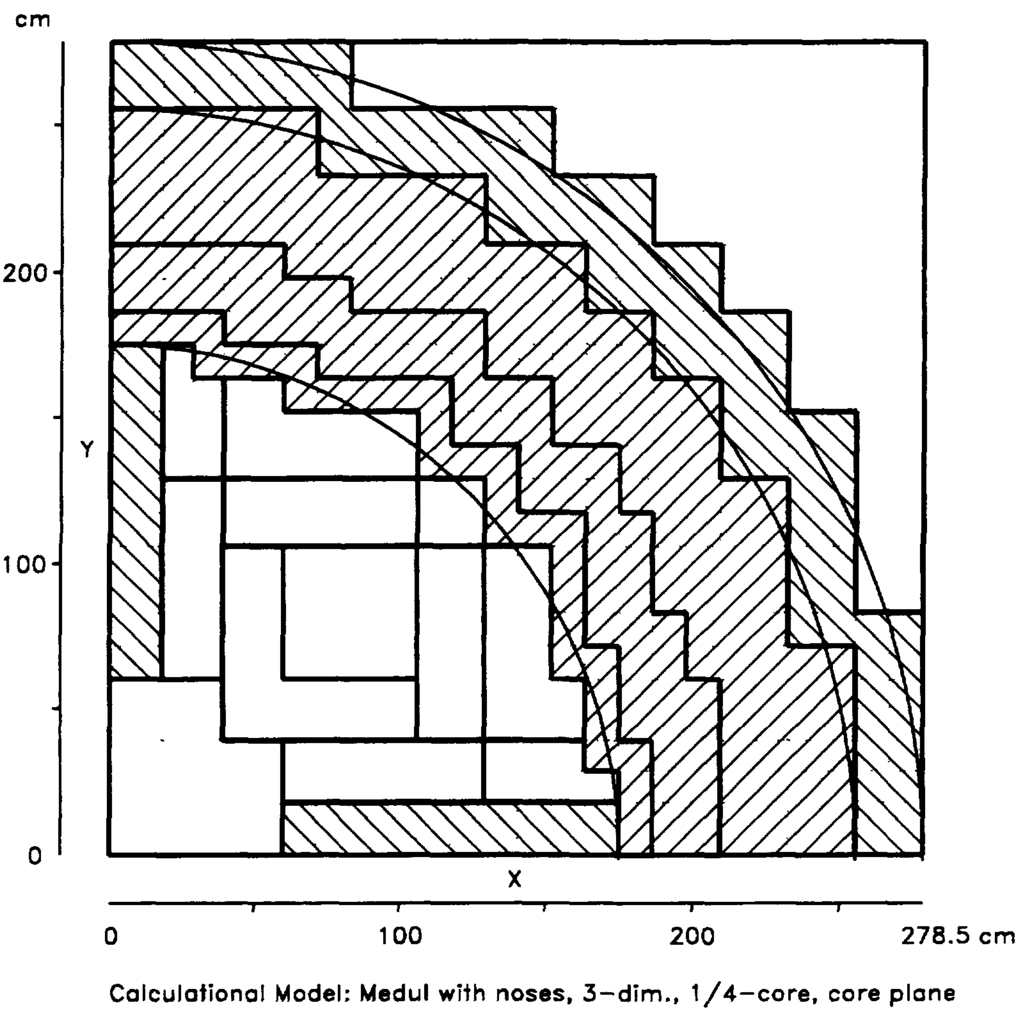

FIG. 8: Calculational model of the reactor (horizontal cross section)

Fig. 8 shows a horizontal cross section of the approximation of the real core geometry for the calculational treatment in an „X-Y-Z"- model. The core structure allows to restrict the explicit calculation to ¼ of the core, applying reflecting boundary conditions at y=0 and at x=0. The hatched areas indicate the reflectors and the graphite noses, where boundaries are approximated by step functions because of the x-y-mesh structure of the calculation. In axial direction, the core is divided into 9 planes, each again divided into 12 regions as it is shown in Fig. 8. Each of the herefrom resulting 108 core regions is homogeneous for the diffusion calculation, but is further subdivided into ten different fuel batches, representing ten different states of burnup. These correspond to the first through the tenth passage of the fuel elements from the top to the bottom of the core. Thus, 1080 states of fuel element burnup are explicitly followed by the code.

Fuel cell calculations, diffusion and burnup calculations are repeated using time steps having a length of some days in the reactor operation, while the fuel management is performed as it was described above. This procedure is ended as soon as the spatial distribution of the neutron flux, the power and the burnup have converged and thus an equilibrium state of the core has been reached. Finally, an „outer" iteration has to be done, allowing a feedback of the resulting fuel and moderator temperature to the evaluation of the thermal neutron spectrum and to the resonance shielding.

For the assessment of the reactor in view of its thermodynamic properties in general and in view of temperature transients following a loss-of-coolant accident the THERMIX-code is included in V.S.O.P. In contrary to the neutronics parts of this code system, the transient THERMIX calculations are restricted to 2 spatial dimensions. It was outlined in chapter 2, that no strong influence of the „noses" on the accidental fuel temperature needs to be expected. Thus, the following procedure was applied in order to get a sufficiently good approximation of the 3-D-Core in a 2-dimensional model:
The neutronics were recalculated in an „r-z"-approximation of the real geometry, replacing the graphite noses by fuel elements and keeping the total power of the core at the original value (220 MW). The steady-state temperature distribution was then calculated on the basis of the resulting 2-d power distribution. The same was done for the time and space dependent decay power of the fuel, which will occur after a depressurization event. The local decay power was calculated on the basis of the german standard DIN 25485. Upon this basis the temperature transients were then evaluated.

3.2. Results

3.2.1. 3-D-Calculations

The results of the 3-D calculations presented in the following were performed for the thermal core power 220 MW. Actually, this power level is a result of an iteration between the neutronics of the core and its resulting thermal behaviour under accident conditions (See chapter 3.22). The maximum fuel temperature appearing during a loss-of-coolant accident has been regarded as the limiting factor for the core power. In view of the retention of the fission products within the fuel elements the temperature shall be restricted to about 1600 °C.

Fig. 9 shows the thermal neutron flux and the power density for a 90°-section of the core, reaching from the middle of one „nose" to the middle of the next one, at different axial positions . One can see the strong peaking of the neutron flux within the graphite noses. Here, the power density results in zero according to the assumption within the code, that the complete fission energy (except neutrino energy) is released just where the fissions occur. The maximum power is in a plane, which is located between 3 and 4 m below the core surface. The total power peaking (maximum/average power density) of the core is 1.9.

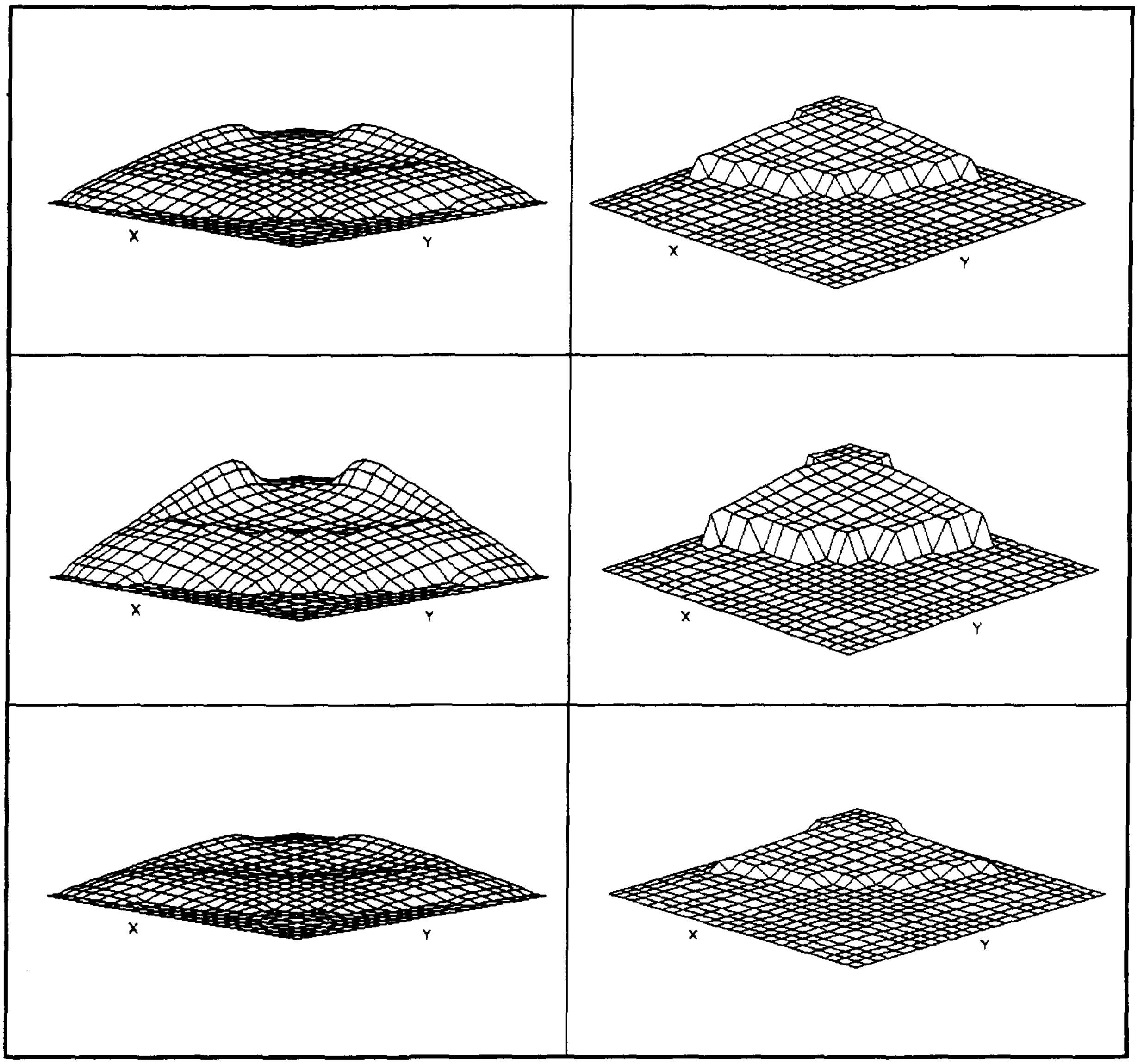

FIG. 9: Thermal neutron flux (left) and power density (right) in the upper core region (upper pictures), at 4 m below core surface (middle) and near the bottom of the core.(90 ° - sectors)

One important safety criterion of a nuclear reactor is a negative feedback between the temperature of the core and its reactivity. An increase in temperature should always result in a decrease of the reactivity, so that the reactor will be stabilized inherently in case of a release of reactivity and a resulting increase of its power. The prompt reactivity effect (fuel temperature coefficient) is dominated by the broadening of the U-238 resonances for neutron absorption. The contributions of the temperature changes of the fuel and of the moderator are listed in Fig. 10. One can see a slight positive contribution of the reflector regions, but the by far dominating effect is the negative contribution of the fuel and of the graphite within the fuel elements.

Temperature coefficients at full core power

	ΔK_{eff} /Kelvin
Fuel	-3.06×10^{-5}
Moderator	-2.68×10^{-5}
Noses	$+0.90 \times 10^{-5}$
Outer reflectors	$+0.74 \times 10^{-5}$
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Total	-4.10×10^{-5}

FIG. 10: Temperature Coefficients

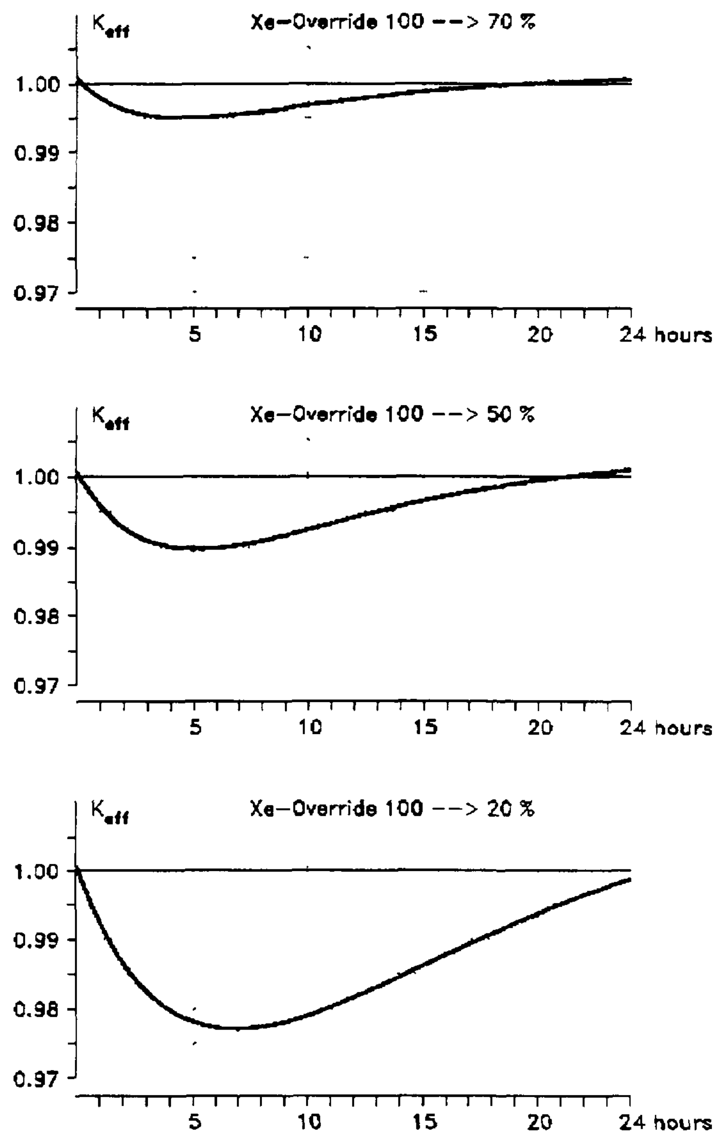

Fig. 11: Reactivity changes caused by load changes

The continuous fuelling of the MODUL-HTR implies that no excess reactivity is necessary in order to compensate burnup effects. Nevertheless, a certain margin is required for the reactor control and to compensate changes of the xenon concentration following changes of the reactor power. Fig. 11 illustrates the neutron multiplication factor after a change from full power operation down to 70, 50 and 20%, respectively. The resulting deviation from $K_{eff}=1$ must be compensated by the control system. It results that if the utility, which operates the reactor, supposes to change from full power down to 20 % of it, then an excess reactivity equal 2.3 % Δ K is required. In case of an erroneous movement of the control system, releasing this reactivity, the fuel temperature would increase until the reactivity is compensated by the negative temperature coefficient. Using the coefficient value of Fig. 10 , this increase would be 560 °C. This, of course, is only a rough number, because of the temperature dependence of the temperature coefficient and of the space dependence of the temperature.

In case of a long-term shut down, several reactivity effects are superposed (Fig. 12). As the reactor is cooled down, the reactivity increases according to the temperature coefficient. As in the beginning J_{135} decays to Xe_{135}, the reactivity decreases and reaches a minimum value after about 9 hours. Then Xe_{135} decays and K_{eff} reaches unity after about 20 hours. The maximum value of K_{eff} appears after about 4 days. Afterwards, there is a slight decrease again, mainly because of the decay of Pm_{149} to Sm_{149}. The maximum reactivity margin, which has to be compensated by the shut down system , is Δ K=0.073.

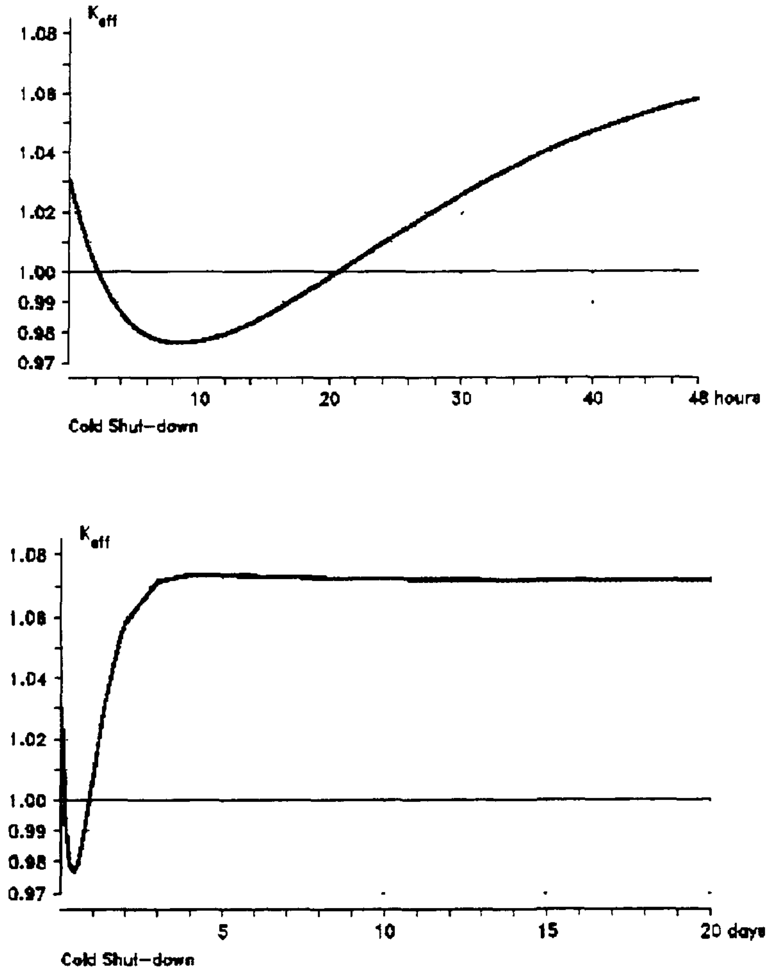

Fig. 12: Reactivity vs. time after reactor shut down

3.2.2 Loss-of-coolant accident (2-d-calculations)

As it was already mentioned in chapter 3.2.1., the reference power 220 MW_{th} of the core, which was the matter of the 3-d-investigations, was the result of a parametric study. The purpose of it was to determine, which power level could be realized in the given core structure under the boundary condition, that in case of a depressurization event the temperature 1600 °C will not be exceeded within the fuel at any time during the resulting temperature transient. The results of this study are outlined in fig. 13. They were obtained in a 2-d-approximation of the reactor, as it has been explained above (Chapter 3.1.).

The lower curve shows the so defined maximum temperature versus the reactor power (T-nominal). If one suggests an uncertainty of two standard deviations for the evaluation of the decay power on the basis of a given spatial power distribution, and in addition adds a 5 % margin to the nominal power at each local position of the fuel elements in the core with respect to uncertainties of the spatial reactor power distribution, then the temperature results according to the upper curve of fig. 13 (Decay-power * 1.1). It can be seen, that here the limiting temperature is reached at the power 220 MW_{th}. Consequently, this power level was taken as the reference in order to perform the final 3-d-investigations.

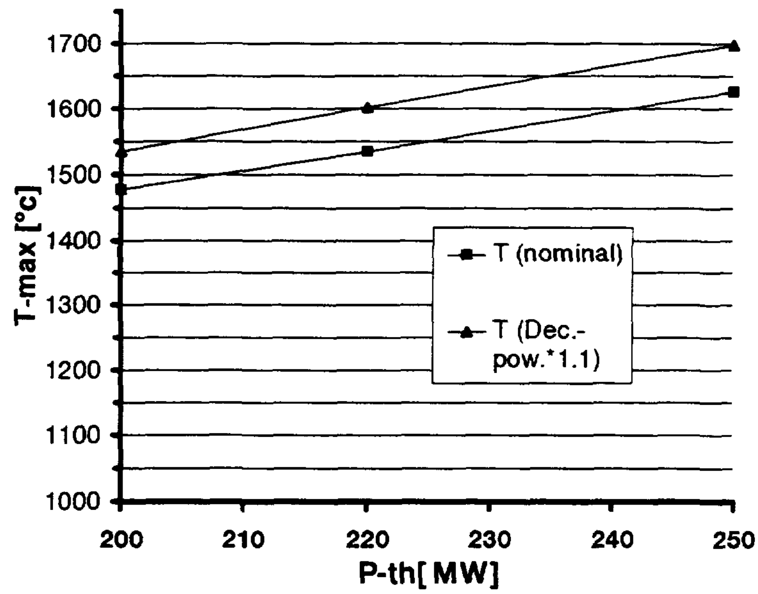

Fig. 13: Maximum fuel temperature of the depressurized core vs. the core power

In case of a loss-of-coolant, the maximum of the fuel temperature is reached after about 30 hours since depressurization occurred. After this time the decreasing decay power in the inner core regions is balanced by the heat transport to the outer parts of the reactor and to the surface coolant system of the reactor cell. Afterwards, the fuel temperature decreases again, while the temperature of the outer reactor components still increases. The pressure vessel reaches its maximum temperature (~ 400 °C) after about two days.

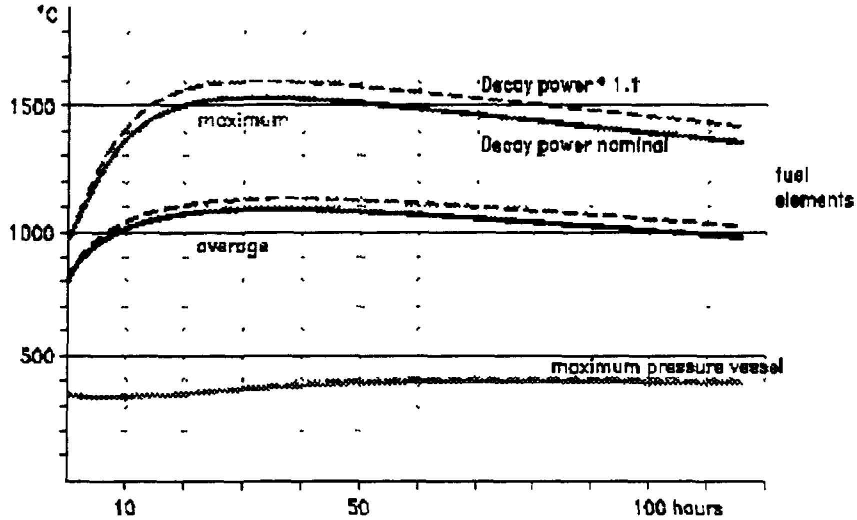

Fig. 14: Temperature of the fuel elements and of the pressure vessel vs. time during a loss-of-coolant accident (220 MW_{th} core power)

4. Control- and Shut-down Systems

Using the nuclear data, that have been produced according to the descriptions of the last chapter, 2-D and 3-D reactor calculations have been performed to evaluate the design and efficiency of control- and shut-down systems for the proposed pebble-bed reactor concept. The calculations were performed using the diffusion code CITATION /3/. The effective cross-sections for the absorber materials were prepared using the methods derived at KFA-ISR for such purposes /4/.

Following the design of both of the german modular pebble-bed reactors a system of small absorber spheres (KLAK) has been chosen as the cold shut-down system. This system has been analyzed experimentally in detail by INTERATOM/SIEMENS for the MODUL-HTR and as a part of this design has been positively reviewed during the site-independent design review by the TÜV-Hannover, Germany. Similar to the HTR-100 design it is proposed to introduce this KLAK system into borings of the four graphite noses.

In standard LWR reactor design besides a shut down system for long-term shut-down a fast acting 'first' shut-down system is necessary, which has to guarantee the shut-down of the reactor to a hot subcritical situation from any operational and accidental situation. Because of the excellent inherent safety properties of a HTGR such a system is not required a priori for such reactors from safety arguments. It is another question, whether it could be useful from operational considerations. In any case a facility for operational control is necessary. It is proposed to use a system of movable control rods or similar absorbing systems for this purpose. These may either be inserted into rear borings of the noses as shown in fig. 15 or may be placed in a more conventional way around the reflector perimeter as foreseen for the german designs. The final decision will depend on the reactivity efficiency of these systems, on the requirements (shut-down or control only) and on additional conditions such as the distortion of the power shape during normal operational transients.

The basis of the calculations was a KLAK system with four different KLAK positions in the front part of the graphite noses. The dimensions of the rectangular borings were chosen as $35x9\ cm^2$. The borings cover the complete height of the active core. It was assumed that the

PBMR Core Model (I)

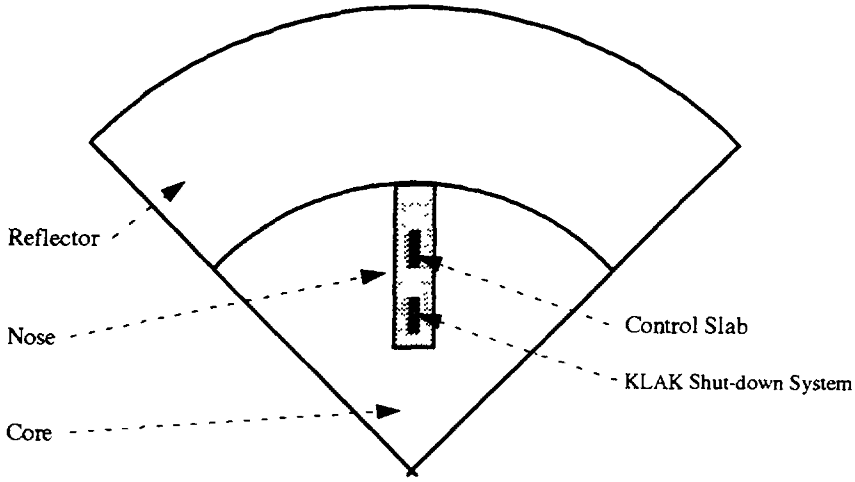

FIG. 15. Core Model with Noses

small absorber spheres will have a filling factor of 0.6 and that they will be made of graphite with 10% boron-carbide. As a first guess for the additional control system a slab-like absorber rod system was assumed, moving in rear borings of the graphite noses. As indicated before, this system may be enhanced by additional control rods in the side reflector.

The results for the efficiency of these absorber systems obtained so far are summarized in fig. 16.

PBMR

Control- and Shutdown Reactivity
System Variant I

Case	k-eff (2-D x-y)	React. / %	k-eff (3-D x-y-z)	React./ %
Full Power, no CS	1.02969	0.00	1.00207	0.00
Control Slabs (total)	0.93474	9.9	0.90820	10.3
Max. Slab missing	0.96529	6.5	0.93798	6.8
Control Slabs (half)	-	-	0.96839	3.5
Max. Slab missing	-	-	0.97232	3.0
Cold LTS, no CS	1.09681	+5.94	1.07227	+6.53
KLAK	0.93958	15.3	0.91795	15.7
max. KLAK missing	0.99193	9.6	0.96883	10.0

FIG. 16. Control- and Shutdown Reactivity

The table shows the effective multiplication factors and the reactivities for both absorber systems. The control system has been analyzed in the hot full power core whereas the KLAK results were obtained in the long-term cold (LTS) core. 2-D (x-y) and 3-D (x-y-z) results are reported.

The total KLAK reactivity efficiency is about 16%, which is by far enough compared to the requirements as given before. However, depending on the licensing requirements the failure of one of the KLAK positions has to be considered. In this case, the efficiency reduces to 10%, which again will be sufficient so far, but has to be validated in the future. Concerning the control system the situation is not so clear. The total reactivity compensation in this case has been calculated to be about 10% with remaining 7% for one of them failing. But these results were obtained for completely inserted systems. Reducing the insertion to half of the core height resulted in a dramatically decrease of efficiency down to 3.5% for all four and down to 3.0% for three of them. This strong reduction is due to a complete distortion of the axial powerdensity shape in this situation as shown in fig. 17. Depending on the part-load requirements (Xe-override) future additional analyses are required to decide whether this system will fulfil the requirements. At least an iterative procedure with the establishing of the equilibrium core design will be necessary.

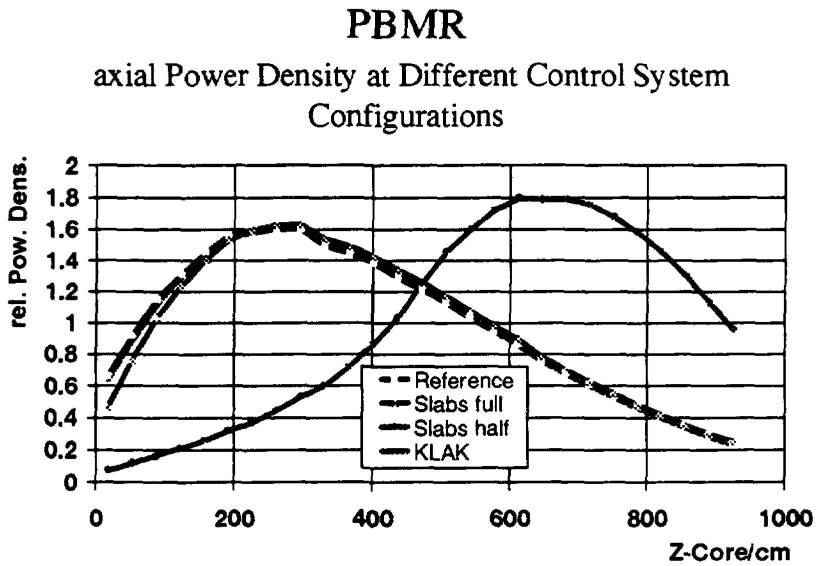

FIG. 17. Axial Power Density with Control Systems inserted

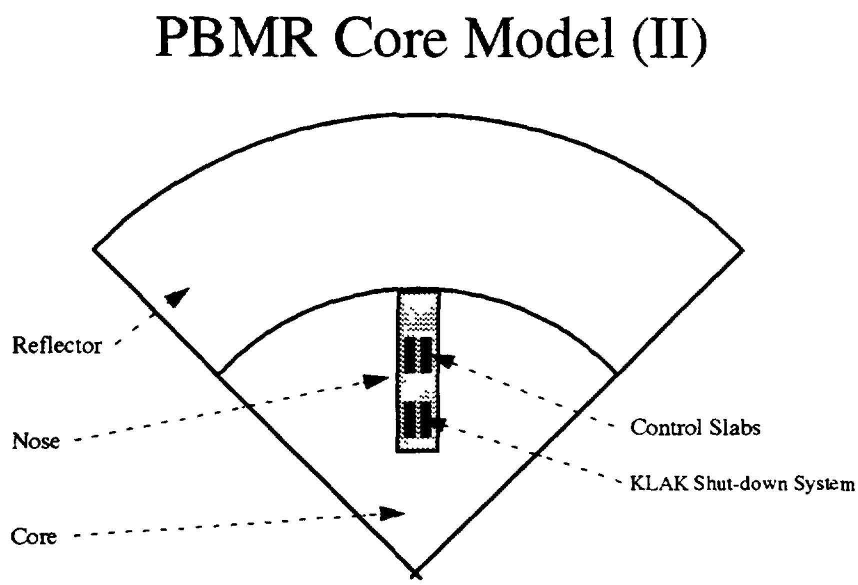

FIG. 18. Alternative Core Model

To show some additional potential for the control- and shut-down system a slightly modified design has been analyzed, too. By separating the KLAK and Slab borings while conserving the total absorber volume as shown in fig. 18 some gain in reactivity efficiency was observed. The results of the corresponding 2-D calculations together with those of the reference variant repeated are given in fig. 19. Therefrom it may be concluded that an additional capacity reserve of about 15%-20% for the reactivity efficiency may easily be activated.

PBMR

Control- and Shutdown Reactivity
Comparison of Variants
(2-D Calculations only)

	Variant I		Variant II	
Case	k-eff (2-D x-y)	React. / %	k-eff (2-D x-y)	React./ %
Hot FP, no CS	1.02969	0.00	1.02969	0.00
Control Slabs (total)	0.93474	9.9	0.92099	11.5
Max. Slab missing	0.96529	6.5	0.95717	7.4
Cold LTS, no CS	1.09681	+5.9	1.09681	+5.9
KLAK	0.93958	15.3	0.91305	18.3
max. KLAK missing	0.99193	9.6	0.97839	11.0

FIG. 19. Comparison of Control Model Variants

5. Open Questions, Planned Investigations and Additional Options

In the course of the research reported here, some assumptions had to be made on items, which have not yet been clarified. These may have a certain impact on the layout of the reactor core and may be to a certain degree also on its safety features.

The calculation of the spatial power distribution was done for the core being free of any control devices. In order to allow load changes of the power plant according to the demand of the grid, a certain amount of excess reactivity has to be available for a release in order to compensate the combined so called 'Xe-override' reactivity (see chapter 3.2.1.). In view of reactor safety, this reactivity should be small. Once that a reactivity requirement will be fixed, the resulting position of the control devices during full power operation has to be evaluated. This again will influence the power peaking of the reactor core and the local fuel burnup. The transients during a loss-of- coolant event will have to be investigated for different positions of the control devices.

The 2-dimensional model of the core, which was used for the calculation of the heat transfer, can account for the power peaking in axial and in radial direction only, using equal values in azimuthal direction. From the 3-d-calculations of the power we see an additional peaking in azimuthal direction close to the graphite noses. On the other hand, the neutron reflection by the noses tends to flatten the power distribution in radial direction. It has to be validated, that the resulting uncertainty in view of the prediction of accident temperatures are covered by the applied safety margin.

Uncertainties concerning the heat conductivity of the graphite noses presently prevent a final answer to the question, whether they hinder or even improve the heat transfer in radial and in axial direction of the core. Here, additional research is foreseen, in order to answer these open questions.

In parallel to the research, which is reported in this paper, an alternative fuelling scheme for a small Pebble-bed-HTR has been investigated, which aims at a simplified fuel management. This provides only a single pass of the fuel elements through the reactor core. The results of this research are presented in a paper of E. MULDER on the 'OTTO-PAP' within the frame of this meeting.

6. Summary

In the paper presented here the possibilities and limitations in increasing the power output of a modular pebble-bed high temperature reactor with multipath fuel loading scheme and 1-zone loading pattern are discussed. In touring the horizon some basic features and requirements for a inherent safe behaviour during operation and accidents are given and several design modifications are presented compared to the well defined and analysed german HTR-MODUL design.

The design modifications considered include an increase of the core radius and the introduction of graphite constructions, the so-called 'noses', which protrude from the side reflector region into the core similar to the german AVR and HTR-100 constructions. The impact of such modifications on the maximum fuel temperature in a LOCA and on the efficiency of the nuclear control and shut-down systems is considered.

The results obtained so far demonstrate the well-adapted and conservative design of the SIEMENS HTR-MODUL within a 10% safety margin to the higher region. The introduction of graphite noses has a remarkably positive influence on the shut-down and control systems, while the positive effect on the maximum accident temperature depends strongly on the fast neutron dose-related thermal conductivity of the nose graphite. Considering the fact that effective conductivity of the pebble-bed core is maintained at high temperatures, the temperature effect due to the noses are of secondary influence at this point.

Based on these results a core design for a 220 MW_{th} core with noses and a diameter of 3.5 m for a direct cycle gas-turbine application has been presented. It has been shown, that such a design meets the basic requirements of a new generation safety philosophy. The maximum accident fuel temperature remains below the 1600 °C limit and the efficiency of the shut-down and control systems seems to be sufficient. Operational details like the range of unlimited part-load and the system dynamics of the gas turbine cycle have to be discussed further.

According to the results obtained and referring to previous investigations the utmost limit for the maximum power output of pebble-bed cores in a steam cycle system, fitting into a pressure vessel of about 6 m diameter, seems to be close to 230 MW_{th}, if purely cylindrical core geometry is envisaged, and close to 270 MWt_{h}, if graphite noses are taken into account. The corresponding values for a modern turbine cycle system are 200 MW_{th} and 240 MW_{th} respectively.

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OTTO-PAP: AN ALTERNATIVE OPTION TO THE PBMR FUELLING PHILOSOPHY

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Abstract

Once Through Then Out, Power Adjusted by Poison (OTTO-PAP) fuelling of a high temperature pebble-bed reactor offers a simple alternative to the MEDUL (Mehrfachdurchlauf = German for multi-pass) fuelling regime followed in pebble bed reactor designs to date

The prerequisite for a modular reactor unit of maximum power output, subject to observing passive safety characteristics is a sufficiently flat axial neutron flux profile This is achieved by introducing B_4C coated particles of pre-calculated size and packing density within the fuel spheres

In accordance with AVR operating practise the temperature profile is radially equalised by introducing a 2-zone core loading Adding pure graphite spheres loosely into the centre column area of the core effectively reduced the maximum power in the middle

Increasing the reactor diameter is enabled by the introduction of noses A 3-D geometric modeller developed in cylindrical co-ordinates enables a given flow description of the pebbles adjacent to the nose boundaries and in the vicinity of the shut down / control rods After translation of the geometric data the neutronic behaviour of the reactor is followed in 3-D by the CITATION code

This study is aimed towards achieving an optimal core layout with a LEU (Low Enriched Uranium) fuel cycle Physical properties of the OTTO-PAP, 150 MWt reference design is reported, while computations performed observe results obtained by the reference HTR-MODUL design

Introduction, Background

As part of a series of supply side option investigations ordered by ESKOM, South African power utility, a feasibility study was launched to establish the technical and economic feasibility of modular high temperature gas-cooled reactors when coupled to direct-cycle, closed-loop, helium power turbines Apart from the preliminary operational requirements as documented in /1,2/ the unit size is subject to the basic owner requirement that *no physical process, however unlikely, must cause a radiation induced off-site hazard* This is achieved in principle in the Pebble Bed Modular Reactor (PBMR) by demonstrating during a postulated depressurised loss of coolant incident that

- the decay heat produced in the post accident condition is exceeded by the inherent heat removal capability of the reactor vessel,
- peak temperatures within the core never reaches 1600°C, thus remaining well below the demonstrated onset point of fuel degradation, i e fuel temperature maintained at 1600°C for a period in excess of 100 hours /3/

Keeping in mind the abovementioned, it is reasonable to employ the fundamental approach of adherence to available technology, observing the proven operational base, and selecting a commercially viable unit size, albeit small and modular.

Central to this study is the selection of a viable fuelling regime. A unit size of 70 - 100 MWe lends itself to two basic fuelling options currently under consideration, *viz.* MEDUL and OTTO. A third option, i.e. Peu-à-Peu (PAP), is of interest for unit sizes of 5 - 40 MWe. Twenty years continuous operation with only one step defuelling at the end of operation offers unsurpassed simplicity and, while awarding a high degree of proliferation resistance /6/.

In /4/ a 220 MWt PBMR reference proposal based on the HTR-MODUL (200 MWt) and HTR-100 (250 MWt) designs by SIEMENS/Interatom and BBC/HRB respectively, is investigated. This concept refers to well-proven technology demonstrated by the operational histories of the AVR experimental reactor in Jülich /see 5/ and the THTR demonstration plant in Schmehausen.

OTTO employs a special fuelling mode of MEDUL, i.e. a single pass of the fuel elements through the core. This concept is extremely attractive for its simplicity, yet remained limited to a maximum thermal power output of 120 MWt due its characteristic neutron flux shape as depicted in Fig. 1. Advantages of this feature are well-documented by several authors /see 7,8/. The major disadvantage to this fuelling concept remained however, its inability to deliver power within the 150 - 200 MWt range, whilst adhering to the principles of passive safety.

A variation of the abovementioned fuelling regime, OTTO & PAP, broadly coincides with the proposal in /9/ as suggested for the PBMR-SA fuelling. Table I poses a summary of the considerations which prompted the current investigation.

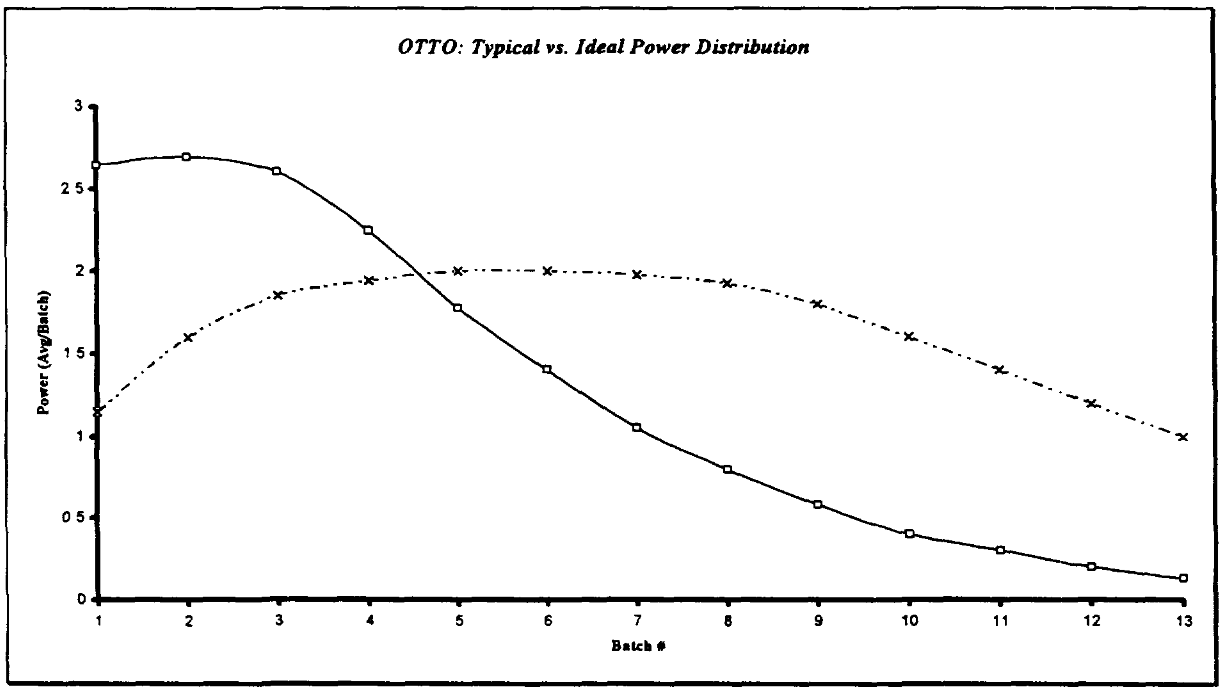

FIG. 1: OTTO - Typical neutron flux distribution in axial direction

TABLE I: CONSIDERATIONS FOR GOING OTTO

PROS	*CONS*
1. Fuel management - Simplified by single pass through the core	1. Unwanted neutron flux profile for unit sizes exceeding 120 MWt
2. Fuel handling - Minimised by single pass through core	2. Operational in-experience - Close synergy is to be observed with MEDUL experience. A MEDUL plant design can incorporate OTTO fuelling regime
3. Measurement equipment - Weight difference will suffice for the separation of fuel and absorber spheres	
4. OTTO fuelling - Is a special case of MEDUL scheme ⇒ Enhanced testing opportunity in a dual-fuelling prototype	
5. Minimised abrasion - Dust production is minimised. This is of importance in direct-cycle operation	
6. Off-line defuelling offers operational freedom	
7. OTTO technology base - Analytical investigations since 1970 provide a solid scientific base	

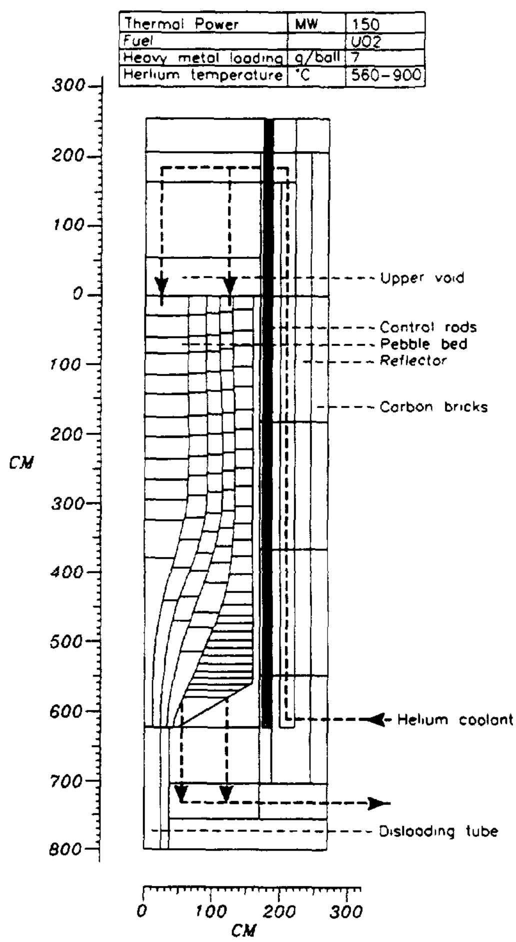

FIG. 2: VSOP model of the OTTO-PAP reactor layout

The scope of this paper includes an account of the development deemed necessary in providing an economical alternative to an HTGR concept in terms of:

- achieving a favourable neutron flux distribution for reaching a power output in the range of 130-200 MWt with OTTO fuelling;
- the geometric modelling capability in ϕ-r-z to account for the neutronic behaviour and influence of control poison due to the introduction of noses;
- enhanced pebble flow model.

Model Description: OTTO Core Design and Fuel Elements

Core Modelling

In the study a 3-D computational simulation of the equilibrium cycle of the OTTO-150 plant is presented. Fig. 2 provides a VSOP model layout of the plant. A comparative listing of the 150 MWt OTTO reference layout and HTR-MODUL is provided in Table II.

TABLE II: SOME REACTOR LAYOUT DATA

LAYOUT PARAMETERS		OTTO-PAP
Burnup	MWd/Kg$_{HM}$	
Heavy metal content	g/FE	7
Core Height/Radius	cm	624/170
Core shape		Cylindrical
Noses		65 cm x 30 cm
Fuel Zoning regime		2-Zone
Number of passes		1
Helium heat up	°C	560→ 900
Decay heat factor		1.1
Thermal conductivity - Pebble Bed (A3-3)	W/m K	Nominal; F(irradiated temp.)
Thermal conductivity - Reflector (AGL IE 1-24)	W/m K	Nominal; F(irradiated temp.)

In Fig. 3 the core geometric model is depicted consisting of two distinct sub-regions regions, i.e. the core-nose and rest-core model, of which the latter, in turn consist of one region bordering the nose side, whilst the second region is delimited by the ring reflector. A grid subdivision of the core-nose model consists of 2 radial and 24 axial zones, with the rest-core model having 5 radial and 30 axial zones. The core is embedded in a graphite reflector shell of 78 cm thickness, which is surrounded, respectively by a carbon brick layer of 22 cm, a core barrel and a moulded steel pressure vessel. Enclosing the reactor pressure vessel the reactor cavity cooling system (RCCS) model is further extended to represent a closed-circuit water cooled panel system, similar to the system employed in the HTR-MODUL. This boundary specification enables the neutronic and thermal-hydraulic evaluation for the specified reactor layout.

For the computational simulation of the reactor life the discretized core is sub-divided into many batches in both the axial and radial directions. Batch positions remain stationary over a burnup period of approximately 8 weeks (load factor of 0.76). Thereafter movement of one step downward follows. Batches at the bottom are unloaded, while the top ones are made up with fresh fuel. This scheme simulates continuous OTTO fuelling by sub-dividing the time scale into relatively large, discrete time steps.

The calculational procedure described above can simultaneously be regarded as representative of Peu-à-Peu fuelling with defuelling steps of 8 weeks. Adaptation of the defuelling period is easily achievable by modifying the downward shuffling to multiple time steps. Defuelling of 3 layers at one time but filling the top layers bit-by-bit represents OTTO-Peu-à-Peu fuelling with a 6 months defuelling period.

Based on a recent study of the flow of spheres through the core of the THTR plant /10/ and experimental data compiled by Kleine-Tebbe /11/ the BIRGIT code flow model could be enhanced to simulate the flow of spheres more accurately in pebble bed HTRs. In Fig. 4 the flow results as compiled by Kleine-Tebbe is depicted. Data obtained experimentally can thus be employed to validate forthcoming sensitivity studies on flow regime variations.

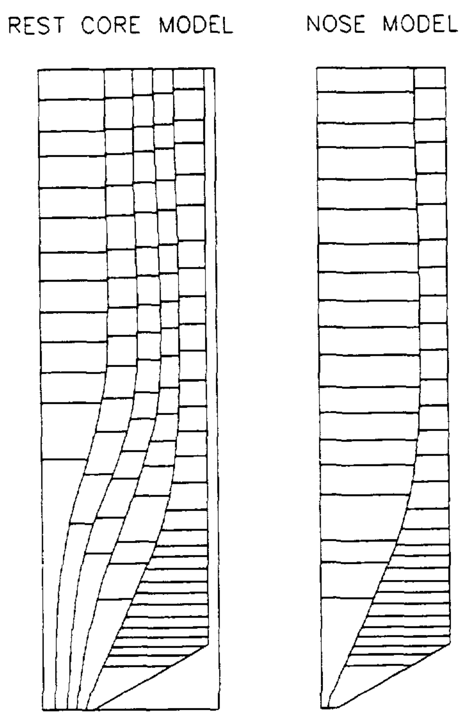

FIG. 3: Discrete core model:- rest-core and core-nose schemes

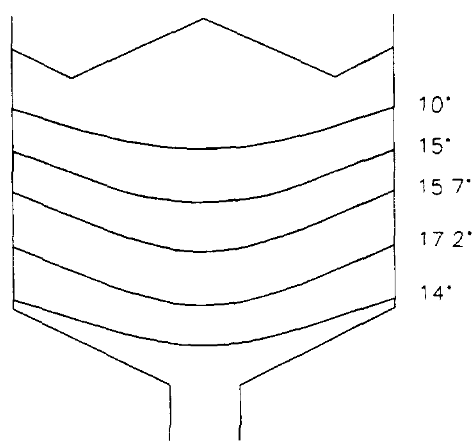

FIG. 4: THTR experimental flow data

Unit Power Increase

The unit electrical power output is achieved by the following four additions:

Increasing the core radius - In /12/ and the noted work by Scherer, *et.al.* confirmation is provided that noses (*nasen*: German for noses) are required in case of the core radius exceeding 1.5m. It is shown that the introduced nose rods provide a necessary and sufficient requirement for effective long term shut down under ambient (50°C) reactor conditions. In Fig. 5 a depiction of the core layout provide nose dimensions as derived from the AVR geometry. The thermal output is roughly increased by 10 MWt/10cm radial increase.

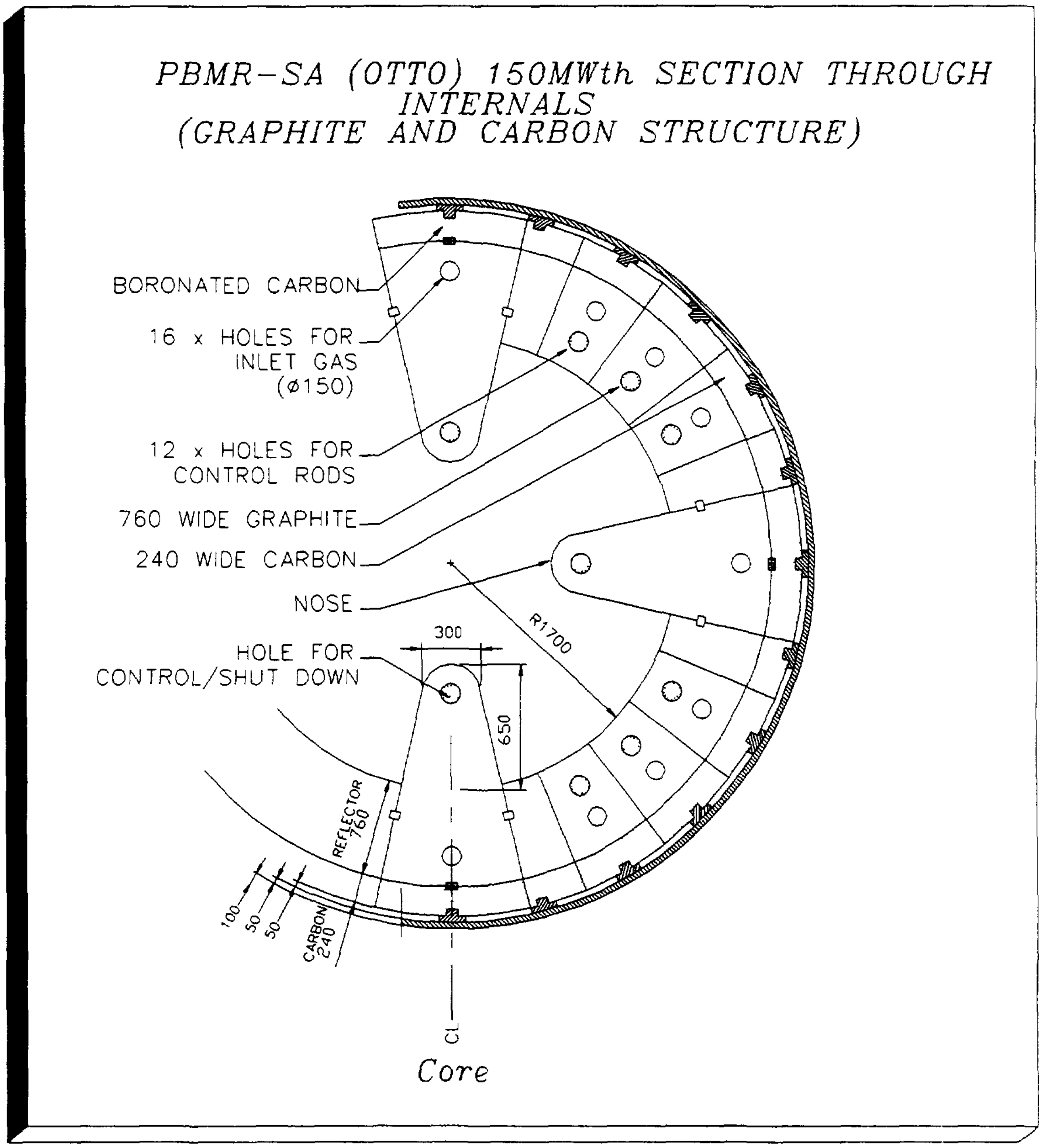

FIG. 5: OTTO-150 - Core layout with noses

Increasing the core height - A parameter variation is performed to determine characteristic sensitivity for the yield in thermal output *versus* increase in height. In summary the tendency in thermal gain is ~10%/m.

Apart from the 3-D calculational requirement necessitated by the resultant core geometry, effective physical modelling of the downward fuel flow through the core dictated the introduction of cylindrical geometry. The FIRZIT code is subsequently developed to perform the 3-D geometrical modelling in ϕ-*r*-*z*. Diffusion calculations in 3-D is performed with the CITATION code. In Fig. 6 a sectional depiction of the OTTO core is posed.

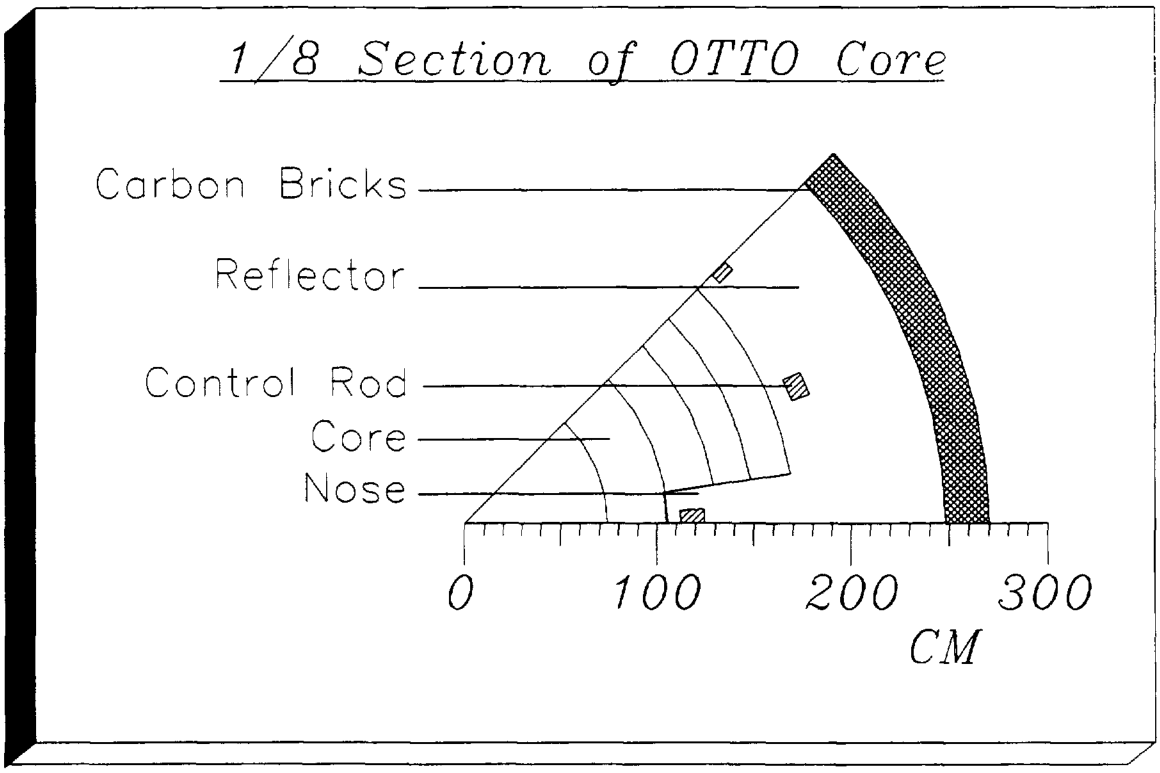

FIG. 6: OTTO-150 - 1/8 Section of the core and reflector model

Introduction of a 2-zone core loading scheme - This concept is well-proven in practise during the AVR operation. The fuelling tubes in the AVR design are perfectly suitable to be employed towards this end. In the note /13/ a mixture of 25% graphite spheres (blindkugeln) is shown to be effectively employed for creating an equalised distribution of the power density from within the centre column of the core radially outward. This is done without upsetting the fuel/moderator ratio in the core. A flatter temperature profile is achieved in this way due the decrease of fissionable material in the inner core area (see Fig. 7). The obtained reduced power density results in a reduction of the decay heat production in the inner core.

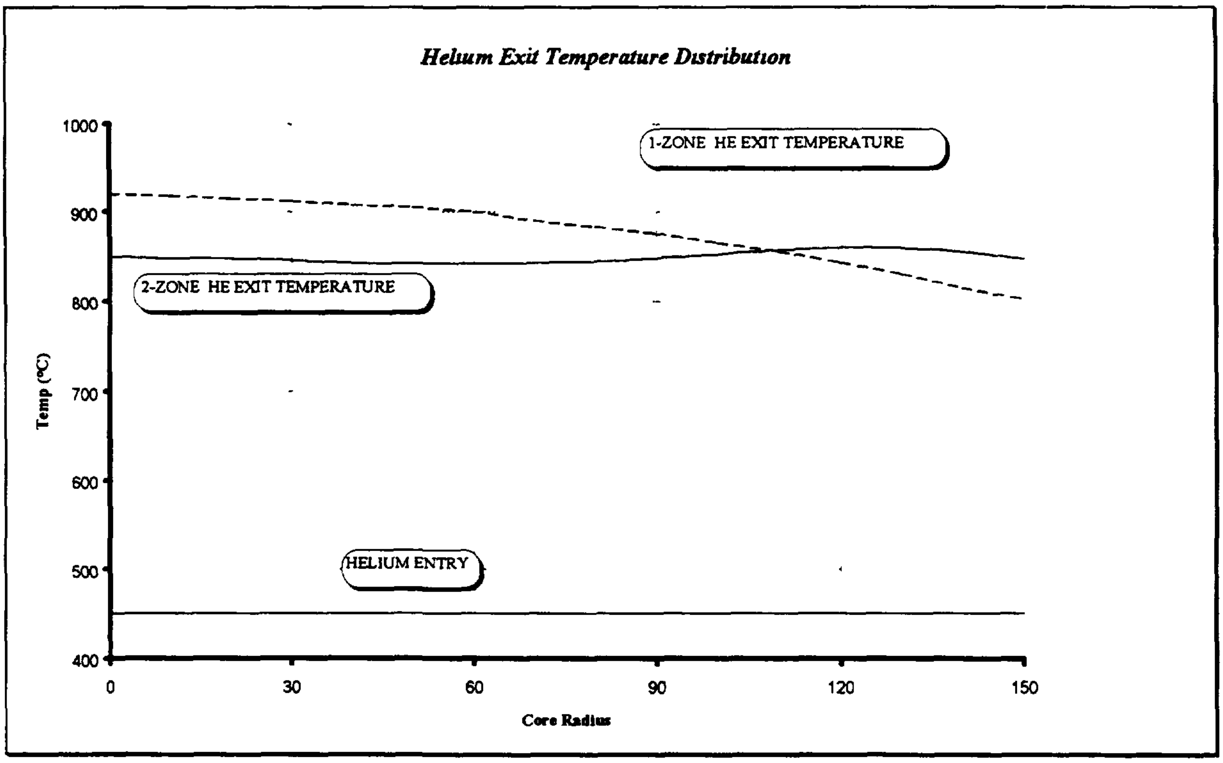

FIG 7 2-Zone versus 1-Zone core layout

Deploying an advanced fuel design by including B_4C coated particles - In the calculational model the shielding of all coated particles, whether fuel or burnable poison, is observed. Self shielding factors (SSF) in the thermal-neutron energy range influencing reaction rates in the particle kernels, coatings and moderator material have been calculated The theory is based on the derivations in /14/ which includes an accurate description of the penetration probability for a neutron traversing the coated particles

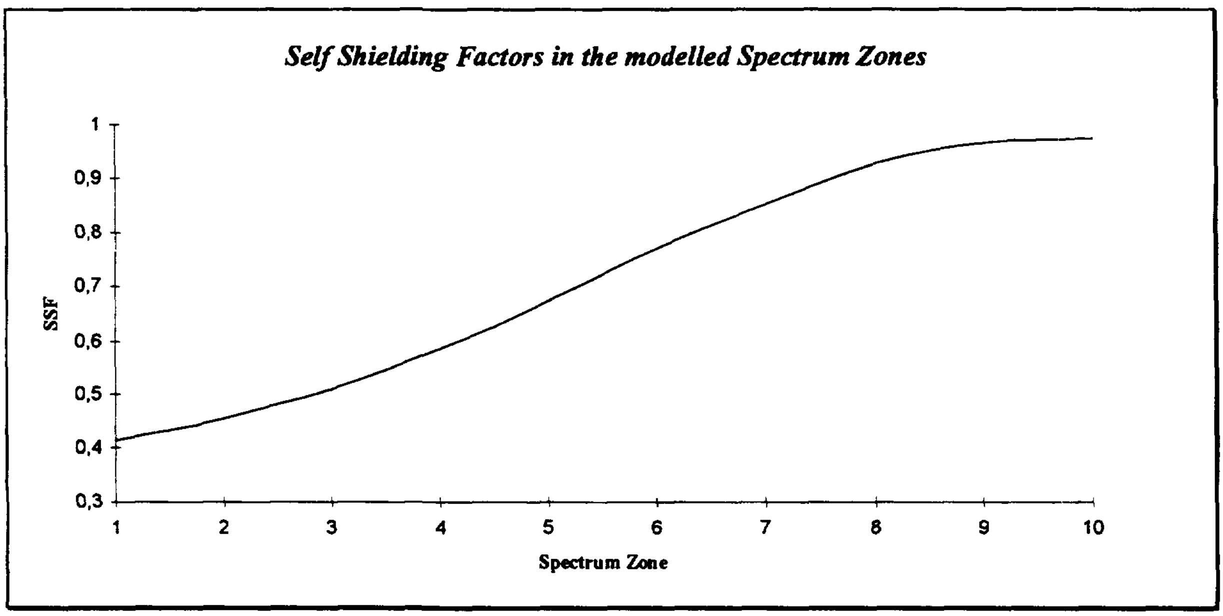

FIG 8 Calculated SSF in Spectrum Zones 1 - 10 ranging from top (1) to the bottom (10) of core

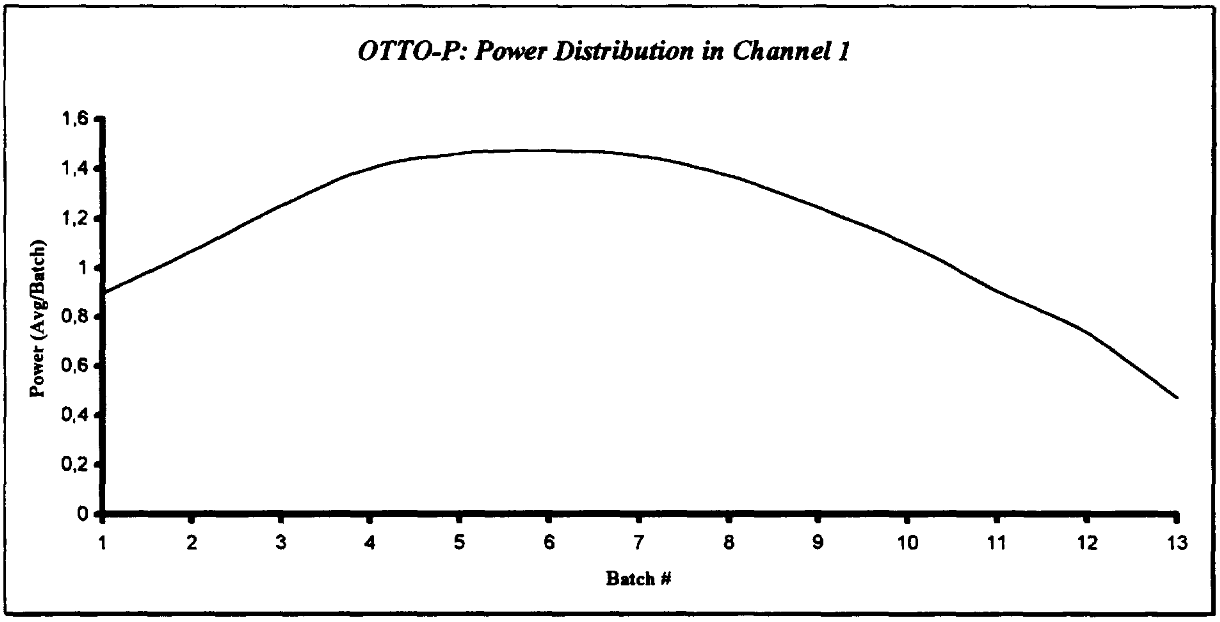

FIG. 9: OTTO-P - In-core (inner column) estimated power distribution

TABLE III: SOME FUEL AND REFLECTOR CHARACTERISTICS OF THE REFERENCE REACTOR

Fuel Elements:			
		First Core	Equilibrium Core
Sphere radius	cm	3	3
Fuel matrix radius	cm	2.5	2.5
Uranium enrichment - N_{U5}/N_U	%	4.66	9.45
Heavy metal loading per fuel sphere	g/sphere	7	7
Fuel spheres / Graphite spheres (inner - outer core zone)	%	70/30 - 100/0	67/33 - 100/0
Coated Fuel Particles:			
Particle diameter	μm	500	
Density	g/cm^3	10.4	
Coating material		C/iPyC/SiC/oPyC	
Thickness	μm	95/40/35/40	
Density	g/cm^3	1.05/1.90/3.18/1.90	
Coated B_4C Particles:			
Particle diameter	μm	400	
Density	g/cm^3	2.45	
Coating material		C/iPyC/SiC/oPyC	
Thickness	μm	95/40/35/40	
Density	g/cm^3	1.05/1.90/3.18/1.90	
Reflector:		Inner	Outer
Thickness	cm	78	22
Graphite density	g/cm^3	1.75	1.55

A pre-study performed of the optimal B_4C kernel size indicated a size within the range of 200 - 400μm. For purposes of this study 400μm was assumed. A SSF distribution through the reactor from top to bottom is depicted in Fig. 8.

In Fig. 9 the discretely calculated power profile is plotted of the axial fuel batches modelled within the inner core channel.

The burnable poison content and type is to be optimised as a function of specific core height and desired burnup.

Table III lists characteristic data of the fuel element reference design, reflector, and thermal shield. For the neutronic calculations the core model is of importance up to the carbon blocks. The carbon blocks are treated with B_4C to enhance a sharp decline of the neutron flux and indeed provide thermal neutron shielding to outlying sensitive construction elements.

Reactor Performance

The OTTO-150 design has been completed for the 7g heavy metal, HTR-MODUL fuel element layout. This pebble bed reactor concept masters the Depressurized Loss of Coolant Accident (DLOFC) by total passive means, i.e. without activating any safety devices. Consequently most of the characteristics of the nuclear performance and fuel cycle are close to the HTR-MODUL (Table IV).

Larger differences are, however, observed in the core related dimensions, helium heating, and the fuelling scheme. This data are significant to the reactor safety performance.

TABLE IV: REACTOR PERFORMANCE DATA

		OTTO-PAP	**HTR-MODUL**
Thermal Power	MW_t	150	200
Core Volume	m^3	50	67
Helium heating	°C	560→900	250→700
Enrichment N_{U5}/N_U	%	9.6	7.8
Conversion ratio		0.40	0.46
Neutron losses:			
Fission products	%	6.1	7.3
Leakage	%	11.6	14.2
Burnable poison	%	6.8	0
Power peaking	KW/Sphere	1.36	1.37
Maximum fuel temp	°C	933	856
He $temp_{Out}$ - Max/Min	°C	890/912	644/788
U_3O_8-Requirement	kg/GWd_t	294	238
Separative work	kg SWU/GWd_t	220	171
Fissile Inventory	kg/GW_t	641	500

In the reference reactor the shut down system consists of 4 and 12 rods located in the noses and the reflector (Fig. 5), respectively. The shut down capability is approximately 3% higher than that of the HTR-MODUL which consists of small absorber spheres, the so-called KLAK (KLeine Absorber-Kugeln), being released into 20 vertical channels in the side reflector (Fig. 10).

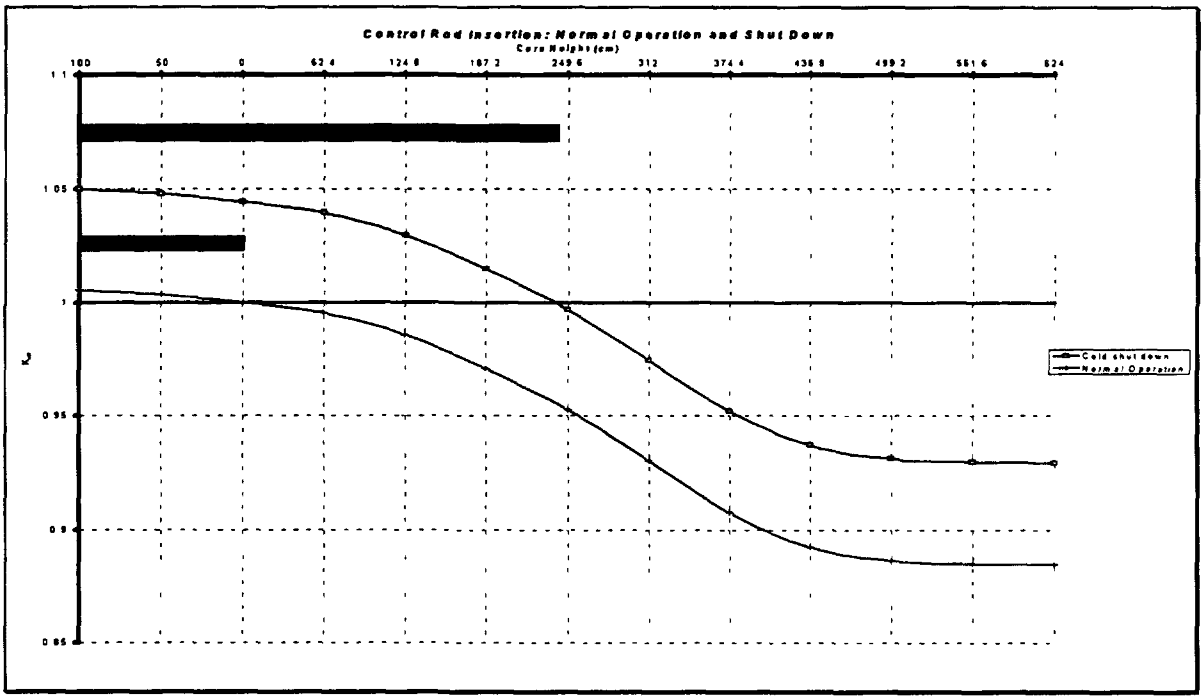

FIG. 10: Control rod insertion - Criticality control

The shutdown requirement is 21% larger than that of the HTR-MODUL. This difference can be attributed to the following three contributors, i.e. the higher operational temperature, a modified neutron spectrum brought about by the boron poisoning, and the presence of the noses which positively contributes to the overall temperature coefficient. As can be seen in Table V the temperature coefficient becomes more negative. This apparently, is due to the modified neutron spectrum caused by the burnable poison.

TABLE V: CONTROL AND SAFETY PERFORMANCE DATA

		OTTO-PAP	HTR-MODUL
Criticality change at shut down	K_{eff}		
Operation →cold shut down	50°C	0.044	0.030
^{135}Xe decay		0.047	0.045
Total		0.091	0.075
Shutdown system		16 rods	20 KLAK
Full insertion	δK_{eff}	0.121	0.118
Temperature coefficient ($\delta K_{eff}/\delta T$)	(x 10^{-5} C^{-1})		
Fuel		-2.8	-4.5
Moderator		-3.7	-0.6
Total		-6.5	-5.1
DLOFC			
Max fuel temp	°C	1572	1535
Time of max temp	h	30	27
Release/Production of decay heat integrated over 2 days	%	71	28

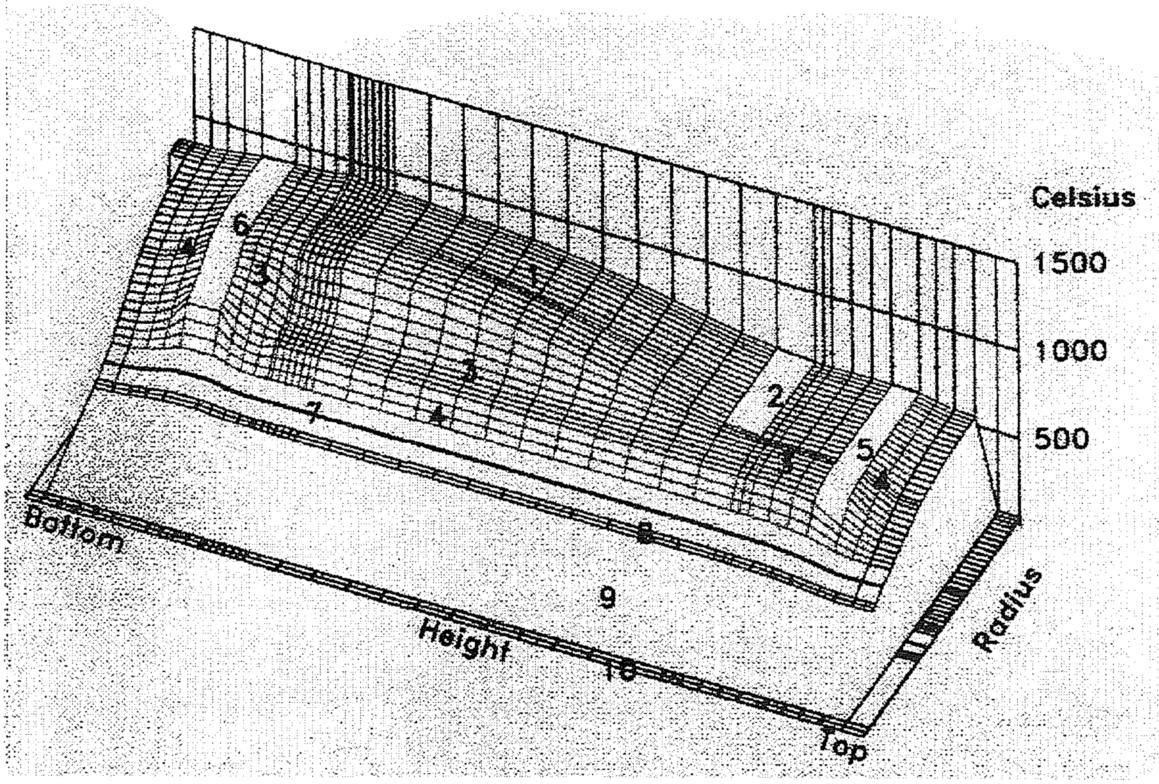

FIG. 11: OTTO-150 - DLOFC temperature distribution in the reactor at accident onset time

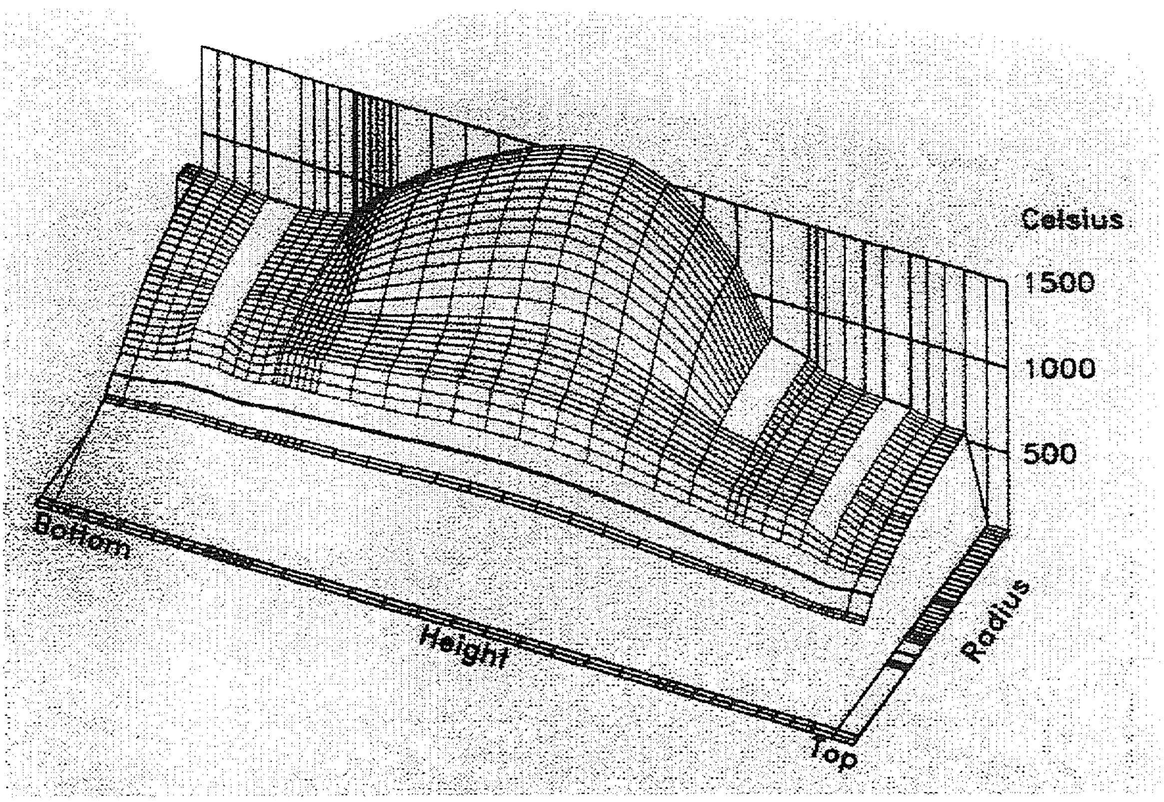

FIG. 12: OTTO-150 - DLOFC temperature distribution at 30 hours into the accident

In Fig 11 the OTTO-PAP model layout is presented

During a DLOFC the passive removal of the decay heat proceeds exactly in accordance with the HTR-MODUL. The maximum fuel temperature, depicted in Fig 12 remains below the limit of 1600°C, which is regarded the limiting temperature for full retention of the radioactive nuclides in the coated particles An important difference is observed in the decay heat domain Whilst in the HTR-MODUL decay heat is mostly stored (72%) in the reactor and internal components over the first 2 days, the OTTO-PAP predominantly releases decay heat (only 29% stored) via the reactor pressure vessel to be removed by the reactor cavity cooling system (RCCS) (see Figs 13a & b) This is due to the higher operational temperature of the reactor, taken as the initiating temperature of the DLOFC transient calculation

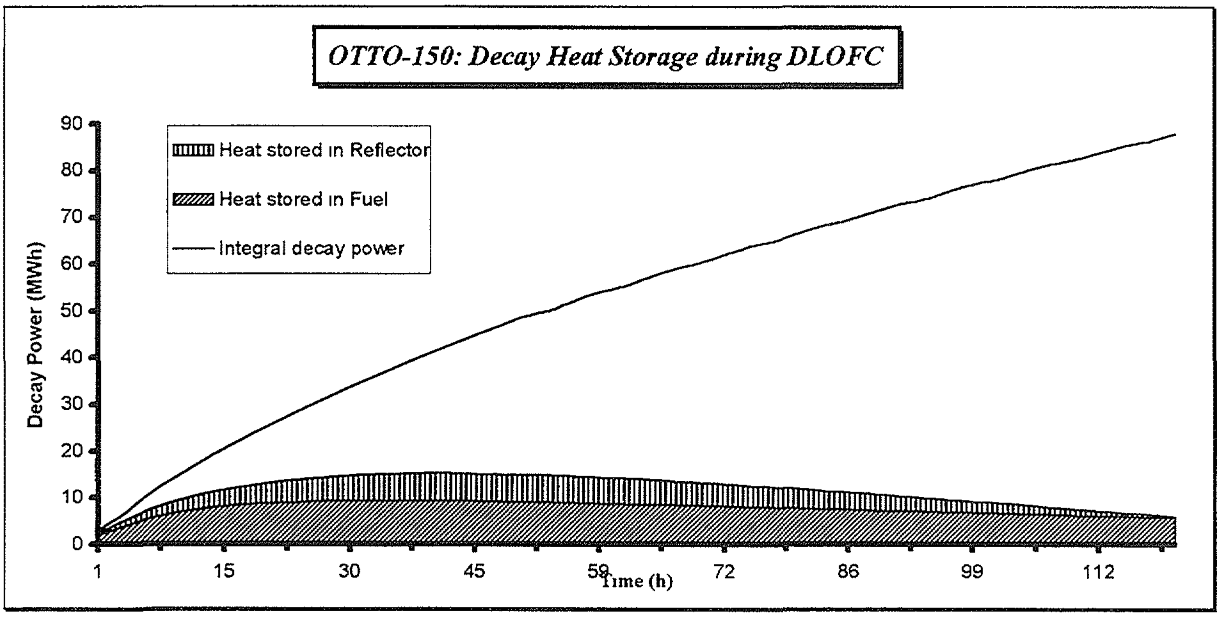

FIG 13a OTTO-150 Characteristic decay heat storage during DLOFC

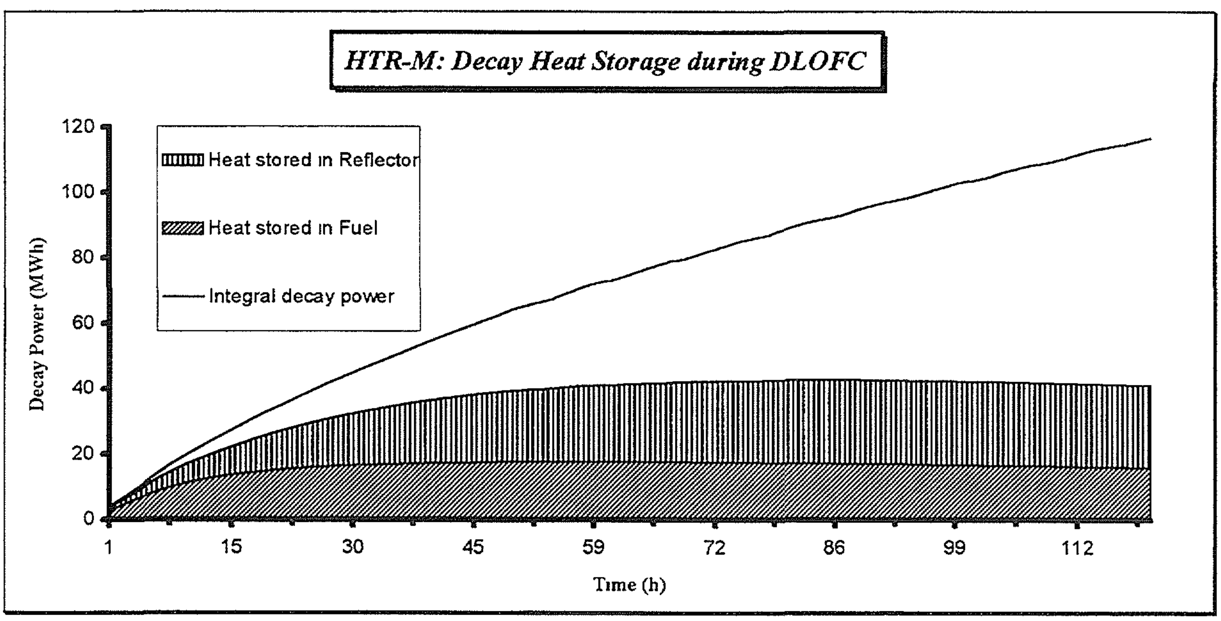

FIG 13b HTR-MODUL - Characteristic decay heat storage during DLOFC

In Fig 14 the temperature behaviour in the fuel and reactor pressure vessel is depicted during the accident, while Fig 13a clearly displays cooling down of the core and reflector after 33 hours since the time of accident inception Fig 13b outlines the heat storage performance differences within the HTR-MODUL reactor

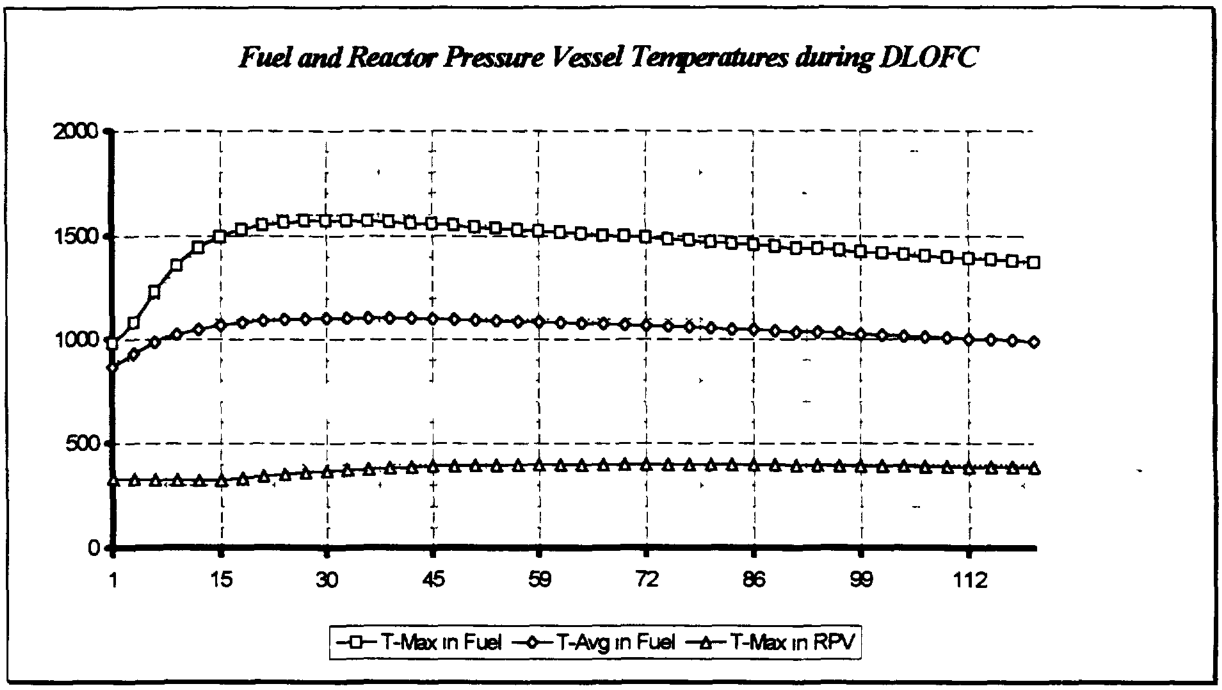

FIG. 14 OTTO-150 - Temperature transients during DLOFC

Conclusions

A PBMR reactor design based on the following philosophy is offered
Technology pertaining to reactor layout and control is considered within the realm of observed dynamic similarity,
The proposed advanced fuel design can be realised (manufactured and qualified) within the existing infra-structural frame of reference /15/

During OTTO-PAP fuelling of a high temperature pebble-bed reactor, fuel spheres are initially being added daily, on-line, in accordance with the fuel burnup Off-line, batchwise de-fuelling is possible for a period to be optimised between six months and two years after reaching an equilibrium core Axially a flat neutronic flux profile resembling that of the MEDUL fuelling regime is achieved by introducing B_4C coated particles of particular size in accordance with a pre-calculated packing density within the fuel spheres. A higher power output together with an optimised overall fuel utilisation is achieved by this manipulation

An optimised radial temperature distribution is achieved by introducing a 2-zone core loading The excellent possibility is offered for power production when coupled to a the power conversion unit (PCU) with Brayton cycle due to the relatively low temperature gradient within the fuel spheres

Physical properties of the OTTO-150 reference design is reported, while computations performed observe results obtained by the reference MEDUL design. Characteristic data for this reactor layout is presented.

Advantages of the OTTO-PAP and Peu-à-Peu fuelling schemes include simplification of the overall fuel management regime, operational ease offered by off-line de-fuelling, and considerable capital saving due to the simplified component requirements for re-fuelling, burnup measurement and instrumentation. The basic fuel elements under consideration are conventional, well-tested with regards to production techniques, temperature and irradiation performance, and burnup.

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HTR PLANT SYSTEM AND COMPONENT DESIGN

GENERATOR TECHNOLOGY FOR HTGR POWER PLANTS

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Abstract

Approximately 15% of the worlds installed capacity in electric energy production is from generators developed and manufactured by GEC Alsthom. GEC Alsthom is now working on the application of generators for HTGR power conversion systems. The main generator characteristics induced by the different HTGR power conversion technology include helium immersion, high helium pressure, brushless excitation system, magnetic bearings, vertical lineshaft, high reliability and long periods between maintenance.

1- Experience

When we speak about manufacturing, experience is one of the major concerns for Power Plant users. Only previous experiences can ensure the real feasibility and safety of an equipment. GEC ALSTHOM is a world leader in Generators manufacturing. It supplies Air-Cooled and Hydrogen-Cooled generators for different power plant applications: Hydro, Nuclear, Combined Cycle, Fossil Fired, Gas Turbine.
It has a world record in generators with a rating of more than 1710 MVA in a single unit (Chooz Nuclear Power Plant in France).

With about 15% of the world installed capacity in electric energy production, GEC ALSTHOM developed expertise in different Power Plant development and manufacturing.

In all continents, many countries are now equipped with GEC ALSTHOM equipment.

GEC ALSTHOM is one of the main suppliers of power conversion systems for nuclear power plants. Its experience in the nuclear field is unique with a total of 117 units manufactured:
- France: 68 units
- United Kingdom: 24 units
- Belgium: 8 units
- South Korea: 6 units
- China : 2 units + 2 under manufacturing
- USA: 3 units
- South Africa: 2 units
- Sweden: 2 units

The large experience of GEC ALSTHOM in Hydrogen-immersed generators permits the application of their technology to the HTGR generator systems.

GEC ALSTHOM is now working in different applications of generators for HTGR power conversion systems.

2- Main technical characteristics of the HTGR Power Convertion system

Main characteristics induced by the different HTGR power conversion technologies to the generator systems are:

A- Helium immersion
B- High pressure
C- Brushless excitation system
D- Magnetic Bearings
E- Vertical lineshaft
F- High reliability
G- Long periods between maintenances

These parameters shall be taken into account for the generator design.

A- Helium immersion

The dielectric properties of helium have to be taken into account. The dielectric strength of Helium is much lower than the air one. The generator design needs to respect insulation distances with a view to avoid dielectric origin problems.
This problem can be solved by pressure increase: it is assumed that the dielectric strenght of Helium at 10 to15 bars pressure is equivalent to air at atmospheric pressure. This is the reason that with a standard design no full voltage in the generator is expected before this minimum pressure value. Nervertheless, the machine can be excited at reduced voltage - for example when it is connected to a startup frequency converter. In fact, there is a direct relation between generator maximum voltage and helium pressure.
Under certain conditions, a special design adapted to a lower minimum pressure can be realized.

B- High pressure

More than the absolute value of the service pressure the pressurisation and depressurisation rates of the generator vessel are one of the sensible points of the system.
According to the compacity of the insulation used, the pressurisation and depressurisation actions versus time may be limited .
Effects of rapid variation of pression on the usual insulation system will have to be determined by tests.
Nevertheless some tests performed in the past on other Helium immersed and smaller electrical machines gave promissing results and showed that (or adaptations of) existing insulation technics can be used.

C- Brushless excitation system

Due to the direct helium flow through turbine and nuclear core, no dust is allowed.
As access to the excitation system wil be exeptional , it will not be possible to use static excitation with brushes. furtheremore, static excitations need a lot of care.
A brushless excitation without direct rotor contact will be compulsary.
GEC ALSTHOM has a large experience on compact brushless excitation systems.

D- Magnetic Bearings

For the same above reason, no oil will be acceptable inside the system. The only technology now available is magnetic bearings.
Magnetic bearings and thrust bearing integration on such a generator is an advantage. Dynamic magnetic bearings can participate on the stability of the lineshaft.

E- Vertical lineshaft

The verticality of the lineshaft is a choice. This choice was taken mainly because of civil works saving reasons (the nuclear vessel and power convertion vessels may be under earth). The shortness of the connection in between the two vessels is also one important point of this choice.
Verticality of the lineshaft can also be an advantageous configuration as far as the magnetic supporting is concerned.

GEC ALSTHOM has a large experience in vertical oriented lineshafts. this was mainly on hydro-generators application.

F- High reliability

Due to the high pressure vessel enclosure, the generator access will be more difficult than in conventional machines.
The reliability of equipment is one of the major concerns.

G- Long periods between maintenances

The limited generator accesses and the maximization of availability will lead to long periods between maintenances.
Maintenance executed each 6 years is now common on GEC ALSTHOM maintenance policy. Additional instrumentation can be installed in the generator with a view to control its behaviour.

3- <u>Implication of these different parameters</u>

The implication of these different parameters will lead GEC ALSTHOM to adapt their conventional proven generator design to this specific application.

Nevertheless, this constraining environment will imply different checks and tests of the capacity of the GEC ALSTHOM technology to comply with these specific requirements.

Some investment in research and development will be then necessary for confirmation of compliance of the existing technology or to change for a suitable one.

4- Main research and development needs

- Generator insulation
  . Dielectric characteristics
  . Behaviour in case of fast variation of pressure

- Diodes
  . Compatibility with dielectric and pressure

- Excitation and starting system
  . In case of generator used as motor

5- Brayton cycle starting technical choice

According to the technical choices on the power conversion system, the Brayton Cycle can be started by using the generator as a motor.
This solution implies having a direct link of the generator with turbine and compressor.

With a view to drive the generator as a motor, a Synchronous Frequency Converter must be used. This system will control the speed of the lineshaft. It will drive it up to the speed when the turbine can sustain the lineshaft. The speed will then increase up to nominal. The generator can then be connected to the grid (synchronisation).

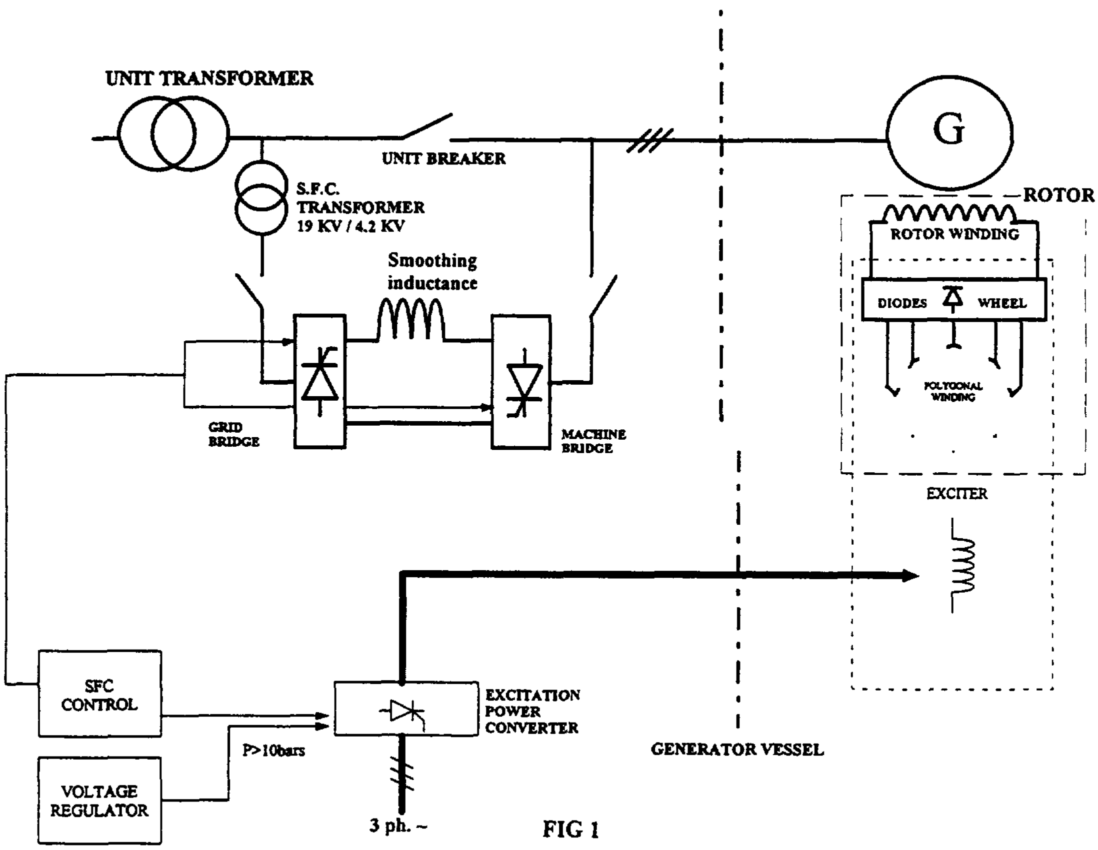


FIG 1

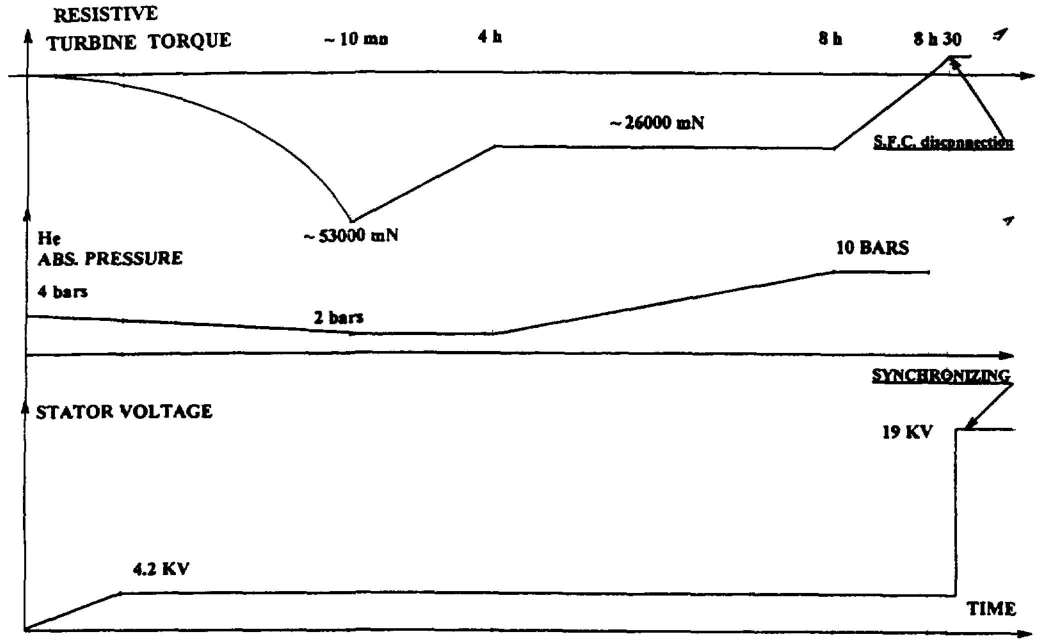


FIG 2a

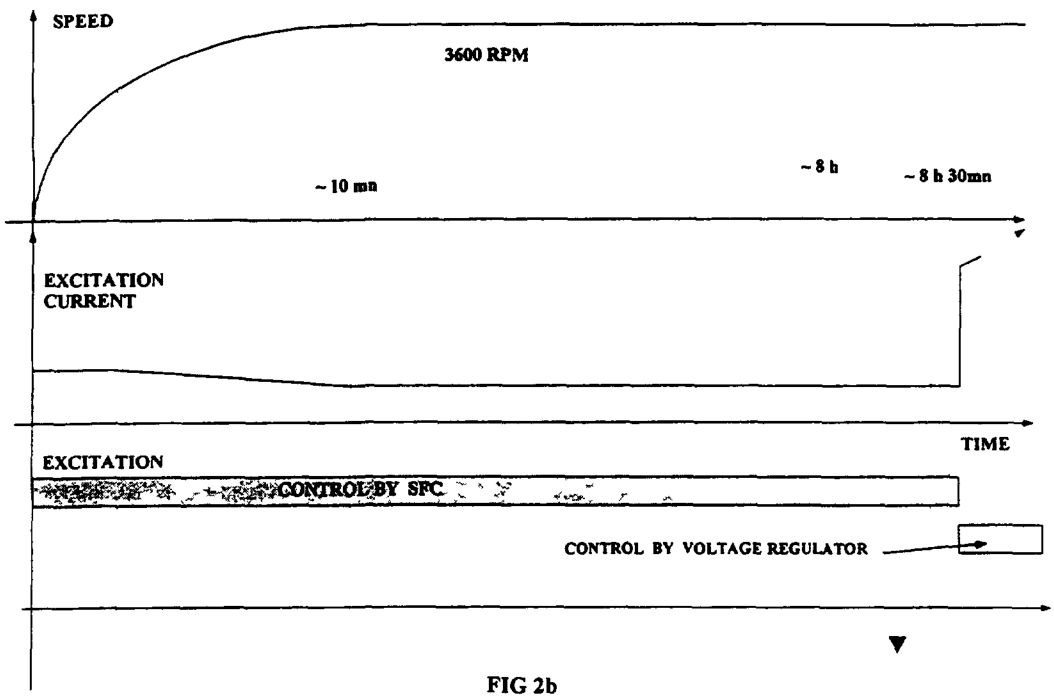


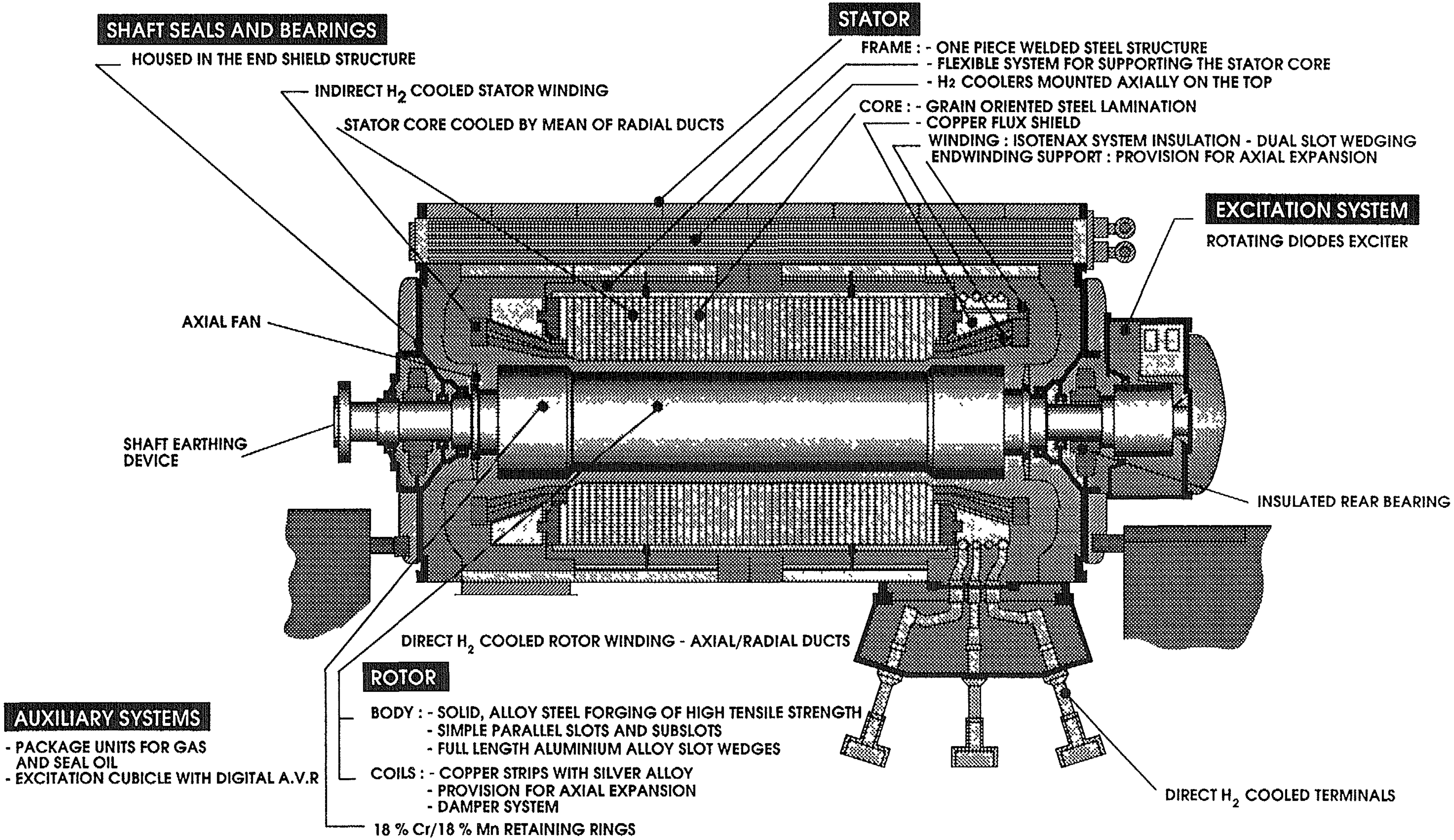
FIG 2b

The frequency converter can be used when stopping the unit the speed can be controlled according to appropriate stages of deceleration and to help core cooling

Finally, it can also stop the lineshaft acting as a brake

Fig 1, 2a and 2b illustrate a simplified schematic single line diagram and the associated starting sequence, both established for the General Atomics/ GT-MHR project

Conventional Hydrogen Cooled Generator
Main Design Features



6- HTGR Generator main technical caracteristics

Main generator caracteristics		Comments
Power	from 30 to 500 MW	according to the power plant project choice
Power factor	usualy 0.85	to be defined by the customer
Voltage	from 10 to 26 kV	function of generator power and dynamic operation of the system
Cooling water temperature at generator inlet coolers	15 to 35°C	according to the country and site location
Efficiency	~ 98 %	efficiency is lower in Helium high pressures due to increased mechanical losses
Speed	3000 or 3600 RPM	according to the frequency of the grid 50 or 60Hz
Over speed	120 %	usual requirements with standard machines
Cooling medium	helium to water coolers	rotor and stator: axial-radial cooling
Helium pressure	normal operation ≈30 bars	according to nuclear core and turbine operation
Temperature	insulation: class F 155°C max.	IEC standards
Bearings	Magnetic bearings	can participate actively on the stability of the lineshaft
Lineshaft orientation	vertical	project choice
Excitation	Brushless	Static excitation to be avoided
Generator used as motor	with S. F. C. drive	project choice
Planned maintenance intervals	≈ every 6 years	according to main power plant maintenance planned period
Design life	30 years	with half-life refurbishment
Turbine-generator Coupling	usually direct coupling	coupling and torsionnal analysis of the lineshaft depends on the flexibilities, inertias and generator short-circuit torque

7- HTGR Generator technical description

The HTGR generator is directly derivated from a standard GEC ALSTHOM machine where existing proven technologies are applied. The Figure 3 shows an example of general arrangement for such a HTGR generator.

A) magnetic core stacking

The magnetic core is made of silicon steel segmental laminations with high permeability and low specific losses. These laminations are varnished on both sides , assembled in annular manner, and separated into packets by radial vents through which the cooling gas circulates.

After stacking the core assembly is securely clamped between two heavy fingered plates and keybars uniformly distributed at the periphery of the magnetic core. A flux shield of high conductivity material is mounted on the clamping plates and protects the stator ends from the leakage flux between the stator and the rotor (please see Fig. 4).

B) Stator armature winding

The armature winding is formed by connecting half-coil bars to make a balanced three phase type winding.There are two bars per slot. The connection between consecutive bars is made by brazing their ends with lateral copper plates .The bars are made of insulated copper strands, transposed according to the Roebel system all along the core length, and ground insulated over their full length (please see Fig. 5).

The standard insulation technology consists of a mica paper tape, glass cloth and bonding epoxy resin. Each bar is insulated over its whole length, straight portion and ends by continuous taping according to the Isotenax process. After wrapping, the bar is moulded and hot-polymerized.

The endwindings are rigidly supported by fibre glass rings which are themselves supported by a number of axial brackets fixed to the stator core endplate assembly (please see Fig. 6).

C) Flexible suspension

The magnetic attraction between the rotor and the stator produces a rotating elliptical deformation resulting in a double frequency vibration whose amplitude depends on the machine's electromagnetic sizing and core frequency.

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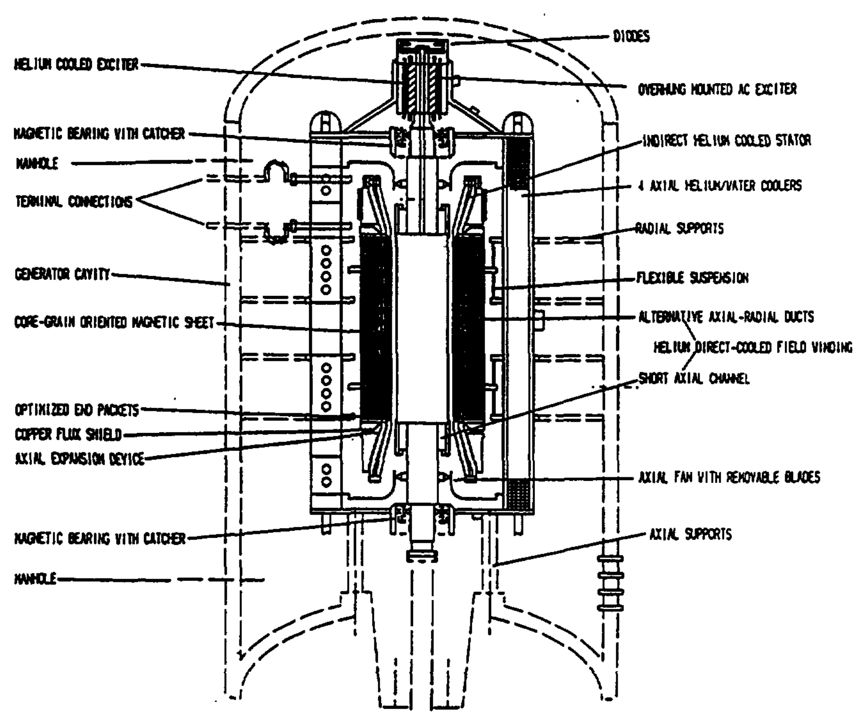
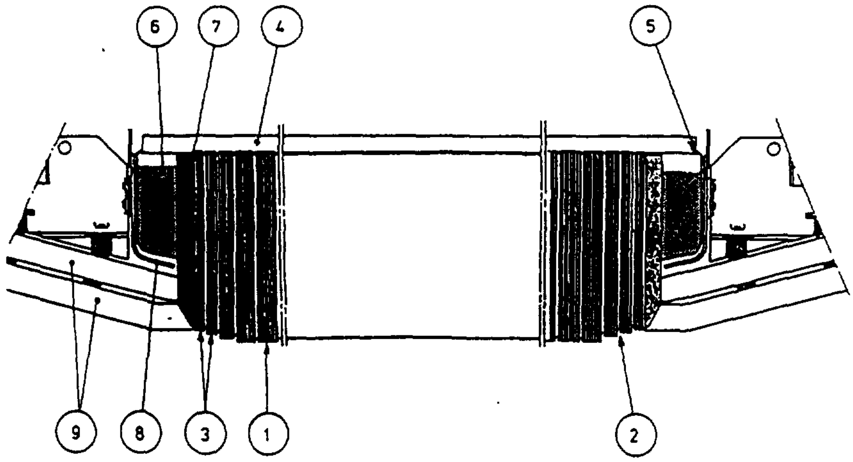


FIG 3



1 - Standard lamination packets
2 - Vent ducts
3 - Stepped lamination packets
4 - Key bars
5 - Welding
6 - Clamping plate
7 - Clamping fingers
8 - End flux shield
9 - Stator bars

FIG4

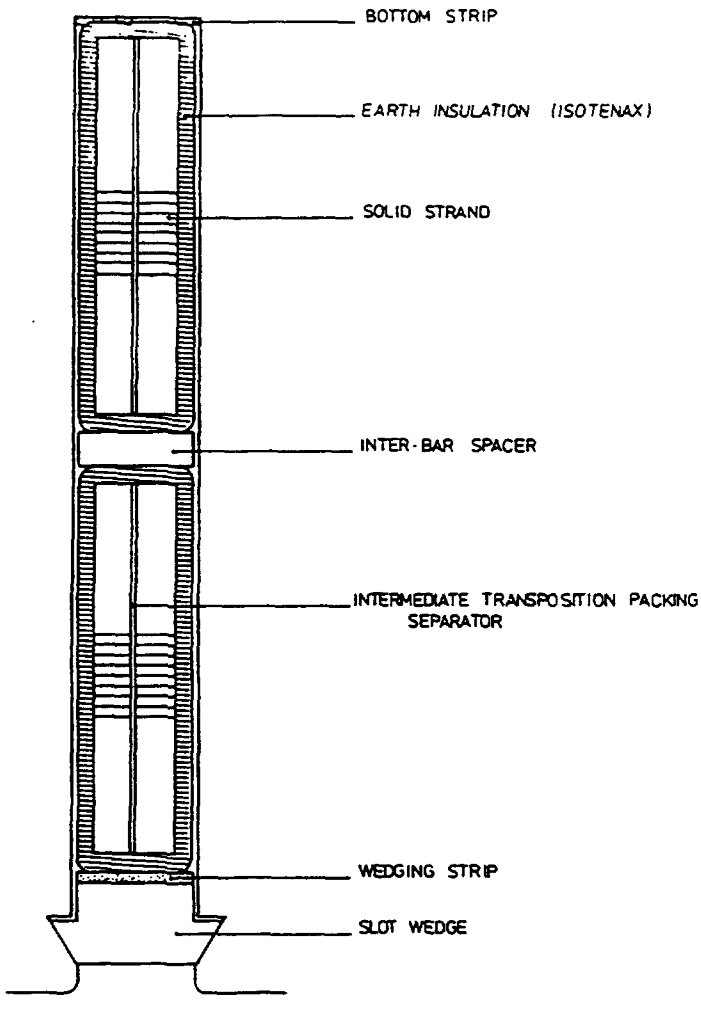
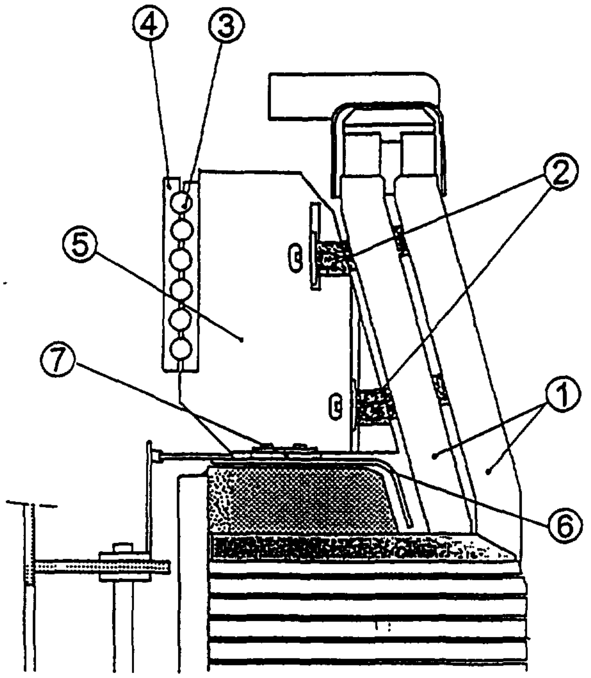


FIG 5

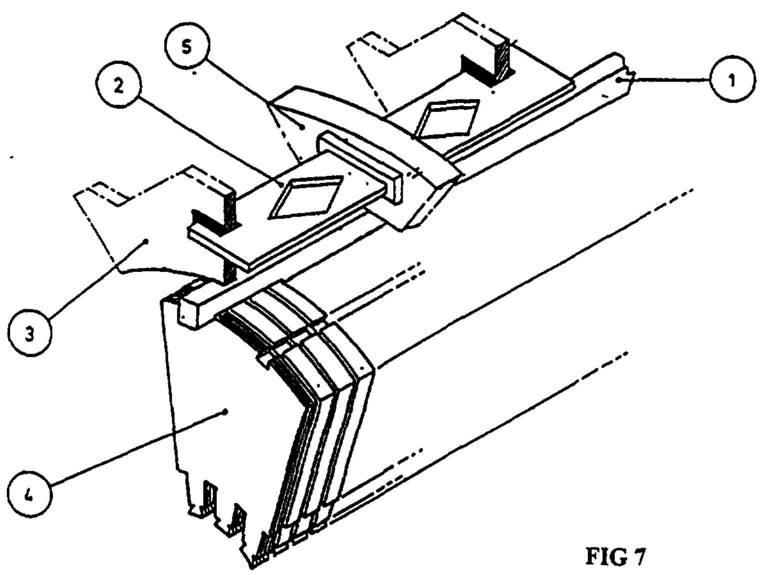
STATOR WINDINGS END SUPPORTS



1.Stator bars
2.support rings
3.circular phase connections
4.insulating block
5.insulating support brackets
6.Flux screen
7.Suspension device

FIG 6

ELASTIC FASTENING OF CORE



1 - Key bar
2 - Spring
3 - Frame web
4 - Magnetic core
5 - Movable web

FIG 7

The core is flexibly mounted to the frame so that vibrations transmitted to the frame are minimised (please see Fig. 7). The flexible suspension also influences the short-circuit torques transmission.

D) Rotor

The rotor is a monobloc steel forging in which some slots are machined to receive the field winding. Each winding turn consists of two strips of alloy copper. The slots are fitted with an insulation to earth consisting of epoxy resin impregnated glass slot liners. Complete coils are obtained by brazing the half coils in the axis of the poles (please see Fig. 8a and 8b).

ROTOR COIL DETAILS

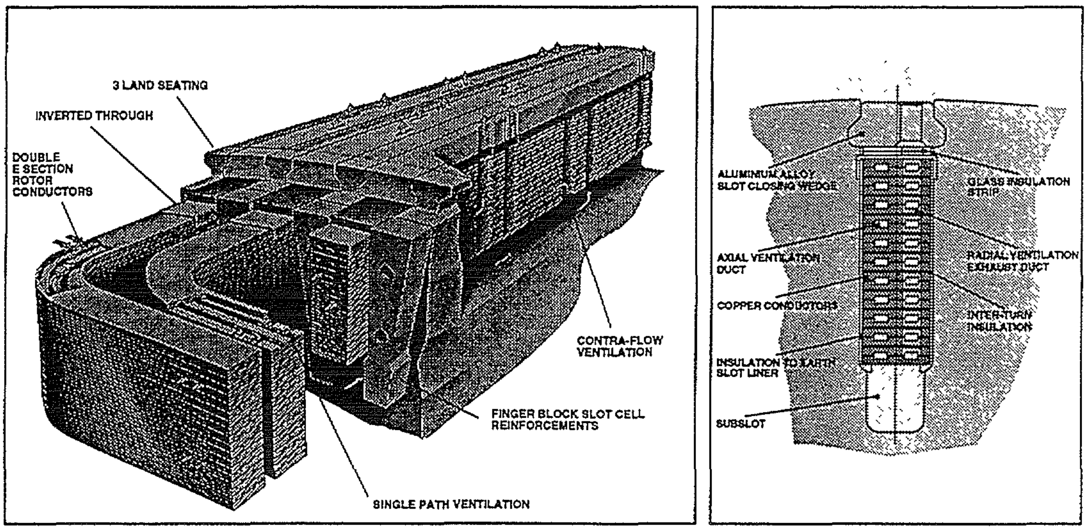


FIG 8a

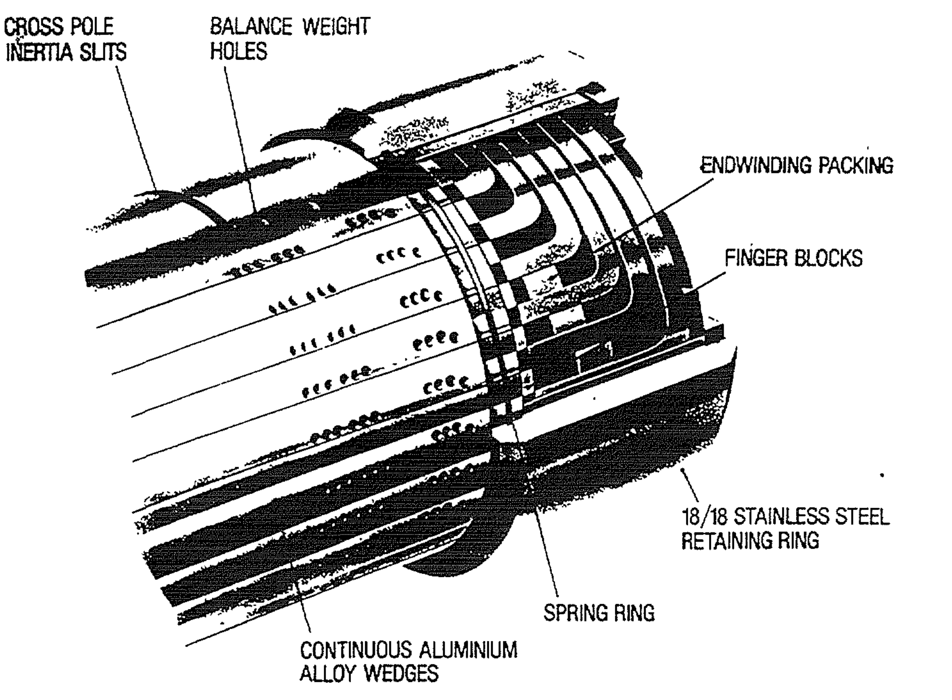


FIG 8b

AXIAL-RADIAL VENTILATION SCHEME

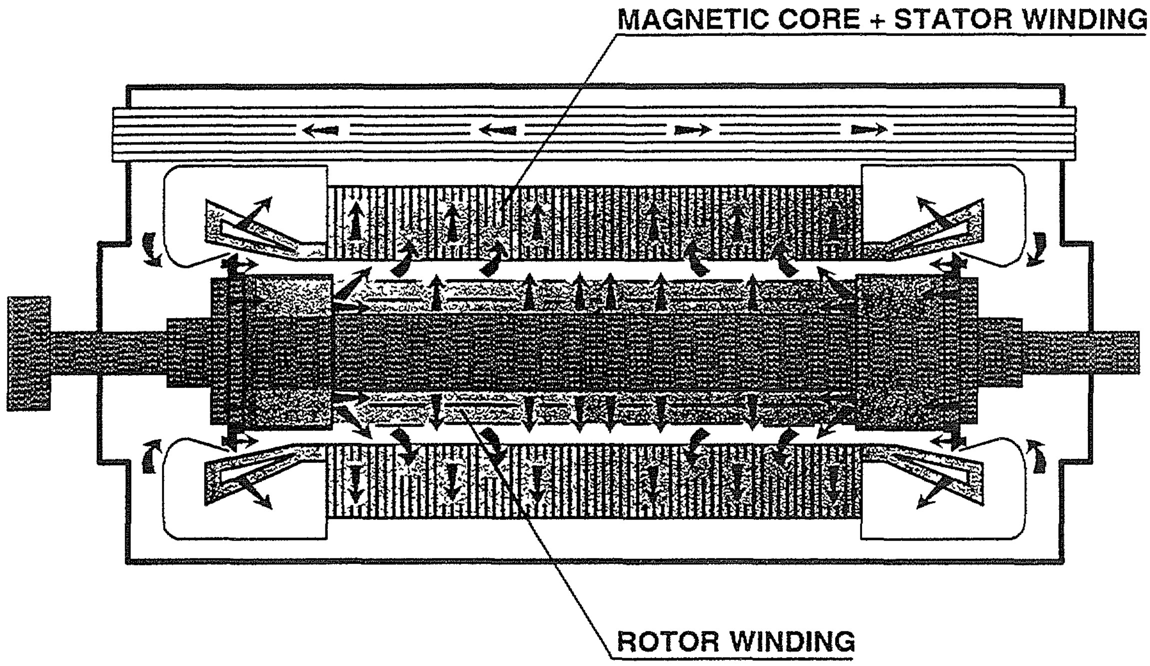


FIG 9

The endwindings are secured by retaining rings and insulating blocks between each coil.The retaining rings are forged in high mechanical properties non-magnetic alloyed steel, and separated from the enwindings' copper by layers of laminated bonded glass

E) Cooling

In high pressure Helium, there is no need for water cooled stator bars before at least 500 MW The stator cooling is simple and reliable, of the indirect type, similar to air cooled machines

The rotor is of direct cooling type, using a combination of radial and axial ducts machined in the rotor forging and in the copper strands, in which the cold Helium flows under the effect of rotation
The general gas circulation in the machine is ensured by two axial fans mounted on each end of the rotor.

All the losses produced in the generator are absorbed by Helium which circulates in closed circuit and is cooled by water exchangers arranged around the stator (please see Fig. 9)

F) Brushless/Bearingless exciter

A brushless rotating diode exciter is an inverted AC generator, the stationary part being the field magnet and the rotating part of the armature The armature winding is multiphase and is connected to the main generator field winding through a diode rectifier bridge mounted in overhang (Fig 10) The rotating armature of this exciter is directly bolted at the rotor shaft end and has no additionnal bearing.

BRUSHLESS EXCITER CROSS-SECTION.

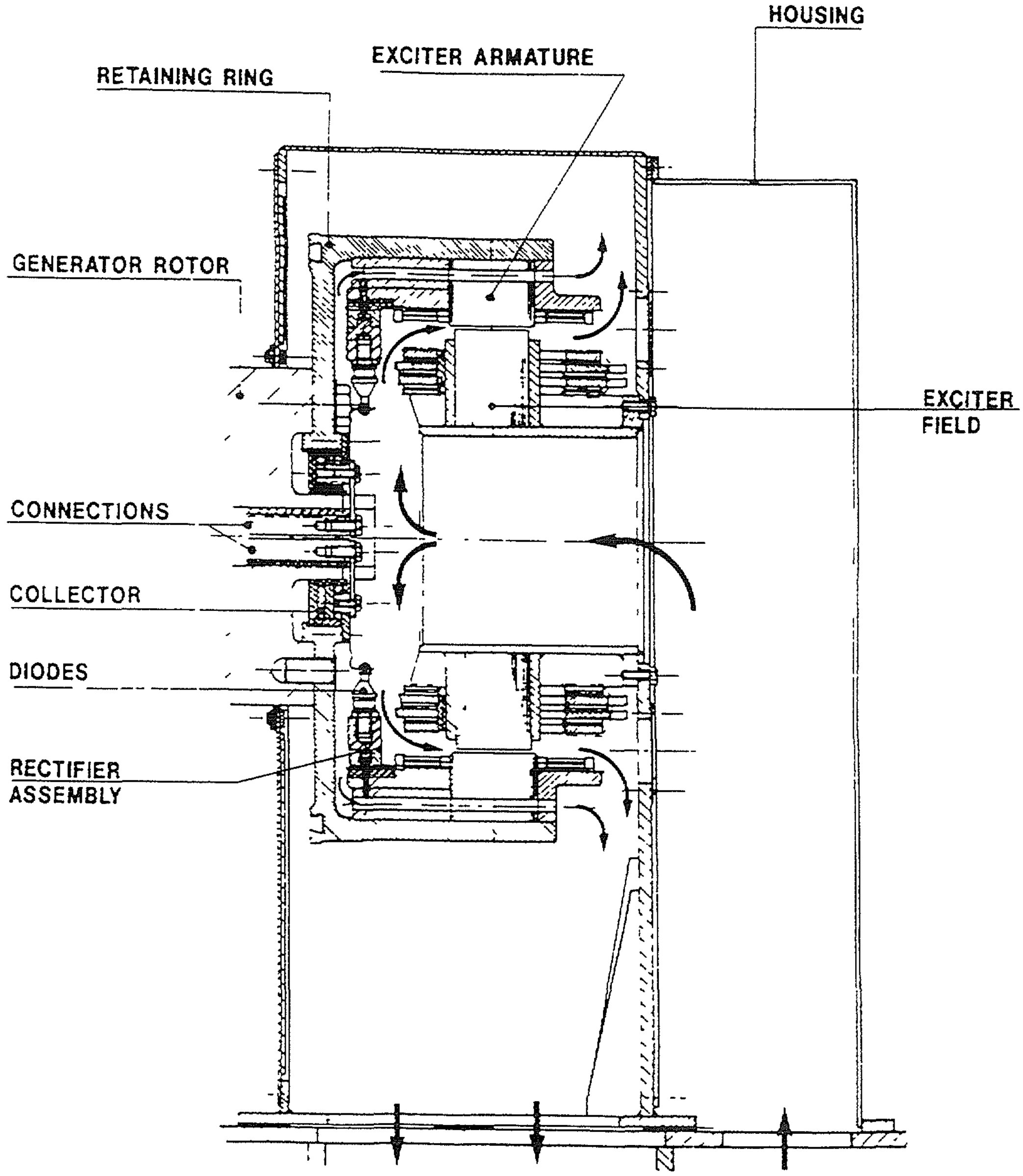


FIG 10

Diodes are electrically oversized. A failure of these components can be detected by its influence on the exciter's field current waveshape.

In the case of association with a static frequency converter, some adaptations are necessary to allow starting from zero speed.

G) Design Adaptations to HTGR

This basic technology will be retained for the HTGR application, but some specific adaptations may be realized in the general dimensioning of the machine

- The generator can be smaller than a conventionnal machine. This is due to the very high cooling capability of pressurized Helium.

- The cooling gas flow is reduced at the minimum possible value to limit the ventilation losses

- The rotor design may be adapted if necessary, in order to obtain a higher overspeed value, lower the friction losses and maintain usual electrical characteristics.

- Under certain conditions, a special stator design, allowing operation at Helium pressures inferior to 10 bars can be obtained.

H) Maintenance

The generator can be equipped with a complete monitoring system to avoid unproductive inspections and removals of the generator:

- Temperature of core
- Temperature of windings
- Conditions of insulation
- Radiofrequency detection
-...

The maintenance operations are of the same type as for conventional generators. With the machine placed at the upper part of the lineshaft, their duration is comparable, with some possible adaptations, as a function of the turbo machine's accessibility and detailed design.

Minor repairs (diode replacement, rewedging of end-windings...) can be done without removing the generator.

For minor insulation touch ups, only the rotor is removed.

For major insulation touch ups, the generator can be exchanged with a replacement unit and the repair is done on site.

XA9846687

DEVELOPMENT OF AN ENGINEERING SIMULATOR AS A DESIGN TOOL FOR THE PEBBLE BED MODULAR REACTOR

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Abstract

The Pebble Bed Nuclear Reactor concept being studied in South Africa has, in many aspects, taken a new approach to and moved away from other designs. Different international entities are supplying expertise on different specific parts of the plant. All of the subsystems are well defined in their own sphere of expertise. However, the interactions between the individual components remained a question mark. An engineering simulator was developed to overcome these uncertainties and to investigate the response of the system. This paper describes the extent and scope of the simulation as well as the results obtained so far in Phase 1.

1. Introduction

The Pebble Bed Nuclear Reactor concept being studied in South Africa has , in many aspects, taken a new approach to and moved away from other designs. A number of international concerns each supply expertise on a specific part of the plant e.g. the AEC in South-Africa on the turbo-machinery, KFA of Germany on the reactor core [3][10] etc. There are many more experts involved in the active and passive heat removal elements as well as the helium inventory system.

All these subsystems are well defined in their own sphere of expertise. The interactions between the individual components however remained a question mark . To overcome these uncertainties it was decided to investigate the response of the system by developing an engineering simulator. This simulator is intended to investigate the above mentioned component interaction. The platform used was the G2 Expert System from Gensym.

Much time and effort was expended in trying to model the major components of the plant as accurately as possible to the experts' design and to incorporate the current design base of all parties into the design tool. This paper describes the extent and scope of the simulation as well as results obtained so far in Phase 1.

2. Plant Overview and Extent of Simulation

The Pebble Bed Reactor Plant is envisaged to consist of the following subsystems:

- *Main/Primary System - Reactor Core and PCU*
- *Helium Inventory System*
- *Shutdown Heat removal system*
- *Active Heat removal system*
- *Passive heat removal*
- *Plant Control system*

- Ventilation & Fire protection System
- Water supply & purification system
- Plant electrical system
- Decontamination & Waste storage
- Spent fuel system
- Fuelling & Defuelling system
- Primary loop initial cleanup system

The focus in the first phase of development clearly falls on the critical elements of the system which also form the largest part of the study. The parts of the overall system modelled were chosen on the basis of the uncertainty the design team encountered during the initial stages of the study.

The most important of these was the Main/Primary System i.e. the reactor core, turbo-machinery, heat removal and power conversion components such as the alternator. In short, all the systems involved in the Brayton cycle. Necessary for the analysis of the Brayton Cycle and component interaction were the following subsystems - Helium Inventory , shutdown- , active- and passive heat removal as well as those parts of the plant control system involved in regulating the cycle. Control systems included are those regulating power, inventory, active cooler temperatures and control rod position.

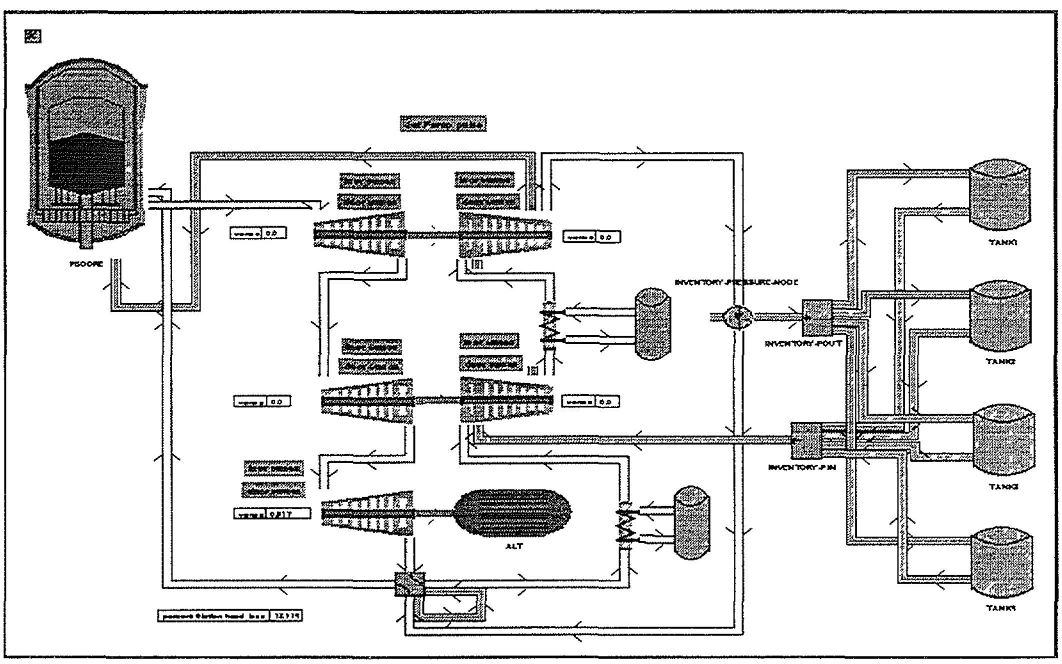

FIG. 1

3. Scope of Simulation

Before discussing the scope of the simulation it is necessary to explain both the purpose and desired results of the engineering simulator.

The purpose of the simulator is threefold:

- Investigate the interaction between the components of the system, included in which was a study on the stability of the Brayton cycle in the current system configuration.
- To study the system's power ramping and step change capability .
- Develop and study the plant control philosophy.

Component models are tailored to function correctly under the operating condition expected during these tests. It was not envisaged to perform detail testing on individual components. The following is a look at the scope/detail of simulation for the unit elements. Equations and calculations are shown where relevant and where the models are not too complex to be reproduction here.

3.1 Reactor core

The core simulation employs a point kinetic neutronics model as supplied by the core experts at KFA in Germany [3][10]. It takes into account reactivity effects such as fuel, rods and moderator reactivity as well iodine and xenon effects. Temperatures of fuel, moderator and pebble shell are modelled and the thermal energy transfer from pebble to gas as well as thermal interaction with the environment are also included.

As yet the model does not incorporate the full thermohyraulics for thermal transfer between pebbles and helium flowing through the core. This would mean splitting up the core into regions / layers, which will most likely be incorporated into the simulator during the next phase of design. The current model is sufficiently accurate to allow analysis of the system.

3.2 Power Conversion Unit

The power conversion unit consists of the following pieces of equipment who's interconnection can be seen in Figure 1. A network of pipes, two (2) turbo-compressor pairs, a power-turbine, alternator , recouperator and active cooling elements. These are individually discussed below:

3.2.1 Network of pipes and pressure nodes

The flow system is modelled as a network of pipes in which a momentum balance is maintained and a set of pressure nodes at the component-pipe interface taking care of the mass- and energy balance of the system [5]. Temperatures are transferred along the pipes at a rate proportional to the mass flow in that part of the system.

The terms Kflow is included for numerical stability only. Kpress is introduced also for stability reasons since the global- instead of the real compressibility was used. Both terms do not influence the dynamics of the system but enhances numerical stability.

3.2.1.1 *Momentum Balance*

The conventional equation for momentum balance is [5]

$$\frac{L}{A}\frac{dW}{dt} = P_i - P_o - dP_{fr} + dP_p - \rho gh \qquad \textbf{Equation 1}$$

where dPfr is given by [5]

$$\begin{aligned} dP_{fr} &= K_{fr}W|W| \\ K_{fr} &= 0.1L/(2A^2) \end{aligned} \qquad \textbf{Equation 2}$$

Applying a filtering coefficient as follows Equation 1 becomes

$$W = W + Kflow * dT(P_i - P_o - dP_{fr} - \rho gh)A/L$$

Equation 3

Kflow was chosen to be 0.1. It has no effect on the steady state condition of the plant.

3.2.1.2 *Pressure calculation at nodes*

The pressure node calculations [5] calculated at the component in- and outlets is as follows:

$$C_i\frac{dP}{dt} = \Sigma W + Kpress * dt(\frac{mRT_{abs}}{V} - P)$$

Equation 4

The term Kpress provides for a correction of the pressure towards the value corresponding to the ideal gas law.

3.2.1.3 *Mass Balance at nodes*

The conventional mass balance formula [5] is used

$$\frac{dm}{dt} = \Sigma W_i - \Sigma W_o \qquad \textbf{Equation 5}$$

Due to the fact that the compressibility used in the pressure calculation is not the true value, the mass variations arising from a set of flows do not correspond exactly to the pressure variations arising from the same set of flows.

The mass calculated above should be used in the pressure/flow calculations, but if a true mass value corresponding to the current pressure is required then the ideal gas law should be used.

3.2.1.4 Energy Balance at nodes

The equation for energy balance [5] is as follows

$$mc_p \frac{dT}{dt} = Q_i + \Sigma W_i c_i (T_i - T) - \Sigma W_o c_o (T_o - T)$$

Equation 6

The mass 'm' used here should be derived from the real absolute pressure using the ideal gas law. i.e.

$$m = \frac{P_{abs} V}{R T_{abs}}$$

Equation 7

The effect of pressure changes on temperature is assumed to be negligible compared to heat and enthalpy flows.

Care must be taken to correctly handle the direction of flow when considering accident conditions. If a flow reverses then it carries the enthalpy of a different upstream node.

3.2.1.5 GLOBAL LOOP PRESSURE

As mentioned earlier, the pressure calculations correctly converge to values that reproduce local pressure variations, but these do not correspond to the ideal gas law value at any given node [13].

For the purposes of this discussion the pressure calculated by the above equations is referred to as the local pressure, while the true pressure corresponding to the ideal gas law can be called the global pressure.

The global pressure in the loop must be calculated by adding an offset to the local pressure. The offset is calculated for a reference node, for which the reactor vessel was chosen. The offset equals the difference at the reference node between the ideal gas law pressure and the local pressure [13]. Each node must add this offset to their local pressure if a true global pressure value is required. The local pressure should not be changed since this would have a serious impact on the pressure/flow calculations. A true global pressure value is typically required for display purposes or for control loops.

3.2.2 Turbines

These components have a full thermodynamic model incorporating calculations of power, torque, work done and outlet temperatures. The flow characteristics are given by a performance map linking the pressure drop across the device with volume flow through it [7][8] as supplied by the experts. Inlet guide vanes are also included to make control on the system possible. It does not include , at this stage, a rotational speed feedback component in its performance map influencing volume-flow. This effectively means that the volume flow (Q) through the turbine is given by Q = f (Pressure ratio, guide vane setting) i.e.

$$Q = f\left(\frac{\mathrm{p1}}{\mathrm{p2}}\right) \qquad \textbf{Equation 8}$$

$$W = Q \times \rho \qquad \textbf{Equation 9}$$

The temperature drop across the turbine and the outlet temperature are given by ,

$$\mathrm{dT} = \mathrm{T1} \times \left(1 - \left(\frac{p1}{p2}^{\frac{k-1}{k}}\right)\right) \qquad \textbf{Equation 10}$$

$$\mathrm{T2} = \mathrm{T1} + \mathrm{dT} \qquad \textbf{Equation 11}$$

The enthalpy drop is found from Equation 8

$$\mathrm{dh} = \mathrm{Cp} \times \mathrm{dT} \qquad \textbf{Equation 12}$$

From Equation 9 & 12 the mechanical power and torque are calculated.

$$\mathrm{P} = \mathrm{dh} \times \mathrm{W} \times \eta \qquad \textbf{Equation 13}$$

$$\mathrm{T} = \frac{60 \times \mathrm{P}}{2 \times \pi \times \mathrm{N}} \qquad \textbf{Equation 14}$$

3.2.3 Compressors

Compressors are modelled [8][11][12] in approximately the same manner as the turbines except that the performance maps incorporate both rotational speed and inlet guide-vanes i.e.

$$\mathrm{Q} = \mathrm{f}\left(N, \tfrac{p2}{p1}, Gv\right) \qquad \textbf{Equation 15}$$

Mass flow is the a function of volume flow and inlet density

$$W = \rho * Q \qquad \textbf{Equation 16}$$

And the work done is given by

$$\mathrm{Yk} = \mathrm{Cp} \cdot \mathrm{T1} \cdot \left(\frac{p1}{p2}^{\frac{k-1}{k}} - 1\right) \qquad \textbf{Equation 17}$$

From these both the outlet temperature and mechanical power and torque are derived,

$$\mathrm{T2} = \frac{\mathrm{Yk}}{\mathrm{Cp} \cdot \eta} \qquad \textbf{Equation 18}$$

$$\mathrm{P} = \frac{\mathrm{Yk} \cdot \mathrm{W}}{\eta} \qquad \textbf{Equation 19}$$

$$\mathrm{T} = \frac{60 \cdot \mathrm{P}}{2 \times \pi \times \mathrm{N}} \qquad \textbf{Equation 20}$$

3.2.4 Alternator

The alternator is represented by a full scale model of a device similar to the one used in Koeberg power station. Included in the calculations are electrical / mechanical power and torque , stator voltage and current, active and reactive power, power factor etc. Synchronisation to both an infinite and finite grid are possible so that scenarios such as con- and disconnection from grid (infinite grid), operation on a separate grid (frequency no longer fixed) as well as incidents such as loss of grid or alternator trip can be analysed. The model is too involved to reproduce here . Detail on this topic can be found in internal reference documents or in any text referencing the Parks transform [6][9].

3.2.5 Recouperator

The recouperator model is capable of providing outlet temperatures under both co- and counter current flows. This allows some process upsets during shutdowns to be evaluated during which the flow in one of the paths may reverse . It is set up by means of a heat transfer coefficient and a transfer surface area for a 20°C long mean temperature difference (LMTD).

3.2.6 Active cooling equipment

This section comprises both the pre- and intercoolers which provide a stable inlet temperature for the compressors. The models used to calculate the cooler's helium outlet temperature are identical to those used in designing the components. They include flow dynamics such as dynamic viscosity and variations in density and transfer coefficients along the length of the exchange unit.
Since process upsets causing flow reversal in the relevant PCU regions are unlikely (the compressor performance maps do not cater for reverse flow), the coolers are also only valid under helium forward flow conditions.

3.2.7 Helium Inventory System

The inventory system is modelled in detail. A system of four(4) pressure vessels together with a set of valves for controlling inventory flow makes up the system. In addition to the normal hydraulic case passive heat loss from the pressure vessels is included. Adiabatic effects both in the vessels and the flow to/from the PCU are not taken into account. This can be tolerated since only a relatively small adiabatic temperature shift results with a ratio of inventory- to process flow in the region of 1/50. For a maximum of 2 kg s^{-1} inventory flow and a 100 kg s^{-1} process flow , the impact of not calculating adiabatic effects is negligible. For further information on the helium inventory system consult [1].

3.3 Passive Heat Removal system

He passive heat removal system is modelled according to the civil structure as a set of layers through which the thermal energy is lost/removed. Layers are the graphite barrel (split in three(3) parts), the pressure vessel, RCCS, pit cooling and concrete structures. Helium leak flow between the barrel and reactor vessel is also included. For detail on the calculation performed in the model, please reference [2].

3.4 Control Philosophies and systems

The control philosophy for this type of nuclear plant can be quite different from those employed in PWR type reactors due to some of the inherent characteristics of the core and PCU. For this reason the philosophies and systems shown here are in a very early stage of development and analysis and should be viewed in that light. Most likely they will change as better insight is gained into the function of the system .

The control philosophy at this point is that all circuits are given a setpoint which is derived from the required electrical power on the alternator [4][11].
The inlet guide vanes on the power turbine are set to provide the correct mechanical power and torque for the alternator. This guide vane setting then has a direct influence on the inventory control system and an indirect effect on the core control rod position controller.
The rod control system keeps the temperature of the helium flowing out of the core as close as possible to a specified setpoint. For a graphical representation see the figure below:

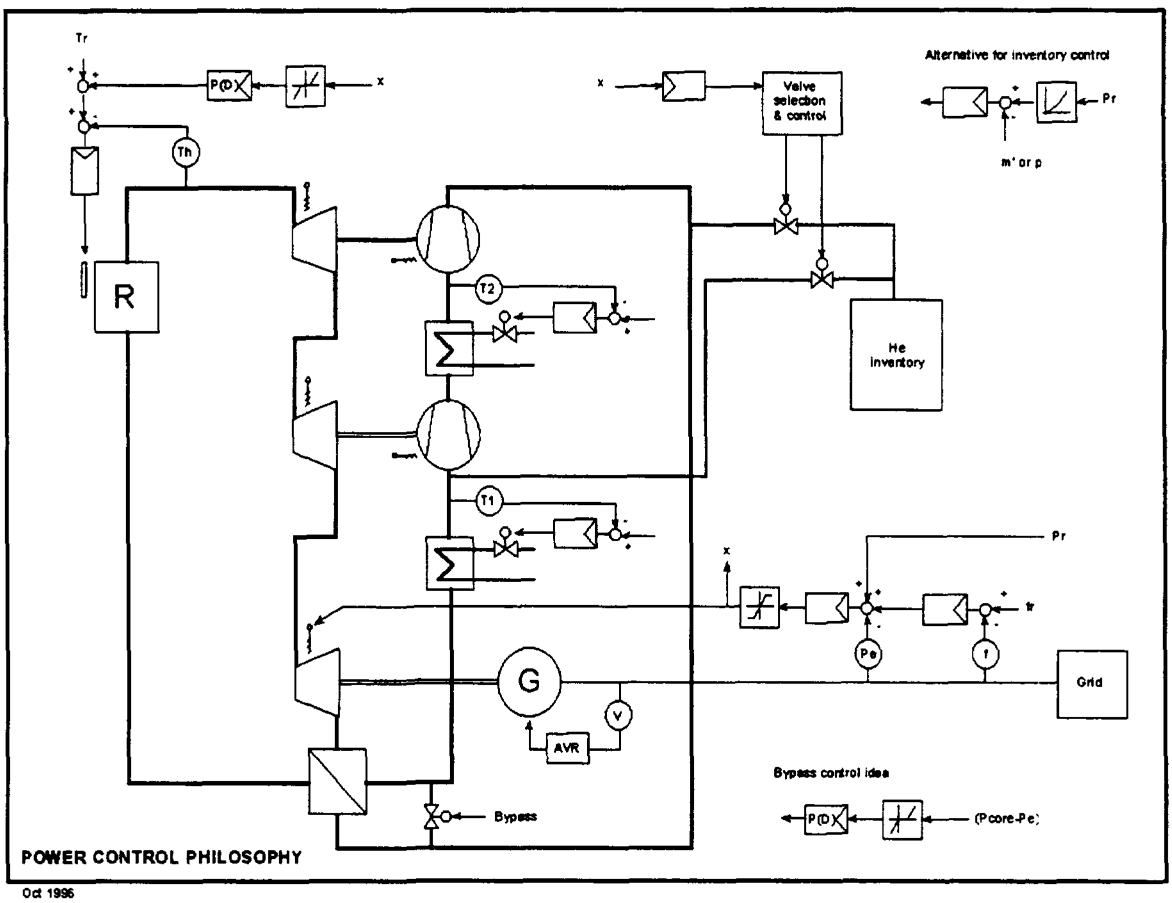

FIG. 2

The individual control loops employ standard PID rate control algorithms, of which mostly the P and I actions are used. The power control structure can also be seen in FIG. 2. The function of these controllers is illustrated in Chapter 4(Results).

4. Results

In this chapter the results obtained for the main objectives for the development of this engineering simulator as stated in Chapter 3 are portrayed.

Two scenarios are illustrated :

- Controlled ramp from 100 - 20% electrical power
- Controlled ramp from 20 - 100% electrical power

4.1 Controlled ramp from 100 - 20% power at 10% inventory per minute

The following graphs were produced by inducing a ramp command on the setpoint for the alternator power. This then filtered down to all other control loops.

FIG. 3 below depicts the reaction of the alternator to the ramp command. It was expected that the active power of the alternator would decrease from 100 % to 20% of nominal power after starting with a full power system at 100MWe.

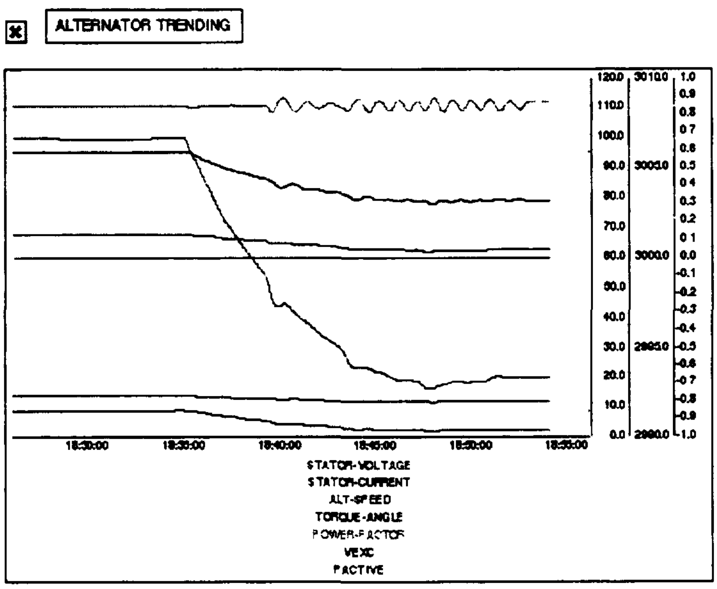

FIG. 3

The graph for the active power can be distinguished by its large variation, i.e.100 down to 20 %. The other trends show the internal variables of the model for interests sake.
As expected, the active power decreased to 20% and stabilised at that point. From this we can see that the control system managed to bring the plant to the desired state within the time of 10% per minute ramp boundaries set for it. The kinks in the active power curve arise from the helium inventory system switching between pressure vessels.

Furthermore, after discussions with the core experts, it was agreed upon that both the neutronic power and the mass flow through the system should decrease by an amount exactly proportional to the desired change in active power. The following figure (FIG. 4) shows the reaction of reactor core parameters such as reactivity, normalised power and temperatures to the same changes:

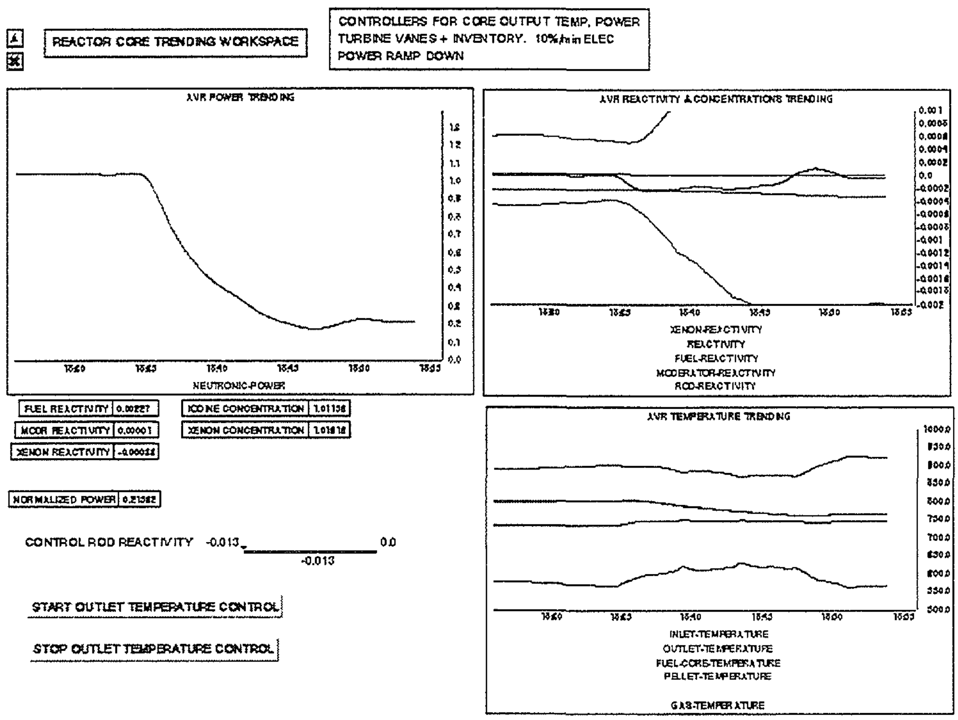

FIG. 4

The top left-hand graph of FIG. 4 depicts the normalise neutronic power, top right the fuel, moderator , rod, xenon and total reactivity. The bottom right graph shows the temperatures of the gas and fuel elements in the core. As can be seen, the neutronic power reduced and stabilised at the expected level. As for reactivity and temperatures , both behaved approximately the same as other simulations performed at KFA, Juelich [3]. The in- and outlet temperatures of the reactor are the lowest and highest trend respectively on the graph at the bottom right of FIG. 4. The mass flow change can be seen to behave correctly in FIG. 5 on the next page.

The two graphs above show mass flow and pressures at the component inlets.
Again both flows and pressures reduced to the desired levels and stabilised.

As the power controller received a ramp reduction command and the vanes on the power turbine responded, the inventory system simultaneously began removing helium from the system into the pressure vessels. This caused a drop in overall system pressure (FIG. 5), reduced cooling in the core (FIG. 4) and reduced mass flow (FIG. 5). Due to reduced cooling and increased temperatures in the core, the neutronic power was lowered by the same amount (FIG. 4). The control rods reacted to keep the reactor outlet temperature as close as possible to the nominal outlet temperature (FIG. 4).

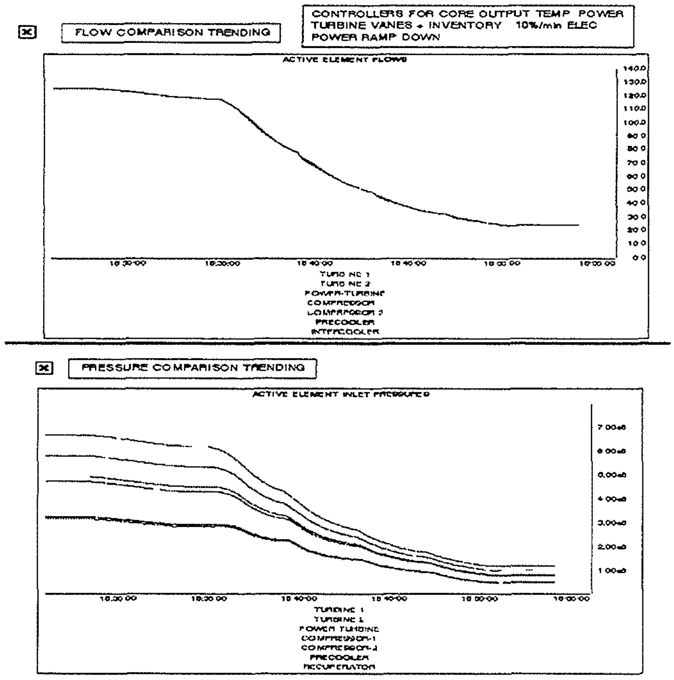

FIG. 5

4.2 Controlled ramp from 20 - 100% power at 10% inventory per minute

The power ramp command to increase power from 20 to 100% was given from the 20% level described in Chapter 4.1 . The expected result of such a command would be that the plant would return to its original state at 100% power (100MWe).

The graphs and trends below show that both the alternator, reactor and other systems return to 100%.

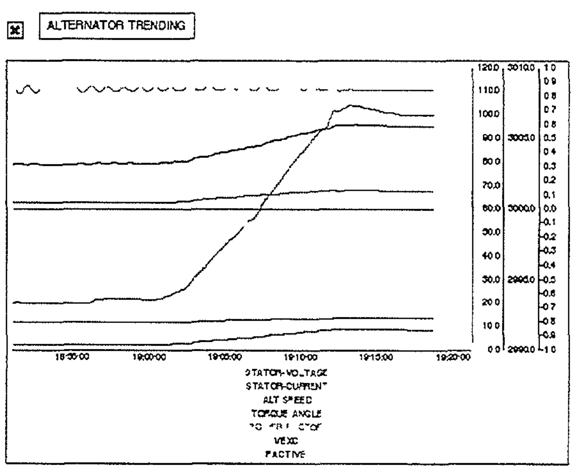

FIG. 6

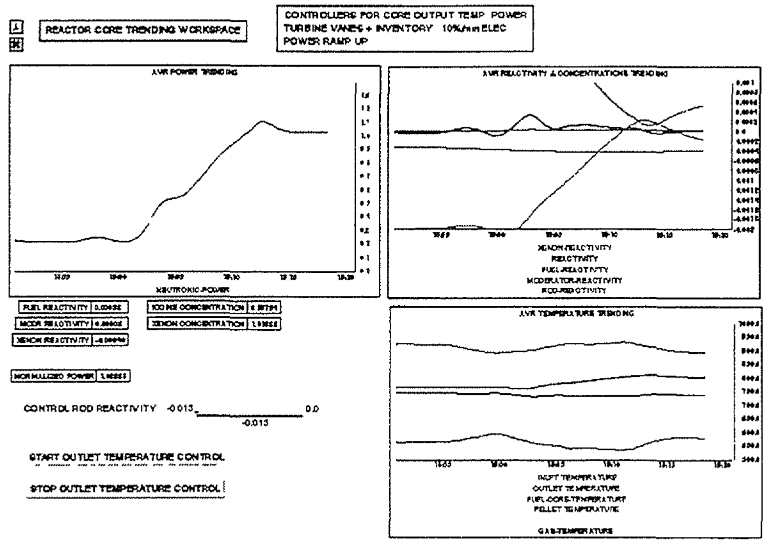

FIG. 7

Once gain we see the points on the flow and pressure curves where the helium inventory system switched tanks.

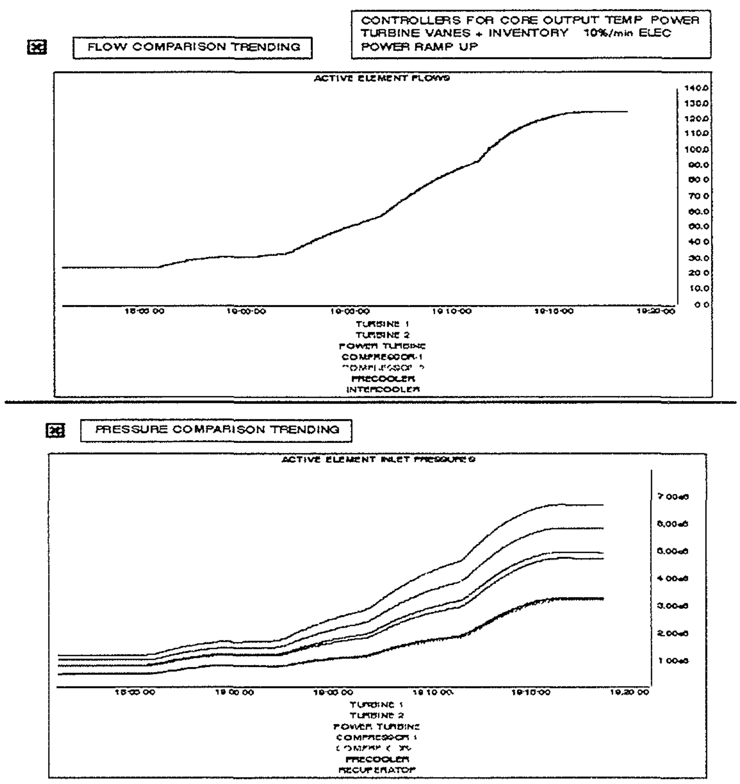

FIG. 8

The above graphs show clearly that the system stabilised at its original power levels.
The action of the control loops and philosophy aids to stabilise the response to the ramp signal and bring all subsystems back to their initial states with little or no overshoot in the required time i.e. at the required ramp rate.

5. Conclusion and Validity of Results

The discussion on both the extent and scope of simulation should show that the engineering simulator is still in an early phase of development. Some of the models may not contain one or the other function, but all omissions are well founded either by consultation with the experts or by a separate study showing the effects to be small to negligible.

The reason for the development of such an engineering simulator was not to retest and evaluate the individual designs, which at least for the core and active cooling elements are being simulated and evaluated in much more detail by the relevant experts. For the purpose of analysing the system as a whole and the interactions between the components, the assumptions and simplifications made still give sufficiently accurate results. Except in accident- and extreme process conditions, the simulator performs the specified transients and has provided valuable insight and information to system- , control- and mechanical engineers alike.

The simulator has proven itself to be capable of performing operational power transients and has shown the reactor/PCU system in the current configuration to be stable and controllable at all power levels as can be seen from the results in Chapter 4. This has boosted confidence in the design. Having revealed much already, the simulator should be of great assistance in evaluating design changes and control philosophies during the next phase of the project.

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STUDIES ON SELF-ACTING HEAT REMOVAL SYSTEMS ON THE BASIS OF PARTIAL BOILING OF WATER FOR STEEL PRESSURE VESSEL COOLING OF MODULAR HTRs

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Abstract

A review is given on the technology of reactor pressure vessel cooling in HTR conceptual designs, plants in construction and existing plants of the HTR Additionally an overview is given on the results of studies on heat removal systems for the cooling of the steel pressure vessel and the reactor cavity for modular HTR designs, as well as for the cooling of hot gas ducts, e g in the project „AVR reconstruction“
For the recent conceptual design and layout work on the Pebble Bed Modular Reactor with a direct Gas Turbine Cycle, PBMR-GT, a self-acting heat removal system on the basis of partial boiling liquids is proposed as a back-up system in the case of the loss of forced cooling of the pressure boundary of the primary circuit in normal operation Analyses for the heat removal values were carried out for different parameters Thermal radiation depends upon the emmisivity of the surfaces of the reactor pressure vessel and of the cooling pannels, and of the arrangement of these two surfaces to each other The influence of the change of the emmisivity on the basis of the surface conditions and temperatures has been evaluated Moreover the effects of heat convection and conduction have been calculated and their influence in the total heat removal has been analysed Results on the evaluation of natural convection of the heat transfer medium, the liquid, preferably water, are reported with variation of the different conditions of evaporation and condensation Moreover the arrangement of the heat transfer loops, in open and in closed form, e g the raiser pannels, down comer tubes, and condensation tubes, have been varied to get the maximum buoyancy for the flow of the heat transfer medium

The arrangement of the raiser pannels has been done in such a way that these can be readjusted from their normal position in the reactor cavitiy, so that sufficient place is available for the normal inspection of the reactor pressure vessel Outlet ducts are also arranged in a flexible structure, so that these can be also removed from their normal place The concrete temperature of the reactor cavity is kept under the value of about 65 °C, by insulating all the structures in the cavity, which are not required for the heat removal In conclusion It is possible to realize a self-acting heat removal system of the base of partial boiling of water for a steel pressure vessel cooling with a heat flux density of about 5 kW/m^2 and thereby with a total heat flux of about 800 kW The proposed system is a back-up system starting its function only in case of the loss of the normal operation forced cooling, which is provided by the gas turbine, and in case of reactor shut-down by an auxiliary circulator cooling loop

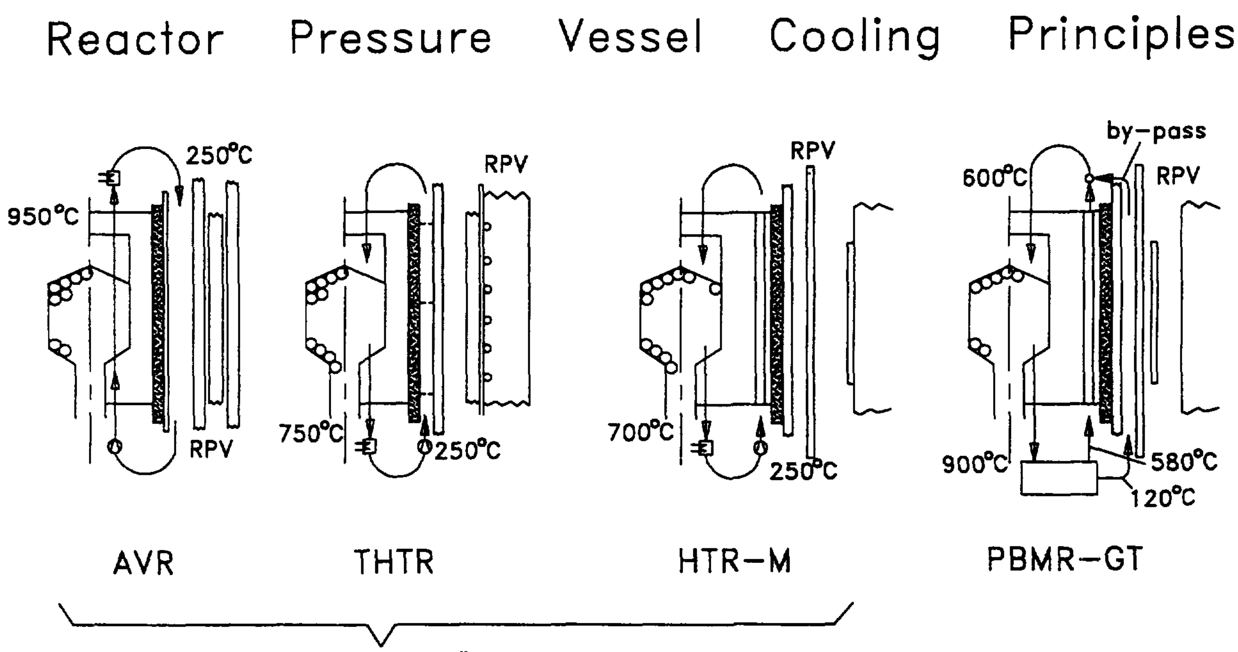
Introduction: Reactor Pressure Vessel Cooling Principles

In the technology of high temperature reactors (HTR), as it has been developed up to now, the concept of the cooling of the reactor pressure vessel follows a general principal, which can be summarized as follows: main-pass cold helium „envelopes“ high temperature ceramic components and „stabilizes“ low temperature metallic components. Only recently a new principle has been introduced into the discussion which, in general, separates the two functions 1) heat transfer from the core to the consuming apparatus and 2) enveloping and stabilizing components of the core, [1], [2]. This separation of functions introduces as new parameter of freedom with respect to the application of high temperature heat for the conversion of that heat into secondary energy carriers by increasing the thermodynamical value of the heat vector. This is explained in the following in some more details.

The concept of the reactor pressure vessel cooling, as it has been developed upto now, is explained in the following for the pebble bed HTR with the existing reactor AVR and THTR and the conceptual design HTR-M (Module), fig. 1. The pebble bed HTR is used as an example only, for HTRs with other types of fuel elements the same principle has been used, for overview see [3]. The principle is:

Main-pass cold helium „envelopes“ high temperature ceramic components, and „stabilizes“ low temperature metallic components.

In AVR, fig. 1, left hand side, the main-pass cold helium has a temperature of 250°C and flows downwards in direct contact with the reactor pressure vessel



main – pass cold He "envelopes" high T ceramic components and "stabilizes" low T metallic components

by-pass cold He RPV cooling

Fig.1:

(RPV) arrangement. The components from inside to outside are: pebble bed, graphite side reflector, carbon brick thermal insulation, core barrel, main-pass cold helium, inner steel pressure vessel, biological shielding, outer steel reactor pressure vessel.

At THTR the main-pass cold helium likewise has a temperature of 250°C. It flows upwards in direct contact with the cast iron thermal shield. The components from inside to outside are: pebble bed, graphite side reflector, carbon brick thermal insulation, metallic support structures in the main-pass cold helium, cast iron thermal shield, reactor cavity with stagnant helium, liner insulation, liner, liner cooling, presstressed concrete reactor pressure vessel.

In the HTR-M the main pass cold helium likewise has a temperature of 250 °C but here the main-pass cold helium, flowing upwards, has direct contact with the colder parts of the side reflector structure. The components from inside to outside are: pebble bed, graphite side reflector structure, carbon brick thermal insulation, core barrel, stagnant helium space, steel reactor pressure vessel, reactor cavity, cavity cooler, concrete cavity structure.

In all these three cases the main-pass cold helium „envelopes" high temperature ceramic components, and „stabilizes" low temperature metallic components. The latter one are in particular metallic components supporting the core structure, as explained above, but are as well e.g. the metallic components for control and shutdown of the reactor. The recently proposed principle for reactor pressure vessel cooling introduces in general an independent cooling for the cooling purposes, e.g. a stream of

„by-pass cold helium for RPV cooling",

as well as other low temperature metallic components, [1]. This is explained in some more detail in fig. 1, right hand side as follows:

For the recent conceptual design and layout work on the Pebble Bed Modular Reactor with a direct cycle Gas Turbine, PBMR-GT, it is proposed to use the heat vector with temperatures of roughly 600 °C to 900 °C in the gas turbine and to use a stream of „by-pass cold helium" with a temperature about 120 °C for the cooling needs. The components from the inside to the outside are: pebble bed, graphite side reflector (with „cold" helium of 600 °C form the gas turbine), carbon brick thermal insulation, core barrel, by-pass cold helium with a temperature of 120 °C, reactor pressure vessel, cavity cooling system, concrete structure of cavity.

The advantage of this proposal is the possibility to increase the efficiency of the conversion process „HTR heat into electricity" considerably, [1]. In the second example, [2], for this proposal the advantage is the better applicability of the high temperature heat vector for process heat applications, e.g. refinement of coal. The disadvantages of the new proposal are 1) not proven and 2) design issue in its own.

Modularization of the core and self-acting heat removal

All modern concepts of HTRs follow the general priniple of the modularization of the core as a general priniple to produce safety by design. The principle is a result of the fundamental problem to improve reactor technology after the accident of Three-Mile-Island by avoiding the possibility of a core-melt. The solution is the HTR-module, [4], with its design principle to allow the afterheat to be transferred out of the core into outside structures, including cooling systems. This general principle can be improved, e.g. by the proposal of a self-acting heat removal system in the reactor cavity, as proposed in many conceptual designs.

For the PBMR-GT, [1], the normal operation by-pass cooling is provided by the gas turbine and therefore additionally a forced decay heat auxiliary removal system has been proposed, so that the temperature of the reactor pressure vessel as well as of the core-barrel can be kept say under 200 °C. This cold helium gasflow flows upwards between the RPV and core-barrel and afterwards it flows downwards through the holes for the reflector rods and also cool these, before it flows to the hot gas chambre at the bottom. Subsequently this helium flows further to the other components of this auxiliary decay heat removal system, such as cooler and circulator. With this system also the core temperature is kept near normal operational temperatures, because the heat dissipates to the side reflector or to the graphite noses, where the forced cooling is effective. By this way no forced cooling through the core is necessary, whereby a minimum helium flow against the thermal buoyancy in the core is required.

In the case of the loss of forced cooling of the pressure boundary of the primary circuit, a selfacting heat removal system is proposed as a back up system, which starts its function only after such failures. This system can be based on the natural convection of only hot water or of partial boiled water. Analyses for the heat removal values were carried out for different parameters and are described in the following.

Heat Transfer between Reactor Pressure Vessel (RPV) and Cooling Pannels (CPA)

Heat can be removed from the reactor pressure vessel by higher temperatures mainly through the thermal radiations. If it is possible to put the heat transfer cooling pannels very near to the RPV, also thermal conductivity can play an important role. Moreover natural convection of the air in the reactor cavitiy also transport heat from the RPV to the CPA.

Thermal radiations depend upon the emissivity of the surfaces of the RPV and of the CPA, and of the arrangement of these two surfaces to each other. Generally the emissivity ε changes with the condition of the surfaces, such as the grade of oxidation and with the temperature of the surfaces. Moreover the arrangement of the both heat transfer surfaces can play a dominant role if these are not parallel or the distance between both surfaces is large or these have different heigts.

Analyses have been done for the variations of emissivity ε_{12} between 0.7 to 0.9, geometryfactor φ_{12} between 0.98 to 1.0, the temperature of the RPV T_1 betweem 250 °C to 450 °C and the temperature of the CPA T_2 between 0 °C to 200 °C. The results of this study are shown in fig. 2. Thereby it can be shown that only with thermal radiations under these conditions heat flux density of about 11 kW/m^2 can be achieved, although it requires high emissivity of about 0.9 of the surfaces.

The emissivity ε_{12} of the rough steel surfaces is given near about 0.79, if the radiations fall perpendicular to each other [5]. However, under normal conditions this value reduces to near about 0.762. Therefore this value is also selected for further analyses.

Cooling water pipes are welded separately with fin plates and thereby these build a concentric wall to the RPV, with relative small distance to each other. This allows to take the geometric factor value near about 0.99. The arrangement of the heat transfer raiser pannels has been done in such a way that these can be readjusted from their normal position in the reactor cavitiy, so that sufficient place is available for the normal inspection of the reactor pressure vessel.

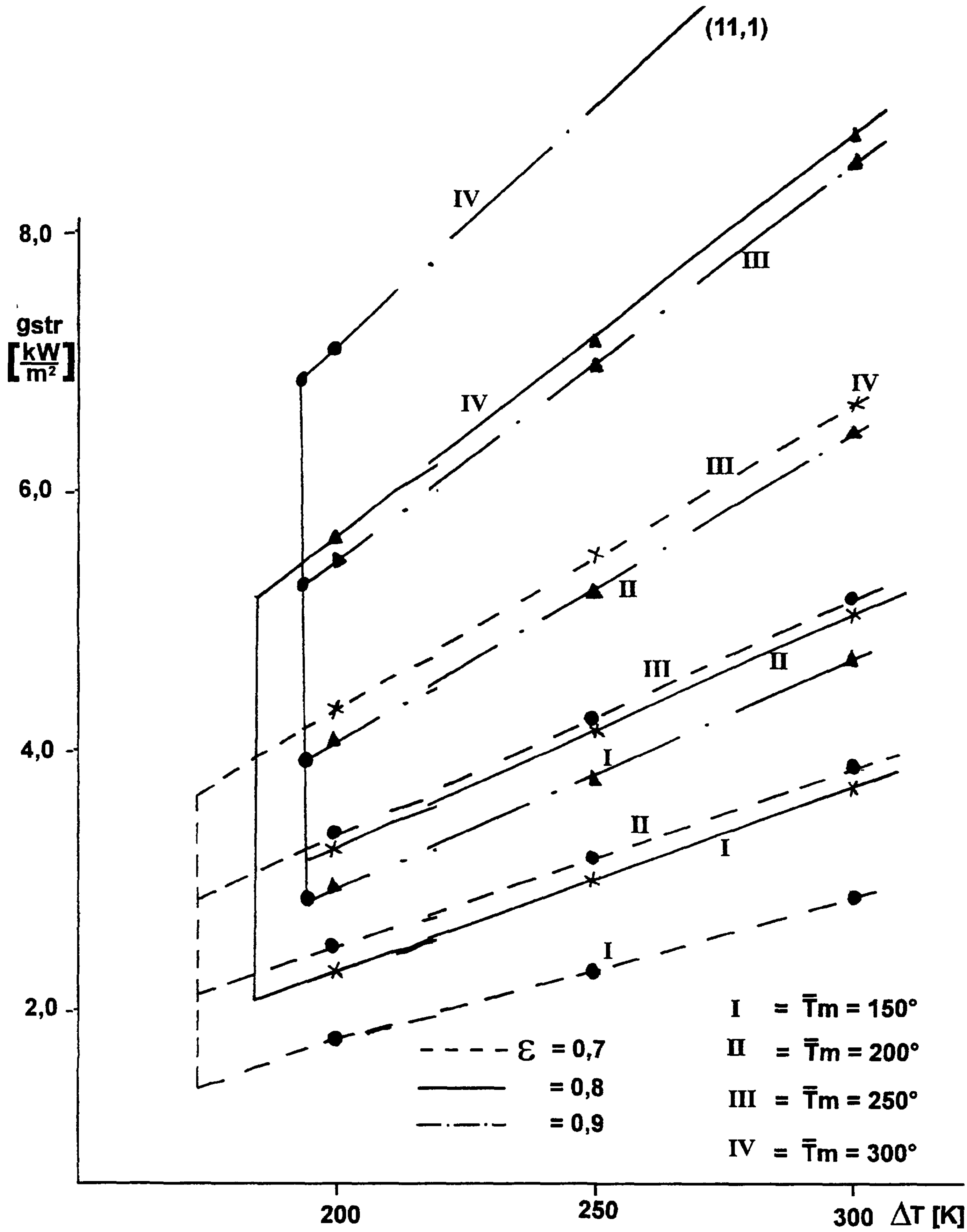


Fig. 2: Pressurevessel Temp.. $T_1 = T_m + \frac{\Delta T}{2}$

pannel Temperature $T_2 = T_m - \frac{\Delta T}{2}$

Table 1: Heat Removal Values under the Assumed Calculating Conditions

Heat Transfer Conditions between Reactor Pressure Vessel (RPV) and Cooling Panels (CPA)	Water Cooling System	Partial Boiling System
Emissivity ε_{12} [-]	0,762	0,762
Geometry factor between RPV and CPA φ_{12} [-]	0,99	0,99
Mean RPV-Temperature [°C]	350	350
Mean CPA-Temperature [°C]	110	110
Heat Transfer Coefficient, Natural Convection $\alpha[\frac{W}{m^2 \cdot K}]$	5,0	5,0
Heat Flux Density $q[\frac{kW}{m^2}]$	5,5	5,5
Mean System pressure p [bar]	5,5	1,5
Water Massflow [kg/s]	12,0	27,0
Steam Massflow [kg/s]	-	~ 0,35
Geometrie of Cooling Panels:		
Downcomer Panels:		
Total Number N_1 [-]	6	6
Total Height H_1 [m]	12,0	11,0
Tube Diameter P_1 [mm]	80,0	100,0
Heat Transfer Panels:		
Total Number N_2 [-]	108	108
Total Height H_2 [m]	9,5	9,5
Tube Diameter D_2 [mm]	50,0	50,0
Collecting Panels:		
Total Number N_3 [-]	6	6
Total Height H_3 [m]	2,5	1,5
Tube Diameter D_3 [mm]	80,0	100,0
Heat Removal Values [kW]	≥800	≥800

The arrangement of the heat transfer pannels can be done in such a way, that these are very close to the RPV, although a small air column prevails inbetween. Thereby heat flux density due to heat conduction of near about 3 kW/m^2 can be achieved. For the present study however this has been neglected.

Natural convection of the air in the reactor cavity under the prevailing temperatures of RPV and CPA has been estimated. The assumed value for the heat transfer coefficient has been selected near to the value of 5.0 W/m^2K.

The results of the analyses show that about 800 kW can be at least transferred from the RPV to the CPA and this heat amount can be further transported to the atmosphere through the cooling medium in the heat transfer raiser pannels. These results are also given in table 1.

Heat Transport from Reactor Cavity to the Atmosphere

In the HTR-Modul three redundant forced water cooling circuits are arranged in the reactor cavitiy to transport the dissipated heat from the RPV to the upper part of the reactor building, and is further given to the secondary cooling water circuit [6].

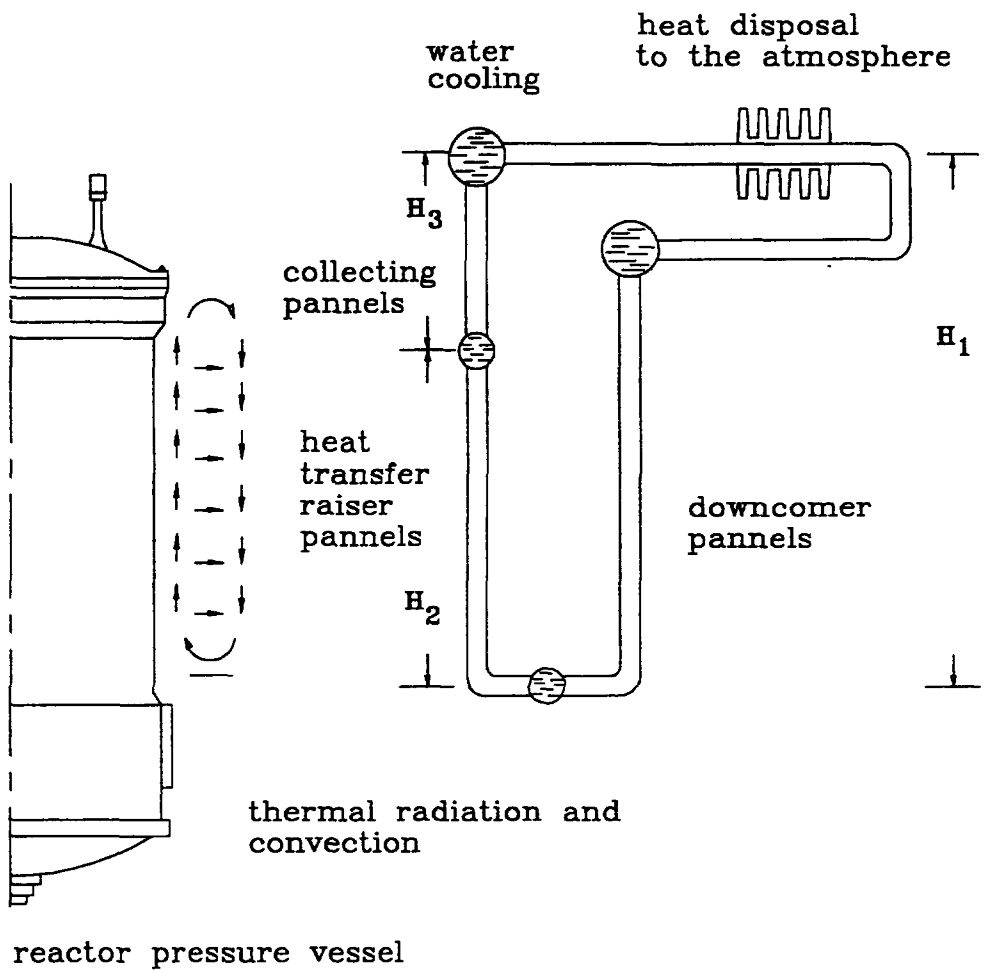


Fig.3: Self-acting Heat Removal System
Water Cooling System

Similar vessel cooling system is also constructed at the site of HTTR-plant [7]. As it represents an engineered safety device, a dual system is used with two circulation pumps.

The present proposal is based on natural circulating of water and therefore it is a self acting system. However, two different systems are investigated, whereas in fig. 3 only water cooling system is shown, on the other hand in fig. 4 the system is based on the potential boiling of water and subsequently its condensation and thereby heat is transported to the atmosphere.

From the water drum outside the reactor building downcomer pannels transport cold water to the bottom water collector inside the reactor cavity. Heat transfer raiser pannels are welded to this bottom water collector and are arranged around the reactor pressure vessel at equal distance to the RPV as well as between each other.

Heat is transported to these raiser pannels, as already described, whereby the water is only further heated or it gets also partially boiled. Hot water or the mixture of hot

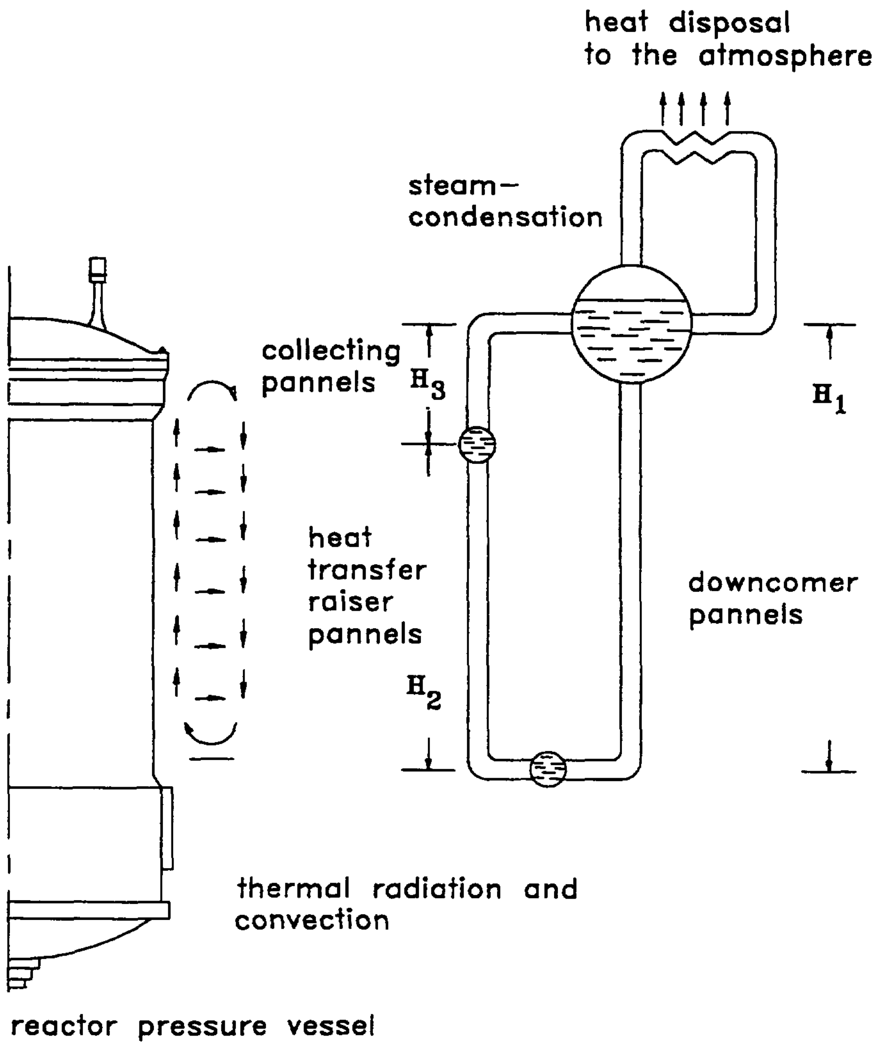


Fig.4: Self-acting Heat Removal System
Partial Boiling of Water

water with steam flows upwards in these raiser pannels and subsequently it is collected in the upper water collector. From here this hot water is further transported through the collecting pannels to the upper water drum. This drum can also contain some steam, if the second system is followed.

From this drum several water tubes with ribbs are arranged around the reactor building to dissipate the heat to the atmosphere. Also cooling tower can be arranged in this system. Safety analyses are still required, before one or the other system is selected, because e.g. the tubes need more place for their arrangement, but on the other hand the chances of their damage due to external events is lower. Cold water flows back to the drum, before it further flows to the reactor cavitiy.

By the water cooling system, as shown in fig. 3, system pressure at the top has been chosen to the value of 5 bars, so that no water boiling in the whole circuit can take place. Further the height difference between the upper water drum and the bottom water collector is selected to the value of 12 m to get sufficient buoyancy for the natural circulation of water in this closed loop.

Pannels diameters in the whole circuit and their arrangement is chosen in such a way, so that low pressure drop through water flow on one side, but at the same time enough heat transfer coefficient in the raiser tubes is possible. At present for this system cold water temperature of 100 °C and temperature increase in the raiser pannels to the value fo 20 °K is selected, so that sufficient heat from the RPV can be removed and at the same time this heat can be dissipated without any cooling tower, only with the arrangement of 114 cooling tubes with ribbs, to the atmosphere with natural air convection. The length of each tube is near about 12 m. Through thermal buoaancy in this circuit about 12 kg/s water flow through it. Results are shown in table 1.

Partial boiling of water in the circuit increases the flexibility of the layout of the second system, as shown in fig. 4, because with the change of phase of water in steam heat transfer coefficient and also at the same time thermal buoyancy in raiser pannels increases. Moreover condensation of this steam in the atmosphere requires also smaller heat transfer surface area in comparison to the cooler tubes in the first system.

For this system, as shown in fig. 4, a system pressure of 1 bar and the water temperature of 100 °C in the upper water drum is chosen. This drum is arranged at the height of about 11 m in comparison to the ground level, where the bottom water collector is placed. Due to this arrangement, the system pressure at this level is near about 2 bars. In the raiser panels, where the water flows upwards, the water temperature from 100 °C has to be raised to the value of saturation temperatur of water in accordance to the prevailing pressure at the level. Calculations have shown that at the height of about 8 m in the raiser panels, the under cooling of the water is completed and at this stage the upward flow of water starts to get partialy boiled. However only very small amount of steam is produced, because the heat transfer area of the raiser panels is reletively small. Moreover in this region, the boundary conditions are changing very fast, because of low heat flux density on one hand and the change of static pressure with the change of height on the other hand. Further the water bubbles have higher velocity than the average water velocity in the raiser panels. These effects are shown in fig. 5 [8], but the minimum system pressure in this diagram is 20 bar. The pressure drop in these panels depends on the area of the water bubbles, and their relative velocity. Correlations for the system pressure near about 1 bar and for small amount of steam flow is not exactly given in this literature and also others. Moreover also thermal buoyancy conditions are changing with the height of the closed circuit. Estimations however have given that about 0.35 kg/s steam is produced at the inlet of the upper drum, which is situated outside of the reactor

Lit.: LEDINEGG - 1966 : Dampferzeugung , Naturumlaufkessel , S. 259

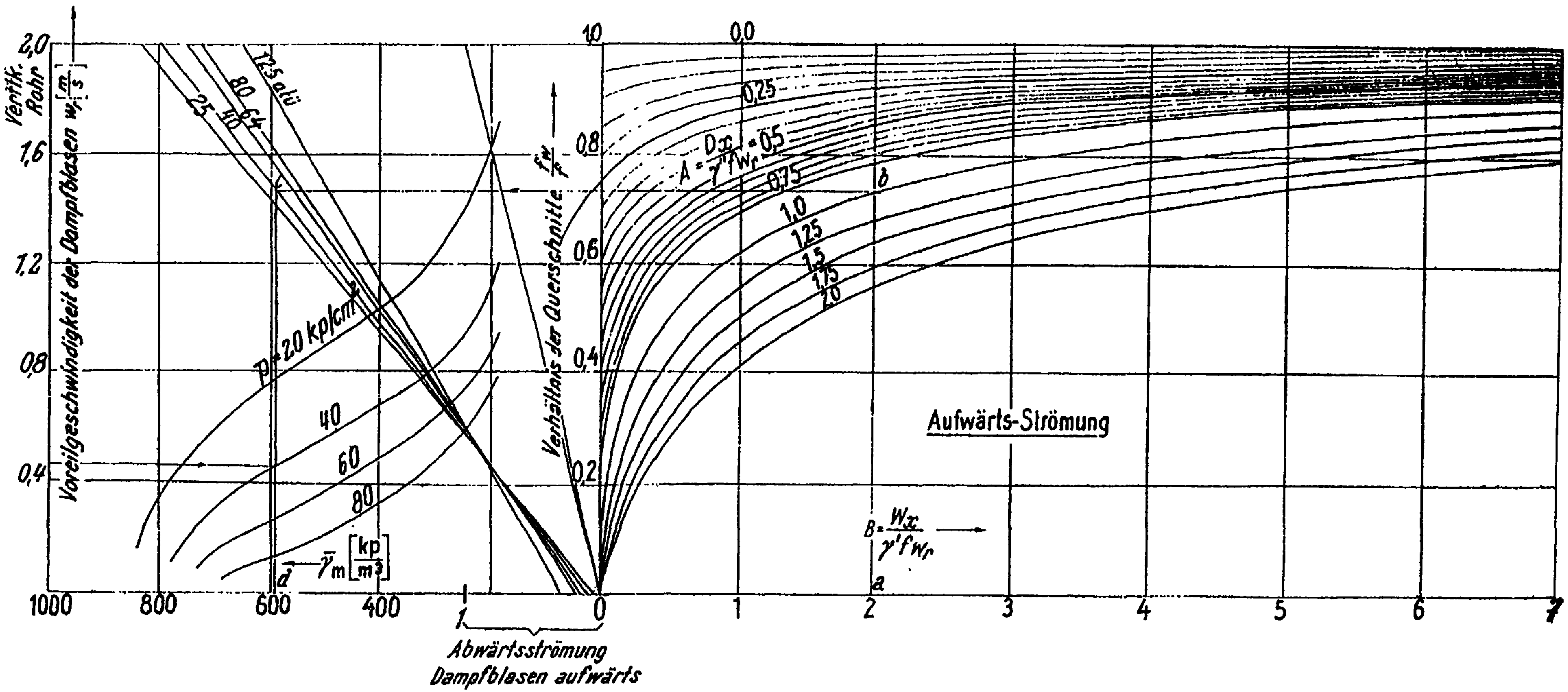


Fig. 5:

Abb. 191a. Ermittlung der Wichte eines Dampfwassergemisches bei Aufwärtsströmung von Dampf und Wasser (Steigströmung, Ausgangsmaßstab rechts vom Nullpunkt) und bei Aufwärtsströmung des Dampfes und Abwärtsströmung des Wassers (Ausgangsmaßstab 0 - 1 links vom Nullpunkt). Der Linienzug $a\ b\ c\ d$ und der von w_r = 0,46m/s über p = 40 kp/cm² nach d führende müssen bei richtiger Annahme von w_r zum gleichen Punkt führen. Gl. (418), (419) building. This steam can be condensed in the condensor tubes, which are arranged near to this drum, whereby heat is dissipated to the atmosphere. The results are shown in tab. 1. The amount of heat removed from the RPV depends on its surface temperature. For the present estimations it is higher than about 800 kW.

Both of these sysems can be combined, so that at lower temperature of the reactor pressure vessel, only natural water circulation takes place and at higher temperatures partial water boiling is possible. However for this system the cooling tower at outside of the reactor building may become necessary to dissipate enough heat to the atmosphere. These studies are further going on.

Conclusion

The principle of reactor pressure vessel cooling allows an improvement by the recently proposed by-pass cooling with cold helium from the gas turbine.

It can be shown that the self acting heat removal system for steel pressure vessel cooling of modular HTRs is possible. Partial boiling of water can be established with a particular system arrangement of the closed circuit.

The concrete temperature of the reactor cavitiy is kept under the value of about 65 °C, by insulating all the structures in the cavitiy, which are not required for the heat removal.

The proposed system is a back-up system starting its function only in case of the loss of the normal operation foreced cooling, which is provided by gas turbine and in case of reactor shut-down by an auxiliary circulator cooling loop. This system has the capacity to remove heat above the value of about 800 kW.

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MATERIAL AND FABRICATION OF THE HTTR REACTOR PRESSURE VESSEL

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Abstract

The High Temperature Engineering Test Reactor (HTTR) is under construction at Oarai Research Establishment, Japan Atomic Energy Research Institute (JAERI) and planned to be critical in October 1997. Fabrication of the HTTR reactor pressure vessel (RPV) at Kure Works, Babcock-Hitachi K. K. took about two years, and the RPV was transported to the Oarai site in August 1994. Pressure test of the primary and secondary cooling system including the RPV was performed successfully in March 1996 [1].

Because temperature of the HTTR RPV becomes about 400 °C at normal operation, 2 1/4 Cr-1 Mo steel is chosen for it. Fluence of the RPV is calculated to be less than 1×10^{17} n/cm^2 (E>1 MeV), and so irradiation embrittlement is presumed to be negligible, but temper embrittlement is not. For the purpose of reducing embrittlement, content of some elements is limited on 2 1/4 Cr-1 Mo steel for the HTTR RPV using embrittlement parameters: J-factor and $\overline{X}$.

In this paper design and structure of the HTTR RPV is briefly reviewed first. Fabrication procedure of the RPV and its special feature is shown. Material data on 2 1/4 Cr-1 Mo steel manufactured for the RPV, especially the embrittlement parameters J-factor and $\overline{X}$, and nil-ductility transition temperatures T_{NDT} by drop weight tests, are shown, and increase in the transition temperature is estimated based on data available in literature. Technology of the HTTR RPV is applicable to RPVs of future commercial High Temperature Gas-cooled Reactors (HTGRs).

1. Design and structure

Table 1 and Fig. 1 show specifications and schematic diagram of the HTTR RPV respectively. The RPV consists of a RPV top head, which includes thirty-one stand-pipes, a top head dome, a top head flange, thermal shields, etc., and a RPV body containing a shell flange, a shell, three stand-pipes, a bottom head petal, a skirt, a bottom head dome, a support ring, radial keys, etc. The RPV top head is bolted to the RPV body by 72 stud bolts. The thirty-one stand-pipes, which include sixteen for control rods, five for irradiation test, three for surveillance test, three for neutron detection, two for in-service inspection of reactor internals, and two for measurement of temperatures and core differential pressure are welded to nozzles on the top head dome. Figure 2 shows arrangement of stand-pipes on the top head dome. The largest seven stand-pipes, N1 to N7, are also used for refueling. Three other stand-pipes for hot plenum temperature measurement and fuel failure detection are welded to nozzles on the shell. Figure 1 shows one of the three stand-pipes. The skirt supports weight

Table 1. Major specification of the HTTR reactor pressure vessel

Design pressure	4.7 MPa [gauge]
Design Temperature	440 °C
Normal operating pressure	3.9 MPa [gauge]
Inlet coolant temperature	395 °C
Inside diameter	5.5 m
Height	13.2 m
Thickness of cylindrical shell and bottom head dome	122 mm [minimum]
Thickness of top head dome	160 mm [minimum]
Number of stand-pipes	34
Material	2 1/4 Cr-1 Mo steel (Normalized and tempered)

and seismic load of the reactor. Horizontal seismic load is sustained by six stabilizers and a stand-pipe support beam as well as the skirt. The support ring support vertical load of the core, and the radial keys hold horizontal movement of the core through core restraint mechanism. The thermal shields are made up of layers of metallic plates, which protect the RPV top head from overheating, especially in loss of forced coolant circulation accidents.

Because temperature of the HTTR RPV becomes about 400 °C at normal operation, 2 1/4 Cr-1 Mo steel, normalized and tempered, which has higher creep rupture strength than Mn-Mo steel for RPVs of Light Water Reactors, is chosen for it. Three kinds of 2 1/4 Cr-1 Mo steel: forgings of Japan Industrial Standard (JIS) specification SFVAF22B, equivalent to ASTM A-336 Gr. F22, Cl. 3, plates of JIS SCMV4-2, equivalent to ASTM A-387 Gr. 22, Cl. 2, and seamless pipes of JIS STPA24, equivalent to ASTM A-335 Gr. P22 are used for components as shown in Fig. 3, which also shows weld lines.

The safety evaluation criteria for temperatures of the HTTR RPV are as follows: maximum temperature shall not exceed 500 °C in anticipated operational occurrences and 550 °C in accidents. The maximum temperature of the RPV in the depressurization accident is calculated to be about 530 °C [2].

2. Fabrication

Figure 4 shows fabrication procedure of the HTTR RPV. As shown in Figs. 3 and 4, two types of stand-pipes exist, designated "Stand-pipe a" and "Stand-pipe b" in Fig. 4: a forging, and a forging and a pipe welded together respectively. The inner nineteen stand-pipes for control rods and irradiation tests, which require dimensional accuracy in manufacturing, are forgings. Heads of the outer twelve stand-pipes on the top head dome and three stand-pipes on the shell are forgings, which are welded to seamless pipes, making up the "Stand-pipe b."

The top head dome is a very large forging, and nineteen nozzles for the inner nineteen stand-pipes above are integrated with itself, that is there are no weld lines between the dome and the nozzles. Since gaps among the nineteen nozzles are so narrow, it is impossible to perform in-service inspection: ultrasonic testing, of the weld lines when they exist. Ultrasonic testing of the weld lines between the stand-pipes and the nozzles is possible utilizing special equipment for it. The other thirteen nozzles including a manhole nozzle, are welded to the top head dome. Regarding these weld lines, it is possible to conduct ultrasonic testing, because there is enough distance between the nozzles, and the weld lines are accessible from

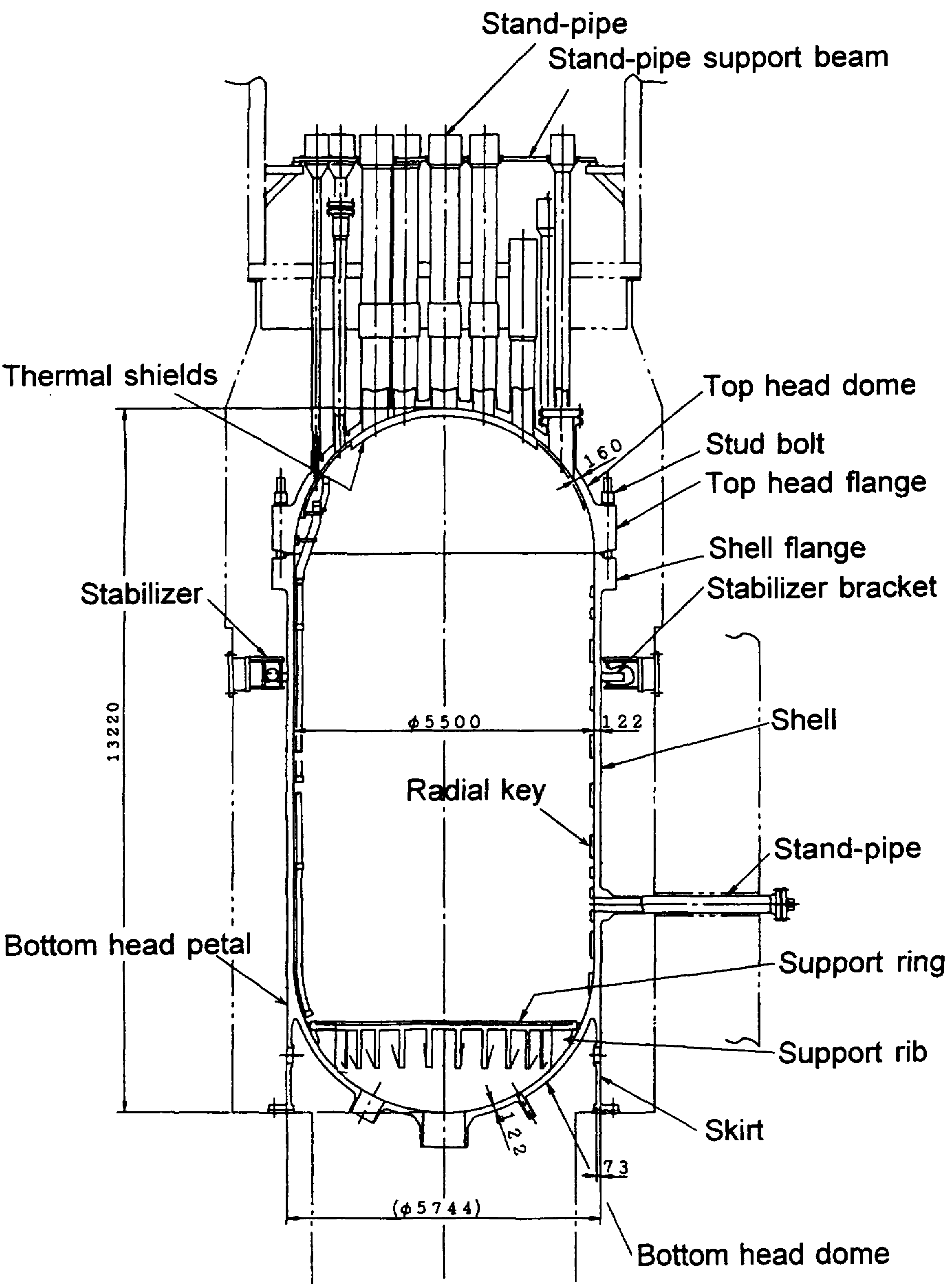
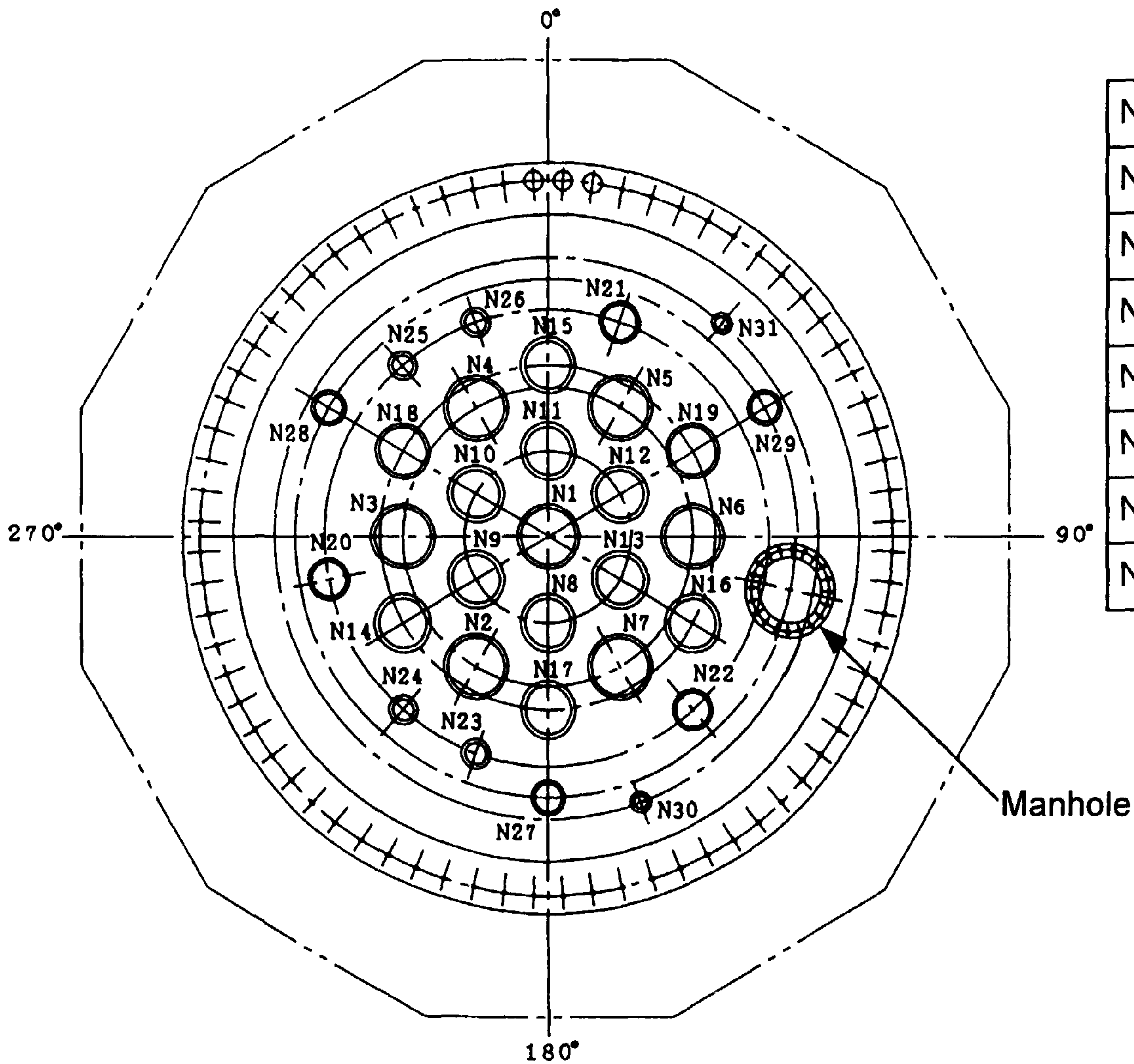


Fig. 1 Schematic diagram of the HTTR RPV



N30, N31 for in-service inspection
N27～N29 for surveillance test
N24, N26 for measurement
N23, N25 for irradiation test
N20～N22 for neutron detection
N17～N19 for irradiation test
N8～N16 for control rods
N1～N7 for control rods and refueling

Fig. 2 Arrangement of stand-pipes of the HTTR RPV

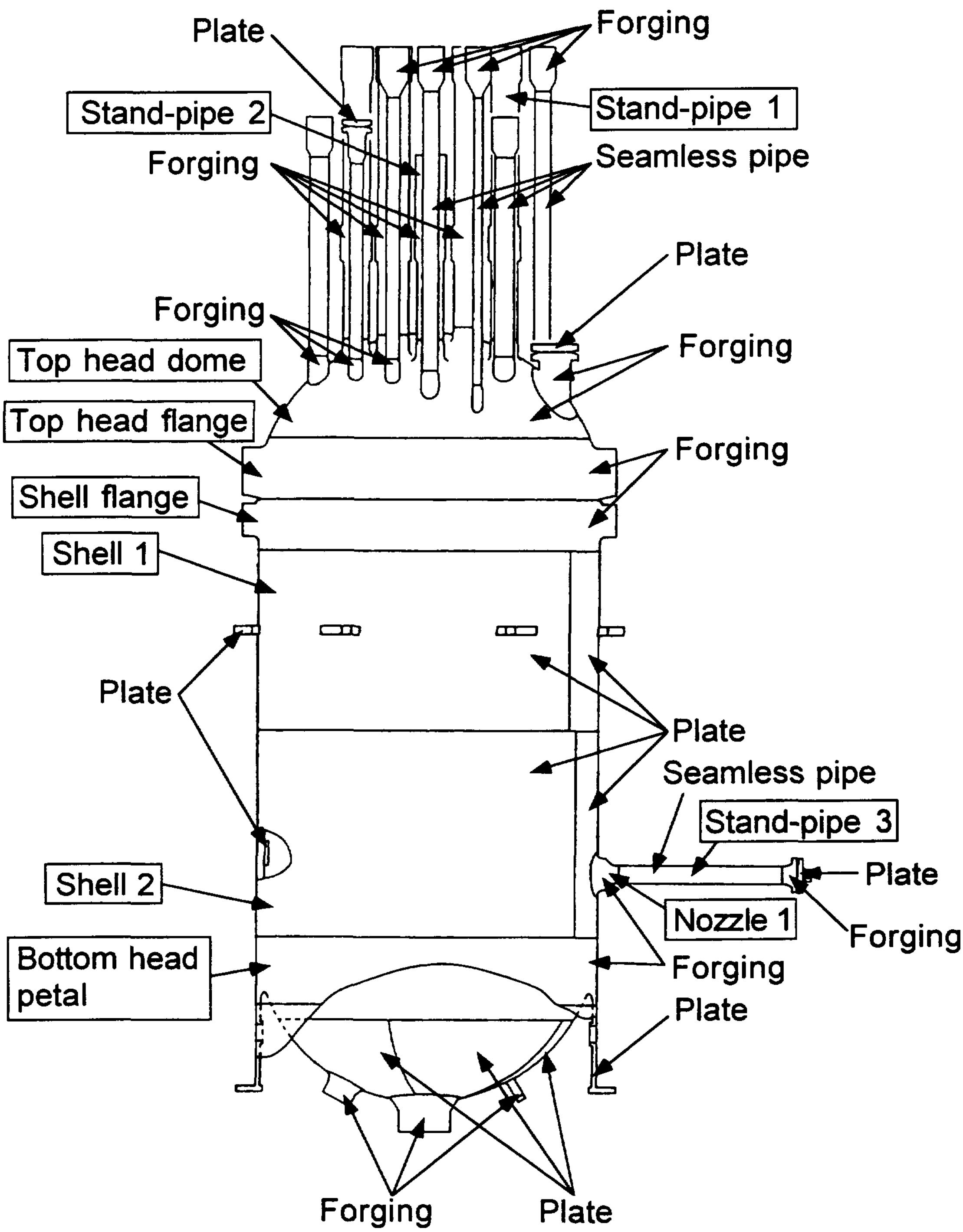


Plate : JIS SCMV4-2, Forging : JIS SFVAF22B,
Seamless Pipe : JIS STPA24

Fig. 3 Material and weld lines of the HTTR RPV

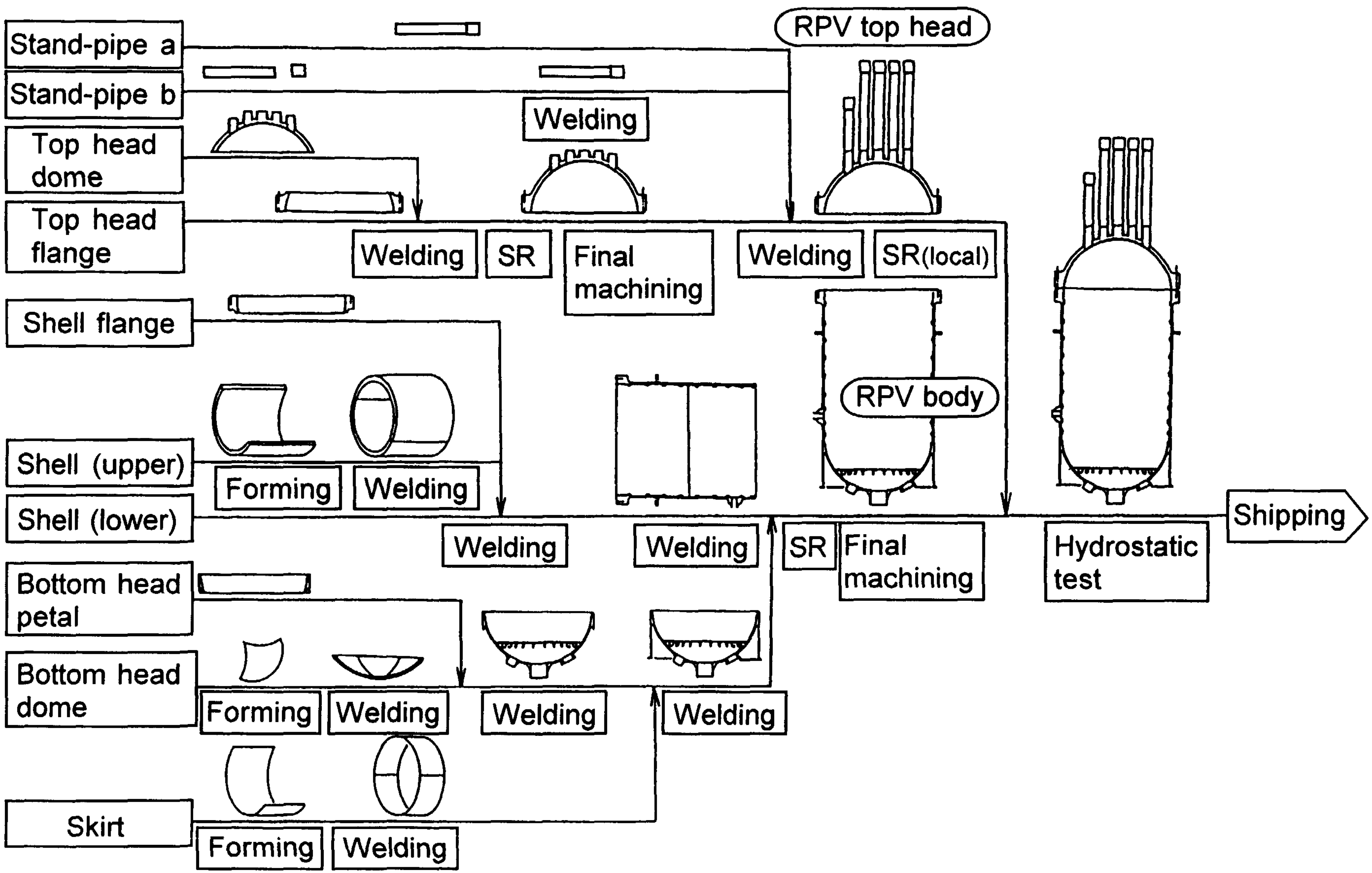


Fig. 4 Fabrication procedure of the HTTR RPV

outside. After welding the top head dome to the upper flange, a large forging, stress relieving (SR) and final machining was performed. Then stand-pipes were welded to the top head dome, followed by local stress relieving as shown in Fig. 4.

Cylindrical shell of the HTTR RPV is made up of four plates formed and welded together as shown in Figs. 3 and 4. The shell was welded to the shell flange, a large forging. The bottom head petal, a large forging, was welded to the bottom head dome, which consists of formed four plates, and the skirt. These were then welded together, building up the RPV body, and stress relieving and final machining was conducted.

For the welding of the large components, automatic narrow-gap (groove) MIG welding, developed by Babcock-Hitachi, K. K. and schematically shown in Fig. 5, was mainly utilized. The narrow-gap MIG welding is superior to ordinary submerged arc welding (SAW) in the following points:

(i) higher ductility of weld and narrower heat-affected zone is attained because of smaller heat input,

(ii) smaller number of path reduces probabilistic occurrence of weld defect,

(iii) smaller amount of welding consumables is needed by adopting square groove.

Hydrostatic test of the RPV at the fabricator was performed before shipping. The RPV was transported to Oarai Research Establishment, JAERI in August 1994. After the installation of the RPV body into the containment vessel, reactor internals started to be installed, which was followed by closing of the RPV top head in October 1995. After that stand-pipe closures with control rods, surveillance holders, etc. were fixed. Pressure test of the primary and secondary cooling system including the RPV was performed successfully in March 1996.

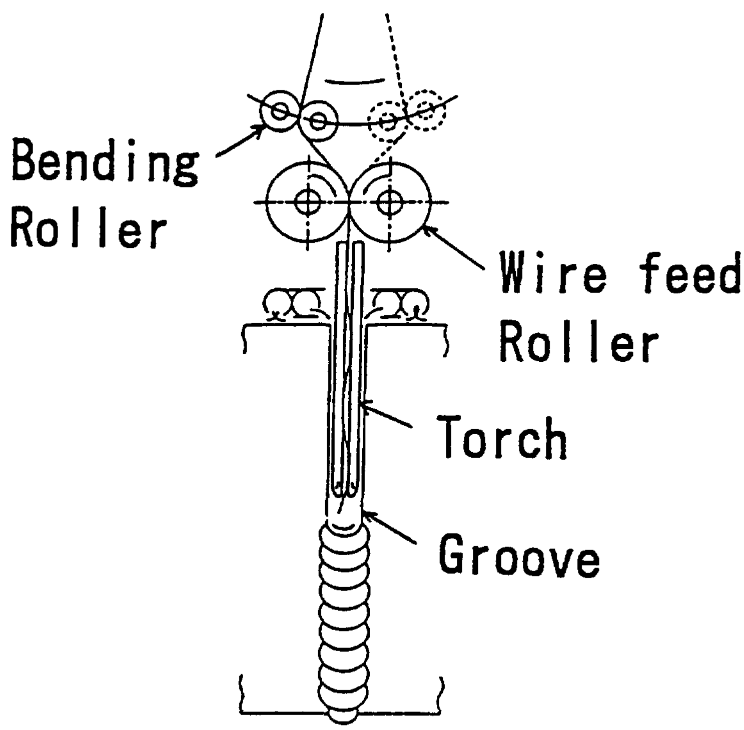


Fig. 5 Mechanism of automatic narrow-gap MIG welding

3. Material

Because fluence of the RPV is calculated to be less than 1×10^{17} n/cm^2 (E>1 MeV), neutron irradiation embrittlement is presumed to be negligible. However, the integrity of the RPV can be impaired due to temper embrittlement at 400 °C. For the purpose of reducing embrittlement, content of some elements is limited on 2 1/4 Cr-1 Mo steel for the HTTR RPV using embrittlement parameters:

$$\text{J-factor} = (\text{Si+Mn})(\text{P+Sn}) \times 10^4 \leq 100 \quad (\text{Si, Mn, P, Sn : wt\%}) \tag{1}$$

$$\overline{X} = (10\text{P}+5\text{Sb}+4\text{Sn}+\text{As})/100 \leq 10 \quad (\text{P, Sb, Sn, As : ppm}) \tag{2}$$

which is called HTGR specification and applied to both base metal and weld metal. J-factor is a commonly used parameter proposed by Watanabe et al. [3]. This factor implies that the impurities Si (silicon) and Mn (manganese) alone in the absence of P (phosphorus) and Sn (tin) cannot cause any embrittlement. Similarly in the absence of Si and Mn, P and Sn, do not cause any embrittlement. It is indicated by Viswanathan and Jaffee [4] that the former coincide with their study but the latter does not, that is Si and/or Mn do not cause temper embrittlement in the absence of the impurity elements P and Sn, however, the combination (P+Sn) is capable of causing significant embrittlement in the absence of Mn and Si contrary to the predictions of the J-factor. Thus it is necessary to employ the other parameter $\overline{X}$.

Base metal

Figure 6 shows J-factor and $\overline{X}$ of nineteen heats of manufactured 2 1/4 Cr-1 Mo steels for the HTTR RPV. J-factor and $\overline{X}$ ranges from 25 to 55 and from 4.1 to 8.3 respectively, which satisfies the HTGR specification. Regarding the plates and four large forgings, the top head dome, top head flange, shell flange, and bottom head petal, which consist main part of the RPV, J-factor and $\overline{X}$ are less than 38 and 5.1 respectively. For these nineteen heats of 2 1/4 Cr-1 Mo steel, J-factor and $\overline{X}$ exhibit almost linear relation. Chemical composition and the embrittlement factors, J-factor and $\overline{X}$, of ten components of the RPV are shown in Table 2, see also Fig. 3 pointing the components with names in boxes.

Figure 7 shows one example of correlation between J-factor and increase in 50 percent ductile-to-brittle fracture appearance transition temperatures (ΔFATT) from the paper by Viswanathan and Jaffee [4]. According to the figure, which arranges long term (20,000 to 60,000 hr) isothermal embrittlement studies in the range 343 °C to 510 °C on numerous heats of commercial, heavy section 2 1/4 Cr-1 Mo steels in a variety of product forms, the ΔFATT becomes less than 10 °C when J-factor is less than 50. Most components of the HTTR RPV fulfill this condition except some medium and small forgings whose values exceed it a little.

Figure 8 shows distribution of nil-ductility transition temperature, T_{NDT}, correlated to J-factor. The figure contains 68 results (some of which overlap each other) of drop weight tests on the components made from the nineteen heats of steels. It should be noted that for all the components reference temperatures, RT_{NDT}, became equal to the nil-ductility transition temperatures, T_{NDT}. The T_{NDT} or RT_{NDT} lies in the range -80 °C to -30 °C, satisfying another HTGR specification that RT_{NDT} should be less than -20 °C. Because T_{NDT} depends on manufacturing process, the results scatter for components of the same heat. Although J-factor is usually related to increase in ductile-to-brittle transition temperatures, it seems to have weak correlation with T_{NDT}.

Table 2. Chemical composition and reference temperature (RT_{NDT}) on some components of HTTR reactor pressure vessel

(wt%)

Forging (ladle analysis)	C	Si	Mn	P	S	Cr	Mo	Cu	Ni	V	Co	Al	Sn	As	Sb	$\overline{X}$ (ppm)	J (wt%)	RT_{NDT} (°C)
JIS spec (SFVAF 22B)	≤0 15	≤0 50	0 30~ 0 60	≤0 030	≤0 030	2 00~ 2 50	0 90~ 1 10											
HTGR spec																≤10	≤100	≤-20
Top head dome	0 14	0 07	0 55	0 003	0 001	2 29	1 06	0 04	0 08	0 005	0 008	<0 005	0 003	0 003	0 0009	5 0	37 2	-50
Top head flange	0 14	0 04	0 54	0 003	0 001	2 29	1 05	0 04	0 09	0 005	0 009	<0 005	<0 003	0 004	0 0010	5 1	34 8	-65
Shell flange	0 14	0 03	0 54	0 003	0 003	2 29	1 00	0 03	0 10	0 005	0 008	<0 005	<0 003	<0 003	0 0010	5 0	34 2	-55
Bottom head petal	0 15	0 05	0 55	0 003	0 001	2 30	1 05	0 03	0 04	0 005	0 007	<0 005	<0 003	0 003	0 0011	5 0	36 0	-35
Nozzle 1	0 13	0 07	0 46	0 006	0 002	2 44	1 05	0 02	0 04	0 006	0 007	0 003	0 003	0 002	0 0006	7 7	47 7	-40
Stand-pipe 1	0 12	0 10	0 45	0 005	0 003	2 32	1 03	0 03	0 04	0 006	0 006	0 003	0 003	0 002	0 0007	6 8	44 0	-60
Stand-pipe 2	0 14	0 10	0 46	0 006	0 004	2 37	1 05	0 03	0 06	0 007	0 007	0 003	0 003	0 003	0 0008	7 9	50 4	-65
Plate (ladle analysis)	C	Si	Mn	P	S	Cr	Mo	Cu	Ni	V	Co	Al	Sn	As	Sb	$\overline{X}$ (ppm)	J (wt%)	RT_{NDT} (°C)
JIS spec (SCMV4-2)	≤0 17	≤0 50	0 30~ 0 60	≤0 030	≤0 030	2 00~ 2 50	0 90~ 1 10											
HTGR spec																≤10	≤100	≤-20
Shell 1	0 15	0 10	0 55	0 003	0 001	2 45	1 06	0 01	0 15	0 010	0 006	0 019	0 001	0 002	<0 001	4 1	26 0	-55
Shell 2	0 14	0 10	0 57	0 004	0 001	2 43	1 05	0 01	0 17	0 010	0 006	0 011	0 001	0 002	<0 001	5 1	33 5	-55
Seamless pipe (ladle analysis)	C	Si	Mn	P	S	Cr	Mo	Cu	Ni	V	Co	Al	Sn	As	Sb	$\overline{X}$ (ppm)	J (wt%)	RT_{NDT} (°C)
JIS spec (STPA24)	≤0 15	≤0 50	0 30~ 0 60	≤0 030	≤0 030	1 92~ 2 60	0 87~ 1 13											
HTGR spec																≤10	≤100	≤-20
Stand-pipe 3	0 11	0 02	0 48	0 005	0 002	2 35	0 99	0 02	0 05	0 010	0 004	0 002	0 002	0 002	0 001	6 5	35 0	-30

Note RT_{NDT} (reference temperature) was equal to T_{NDT} (nil-ductility transition temperature) regarding all the components above

Weld	C	Si	Mn	P	S	Cr	Mo	Cu	Ni	V	Co	Al	Sn	As	Sb	$\overline{X}$ (ppm)	J (wt%)	RT_{NDT} (°C)
JIS spec of wire (YG2CM-A)	≤0 15	0 20~ 0 90	0 40~ 1 40	≤0 025	≤0 025	2 10~ 2 70	0 90~ 1 20	≤0 40										
HTGR spec																≤10	≤100	≤-20
Wire	0 12	0 38	0 86	0 007	0 007	2 33	1 12	0 01			0 002							
Weld metal	0 12	0 22	0 63	0 005	0 006	2 25	1 05	0 03	0 02	0 003	0 003	0 004	0 002	0 002	0 001	6 0	60 0	-35

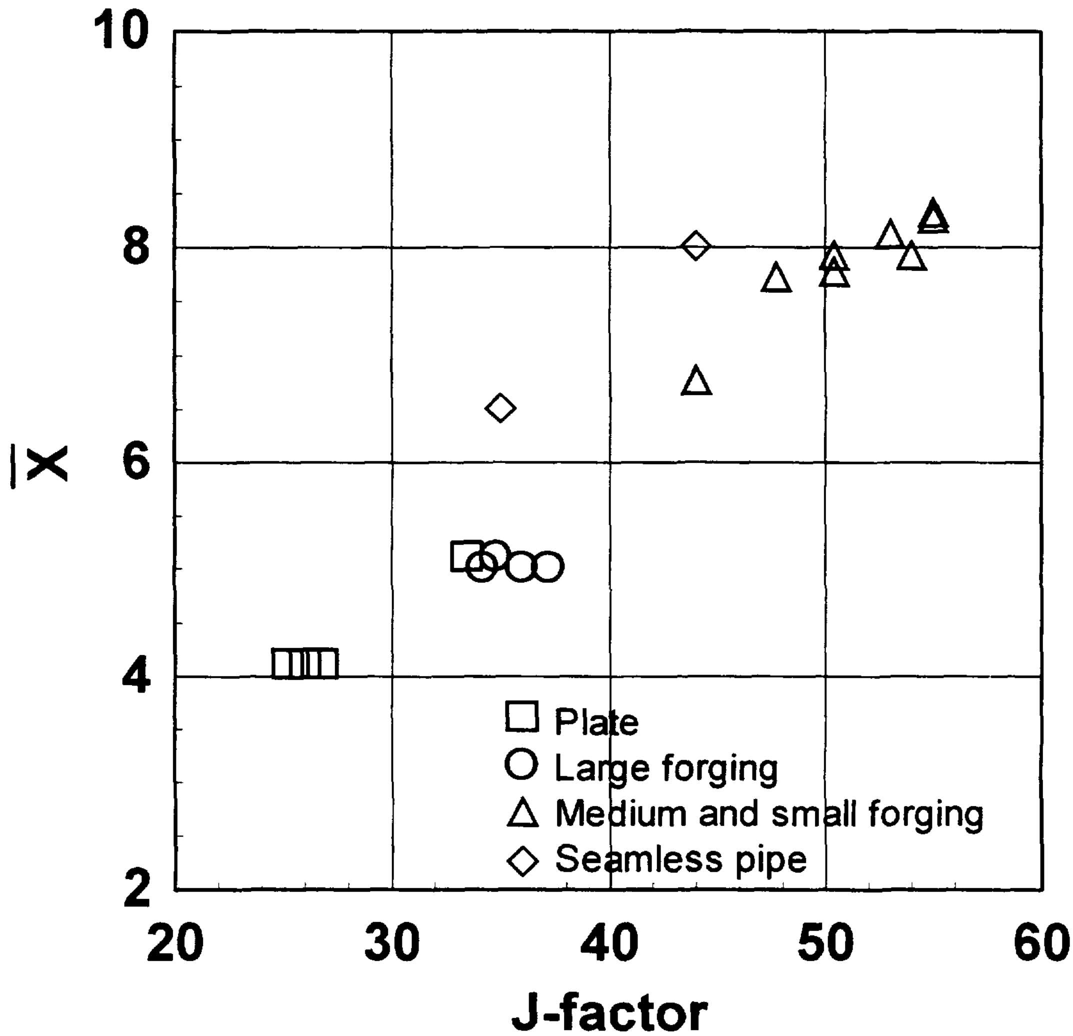


Fig. 6 Distribution of $\overline{X}$ and J-factor on 2 1/4 Cr -1 Mo steel for the HTTR RPV

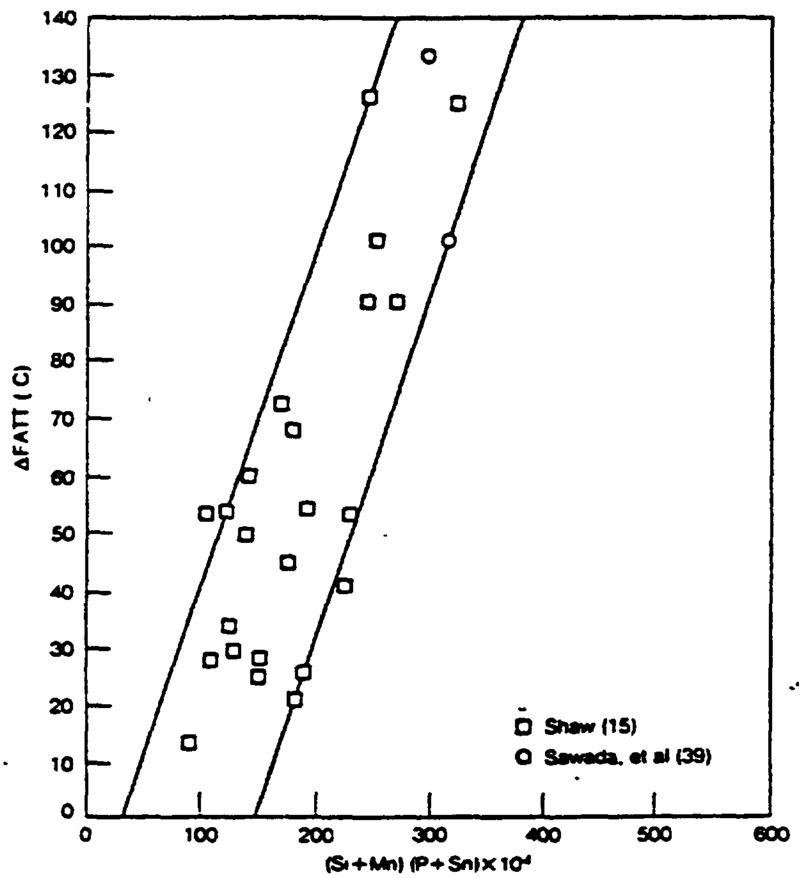


Fig. 7 Correlation between J-factor and ΔFATT by Viswanathan and Jaffee [3]

Weld metal

The embrittlement parameters, J-factor and $\overline{X}$, derived by check analysis of weld metal of the HTTR RPV ranged from 60 to 84 and 6.0 to 9.0 respectively, which satisfies the HTGR specification. The values of J-factor of weld metal are larger than those of base metal, because content of silicon (Si) and manganese (Mn) of the weld metal is higher than that of the base metal, as shown in Table 2. Applying the J-factor to Fig. 7, maximum embrittlement potential (ΔFATT) of the weld metal is estimated to be roughly 30 °C. The ΔFATT at 400 °C, which is temperature of the HTTR RPV at normal operation, is predicted to be lower than the estimated value, because Fig. 7 includes data at as high as 510 °C.

Thus it is concluded that increase in reference temperatures RT_{NDT} on the HTTR RPV due to irradiation and temper embrittlement is presumed to be small for both the base and weld metal.

Fracture toughness requirements

For the purpose of preventing brittle fracture, Japanese regulation on fracture toughness requirements on RPVs for Light Water Reactors, which is based on ASME Code Section III and Nuclear Regulation Commission 10CFR Ch. 1 Part 50 Appendix G, is applied to the HTTR RPV. For example, the temperature of the closure flange regions that are highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 67 °C (120 °F) for normal operation, when pressure exceeds 20 percent of the preservice system hydrostatic test pressure. Because reference temperature of the weld

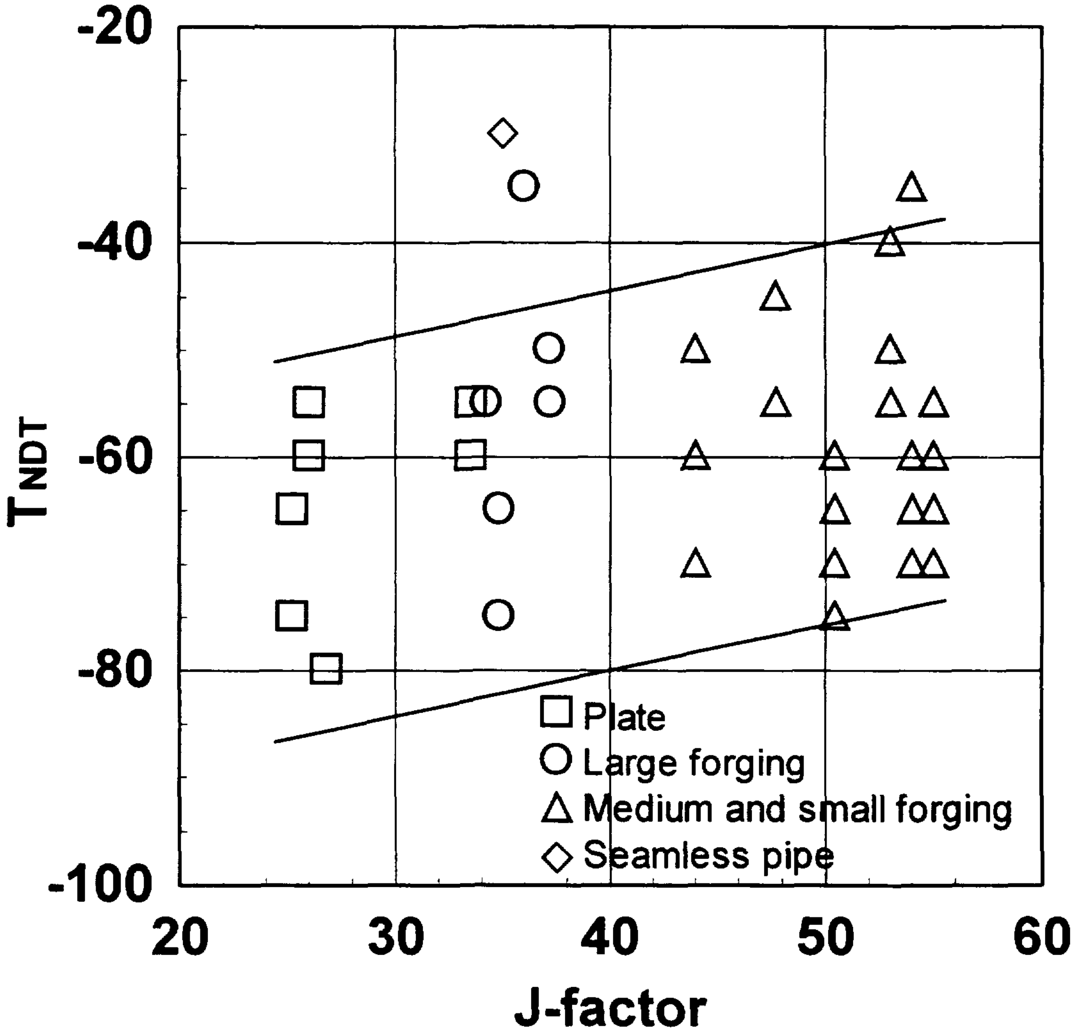


Fig. 8 Relation between T_{NDT} and J-factor on 2 1/4Cr-1Mo steel for the HTTR RPV

metal is -35 °C as shown in Table. 2, temperatures of the top head and shell flange of the HTTR RPV must exceed 52 °C:

-35 + 67 + 20 (predicted increase in reference temperature) =52 (°C)

at normal operation including start-up of the reactor.

Though primary coolant of the HTTR can be preheated by three primary gas circulators before start-up of the reactor, temperature of the primary helium coolant with small heat capacity decreases rapidly when secondary (water) cooling system start operating. Thus special start-up procedure has to be taken for the HTTR, which is still being considered.

Surveillance test

Surveillance tests on materials of the HTTR RPV will be performed in order to examine transition of mechanical properties of the materials due to irradiation and temper embrittlement. In addition to the mandatory tensile test and Charpy impact test, fracture toughness test for determination of J_{IC}, and test based on Magnetic Interrogation Method [5] for assessment of material deterioration and remaining life of the RPV. Surveillance test items are shown in Table. 2.

Surveillance test specimens are stored in 12 holders, which are installed close to inner surface of cylindrical shell of the RPV. During the operation of the HTTR for 20 years, the specimen holders are taken out four times: three holders at a time. The post-irradiation tests are planned to be performed at JMTR hot laboratory, JAERI. Temperature and dose of the specimens are measured by temperature monitors and dosimeters stored in the holder respectively.

4. Conclusion

Fabrication, examination and testing of the HTTR RPV completed successfully in March 1996. By limiting content of elements on 2 1/4 Cr-1 Mo steel, especially impurities of Si, P, Sb, and Sn, temper embrittlement as well as irradiation embrittlement of the RPV is presumed to be small.

Table 3. Surveillance test items of the HTTR reactor pressure vessel

		Plate		Forging
	(base metal,	H. A. Z.,	weld metal)	(base metal)
Tensile test	O	O	O	O
Charpy impact test	O	O	O	O
Fracture toughness test	O	O	O	O
Magnetic Interrogation Method [5]	O	O	O	-

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STATUS OF GAS COOLED TEST REACTOR PROGRAMMES

XA9846690

HTR-10 ENGINEERING EXPERIMENTS

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Abstract

The Institute of Nuclear energy Technology has undertaken a programme of engineering experiments to verify the design characteristics and performance of the HTR-10's components and systems. These include depressurization tests of the hot gas duct from operating pressure, tests of the control rod drive mechanism, performance validation of the hot gas duct under operating pressure and temperature conditions, two phase flow stability testing for the once through steam generator and performance testing of the fuel handling system. This paper provides a review of these engineering experiments.

1. INTRODUCTION

With the first concrete of the reactor building fundament being poured on June 14, 1995, the construction of a pebble bed type high temperature gas-cooled reactor (HTGR) started in China. It is a test reactor with 10MW thermal power output (termed HTR-10). The test reactor is located on the site of Institute of Nuclear Energy Technology of Tsinghua University (INET) in the northwest suburb of Beijing, about 40km away from the city.

Design of the HTR-10 test reactor represents the features of HTR-Modular design. Reactor core and steam generator are housed in two steel pressure vessels which are arranged in a "side-by-side" way. The two vessels are connected to each other by a connecting vessel in which the hot gas duct is designed. All these steel pressure vessels are in touch with the cold helium of about 250 °C coming out from the circulator which sits over the steam generator tubes in the same vessel.

Fuel elements used are the spherical fuel elements (6cm in diameter) with coated particles. The reactor core contains about 27,000 fuel elements forming a pebble bed which is 180 cm in diameter and 197 cm in average height. Spherical fuel elements go through the reactor core in a "multi-pass" pattern. Graphite serves as the main material of core structures.

Decay heat removal of the HTR-10 is designed on a completely passive basis. At a loss of pressure accident, against which no core cooling is foreseen at all, decay power will dissipate through the core structures by means of heat conduction and radiation to the outside of the reactor pressure vessel, where, on the wall of the concrete housing, a surface cooling system is designed. This system works on the principle of natural circulation of water and it takes the decay heat via air coolers to the atmosphere.

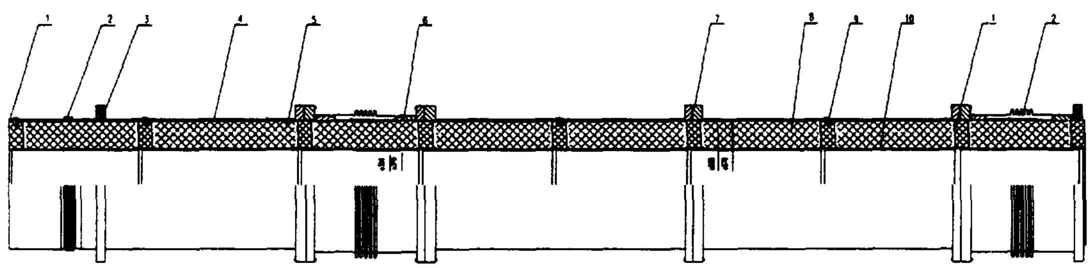
There are two reactor shutdown systems, one control rod system and one small absorber ball system. They are all designed in the side reflector. Both systems are able to bring the reactor to cold shutdown conditions. Since the reactor has strong negative temperature coefficients and decay heat removal does not require any circulation of the helium coolant, the turn-off of the helium circulator can also shut down the reactor from power operating conditions.

However in the HTR-10 design some modifications from the HTR-Module were made to satisfy Chinese conditions. For example, the steam generator is composed of a number of modular helical tubes with small diameter, pulse pneumatic discharging apparatus are used in the fuel handling system and step motor driving control rods are designed. These modifications would cause some uncertainty in our design. It is necessary to do engineering experiments to prove these new or modified idea. Therefore a program of engineering experiments for HTR-10 key technologies is being conducted in INET. The main aims of these engineering experiments are to verify the designed characteristics and performance of the components and systems, to feed back on design and to obtain operational experiences.

Those engineering experiments are depressurization test of the hot gas duct at room temperature and operating pressure, performance test of the hot gas duct at operating helium temperature and pressure, performance test of the pulse pneumatic fuel handling system, test of the control rods driving apparatus and two phase flow stability test for the once through steam generator.

2. DEPRESSURIZATION TEST OF THE HOT GAS DUCT

The hot gas duct, is shown in Fig 1, is composed of inner tube, outer tube, insulation layer and corundum bricks etc.. The outer tube acts as the supporting structure of the inner tube and insulation layer. Both ends of the outer tube are welded to the corrugated pipes to absorb the thermal expansion. The hot gas duct is divided into five sections in order to be easy to be installed. Each section is connected by slide joint. Corundum bricks are used as insulating material between two inner tubes and fixed to the outer tube.

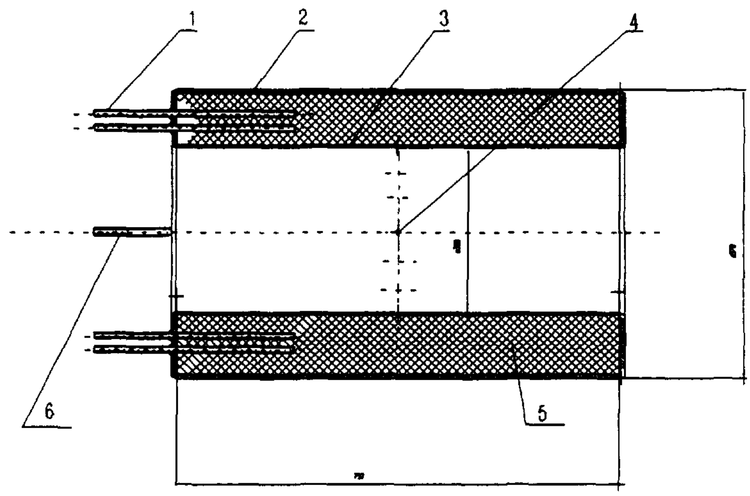


1 End
2 Corrugated pipes
3 Flange at the end
4 Outer tube
5 Insulation layer
6 Outer tube at the end
7 Flange
8 Stainless steel foies
9 Bolts
10 Inner tube
11 Corundum bricks

FIG.1 Structure of the Hot Gas Duct

At the depressurization accident the pressure in insulation layer would be released only through area of exhaust in slide joints between two inner tubes of the hot gas duct. The pressure difference between the insulation layer and outside the inner tube would be existed due to velocity difference of helium gas flow in insulation and in inner tube at depressurization accident, and may cause damage of the inner tube. In order to research on the effect of depressurization rate and area of exhaust in slide joints on the pressure difference an experimental equipment was set up at INET.

The test section is shown in Fig 2. Its diameter is the same as one of the hot gas duct. But only two sections without slide joint between them are taken for the test and several small holes at the middle of the test section are equipped to simulate area of exhaust in slide joints. The test was carried out at air of 0.8 MPa, nitrogen of 1.0 MPa and 3.0 MPa as well as helium of 0.8 MPa, respectively.



1 Pressure measure tubes
2. Outer tube
3. Inner tube
4 Holes and bolts
5. Insulation layer
6 Pressure measure tube

FIG. 2 Test Section for the DepressurizationTest

The effect of depressurization rate on the pressure difference between the insulation layer and outside the inner tube is shown in Fig 3. It is clear that maximum pressure difference is raised with increase of depressurization rate.

The effect of area of exhaust in slide joints on the pressure difference between the insulation layer and outside the inner tube is shown in Fig 4. It can seen that maximum pressure difference is increased with decrease of area of exhaust in slide joints and a critical area of exhaust in slide joints exists, that is, maximum pressure difference would rapidly increased if area of exhaust in slide joints would be less this critical one. It is also related with depressurization rate.

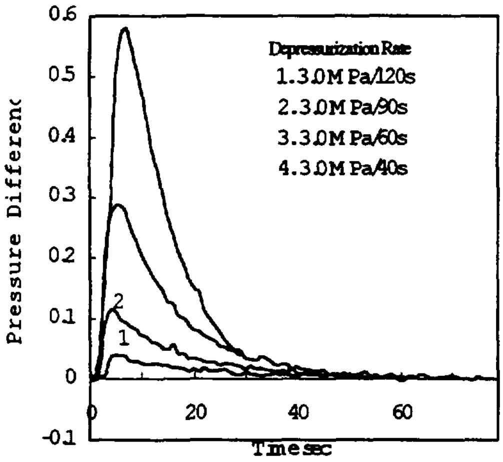


Fig 3 Effect of Rate on Pressure Differenc

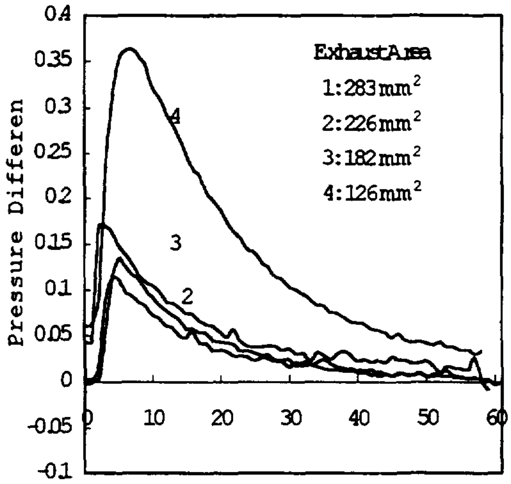


Fig 4 Effect of Area on Pressure Difference

The effects of depressurization rate and area of exhaust in slide joints on maximum pressure difference at nitrogen 3.0 MPa is listed in Table 1.

Table 1 Effects of Depressurization Rate and Area of Exhaust in Slide Joints on Maximum Pressure Difference at Nitrogen 3.0 MPa

	440 mm^2	340 mm^2	283 mm^2	226 mm^2	182 mm^2	126 mm^2	25 mm^2
40 sec		0.476MPa	0.579MPa				
60 sec	0.107MPa	0.180MPa	0.286MPa				
90 sec			0.130MPa	0.134MPa	0.170MPa	0.364MPa	0.726MPa
120 sec			0.039MPa	0.054MPa	0.116MPa	0.236MPa	

The effect of different medium (helium and nitrogen) on the pressure difference is shown in Fig 5 and 6. It can be seen that at same depressurization rate and area of exhaust in slide joints maximum pressure difference at helium atmosphere is less than one at nitrogen atmosphere.

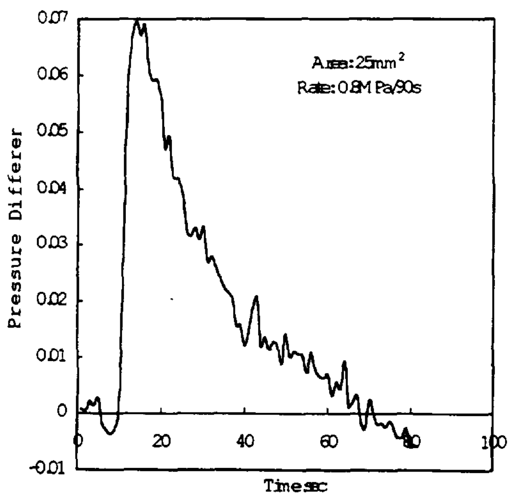


Fig 5 Effect on Pressure Difference at Helium

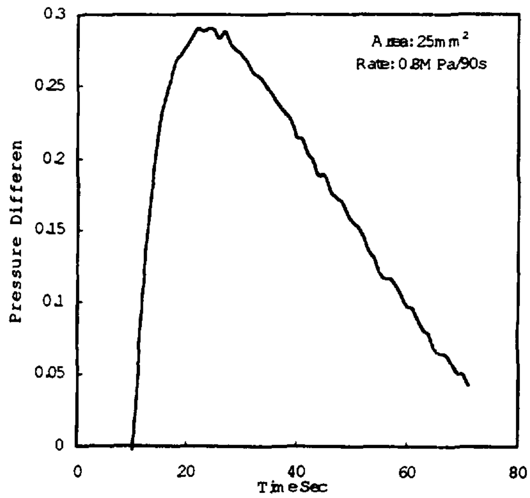


Fig 6 Effect on Pressure Difference at Nitrogen

At depressurization accident in the HTR-10 the maximum depressurization rate is 3.0 MPa/120 sec, the minimum area of exhaust in slide joints is equivalent area of 25 mm^2. The maximum pressure difference would be 0.72 MPa for nitrogen atmosphere and 0.17 MPa for helium atmosphere. Based on calculation the integrity of the hot gas duct can be maintained at this condition.

3. PERFORMANCE TEST OF THE HOT GAS DUCT

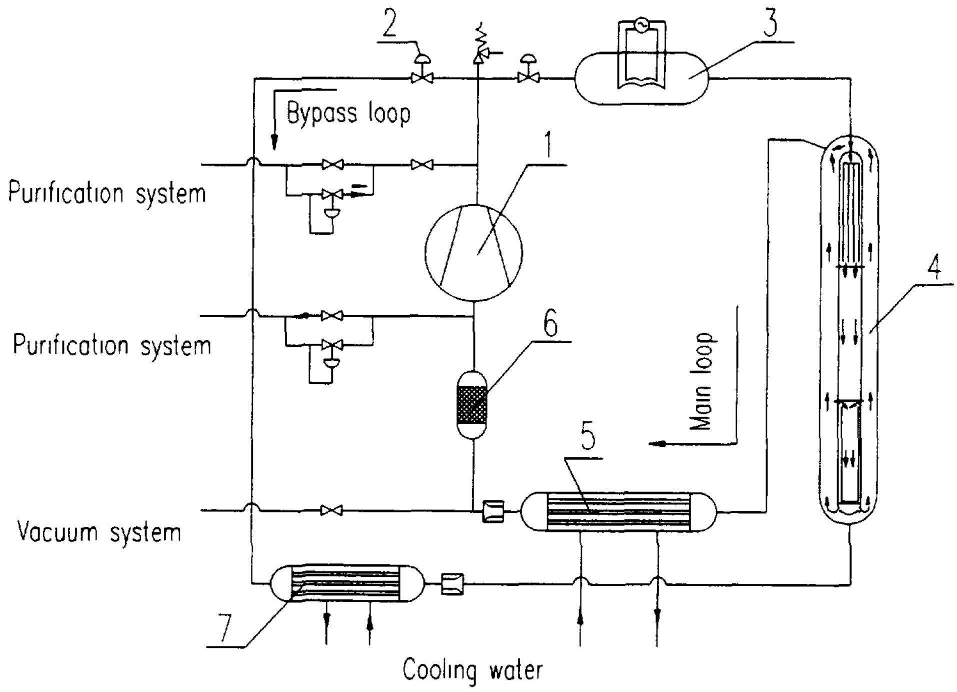
In order to master the technique and experience of designing, constructing and operating helium system and provide a facility for the research and development of the helium technique and helium components, a helium test loop (HETL) has been installed. The HETL consists of helium gas loop, water cooling system, helium purification system, helium storage and pressure control system, sampling and analyzing system of trace impurity in helium gas, measuring and data acquisition system, etc. The major technical parameters of the HETL are shown in Table 2.

Table 2 Technical Parameters of the HETL

Operational pressure	4.0	MPa
Helium gas flow rate	30 ~ 300	M^3/h
Pressure head of the helium circulator	120	kPa
Power supply	650	kW
Helium flow rate of the purification system	2.3	g/s
Analyzing sensitivity of trace impurity in helium	<0.4	cm^3/M^3
Mean helium leak rate for each discharging connecting	$<6 \times 10^{-5}$	$Pa.M^3/s$

Fig. 7 shows a flow sheet of the helium gas loop. It includes main loop and bypass loop. They consists of helium circulator, median temperature electric heater, main cooler, bypass cooler, filter and regulating valves etc. The circulator is a type of two stage centrifugal blower with gas bearing. The maximum helium gas flow rate, maximum operating temperature and pressure are 300m^3/s, 116 °C and 8 MPa, respectively. The revolution, rated voltage supply and power input are 3000-12000 r.p.m, three-phase 500V and 115 kW, respectively. The medium temperature electric heater is an once-through tube-shell type and its heating tube is made of Incolloy-800 alloy. The helium gas flows inside the tube and is heated to rated temperature. The main cooler and bypass cooler are tube-shell and U-tube type, and the helium gas flows inside the tube.

The hot gas duct test section with a triple tube structure includes an inner electrical heater, a hot gas duct model and a cold-hot helium gas static mixer. The schematic structure of the dot gas duct test section is shown in Fig. 8. The helium gas coming from the median temperature heater with a maximum temperature of 400 °C flows into the inner electrical heater from its entrance, and is heated up to 700 °C (or higher), and then flows through the inner passage of the hot gas duct model. Cold helium gas stream coming from the bypass cooler with a temperature of about 70 °C is led into the inlet of the

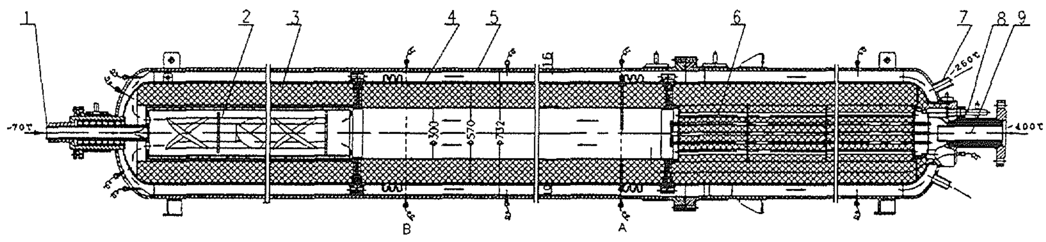


1 Circulater 2 Regulating valve 3 Electric heater 4 HGD test section
5 Main cooler 6 Filter 7 Bypass cooler

FIG .7 Flow Sheet of the Helium Gas Loop

mixer through the low temperature helium entrance and an annular passage. The hot helium gas and the cold helium gas are mixed in the mixer, and then the mixing helium gas with a temperature of about 250 °C passes through the annular passage between the pressure tube and the outer tube of the hot gas duct. The hot gas duct model is designed with the same configuration and radial dimensions as the hot gas duct for the HTR-10. Incolloy-800 alloy was chosen for the inner tube, as it may be exposed to helium gas flow at temperature of 700 °C (or 950 °C). The outer tube and the pressure tube is made of 16Mn. Fibrous ceramic insulation material, the main composition of which is Al_2O_3 and SiO_2, is divided into two sub-layers by stainless steel foils and is packed into the space between inner and outer tube.

Flow rate, temperature, pressure and differential pressure are measured. Flow rates are measured by two orifice meter which are installed in main loop and bypass loop. Pressure and differential pressure are measured using transmitters. At the cross sections A and B, 610 mm far from two ends of the hot gas duct model, 16 thermocouples are installed for measuring the temperature of helium gas, liner tube, insulation layer and outer tube. The temperatures of helium gas at the inlet and outlet of inner heater, helium gas in the annular passage and pressure tube are also measured by thermocouples.



1 Entrance for low temperature He 2. Cold-hot He mixer 3 Internal insulation 4 HGD model 5. Pressure tube

6 Inner heater 7 Outlet for He 8 Electrode assembly 9 Entrance for median temperature

Fig 8 Schematic Structure of the HGD Test Section

The effective thermal conductivity of insulation layer was measured. A relationship between the effective thermal conductivity and the average temperature of the insulation layer at helium pressure of 3.0, 2.5 and 1.5 MPa is shown in Fig 9. The empirical equation of the effective thermal conductivity were obtained as follows:

$K_{eff} = 0.3468 + 0.0003T$ (°C) W/m/°C

Where T is the average temperature of the insulation layer.

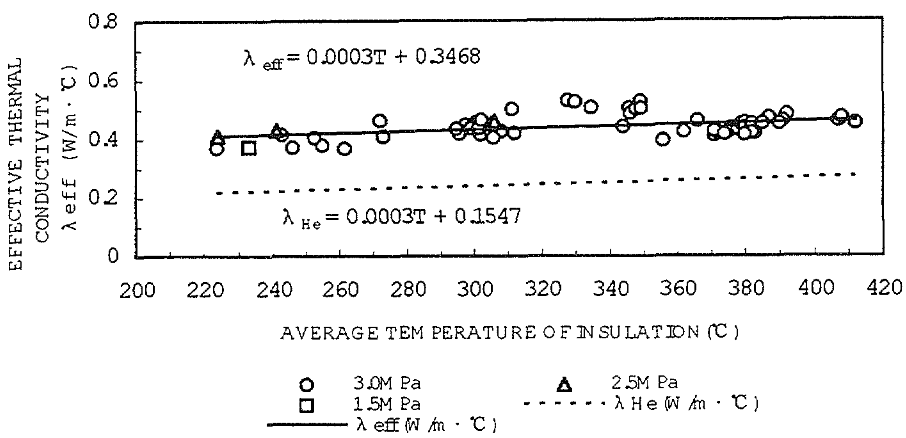


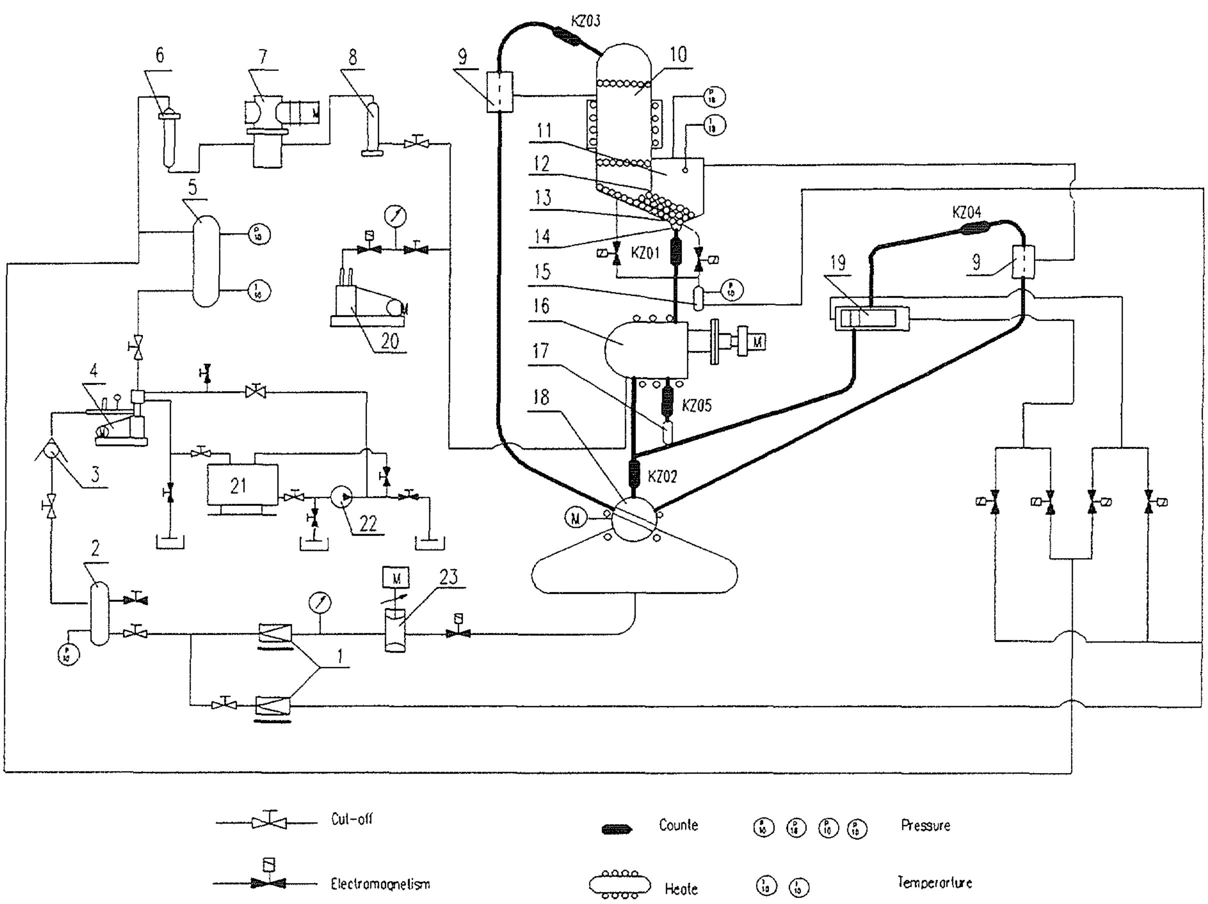
FIG.9 Relatoin Between λ_{eff} and Average temperature of Insulation

The measured effective thermal conductivity are 1.5 ~ 2 times as large as the thermal conductivity of helium gas at the same condition of temperature and pressure.

The hot gas duct model has been operated more than 100 hours at the temperature of 700 °C and the pressure of 3.0 MPa, and has been borne over 15 times temperature cycle between 300 and 700 °C, and 20 times pressure cycle between atmospheric pressure and 3.0 MPa. No any deterioration of thermal performance was detected.

4. PERFORMANCE TEST OF THE PULSE PNEUMATIC FUEL HANDLING SYSTEM

The fuel elements in the HTR-10 are successively fed to and removed from the reactor core by the refueling and discharge facilities. After passing through the core, the fuel elements are removed from the fuel element discharge tube via a pulse pneumatic single- exit gate (a reducer), which is placed inside the reactor pressure vessel. With the aid of the separator, fuel elements which are not geometrically satisfactory drop into a fragments container, while the intact fuel elements are transported to the elevator set which includes a burn-up measurement facility inside. This determines whether the fuel element has reached its designed burn-up and has to be finally discharged or can be returned to the core.



1. Pressure reducing valve	2. H. P. tank	3. Check valve	4. Compressors
5. L. P. tank	6. Drying cylinder	7. Air cooler	8. Filter
9. Buffer	10. Discharging tube	11. Tank	12. Fender
13. Large funnel	14. Small funnel	15. Pulse gas storage	16. Separator
17. Fragments tank	18. Elevator	19. Reducer	20. Vacuum pump
21. Water tank	22. Pump	23. Regulating valve	

FIG.10 Schematic of the Fuel Handling System Apparatus

In order to avoid operating difficulty with mechanical facilities a new fuel handling system has been developed in our institute. Its features are characterized by pulse pneumatic discharging, gas tight magnetic driving and elctro-induction balls counters.

The test at room temperature has been carried out. Its main task is to prove its design concept for fuel discharging. More than 100,000 balls were discharged by pulse pneumatic discharging way.

Full scale apparatus for test of whole fuel handling system and its components at helium temperature of 180 °C and low pressure was installed. The schematic of the full scale apparatus is shown in Fig. 10. Main components in this apparatus are prototype ones. It consists of the discharge tube, a reducer, a failed ball separator, an elevator, graphite ball detectors, pressure reducing valves, adjusting valves, heaters and so on. Its helium auxiliary system consists of the helium compressor, helium storage tanks, air coolers, filters and vacuum pumps etc.. The power loading control (PLC) is used for automatic control, mosaic simulation panels for simulating display, the computer for data management.

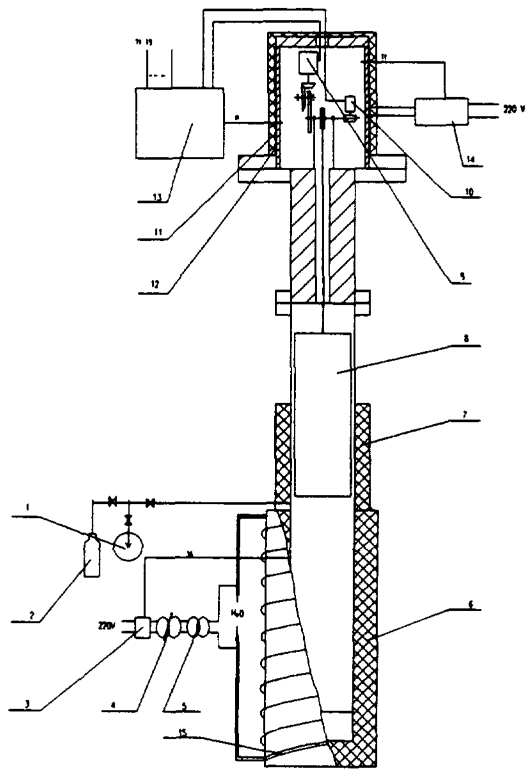
Main parameters of full scale apparatus are listed in Table 3. The operation at temperature of 150 °C is being carried out.

Table 3 Main Parameters of Fuel Handling System

Diameter of discharging tube	500	mm
Height of discharge tube	4200	mm
Maximum charging height	4000	mm
Width of tank	500	mm
Operating temperature	150-180	°C
Operating pressure	0.2	MPa
Circulating balls number	60	balls/hr
Volume of pulse gas storage tank	0.05	M^3
Operating pressure of pulse gas	0.4	MPa
Pulse frequency	2	Hz/min.
Revolution of separator	1	r/min.
Diameter of graphite balls	60	mm

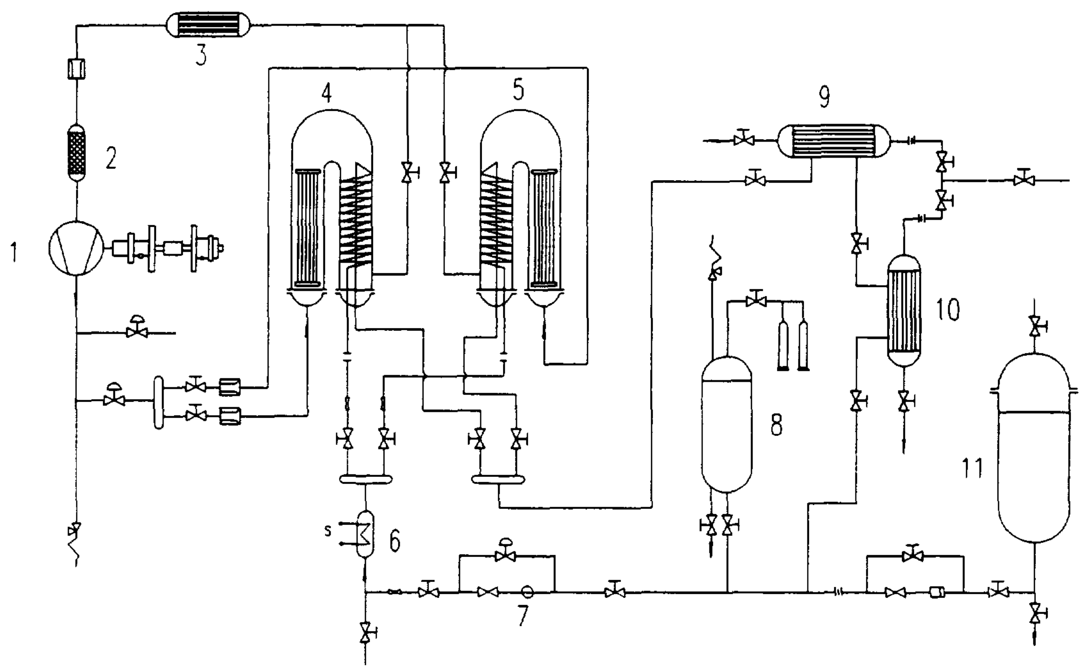
5. TEST OF THE CONTROL RODS DRIVING APPARATUS

The experiment on the control rods driving system at room temperature was done in order to verify its main performance. The performance of the control rod at operating temperature and helium atmosphere of low pressure will be tested at a full scaling apparatus.The schematic of the control rods driving apparatus is shown in Fig.11. It consists of a step motor, a gearbox, a chain-chain wheel, a speed restriction, an indictor for position of the control



1. Vacuum pump
2. Helium cylinder
3. Temperature controller
4. Voltage regulator
5. Transformer
6. Insulation
7. Insulation
8. Control rod
9. Step motor
10. Synchronometer
11. Insulation
12. Heater
13. Control board
14. Temperature controller
15. Eddy heater

FIG. 11 Schematic of Control Rods Driving System



1. Helium blower 2. Filter 3 Heater exchanger 4.5 Test section 6 Preheater 7 Circulating pump 8 Pressurizer tank 9 Condenser 10 Heater exchanger 11 Water storage tank

FIG. 12 Flowsheet of Two Phase Flow Stability Loop rod, a control rod and a vessel. Following test will be done by the means of this apparatus: control rod movement, maximum fall speed of the control rod, indictor performance and life time. The control rods driving apparatus is under installation and whole test will be finished at middle of next year.

6. TWO PHASE FLOW STABILITY TEST FOR THE ONCE THROW STEAM GENERATOR

The main purposes of the two phase flow engineering test facility are: to study on the stability of the HTR-10 steam generator at operational conditions and to determine maximum throat diameter of the throttle appertains, to study flow resistance on the water side and average heat transfer coefficient on the helium gas side of the steam generator with small curvature radius of the helical tube.

The two phase flow engineering test facility of the steam generator consists of the helium loop, primary water loop and secondary water loop. In the helium loop there are two full scale test sections. The schematic for two phase flow engineering test facility of the steam generator is shown in Fig. 12.

The helium loop includes a helium blower, a filter, regulator valves, an electric heater, two test sections, a main cooler, helium flow meters and the helium purification system. Helium coming from the helium blower is heated by the electric heater to 600-700 °C, then led to the test section (steam generator bundle model). In the test section helium gas transfers its heat to the primary loop water as does in the HTR-10' steam generator. Cooled gas goes through the main cooler, the filter and returns to the helium blower to make a helium gas circulation.

Low voltage, large current heating pattern is used for the electric heater. The heating power is 2 x 180 kW.

The primary water loop mainly consists of a circulating pump, a pre-heater, inlet throttle apparatus, steam generator bundle, a condenser, a heat exchanger, a pressurizer, and a water tank. Deionized water is used in the primary water loop.

The function of the secondary water loop is to remove the heat from the primary loop water. The secondary loop consists of a water tank, a sealed pump and an air cooling tower.

The two phase flow engineering test facility of the steam generator is under installation and will be operated at beginning of next year

7. CONCLUSION

A program of engineering experiments for HTR-10 key technologies is being conducted in INET. The main aims of these engineering experiments are to verify the designed characteristics and performance of the components and systems.

The experiment for the hot gas duct at both room and operation condition was carried out and the obtained results are satisfied. The experiments for the fuel handling system and the control rods driving system at room temperature were completed and the experiments at operation temperature are being carried out.

XA9846691

CHARACTERISTICS OF HTTR's STARTUP PHYSICS TESTS

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Abstract

The High Temperature Engineering Test Reactor (HTTR) which is under construction by Japan Atomic Energy Research Institute (JAERI) is a graphite-moderated and helium gas-cooled reactor with an outlet temperature of 950°C and a thermal output of 30MW. The first criticality is expected at the end of October 1997. The start-up physics tests (SPTs) are planned in the period from mid 1997 to the end of 1998. Characteristic items of the SPTs are 1) Criticality approach, 2) Tests on a preliminary annual core, 3) Measurement of scram reactivity, 4) Excess reactivity test, 5) Measurements along with a 2-step-scram reactor shutdown procedure.

1. Introduction

The High Temperature Engineering Test Reactor (HTTR) is developed to establish and upgrade the technological basis for advanced high temperature gas-cooled reactors (HTGRs) and to conduct various irradiation tests for innovative high temperature basic research. The HTTR is a test reactor with a thermal output of 30MW and a coolant outlet temperature of 850°C at rated power operation and 950°C at high temperature test operation. It has the capability to demonstrate nuclear process heat utilization. The fuel loading will begin in the middle of 1997. The first criticality is expected at the end of October 1997. The start-up physics tests (SPTs) are planned in the period from mid 1997 to the end of 1998, and full power operation will be achieved in the beginning of 1999.

The SPTs have been selected from two viewpoints. One is to carry out the function tests necessary for commissioning. These tests consist of the measurement of significant values concerning safe reactor operation and shutdown so as the control rod reactivity, shutdown margin, and excess reactivity. The other is to get an insight into HTTR's characteristics. To determine scram reactivity, a new approach using a delayed integral counting method will be tested and compared to other evaluation methods. Table 1 lists all planned SPTs. This paper describes characteristics of the SPTs mainly related to nuclear properties.

2. Overview of the HTTR design

Table 2 shows the major specifications of the HTTR. The active core, 2.9 m in height and 2.3 m in equivalent diameter, consists of 30 fuel columns and 7 control rod (CR) guide columns. 9 more CR columns are distributed among the adjacent reflector graphite columns. Vertical and horizontal cross sections of the HTTR are shown in Fig. 1 and 2, respectively. Each fuel column consists of 9 graphite blocks of which the inner 5 contain the fuel. Fig.3 shows the structure of the pin-in-block type fuel used in the HTTR. The fuel rod consists of a graphite sleeve with 14 fuel compacts. The fuel rods are inserted into the coolant channels of the fuel blocks. The fuel compacts contain the Coated Fuel Particles (CFPs). The coolant flow is downward. Burnable Poison rods (BP) made of boron carbide are inserted into the vertical holes below the dowel pins of the fuel blocks.

3. Physics Tests

Physics tests are basically divided into tests before reaching first criticality and those that need the critical condition. The latter can be divided in the tests carried out at zero power level, i.e., at room temperature and at a condition where the core is kept isothermal at elevated

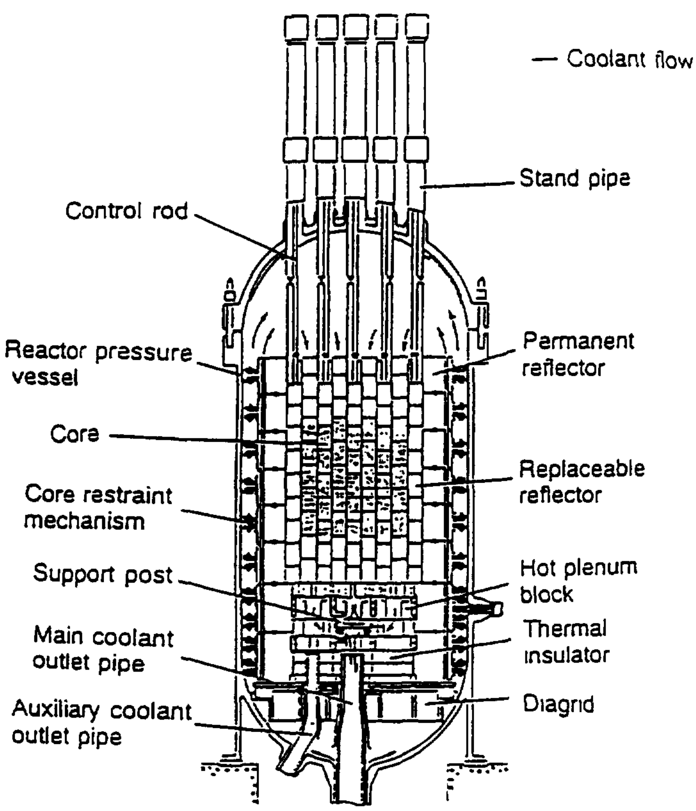


Fig.1 Vertical cross section of the HTTR

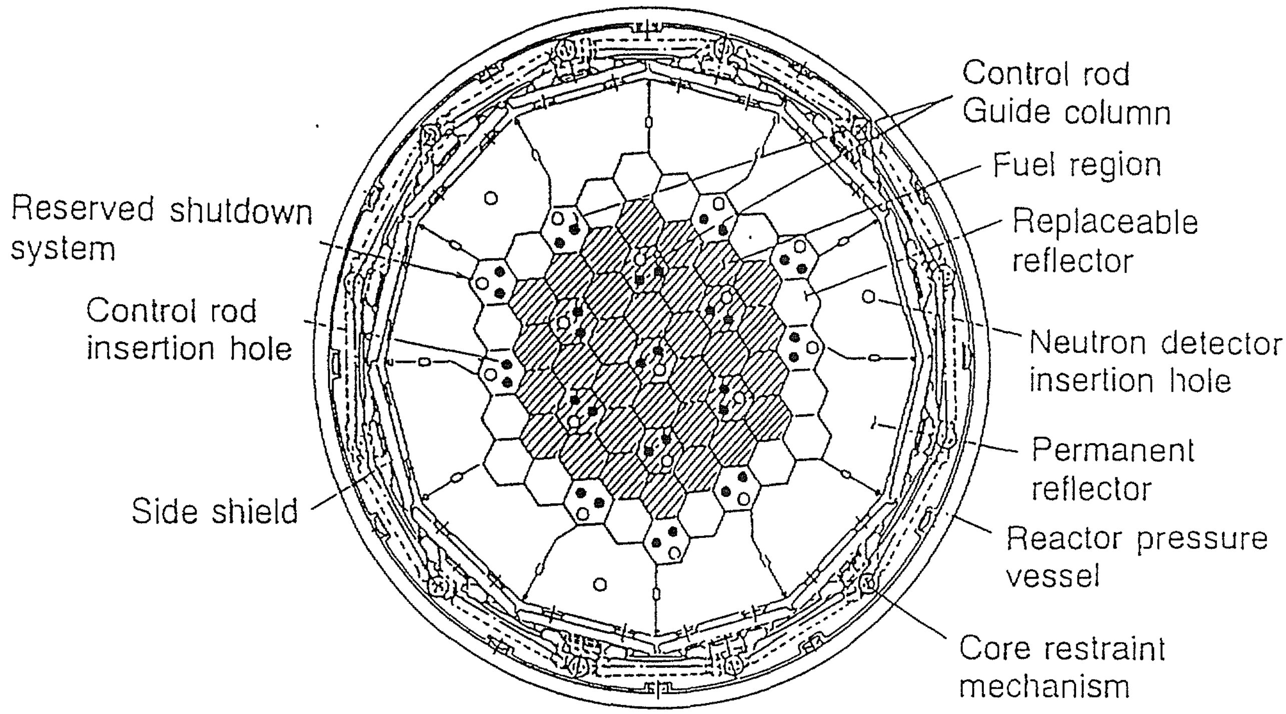


Fig.2 Horizontal cross section of the HTTR

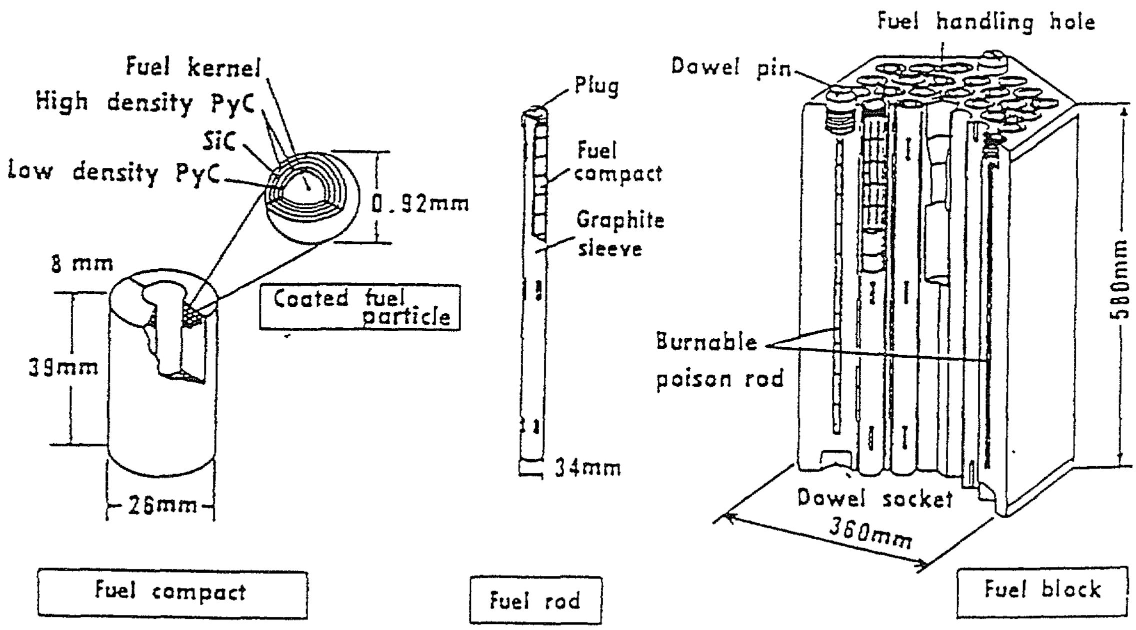


Fig.3 Block-type fuel of the HTTR

temperatures by operation of the primary He circulators (PGCs), and the tests during reactor operation at different power and temperature levels.

The tests at zero power level will not only be done with the fully loaded core. There is increased interest now in annular shaped cores in regard of future HTGRs since this core form allows higher reactor powers than cylindrical cores maintaining the typical passive safety characteristics of the modular HTGR concept. Therefore, the first fuel columns of the HTTR will be arranged in an annular way aligning column by column so as to close the circle. The first criticality is supposed to be reached before the circle is closed, so that, after its closure, the HTTR, at that intermediate stage, will be the first power HTGR with an annular core offering the unique opportunity to carry out critical measurements and to generate experimental data needed for the validation of still-to-improve computer models for annular core description.

3.1 First fuel loading and approaching criticality

Preparations, initial conditions: At first, the whole core consists of graphite dummy blocks. Temporary neutron flux instrumentation and a temporary neutron source are provided. For details, see Fig.4. The primary system contains He at normal atmospheric pressure and is not in operation; the core is at room temperature.

Procedure: Column by column, the dummy blocks are replaced by fuel assemblies in clockwise sense starting with fuel column No.1 (Fig.4). The approach to criticality is monitored by inverse multiplication measurements. The core is supposed to become critical with 12 columns loaded and the control rods (CRs) fully or nearly fully withdrawn. The first criticality is regarded as achieved when, by maintaining the just critical condition, the temporary neutron source has been removed.

After having established the number of fuel columns needed for first criticality in the described annular arrangement, more columns are filled with fuel, and after each 3 columns loaded, the respective reactivity increase is determined by IK and scram reactivity measurements. These measurements are not only carried out until the annular core is established (18 columns) but are, after the completion of the annular core physics tests described below, continued until all fuel is loaded (30 columns, Fig.4). (These reactivity addition measurements already belong, of course, to the critical measurements.)

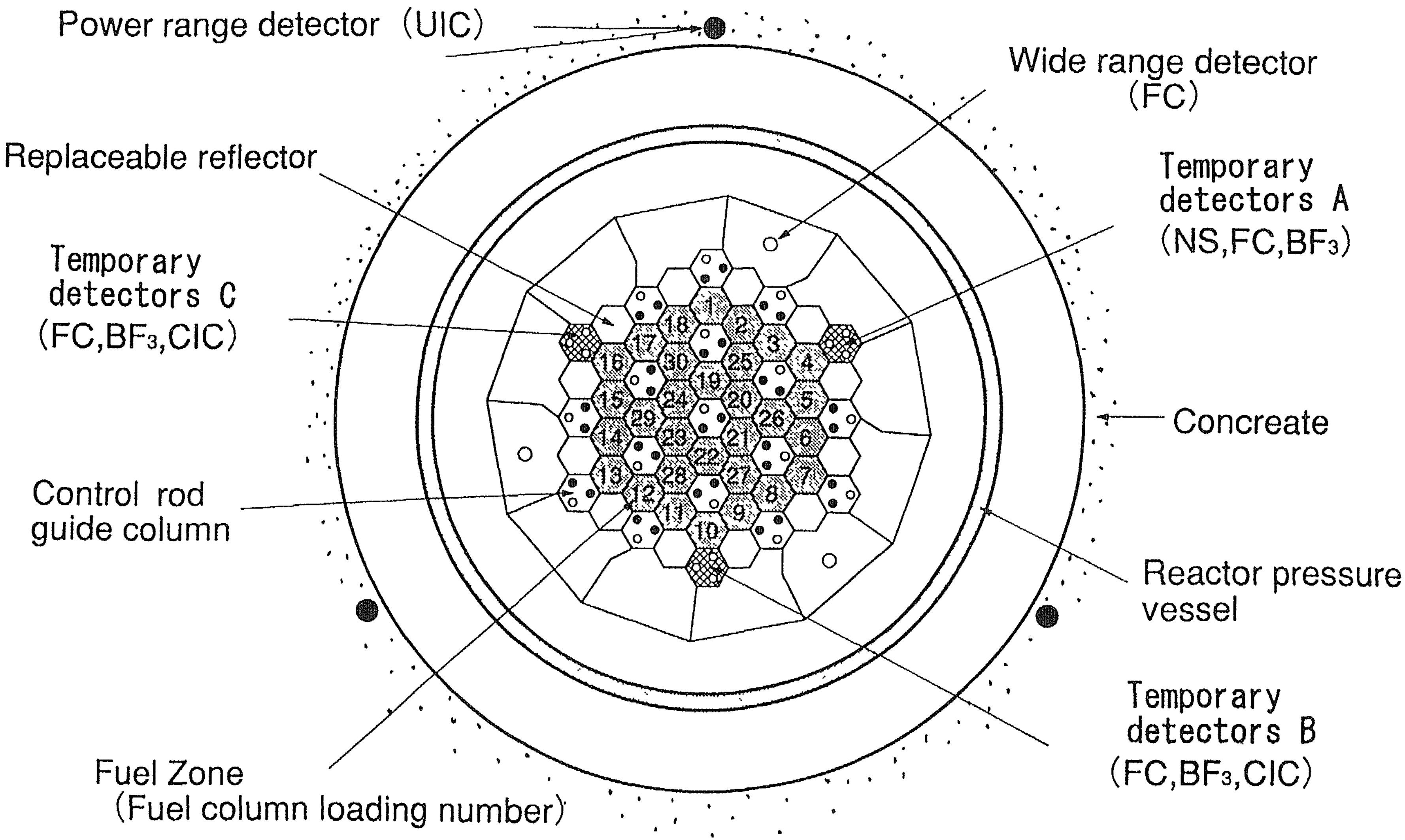


Fig.4 Loading numbers in criticality approach

3.2 Tests at zero power level with the fully loaded core

Although zero power tests with the annular core have, naturally, to be carried out before the core is fully loaded, this chapter regards the tests with the fully loaded core first because this concerns the initial HTTR core, and the measurements are more important for the HTTR, particularly in regulatory terms. Furthermore, the annular core tests are comprised in the tests described in this chapter and can, therefore, be regarded in less detail in the chapter below.

Preparations, initial conditions: For most of the tests described here the conditions are the same as described in 3.1. Only the measurement of the cold temperature coefficient of reactivity requires different conditions. This test needs the operation of the primary circulators (PGCs) as heat source to establish an isothermal core at various temperatures up to about 200°C. This conventional core heat-up can only be done with a pressurized primary system. Therefore, helium will be added and the pressure risen to about 1.7 MPa at room temperature. This value is also required for the below described rise-to-power tests. Before the primary system is pressurized, the temporary detectors will have to be removed and the system be appropriately sealed off.

Procedures: The tests at zero power are the central point of the HTTR start-up test programme and comprise various very different measurement methods. The following description of the procedures is therefore divided up into several subsections.

- Rod worth measurements with calculated corrections

Most of the zero power tests are CR reactivity worth measurements (see Table 1) carried out by the IK method. At first, integral worth curves for each CR (Fig.4) are constructed using an up-and-down method with another CR for reactivity compensation. All individual integral CR worth curves are added, giving a first integral CR bank worth curve. Systematical deviations in the individual CR worths caused by reactivity compensation and the deviation in the bank worth by just summing-up the individual results will be corrected for by calculated factors. The corrected integral rod worth curves and the known critical CR bank position are then used to evaluate the full reactivity control effect of the CRs, i.e., their full shutdown margin, the cold shutdown margin (one CR stuck) and the overall excess reactivity. (The excess reactivity is identical to the amount of reactivity bound by the CR bank in the cold core when inserted to the cold critical position.) The excess reactivity is also determined from the measurements during fuel loading, described in 3.1, using appropriate calculated correction factors. The obtained values are compared to the license requirements (Table 1).

Table 1 List of HTTR start-up physics tests

He atmosphere

Item	Purpose	Method	Note
1. Criticality tests Minimum core at room temperature			
1-1 First criticality test	For license	Inverse multiplication method	
1-2 Initial Core Construction test	For license	Measurement of added fuel reactivity by IK method and scram reactivity measurement	Performed from the first criticality to the complete fuel loading
2. Annular core tests 18 columns core at room temperature			
2-1 CR reactivity worth test	For research of annular core physics	Obtain CR worth curves by IK method	Evaluation of correction factor for interaction between CRs
2-2 Measurement of scram reactivity	Same as above	Rod drop method with delayed integral counting (new) and/or IK method	New method test
2-3 Excess reactivity test	Same as above	Fuel addition method and/or Absorption substitution method (from CR worth)	Evaluation of correction factor for interaction between CRs and sum-up effect
2-4 Neutron flux distribution measurement	Same as above	Measurement of axial neutron flux distribution inside reflector by temporary detectors	
2-5 Measurement test for kinetics parameter	Same as above	Reactor noise analysis	
3. Zero-power tests 30 columns core at room temperature			
3-1 CR reactivity worth test	For license	Measurement of CR worth curves by IK method	
3-2 Reactivity control effect test of CRs	For license >0.18 Δk/k	Summation of all CR worth obtained in 3-1	Without one pair of CRs Evaluation of correction factor for interaction between CRs
3-3 Measurement of maximum reactivity insertion rate	For license <2.4E-4 Δk/k/s	Evaluation from CR worth curves and CR drawing speed	
3-4 Excess reactivity test	For license <0.165 Δk/k	Fuel addition method	Evaluation of correction factor for interaction between CRs
3-5 Reactivity shutdown margin test	For license >0.01 Δk/k	Subtraction of value of 3-4 from value of 3-2	Without one pair of CRs
3-6 Measurement of scram reactivity	For research	As 2-2	
3-7 Neutron flux distribution measurement	For research	As 2-4	
3-8 Nuclear power correlation test	Adjustment of the wide range detectors	Reactor noise analysis and/or use of calibrated fission chamber	
3-9 Measurement test for kinetics parameter	Evaluation of β_{eff}/l	Reactor noise analysis	
3-10 Measurement of temperature coefficient of reactivity	For license <0	Reactivity measurement at various core temperatures	Core heated by primary circulator energy
4. Rise-to-power tests 30 columns core at various power and temperature levels			
4-1 Power coefficient measurement	For license <0	Measurement of CR position at various reactor power levels	
4-2 Reactivity Measurement for 2-step scram a) Measurement at first, reflector CR scram b) Subcriticality assessment in the period from first-step to second-step scram c) Measurement at second, core CR scram	For license (Research)	As 2-2	Scram shutdown procedure Scram of core CRs delayed until coolant outlet temperature < 750°C

CR . control rod pair

The maximum differential CR worth found for an individual CR is combined to the known CR driving speed, giving the maximum reactivity insertion rate (for license).

Also, the scram reactivity will be measured. For accuracy studies, both a rod drop evaluation method with delayed integral counting (details in Reference 1) and the IK method will be applied for the evaluation. The delayed integral counting method has been developed to correct systematical errors for long rod insertion times (like for HTTR).

- Low-deviation rod worth measurements

Rod worth measurements are bound to systematical deviations because of the necessary displacement of compensation rods. This is a major disadvantage in, e.g., the determination of the subcriticalily. In the above described methods, the problem is overcome by calculated corrections. To avoid these corrections and for comparison, major parameters like the cold shutdown margin, i.e., the subcriticality with all CRs inserted and the excess reactivity are also evaluated by different rod worth measurement techniques in which the deviation behavior is studied experimentally and low-deviation rod combinations are assessed. This method has been employed before. It uses simple flux shift considerations that, although only qualitative in principle but when backed by some investigating measurements, allow approximately to judge the deviation incorporated in a certain rod combination. For more details, see Reference 2.

- Axial flux distribution

For these measurements, the temporary fission chambers (FCs, with ^{235}U) inserted into the 3 outer irradiation columns (Fig.4) are moved along the columns and their signals are read at various positions. The CRs are withdrawn bank-wise.

- Reactor noise analysis

HTTR will be the first power HTGR at which reactor noise measurements will be carried out. These tests, common in LWRs, are difficult to evaluate in case of an HTGR because of the longer neutron lifetime l. However, by an oscillating movement of the control rods at low frequency etc., it is hoped that sufficiently clear noise signal can be generated and the reactor transfer function be obtained in high enough resolution.

As detectors, two of the temporary compensated ionization chambers (CICs, Fig.4) will be used and it will be tried to evaluate β_{eff} / l as well as the reactor power to obtain a first power calibration of the wide range neutron detectors (WRMS).

- Temperature coefficient of reactivity (cold)

These measurements are carried out as the last ones within the zero power tests because they need the pressurized condition. By appropriate operation of the PGCs the core will be made isothermal at about 70°C, 110°C, 150°C and 200°C. The reactivity difference between the various temperatures, including room temperature, is assessed by rod worth measurements. Here also, as described above, both a method with calculated correction and a low-deviation method will be applied. To determine the respective temperature difference, all reflector temperatures and the coolant inlet and outlet temperatures, measured with thermo- couples, are averaged.

3.3 Annular core tests

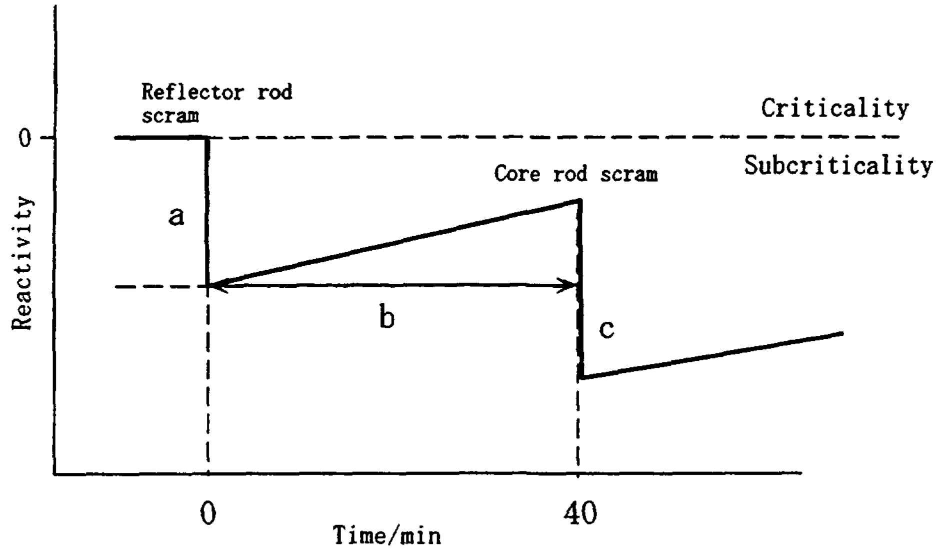
Most of the tests at zero power with the fully loaded core, described in the previous chapter, will, at an earlier stage, also be carried out with the annular core (see Table 1) under the same conditions. Since the primary system is still depressurized operation of the PGCs is practically useless so that no temperature coefficient can be measured.

3.4 Rise-to-power tests

These tests are mainly license-related. It will have to be assessed that there is a safe negative reactivity feedback for all reactor power and temperature levels between cold shutdown and high temperature operation (950°C), and that a scram at high temperature operation in the prescribed two-step CR insertion manner will result in a safe shutdown at all times.

Preparation, initial conditions: The primary system is pressurized, as already for the temperature coefficient measurements (3.2), and the coolant flow is set to the fixed value of 10.2 kg/s (prescribed value for high temperature operation). It will be kept constant at that level until the scram is triggered.

Procedures: At the fixed coolant flow the CR bank will slowly be withdrawn and the reactor power and temperature increased to 30 MW, 950 °C, respectively with several stationary intervals. The reactivity difference between the various power and temperature levels will be read from the calculationally corrected integral CR bank worth curve obtained in the zero power tests (3.2) and related to the respective differences in core power and coolant outlet temperature, thus obtaining values for the power and coolant outlet temperature coefficients of reactivity.



Reactivity evaluation

Section	a	b	c
Case 1	Rod drop method*	IK method	IK method
Case 2	IK method	IK method	IK method

* with delayed integral counting

Fig. 5 Reactivity evaluation methods for the 2-step-scram reactor shutdown procedure

Table 2 Major specifications of the HTTR

Thermal power	30MW
Core structure	Graphite
Core equivalent diameter	2.3m
Core height	2.9m
Average power density	$2.5W/cm^3$
Fuel	UO_2
Fuel type	Pin-in-block Coated Fuel Particle (CFP)
Fuel loading	off-load, 1 batch
Maximum fuel temperature	1492°C
Excess reactivity	15%Δk
Uranium enrichment	3 - 10wt%
(Average)	about 6wt%
Fuel burnup (average)	22GWd/t
Burnup days in a cycle	660days
Reactivity coefficients	
Fuel temperature coefficient	$-(1.5 \text{ to } 4.6)\times10^{-5}\Delta k/k/°C$
Moderator temperature coefficient	$(-17.1 \text{ to } 0.99)\times10^{-5}\Delta k/k/°C$
Power coefficient	$-(2.4 \text{ to } 4.0)\times10^{-5}\Delta k/k/MW$
Prompt neutron lifetime	0.67 - 0.70ms
Effective delayed neutron fraction	0.0047-0.0065
Inlet coolant temperature	395°C
Outlet coolant temperature	850°C/950°C
Power peaking factor	
radial	1.1
axial	1.7
Coolant	Helium gas
Primary coolant pressure	4MPa
Coolant flow in core	Downward
Effective core coolant flow rate	88%

At high temperature operation (30 MW, 950 °C) a scram will be triggered. At first, only the reflector CRs are dropped. The control system prevents the drop of the core CRs until either the coolant outlet temperature has fallen below 750°C or a time of 40 min has elapsed (CR protection). After the first scram, the normal core cooling procedure is followed, i.e., the PCS (primary cooling system) is shut down and the ACS (auxiliary cooling system) started up. The main outcome of this test is, as already mentioned above, the confirmation that the reactor is always subcritical. The scram reactivity of the reflector CRs will be evaluated. Also, it will be tried to evaluate the reactivity course between the two scrams as well as the reactivity of the second scram. For the evaluation of the first scram, again the two methods, as described above, will be employed. The other reactivity changes will be evaluated by the IK method (Fig.5).

4. Conclusions

Although to a great deal related to license requirements the HTTR start-up physics test programme is rather comprehensive and includes also some novel evaluation approaches. It will contribute to deepen insights into HTGR behavior and deliver important data for annular core description. In planning tests at power operation, however, some operational restrictions, like the requirement of equal rod withdrawal for reactor power levels >9MW and the fixation of the coolant flow to the full-power operation requirements (850°C or 950°C), had to be observed. It is hoped that, after a first successful operation period of the HTTR, these regulatory restrictions can be reviewed and by that some more interesting testing made available.

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STUDY OF A HIGH TEMPERATURE GAS COOLED REACTOR HEAT UTILIZATION PLANT

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Abstract

A number of nuclear power plants have been successfully constructed and operating in Japan. The nuclear-generated electricity is expected to be increasing constantly and to account for 42% of total electricity supply in FY 2010, which is now about 30%. Since about 40% of the primary energy supply is consumed for the electricity production in Japan, the nuclear energy would account for only 20% of the primary energy supply even if the nuclear-generated electricity could account for 50% of the total electricity supply. In order to preserve the global environment and to secure the stable energy supply, it is most effective to increase the use of the nuclear energy. However, considering the situation described above, if the nuclear energy is applied only to electricity generation, the effect is limited. Therefore, it is necessary to utilize the nuclear energy to wide filed other than the electric power generation. This is very important especially in Japan where most of the energy supply depends on imported fossil fuels and in the developing countries where the energy demand is increasing rapidly.

From this point of view, FAPIG organized a working group named "HTR-HUC" (HTR Heat Utilization Core Group) in order to study the heat utilization of HTGR in the field other than electric power generation. In this study, we considered that the nuclear heat is utilized most effectively in three fields: in replacing fossil fuel, in producing clean fuels from fossil resources and in producing hydrogen from water as the ultimate clean energy carrier. We selected the following three types of plants, ① Co-generation Plant : a chemical plant with the modular HTGRs of the outlet temperature of 700℃,

② Coal Gasification Plant : a coal gasification ammonia production plant with the modular HTGRs of the outlet temperature of 950℃, ③ Hydrogen Production Plant : two types of the hydrogen production plants (the thermo-chemical process and the high temperature steam electrolysis) with modular HTGRs of the outlet temperature up to 1050℃.

For each plant, a system outline and basic plan as well as costs, environmental effects and management methods were studied. We have presented a paper on the same topic at the Technical Committee Meeting at Petten on November 28 to 30, 1994. At that time, we have mainly explained on the outline of the plant we have chosen for the study. Now our study has been completed and we would like to present the results of study including additional information we made since the last presentation. We will present, in addition to the plant concepts, the results of studies on costs, environmental effects and operational and organizational aspects.

1. Introduction

Japan has one gas-cooled reactor and is going to have another gas-cooled reactor. One is the Tokai Commercial Power Plant (166MWe), which has been in operation for about 30 years, and the other is the High Temperature Engineering Test Reactor (HTTR), which is under construction and will reach the first criticality in autumn, 1997. FAPIG joined in the construction of both plants and has been supporting the maintenance of the Tokai plant. Through these experiences, FAPIG is confident of the excellent features of gas-cooled reactors, especially the modular HTGRs.

The modular HTGRs are promising candidate for next generation reactors, because they can supply high temperature energy up to 1000℃, which allows it to cover over 70% of industrial requirements, and they have various excellent safety features. So FAPIG organized the HTR-HUC Working Group and has been investigating the introduction of this type of reactor into industry, especially in process where heat is needed.

2. Selection of the reference plants

We started this work by getting information on energy consumption in large energy consuming plant from the member companies of FAPIG to allow us to make realistic model plant for the study.

Based on the information obtained and the prospects of the energy supply in future, we selected the following three types of plants for our study.

(1)Co-generation plant : to supply electricity and process steam to industries in place of conventional fossil fueled power plants.

(2)Coal gasification plant : to reform coal, which is expected to provide long term supply, into cleaner fuels or raw materials.

(3)Hydrogen production plants : to produce hydrogen, which is expected as the ultimate clean energy carrier, from water using thermo-chemical process and high temperature steam electrolysis

As the reference reactors, we chose the 450MWt MHTGR[1] for steam cycle and the 170 to 200MWt HTR-module[2] for high temperature process heat.

3. Co-generation Plant

We selected a chemical plant as a reference co-generation plant because it requires large amount of energy of 200℃ to 500℃. Based on the data of an existing plant, we decided the capacity of the chemical plant. The capacity of the plant was modified slightly to match that of the MHTGR co-generation system.

The system diagram of the total plant and the heat/mass balance in the steam-turbine system are shown in Fig.1 and Fig. 2. The main specifications of the nuclear reactor system are shown in Table Ⅰ.

Table Ⅰ. Main Specifications of Reactor System for Co-generation

Item	Specification
Type of Reactor	MHTGR
Thermal Output of Reactor	450MW×2
Inlet & Outlet Coolant Temperature	288/704℃
Coolant Pressure	7.09MPa
Steam Temperature	540℃
Steam Pressure	17.8MPa
Power Generated	117MWe×2

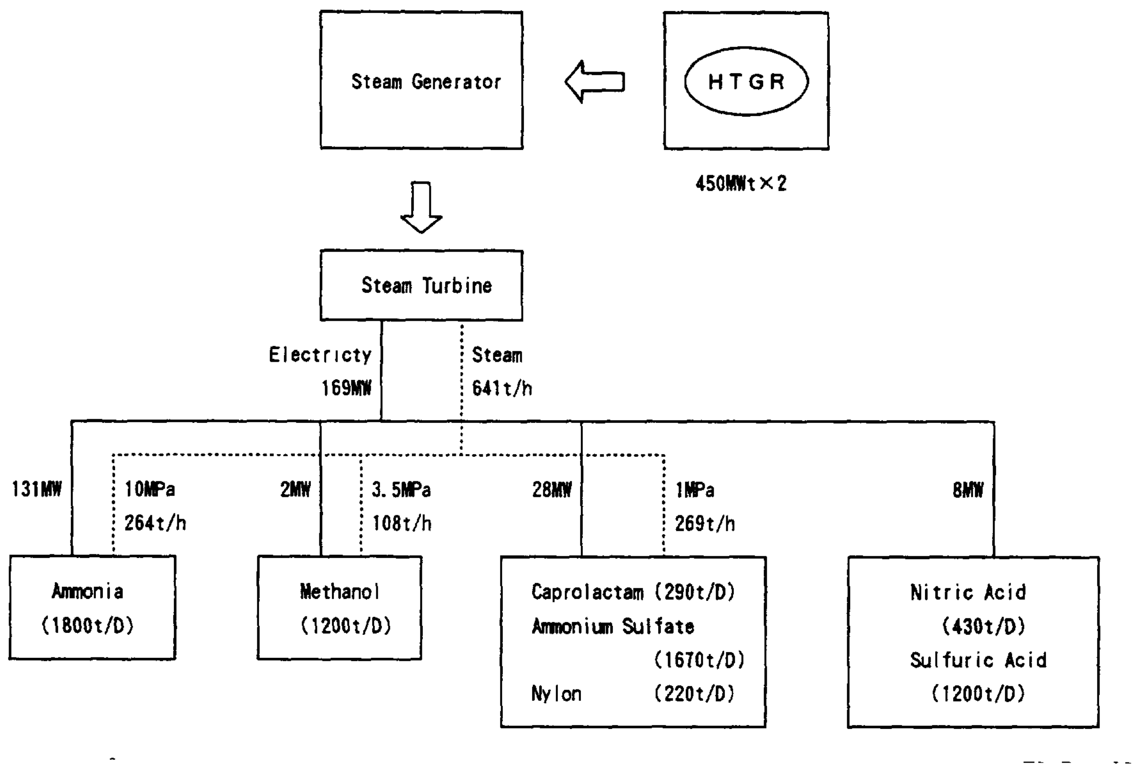

Fig. 1. System Diagram of HTGR Co-generation Chemical Plant

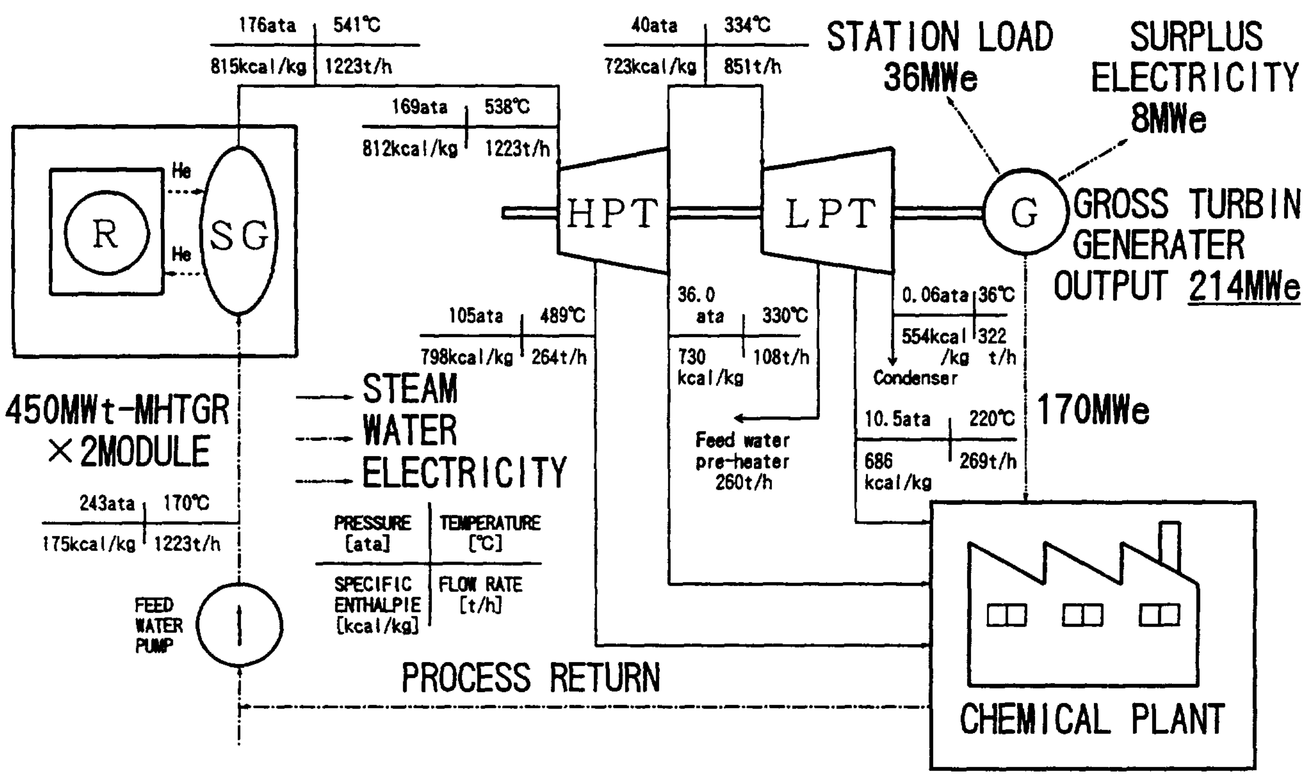

Fig. 2. Heat/Mass Balance of HTGR Co-generation Chemical Plant

Fig. 3 shows the plot plan of this plant. An area of 800m×600m is required for the total system. A bird's-eye view of the plant is shown in Fig. 4.

4. Coal Gasification Plant

We chose ammonia as the product of the coal gasification plant. The coal is gasified by a steam gasification process with a fluidized bed gasification furnace with a gasification temperature of 750℃, and the product gas is supplied to an ammonia plant with a capacity of 1500 ton/day. The most important subject to be discussed is how to use helium enthalpy most effectively for coal gasification. We adopted a two-step method of gasification in which a pre-reforming fluidized bed uses as much of the helium energy as possible for gasification.

The main specifications of the nuclear system are shown in Table Ⅱ. The system flow diagram is shown in Fig. 5. The main specifications of the ammonia synthesis system are shown in Table Ⅲ.

Fig.6. shows the plot plan of this plant. An area of 430m×420m is required for the total system. A bird's-eye view of the plant is shown in Fig. 7.

Table Ⅱ. Main Specifications of Reactor System for Coal Gasification

Item	Specification	
Type of Reactor	HTR-M	
Thermal Output of Reactor	170MW×4	
Inlet & Outlet Coolant Temperature	300/950℃	
Coolant Pressure	4.05MPa	
Intermediate heat exchanger		
	Primary	Secondary
Helium Flow Rate (kg/s)	50.3	47.3
Inlet & Outlet Gas Temperature (℃)	950/292	200/900
Inlet & Outlet Gas Pressure (MPa)	4.04/4.02	4.44/4.25

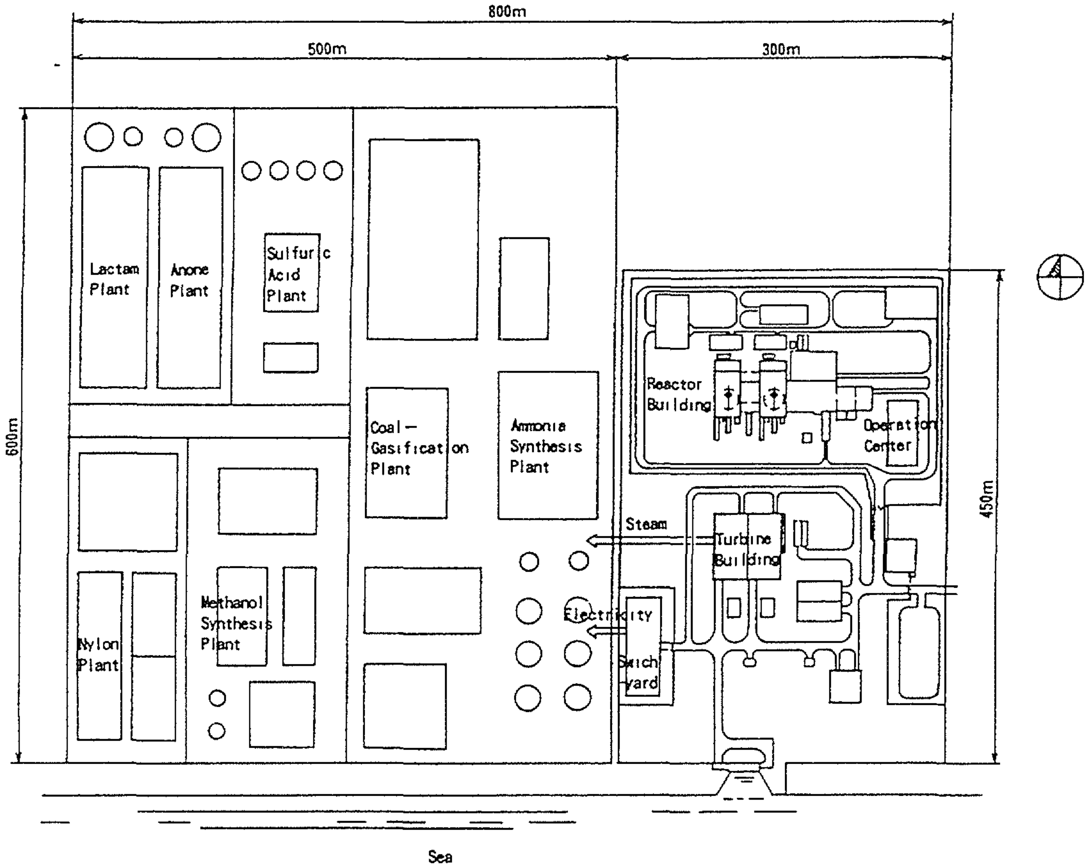

Fig 3 Plot Plan of HTGR Co-generation Chemical Plant

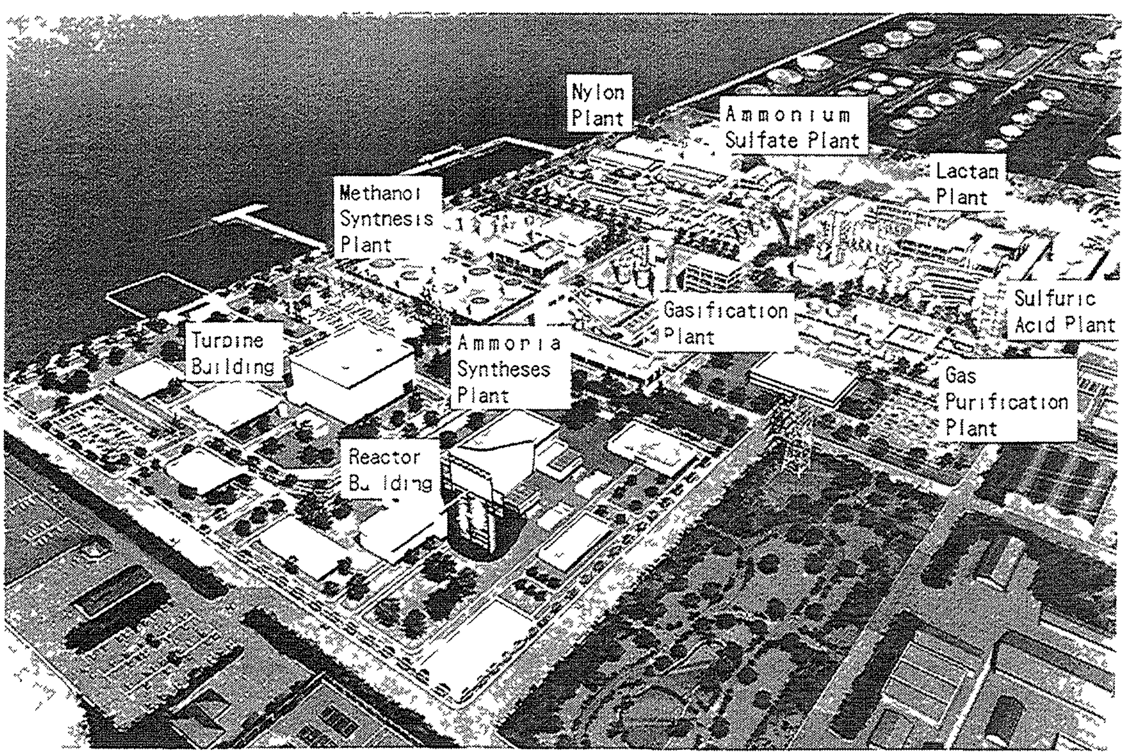

Fig 4 Bird's-eye View of HTGR Co-generation Chemical Plant

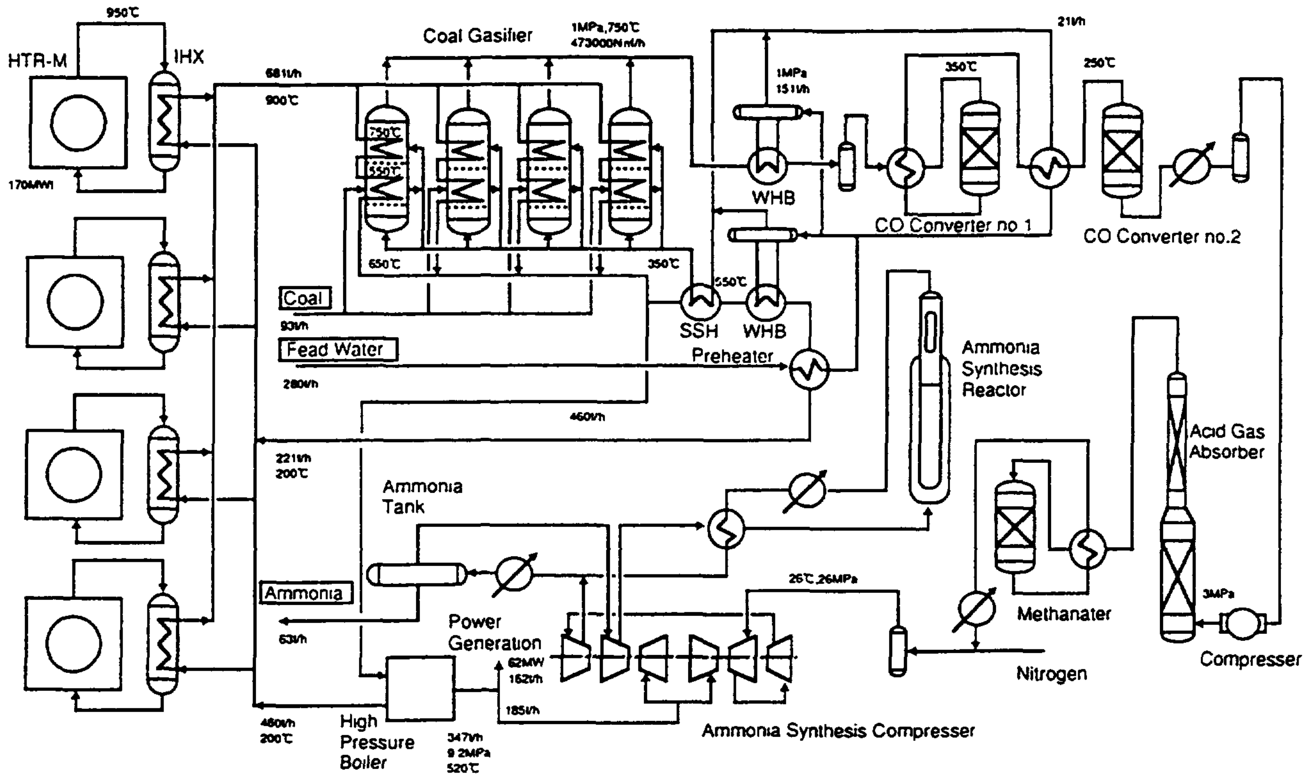

Fig. 5. System Flow Diagram of HTGR Coal Gasification Ammonia Production Plant

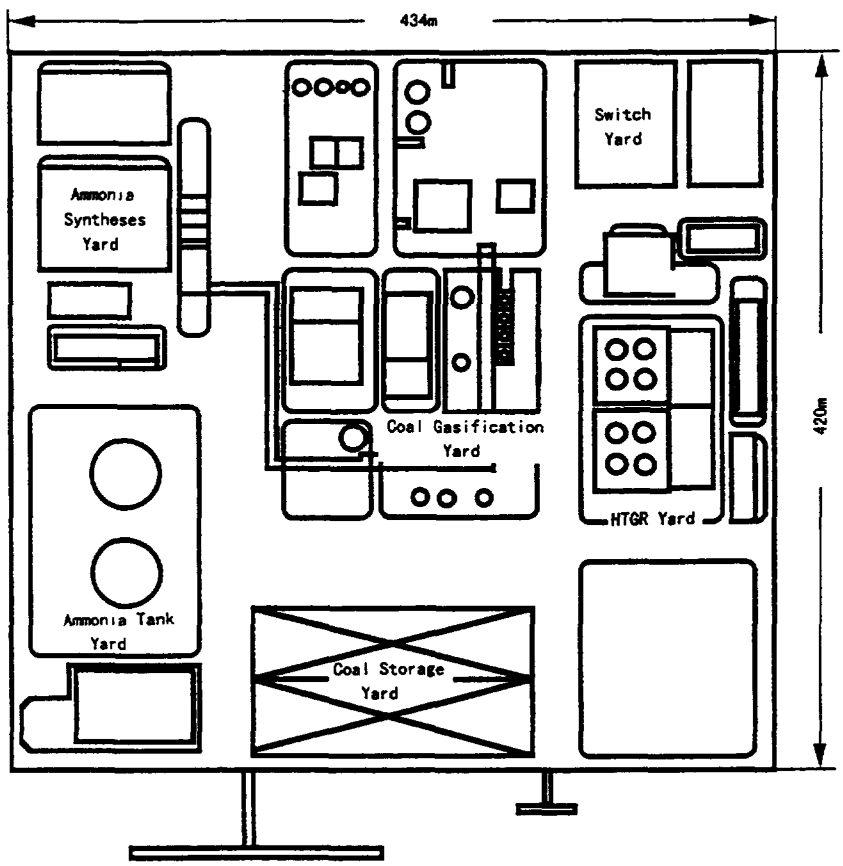

Fig. 6. Plot Plan of HTGR Coal Gasification Ammonia Production Plant

Table III. Main Specifications of Coal Gasification Ammonia Production System

Item	Specification
Ammonia Production Rate	1,500t/d
Coal Gasification Rate	2,200t/d
Number of Gasifiers	4
Type of Gasifiers	Fluidized Bed
Helium Temperatures	
at Gasifier (inlet/outlet)	900/650℃
Gasifier Operating Temperature	750℃
Gasifier Operating Pressure	1.01MPa
Steam Carbon Ratio	3

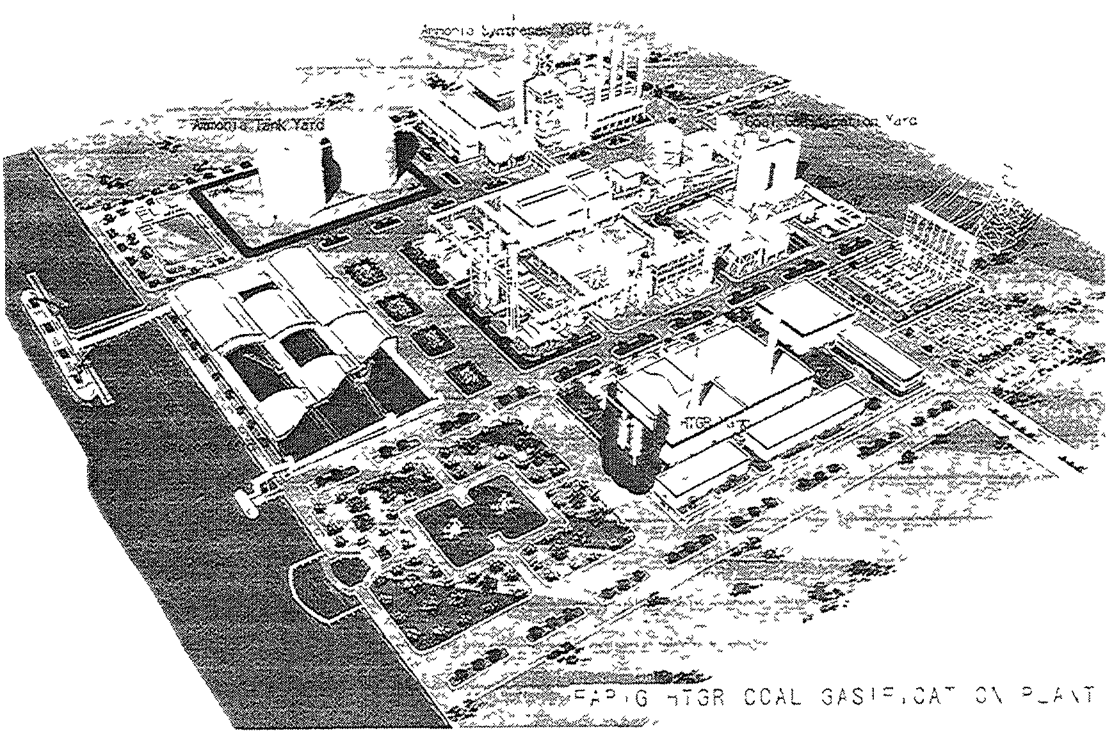

Fig. 7. Bird's-eye View of HTGR Coal Gasification Ammonia Production Plant

5. Hydrogen Production Plant

A hydrogen production system using the nuclear heat from HTR-Ms has been studied. This study is based on the state-of-the-art engineering expertise expected to exist 30 years from now. In this study we selected the Iodine-Sulfur (IS) process and the high temperature steam electrolysis process.

The IS plant is composed of four nuclear reactors and one IS process, which is connected to the reactors through IHXs, because of the complexity of secondary systems such as chemical reactors. Fig. 8 shows the flow diagram of the IS plant.

The high temperature steam electrolysis plant is composed of four nuclear reactors, four high temperature steam electrolysis processes and one gas turbine power generator. Fig. 9 shows a block diagram of this system.

Table IV shows the main specifications and performance of the IS and high temperature steam electrolysis hydrogen production plant.

Table IV. Main Specifications of Hydrogen Production plant

Item	IS Process	High Temperature Steam Electrolysis Process
Nuclear Reactor (units)	4	4
Thermal Power (MWt)	800	800
Hydrogen Production Process	IS Process	Electrolysis Cell
	1 System	4 System
Combined Process using Low Temperature Heat	SG 1 unit	Gas turbine 1 unit
Hydrogen Production Rate (Nm^3/hr)	37,090 (20,400)*	110,000
Electric Power (MWe)	50.8	21.4
Hydrogen Production Efficiency (%)	40 (22)*	49
Plant Thermal Efficiency (%)	34	51

*numbers in parenthesis are deducted by laboratory results

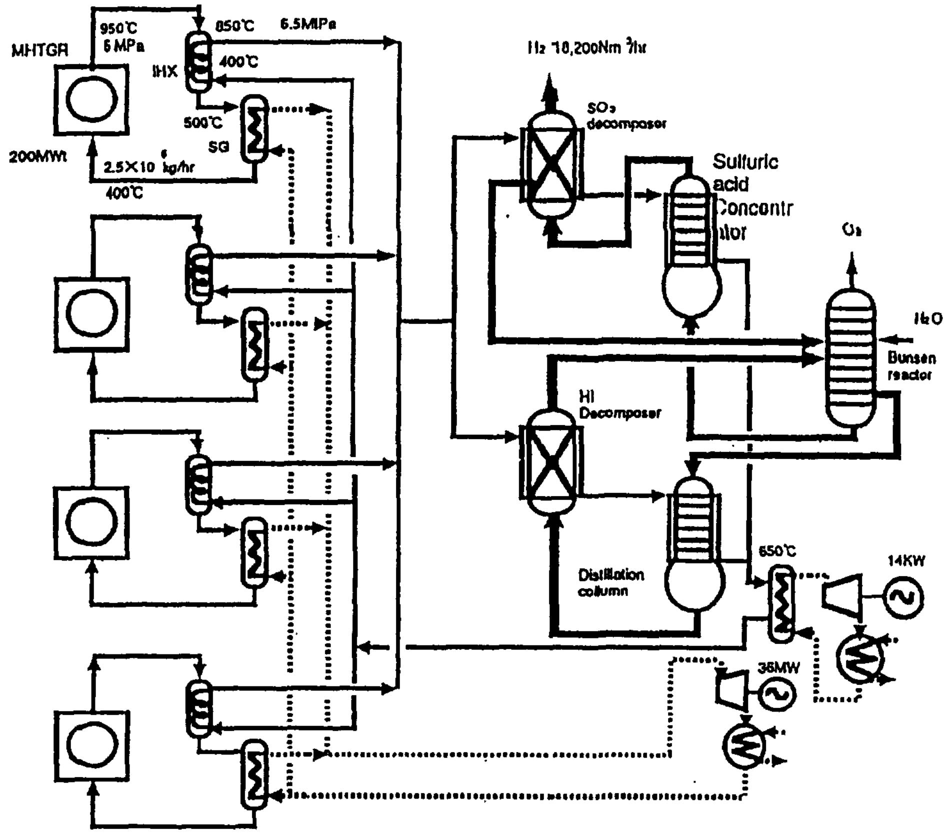

Fig. 8. System Flow Diagram of IS process for Hydrogen Production

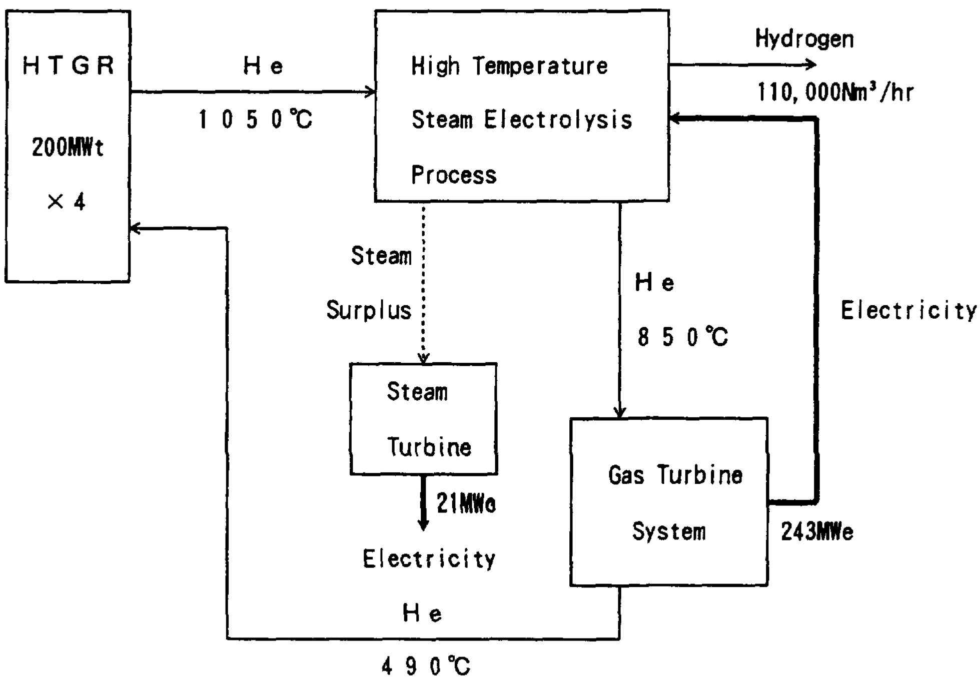

Fig. 9. Block Diagram of High Temperature Electrolysis Hydrogen Production Plant[3)]

6. Cost of Power Generation

Plant construction costs as well as the cost of generating power with HTGR has been studied under Japanese conditions. For this study we selected a steam cycle MHTGR for power generation, because the detailed date necessary for cost estimation are available only with steam cycle MHTGR at present.

The overnight construction cost of the standard 4×450MWt NOAK plant with 4 turbines was estimated to be 336 billion yen or 486 thousand yen/kWe in 1994. The generation cost was calculated using the parameters shown in Table V. The results of those cost estimation are also shown in the table.

We roughly estimated the electricity and steam cost based on the cost estimate described above. Fig. 10 shows the cost allocation to electricity and steam for the reference co-generation chemical plant. It can be seen from the figure that the price of steam will be about 5.7 yen/Mcal when the price of electricity is 9 yen/kWh.

Table V. Parameters Used to Estimate Generation Cost

Item	Specification
Net Electrical Output	693MWe
Net Thermal Efficiency	38.5%
Capacity Factor	84%
Amortization Period	16 years
Fixed Expenses	9.47%
O/M Cost Rate	4.15%
Fuel Cycle Cost Rate	2.53%
Fuel Burnup	80,000MWD/t
Decommissioning Cost	27.5 billion yen
Generation Cost	10.8 yen/kWh

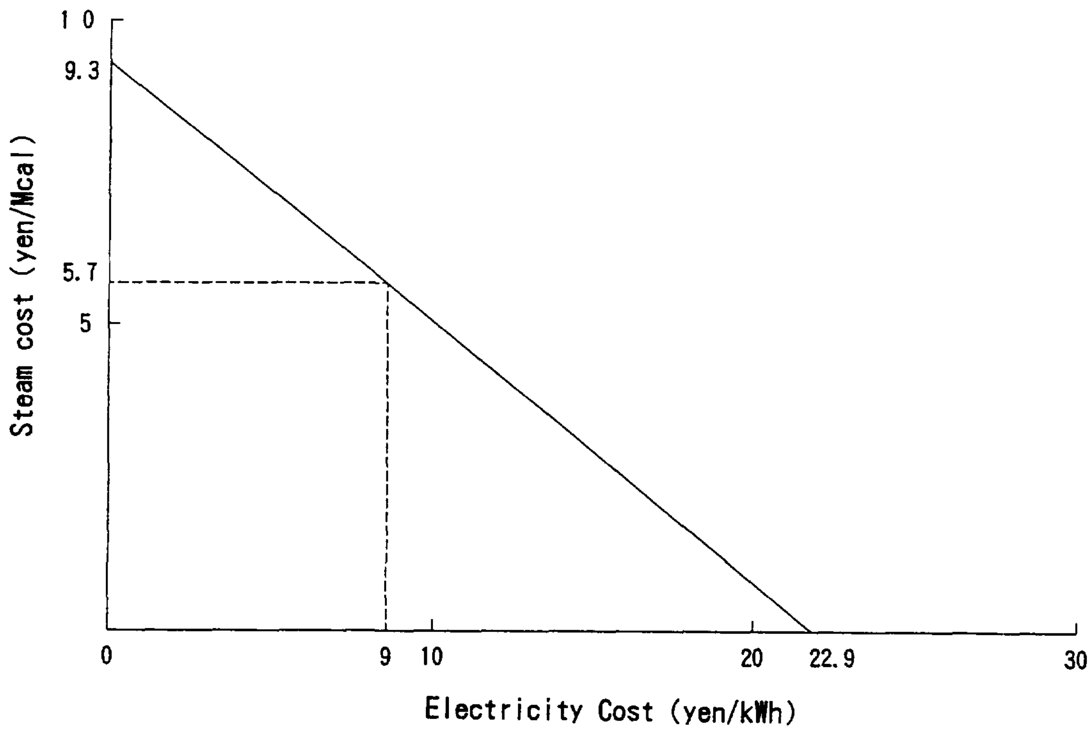

Fig.10. Cost Allocation to Electricity and Steam for HTGR Co-generation Chemical Plant

7. Management of Construction and Operation

The most realistic way to apply MHTGR in industry for the first time is, we think, in the form of replacing the existing power boilers of an industrial plant with an HTGR of equal capacity.

Because the construction and operation of a reactor requires more sophisticated technology and economics than those requires for conventional power units, it will be essential to utilize existing knowledge and experience on the nuclear power plants. We believe it would be better to organize a joint operating company with the user, the electricity generating company, the local community and the reactor vender. This company should constructed, own and operate the HTGR plant.

8. Global and Environmental Effects

Nuclear reactors are the only established, large energy source that is nearly CO_2 free even taking the energy used for manufacturing and disposing spent fuels etc. and the associated CO_2 release into consideration. HTGRs are outstanding because of their inherent high efficiency and extreme safety. HTGRs generate less thermal disturbance in the environment and opens the way to the diversified use of nuclear energy. This, in turn, opens the way for developing countries and smaller utilities or industries to select a reactor as an energy source. This means that HTRs can contribute to widening the range of available energy sources, to conserve the environment and preserve fossil fuels.

In the case of the reference co-generation plant, replacing the coal fired boiler of the chemical plant with 2×450MWt MHTGRs with steam generators will allow about 160 ton/h of coal to be saved and will reduce CO_2 emissions by about 2.6 million ton/year. For the reference coal gasification ammonia production plant, the amount of CO_2 emission is reduced by about 2.1 million ton/year than that from the current coal fired gasification plant (with a partial combustion process).

9. Conclusion

The used of the heat of HTGRs was studied in three fields of application: co-generation, coal gasification and hydrogen production. We studied the cost of HTRs for power generation and co-generation and found the possibility of competitiveness with LWRs in small medium sized power plants. Our results have convinced us HTGRs have possibility of the reasonable competitiveness in the market for heat for industrial use. Because HTRs can provide various industries with high temperature nuclear heat, we believe that the development of HTR will benefit the prosperity of the world.

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TEST PROGRAMMES OF HTTR FOR FUTURE GENERATION HTGRs

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Abstract

Test programs of the High Temperature Engineering Test Reactor (HTTR) for future generation High Temperature Gas-Cooled Reactors (HTGRs) have been established considering design and development status of HTGRs in Japan and the world. Test programs are divided into six categories, thermal hydraulics, fuel, safety, high temperature components, core physics and control- instrumentations. All programs are related to the technology of future generation HTGRs and will be submitted to a new Coordinated Research Program (CRP) so that all participants from the world in test programs of the HTTR can use measured data for their future generation HTGRs.

This paper describes test programs of the HTTR for the development of future generation HTGRs after explanation of a future generation HTGR in Japan.

1. Introduction

A High Temperature Gas-cooled Reactor (HTGR) is expected to be of prime importance for broadening nuclear energy use in the future because it would supply high temperature heat and have high thermal efficiency, outstanding inherent safety characteristics and high fuel burn-up. Design and development of future generation HTGRs have been conducted in many countries. Recently, in South Africa, Netherlands and China, practical design works have been performed for the development of commercial HTGRs. In Japan, the High Temperature Engineering Test Reactor (HTTR) at Oarai Research Establishment of Japan Atomic Energy Research Institute (JAERI) is at a final stage of construction, and after its first criticality in October 1997, various kinds of tests under high temperature conditions will be conducted for future generation HTGRs. In addition to the construction of the HTTR, design studies of future generation HTGRs have been started.

HTGRs have several advantages in safety concept. An Advanced Light Water Reactor (ALWR) is designed based on passive safety concept and accident managements to prevent or mitigate severe accidents causing core melt down. On the other hand, HTGRs can achieve the same or higher degree of safety using simpler passive systems and prevent or minimize risk of severe accidents because of their salient inherent safety features. In a severe accident free HTR (SFHTR)[1], high degree of safety is achieved due to 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. No severe accidents occur even in the worst scenario.

However, in previously designed HTGRs, safety is confirmed by their safety analyses and probabilistic safety assessments. In contrast, L. M. Lidsky in Massachusetts Institute of Technology proposed demonstarable safety [2]. Demonstrable safety is essential for public acceptance and licensing because the public has little trust in probabilistic safety assessments, and experience in HTGR history is not enough to convince the regulatory of trusting these safety assessments. Safety must be demonstrated by nearly full-scale worst case test. In the SFHTR, nearly full-scale worst case tests (safety demonstration tests) will be carried out to obtain licensing before commercial operations.

The High Temperature Engineering Test Reactor (HTTR)[3] is the best reactor to demonstrate passive safety systems designed for HTGRs such as a vessel cooling system (core cavity cooling system) work well and to confirm safety demonstration tests can be carried out in the SFHTR. Also various tests concerning fuels, high temperature components for future generation HTGRs will be conducted. In this paper, first a conceptual design of the SFHTR is described and then test programs of the HTTR for future generation HTGRs are presented in section 3 except a test program for core physics presented in another issue in this conference[4].

2. Conceptual design of SFHTR

2.1 Safety concept of SFHTR

There are two types of SFHTRs, modular SFHTR (MSFHTR) and small SFHTR (SSFHTR). Both SFHTRs are designed based on the same safety concept. Severe accidents are defined as any conditions beyond designed base accidents causing core damages with the large amount of fission product releases to the environment although all severe accident sequences are very low in probability. The safety concept of the SFHTR is to avoid or minimize the risk of severe accidents at least two orders lower than current reactors. In the SFHTR fuel temperature rise exceeding its limit by complete loss of forced coolant (depressurization accident) without scram or control rod complete withdrawal can be avoided because of large heat capacity of the core and low power density. To prevent the risk of fuel oxidation, the development of no-burning fuels is inevitable, and several laboratory scale tests have been started in JAERI. However, in the first design stage of the SFHTR a confinement containing inert gas is installed to prevent fuel oxidation during depressurization accidents.

Furthermore, nearly full-scale worst tests will be carried out to obtain licensing before commercial operations. Fig. 1 shows the difference between the licensing procedure of the HTTR or current reactors and the SFHTR.

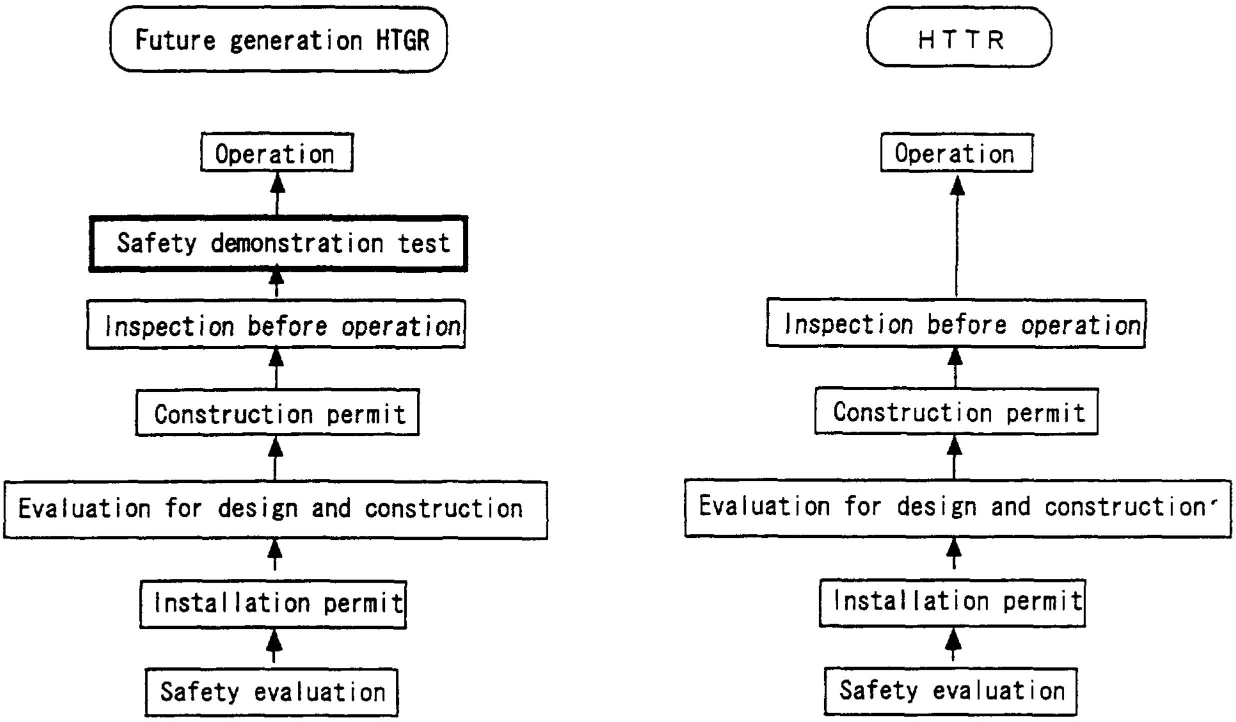

Fig. 1 Licensing procedure for future generation HTGR and HTTR

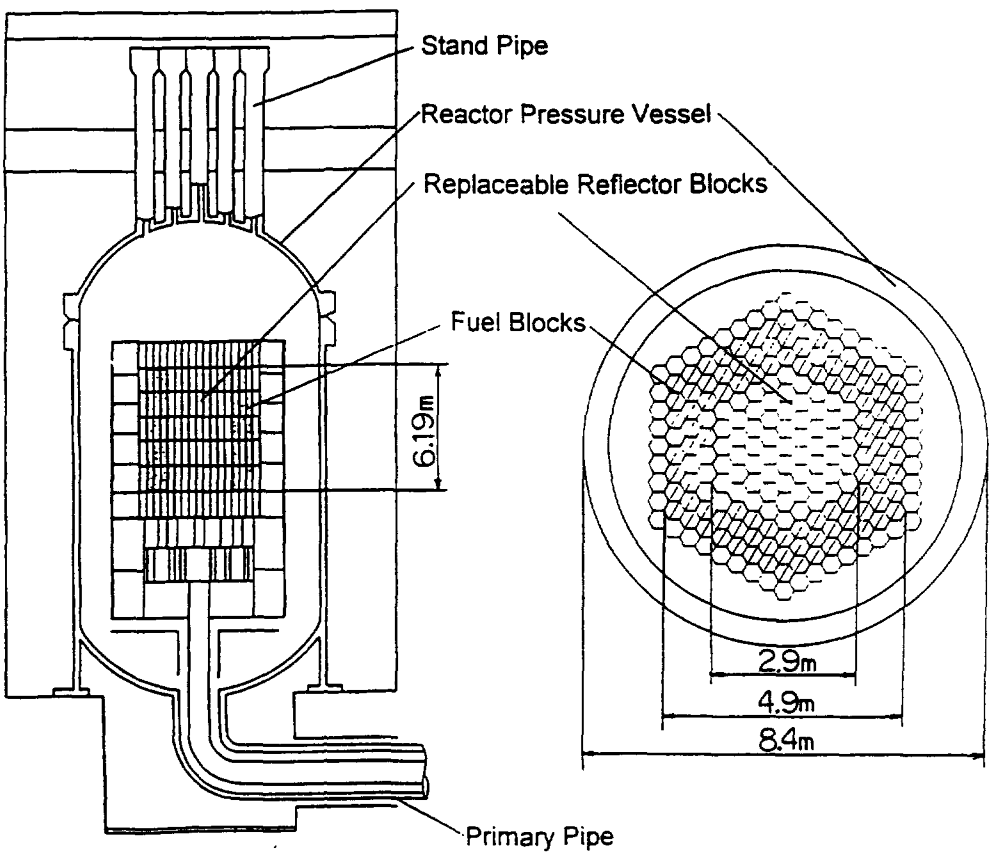

Fig. 2 Vertical and horizontal cross sectional views of the modular SFHTR

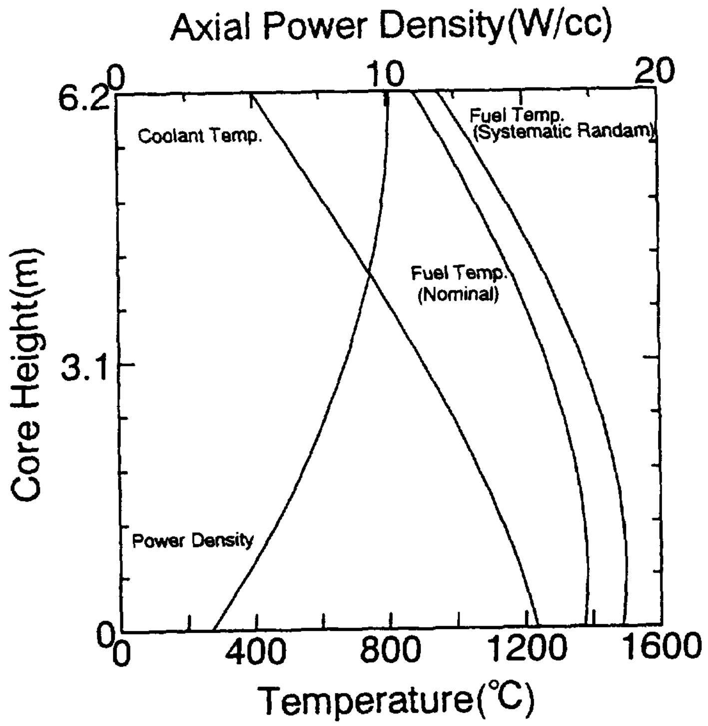

Fig. 3 Fuel temperature distribution of the MSFHTR

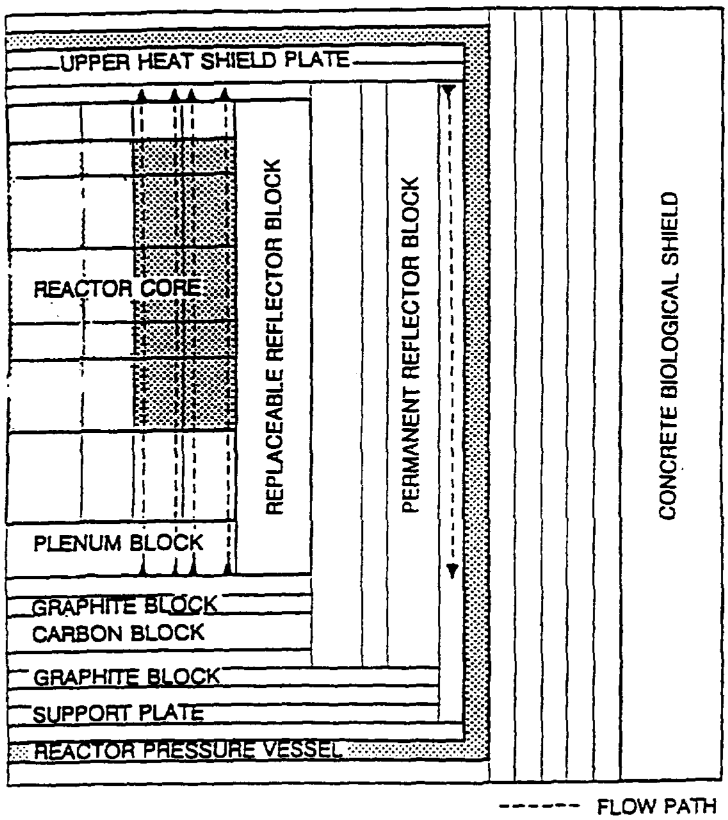

Fig. 4 Analytical model for depressurization accident by TAC-NC

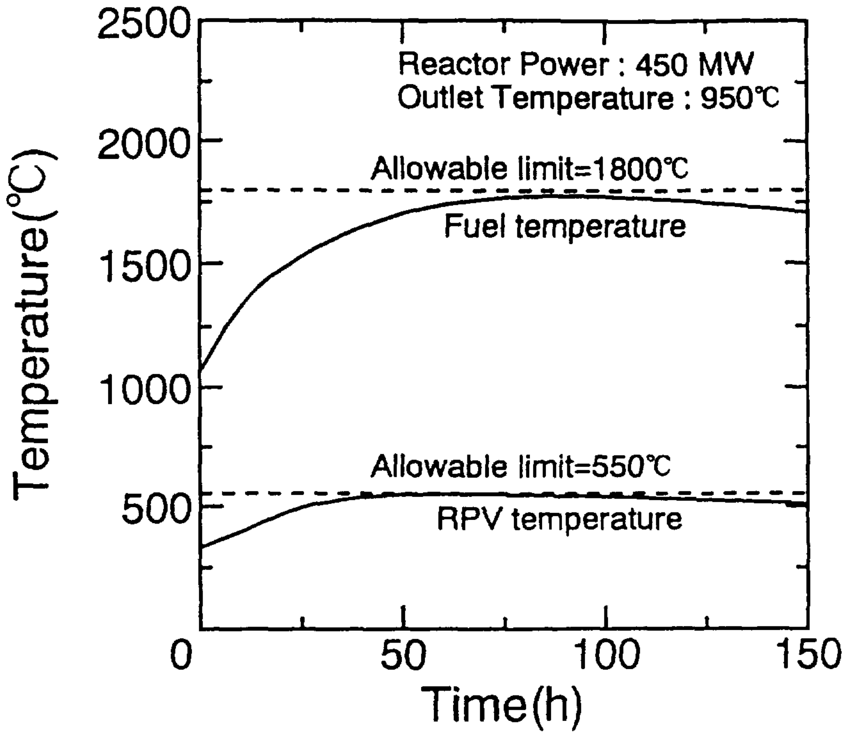

Fig. 5 The maximum fuel temperature transient during depressurization accident

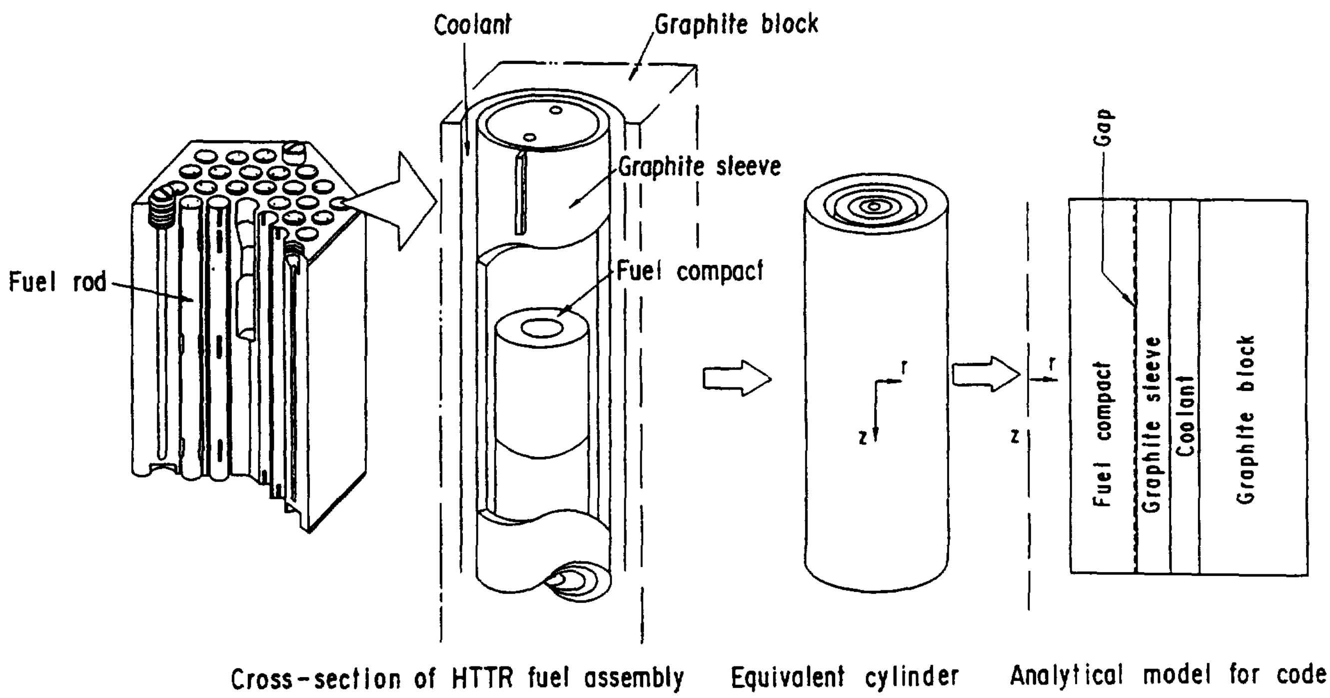

Fig. 6 Analytical model for reactivity addition accident by BLOOST-J2

Table 1 Major Specification of modular and small SFHTR

	Modular	Small
Thermal Output	450MWt× 4 Unit (Commercial)	Max. 200MWt
Outlet Gas Temp.	850-950°C	
Inlet Gas Temp.	350-550°C	
Helium Gas Press.	6.0 MPa	
Fuel block	Prismatic Block	
Fuel	Low Enriched Uranium	
Power density	Average 6 w/cc	2.5 w/cc
Fuel coating material	ZrC	
Burn-up	100GWd/t	150GWd/t
Allowable Temp.	1800°C	
Refueling	Axial shuffling Refueling every year	1-2 times during life time
Internal Graphite	ISO-63	
Control rod clad material	C/C composite	
Pressure Vessel	9Cr-1Mo-V	
Safety System	Complete Passive Cooling	
Confinement	Inert gas	
Utilization System	Gas Turbine or Hydrogen production	Gas Turbine & desalination & district heating
Plant Life Time	40 years	

2.2 Design study of SFHTR

2.2.1 Modular SFHTR

2.2.1.1 Core design

Table 1 shows major specification of the MSFHTR. The core consists of 108 columns (five blocks in each column), and annular arrangement (see fig. 2) is adopted to prevent fuel temperature increase during depressurization accidents. A pin-in block type fuel (see fig.6), in which diameter of fuel rods, the number of fuel rods, width of the flow path around fuel rods etc. are modified from those of the HTTR, is piled up inside the core.

Nuclear design codes DELIGHT[5] and CITATION[6] verified for the HTTR design are used to obtain power distribution and fuel burnup. The fuel temperature is calculated using the power

Table 2 Fuel burnup after 5 years operation of modular SFHTR

Items	Inner ring	Intermediate ring	Outer ring
1st fuel block (top)	65	53	51
	72	57	56
2nd fuel block	117	94	93
	116	92	92
3rd fuel block	131	107	106
	123	98	98
4th fuel block	133	108	108
	123	98	93
5th fuel block(bottom)	138	111	111
	131	104	105

Unit (GWd/t)

Table 3 Fuel burnup after 15 years operation of small SFHTR

Items	Center	Ring 1	Ring 2	Ring 3
Top fuel block	52	45	29	26
2nd fuel block	119	119	91	91
3rd fuel block	154	161	131	138
4th fuel block	172	184	159	167
5th fuel block	154	164	139	150
6th(bottom)	141	149	123	130

Unit (GWd/t)

distribution obtained from nuclear analyses by these codes. Fig. 3 shows the calculated maximum fuel temperature distribution during normal operations. The maximum fuel temperature is lower than the temperature limit of 1500°C during normal operations. Table 2 shows the fuel burnup after operations. Average burnup is more than 100GWd/t because of axial shuffling[7]. In the axial shuffling, the fuel blocks and top reflector blocks in a column are removed by a refueling machine. The fuel block in the core bottom is removed from the core and a new fuel block is set at the top. A fuel cycle is five years and 1/5 of total fuel blocks are refueled.

2.2.1.2 Safety evaluation

In the HTTR safety evaluations, various analyses were carried out to obtain fuel temperatures, boundary temperatures, boundary pressures and so on. According to the results of the HTTR safety evaluations, the two worst events that have a possibility to cause fatal fuel failure are selected in the safety analyses for the SFHTR. They are a depressurization accident characterized by a primary pipe rupture, and an abrupt reactivity addition by a stand pipe rupture.

(1) Depressurization accident

The two-dimensional heat transfer code TAC-NC [8] is used for analyzing temperature transient during depressurization accidents. An analytical model for the core, internal structure, Reactor Pressure Vessel (RPV) and so on shown in Fig. 4. The major features of the analytical model are as follows;

1) Two-dimensional time dependent heat conduction equation is solved.

2) Radiation and convection heat transfers are also considered.

3) Heat transfer by natural circulation can be solved.

Fig. 5 shows the maximum fuel temperature transient during the depressurization accident. It does not exceed the temperature limit of 1800°C.

(2) Reactivity addition accident

The core dynamics code BLOOST-J2 [9] is used for analyzing fuel temperature transient during this accident. Fuel elements and blocks are represented by an equivalent cylindrical single channel shown in Fig. 6. The major features of the analytical model are as follows;

1) A one point kinetics equation with six groups of delayed neutron precursors is solved in the core dynamics.

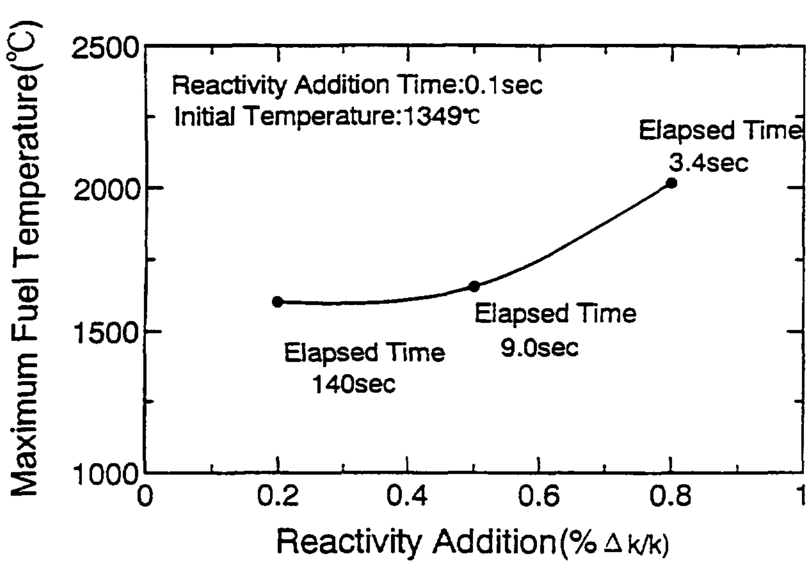

Fig. 7 Relationship between reactivity and maximum fuel temperature

2) Two-dimensional time-dependent heat conduction equation is solved.

3) One-dimensional quasistatic heat transfer equation is solved for the primary coolant.

4) Heat generation due to nuclear fission reaction in the UO_2 fuel kernel of a coated fuel particle is assumed to be generated uniformly in the fuel compact because the heat generated in the fuel kernel is conducted instantly to the graphite matrix around the fuel particle.

The analysis is carried out as a parameter of the reactivity addition. The reactivity is assumed to be added within 0.1 seconds. Fig. 7 shows the relationship between the maximum fuel temperature and the reactivity addition.

The analysis proved that even when 0.7 %Δk/k of reactivity is added, the fuel temperature does not exceed the limit temperature. The excess reactivity of the MSFHTR is designed not to exceed the 3-5%Δk/k, and 10 pairs of control rods are installed in the core. Therefore, if a pair of control rods is withdrawn from the core, the maximum reactivity addition is 0.3-0.5%Δk/k and the fuel temperature is approximately 1660°C.

2.2.2 Small SFHTR

The small SFHTR(SSFHTR) is designed to be appropriated for a remote area not having electrical network. Its thermal power is at most 200MWt and used for dedicated applications such as desalination, district heating as well as electricity generation. Table 1 shows the major specification of the SSFHTR. The most important issues to be considered in designing the small SFHTR is that it must be modularised easily and constructed in a shorter time than larger plants, thus reducing its capital cost, however it has higher safety and severe accidents will never occur. The problem of block type HTGRs such as the HTTR is that they need a large size complicated refueling machine. It raises capital cost and makes availability low. Therefore, a fuel cycle of the small SFHTR is designed to be more than 15-20 years with a discharged burnup of 150MWd/t so that each plant need not its own refueling machine. In our preliminary design study of the SSFHTR with 50MWt, it is confirmed that a fuel cycle of 17 years is possible. Table 3 shows fuel burnup after operations calculated by the DELIGHT and CITATION. The maximum power density is kept lower than 9w/cc in this case so that the maximum fuel temperature does not exceed the temperature limit. More than 20 years are feasible in the final design after optimizing the amount of burnable poison (BP), location of BP, fuel enrichment and so on. In the next year, safety evaluations during depressurization and reactivity addition accidents without scram will be carried out to confirm the safety of the SSFHTR.

3. Test programs of HTTR for SFHTR and future generation HTGRs in the world

3.1 Objectives

Test programs of the HTTR for future generation HTGRs have been established considering design and development status of HTGRs in Japan and the world. Test programs are divided into six categories, safety, thermal hydraulics, fuel, high temperature components, core physics and control-instrumentations. All programs are related to the technology of future generation HTGRs and will be submitted to a new Coordinated Research Program (CRP) so that all participants from the world to the HTTR tests can use measured data for their future generation HTGRs.

3.2 Safety

In the safety demonstration tests [10] by using the HTTR, the postulated worst two events in the SFHTR are simulated. They are a loss of forced cooling accident (depressurization accident) and control rod withdrawal accident simulation. The safety demonstration tests will be started in 1998 in order to confirm the safety concept of the SFHTR and the other future generation HTGRs.

3.2.1 Loss of forced cooling accident simulation(partly licensed)

This test simulates conduction cooldown behavior during depressurization accidents. In this test, circulators for a primary cooling system stop at a rated power operation, and no active direct core cooling system starts after that. During depressurization accidents in the SFHTR, the natural circulation occurs in the RPV and transfers heat from the core to outside. On the other hand, this test cannot simulate the natural circulation because a guillotine break of the primary pipe cannot be simulated. However, the flow rate of the natural circulation is low in low pressure condition and heat transfer by the natural circulation is limited. It is already confirmed that the maximum fuel temperature between the depressurization accident with and without the natural circulation is not different in the HTTR safety analysis.

3.2.2 Control rod withdrawal accident simulation(partly licensed)

This test simulates the fuel temperature rise by reactivity addition. In a preliminary test, a pair of control rods is withdrawn from the core by a control drive mechanism. In a secondary test, a capsule containing a reactivity absorption material such as B_4C ball is installed in the center of the core and immediately ejected from the core.

The maximum fuel temperatures during both simulation tests are measured by the temperature monitoring elements installed in the fuel blocks.

3.3 Passive heat removal system

In HTGRs, even in depressurization accidents and no direct core cooling is expected, the core temperature does not exceed its temperature limit. A passive heat removal system (Vessel Cooling System (VCS) in the HTTR) installed around the RPV removes the heat from the RPV by radiation and natural convection and keeps the core temperature lower the temperature limit (see fig. 5).

In the HTTR, the amount of heat transfer from the RPV to VCS will be measured in 0, 30, 50, 100% of rated power condition and transient condition after a reactor scram. RPV and VCS temperatures will be also measured to confirm that there is no hot spot on the RPV and to evaluate how much amount of heat is transferred by radiation or by natural convection.

In addition to above tests, a plan to change the water cooling VCS to a complete passive air cooling VCS has been studied for tests in future. The best system to cool the reactor core and keep the fuel temperature lower than the limit will be selected after these tests.

3.4 Fuel

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

(1) Initial fuel performance test

Performance of the first loading fuel of the HTTR will be examined by fractional releases of gaseous fission products.

(2) Fuel temperature measurement test

Fuel temperature will be measured by melt wires. The validation of the temperature calculation codes is carried out by comparison of the measured and calculated temperatures.

(3) Fission product plateout measurement test

The release fractions and plateout distribution of condensable fission products will be investigated by the measurement of plateout distribution in the primary cooling system of the HTTR.

(4) Organic iodine measurement test

Fractions of organic iodine in the HTTR will be measured by sampling of the primary coolant as a function of coolant temperatures during the power up test. Based on measured data, mechanism of the organic iodine formation in the HTGR system will be investigated.

(5) Dust sampling test

Dust in the primary coolant will be source term in depressurization accidents of the HTGR. Amount, composition and size of dusts will be measured and basic data will be accumulated during the start up test of the HTTR.

After tests for initial loading fuels, tests for ZrC coated fuel specimen will be conducted.

3.5 Thermal hydraulics

As for thermal hydraulics, a total code system to evaluate temperatures of fuel and internal core structures will be developed and validated in the HTTR. This code system would be used to design future generation HTGR in the Japan and world. It includes steady-state fuel temperature evaluation code (FLOWNET[11], TEMDIM[12]), in-core transient heat transfer analysis code (TAC-NC[8], SSPHEAT[13]) and plant dynamic code (ACCORD[14]). During operations, direct comparison between experimental and analytical fuel temperatures cannot be conducted because fuel temperature is too high to be measured. However, inside and outside temperatures of permanent reflector blocks around the reactor core, inlet and outlet helium gas temperature, helium gas temperatures in hot plenums, etc. will be measured and these experimental data will be useful for validation of fuel temperature evaluation codes. After operations, temperature monitors installed in fuel blocks will be removed, and analytical and experimental fuel temperatures will be directly compared.

Also dynamic tests such as step and ramp change of thermal power, reactor scram and so on will be carried out to validate plant dynamics codes and in-core transient analysis code.

3.6 Control-instrumentation

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

3.7 High temperature components

An intermediate heat exchanger (IHX) was newly developed for the HTTR. Thermal performance tests in steady state condition will be carried out to obtain heat transfer coefficient in the tube side, heat transfer coefficient on the shell side, over all heat transfer characteristics, temperature of manifold, heat transfer tube, radiation promoter effect, etc. Also in transient condition, temperature of manifold, heat transfer tube, etc. will be measured, and stress and strain evaluation using real temperature will be conducted to investigate deterioration of thermal performance after the first high temperature operation to prolong life time.

4. Summary

In the HTTR, various tests will be performed for the development of future generation HTGRs in the world and Japan. Test programs and test results are open to the world and are extremely important to the HTGR development because the other HTGRs are not operating except the HTR-10 in China will be operated in the near future. Especially, it is important to confirm safety demonstration tests are feasible using reactor itself and demonstrable safety concept of the SFHTR can be accepted for licensing of future generation HTGRs.

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PRESENT STATUS OF INITIAL CORE FUEL FABRICATION FOR THE HTTR

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Abstract

This paper describes present status of initial core fuel fabrication for the HTTR in Japan.

The High Temperature Engineering Test Reactor (HTTR) is being constructed satisfactorily, aiming the first criticality in October 1997. Fuel fabrication for the HTTR was started in June 1995 at Tokai works of Nuclear Fuel Industries, Ltd. (NFI). The fabrication technologies have been developed through cooperation works of NFI and Japan Atomic Energy Research Institute.

The initial core fuel of the HTTR consists of 150 fuel assemblies. An HTTR fuel assembly is so-called a pin-in-block type of hexagonal graphite block containing 31 or 33 fuel rods. A fuel rod consists of a graphite sleeve and of 14 fuel compacts. In a fuel compact about 13,000 TRISO coated fuel particles are dispersed densely.

Fabrication of the fuel is being carried out at NFI's facility, and a half of the initial fuels have been completed. The results of inspection for the size, the density and the sphericity of coated fuel particle, and the as-manufactured particle failure fractions in fuel compacts, which are most important inspection items from the view point of quality assurance, are quite satisfactory.

Transportation and assembling of the fuel rods will be carried out at the reactor site from the middle of 1997.

1. Introduction

The High Temperature Engineering Test Reactor (HTTR), which is the first High Temperature Gas-cooled Reactor (HTGR) in Japan, is now being constructed at the Oarai Research Establishment of Japan Atomic Energy Research Institute (JAERI) and the first criticality is planned to be attained in October 1997. After the completion, it will be the only HTGR under operation in the world. Also, it will be the first HTGR that will demonstrate the high temperature nuclear heat utilization.

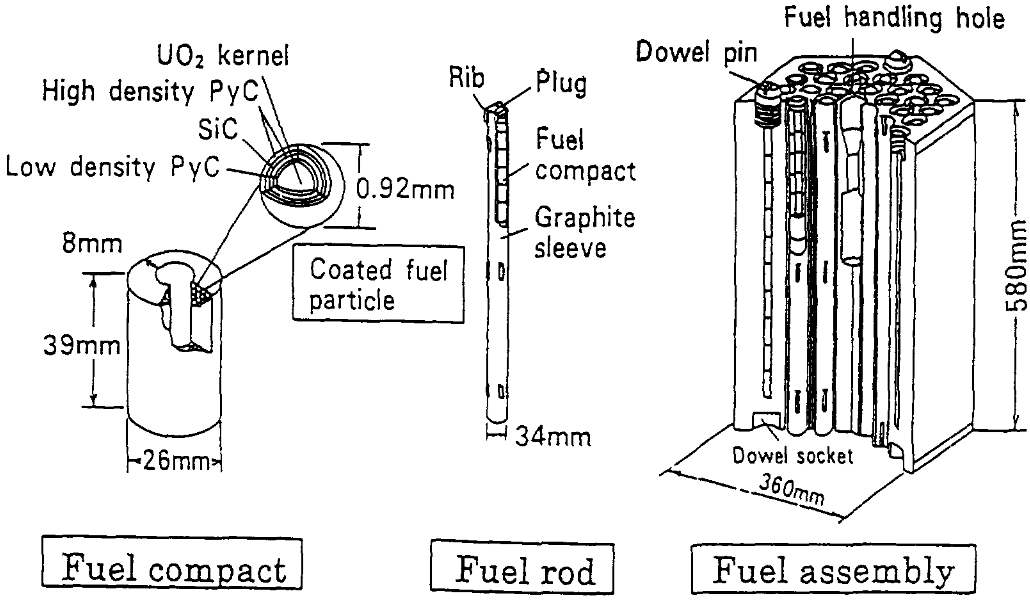


Fig. 1 The HTTR fuel configuration

Nuclear Fuel Industries, Ltd. (NFI) started the fabrication of initial core fuel for HTTR in June 1995, and has finished about a half amount of fabrication of the total fuel. The fabrication of all fuels, about 900 kgU fuels, will be completed in the end of 1997.

2. Fuel Fabrication

2.1. Fuel Design[1]

The initial core fuel of the HTTR consists of 150 fuel assemblies. An HTTR fuel assembly is so-called a pin-in-block type of hexagonal graphite block containing 31 or 33 fuel rods. A fuel rod consists of a graphite sleeve and of 14 fuel compacts. In a fuel compact, about 13,000 TRISO coated fuel particles are dispersed densely. The fuel configuration is shown in **Fig.1**.

The fuel temperature is limited below 1495℃ under normal operation conditions and below 1600℃ under abnormal transient condition. The fuel burnup is limited to be 3.6% FIMA as for the HTTR initial core fuel. The safety design criteria for the initial core fuel are as follows.

(1) The initial failure fraction in the coating layers of the fuel fabrication shall be regulatorily less than 0.2% in terms of the sum of heavy metal contamination and SiC defects, while the expected one is less than 10^{-4}.

(2) The coated fuel particles shall not fail systematically under normal operating conditions, that is, in the safety analyses,

(a) The penetration depth of the Pd/SiC interaction shall not exceed the thickness of SiC layer of 25 μm, because the full-penetrating Pd/SiC interaction is thought to lead loss of fission fabrication retention in SiC coating layer.

Table 1. Major specification

Item	Specification
UO_2 Kernel	
Diameter	600 ± 75 μm
Density	> 10.3 g/cm³
O/U ratio	1.99 ~ 2.02
Sphericity	< 1.2 (over 90%)
Coated Particle	
Diameter	920^{+100}_{-80}
Thickness	
1st Layer	60 ± 36 μm
2nd Layer	30 ± 12 μm
3rd Layer	> 17 μm
4th Layer	45 ± 20 μm
Density	
1st Layer	1.10 ± 0.15 g/cm³
2nd Layer	1.85 ± 0.10 g/cm³
3rd Layer	> 3.18 g/cm³
4th Layer	1.85± 0.10 g/cm³
OPTAF	
2nd & 4th layer	< 1.04
Compact	
Failure Fraction	
Through-Coating	<1.50 x 10^{-4}
SiC Layer	<1.50 x 10^{-3}
Dimension	
Outer diameter	26.0±0.1mm
Inner diameter	10.0±0.1mm
Length	39.0±0.5mm
Fulfillment ratio	30 ± 3 vol%

(b) The distance of kernel migration shall not exceed the sum of thickness of the first layer and the second layer thickness, 90 μm, to avoid failure of SiC layer.

(3) The fuel shall be designed so as to maintain its intactness even in consideration of irradiation-induced damage and chemical attack through the full service period, that is, the additional failure fraction in coating layers of fuel particles shall be regulatorily less than 0.2% through the full service period.

(4) The coated fuel particle shall maintain its intact below 1600℃ at any anticipated transient.

The fuel specifications of the initial core fuel are as shown in **Table 1**.

2-2. Process

UO_2 kernels are fabricated by the well-known gel precipitation process. Uranyl nitrate solution mixed with additives is dropped into ammonium solution to form ammonium diuranate (ADU) spheres. After washing and drying, these are calcinated and sintered into UO_2 kernels. UO_2 kernels are coated in a fluidized bed to form TRISO coated particles. All UO_2 kernels and TRISO coated particles are classified by vibrating table to exclude odd shape particles. TRISO coated particles are overcoated with graphite powder and binder resin mixture, and hot-pressed to make up green fuel compacts. The green compacts are baked in a furnace to be fuel compacts. Fourteen fuel compacts are inserted in a graphite sleeve to make a fuel rod. Then fuel rods are assembled into graphite blocks. The Sleeves and the blocks are made of IG-110 grade graphite.

Fabrication process of the HTTR fuel is shown in **Fig. 2**.

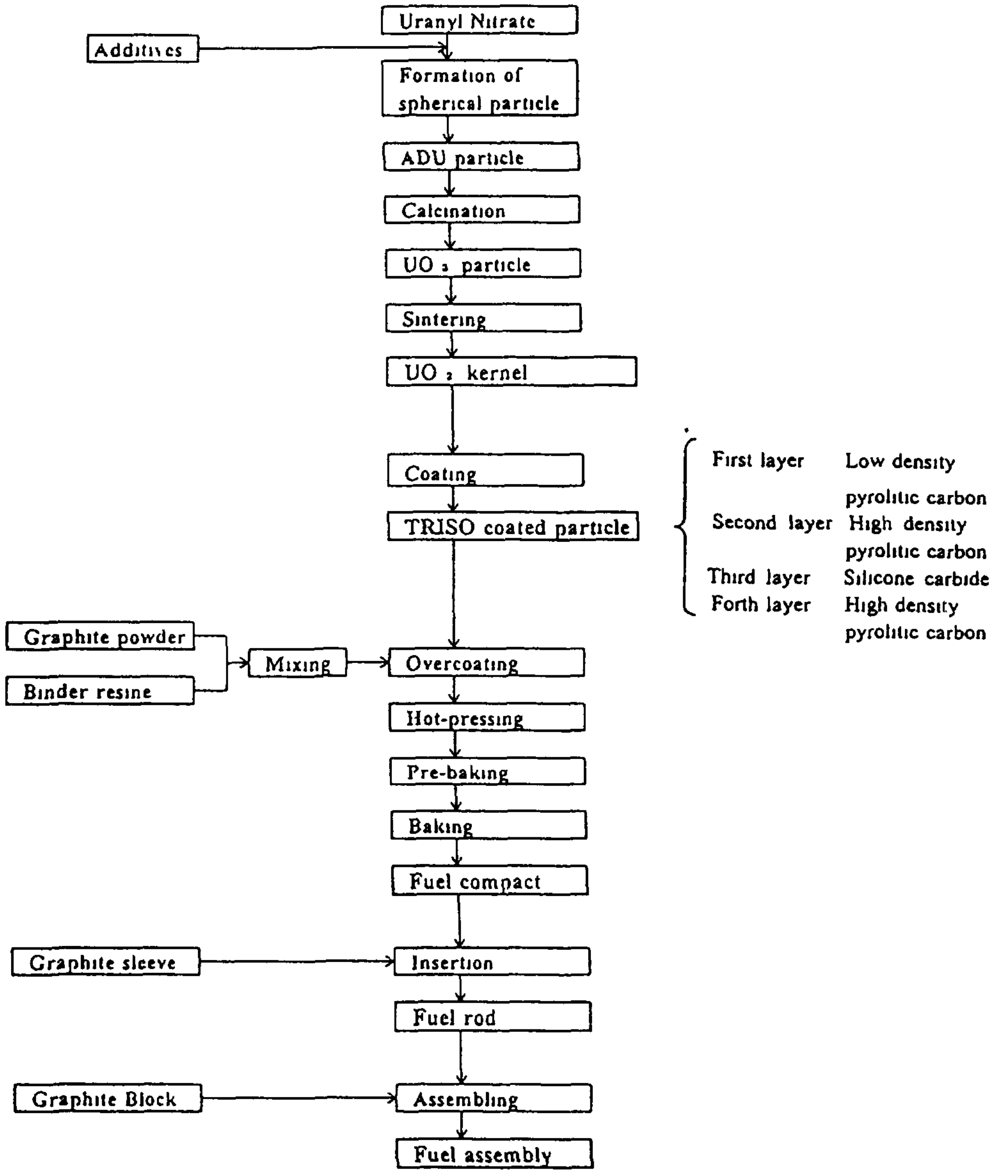


Fig. 2 Fabrication process of the HTTR fuel

2-3. Plant

NFI started its research and development work for HTGR fuel fabrication technology in the end of 1960's, and has proceeded cooperatively with JAERI during this quarter of a century. NFI fabricated 260 kgU fuel for VHTRC (Very High Temperature Reactor Critical Assembly) of JAERI as the first mass fabrication from 1985 to 1986 by a pilot plant with 200 KgU/year in fabrication capacity.

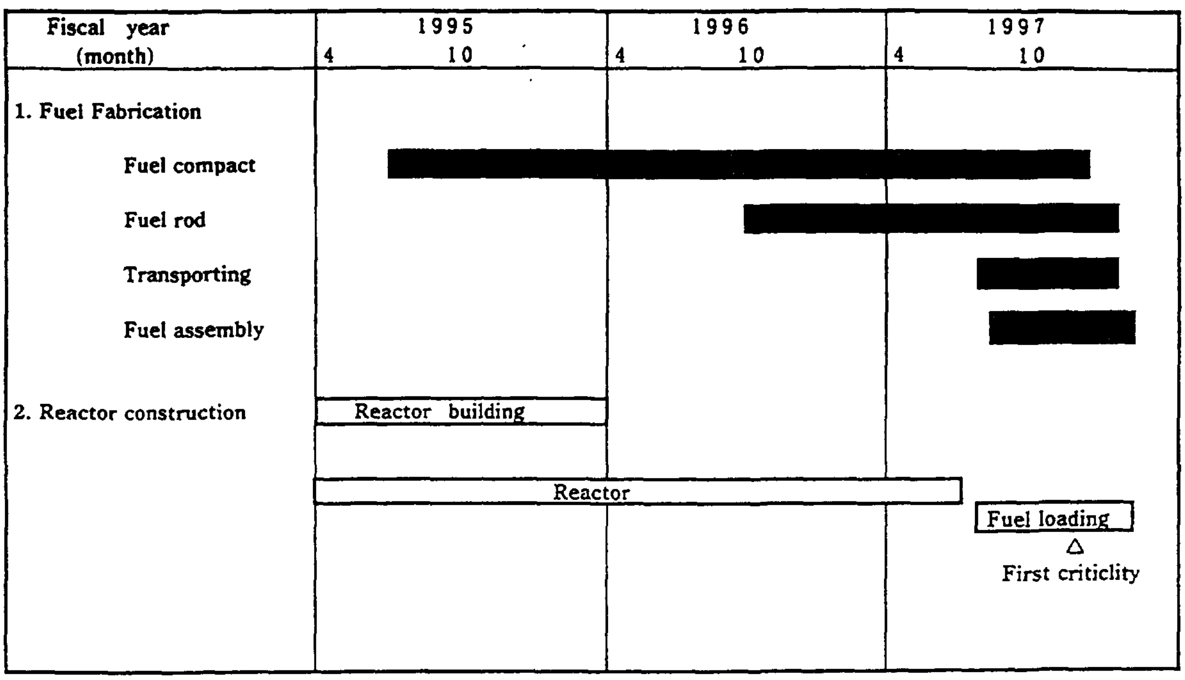
NFI completed the construction of the new HTGR fuel plant at Tokai in October 1992. In the plant the fabrication technologies gained through the long-term development works was applied. The main facility building is made of ferroconcrete with two floors and 1,500 square meters area. There are two fabrication lines for the fuel kernel and coating processes. Day-shift capacity of these is totally about 1,500 KgU/y. There is one line for fuel compact fabrication process which capacity is about 40,000 compacts/y at now. Two coaters designed originally by NFI are prepared. Fabrication capacity of each coater is max. 3.4 KgU/batch in standard operation. Hot-pressing equipment is operated full-automatically from weighing overcoated particles to lining up green compacts on a tray.

2-4. Schedule

Before starting the fuel fabrication of initial HTTR fuel performed qualification test and JAERI certificated the quality level. Fuel fabrication in NFI's facility will be finished at the end of 1997. Assembling of fuel will be started at the middle of 1997.

Table.2 shows the fuel fabrication schedule.

Table 2. Fuel fabrication schedule



3. Quality Control and assurance system[2]

It is impossible to inspect all fuel products because initial HTTR fuel contains a large number of coated fuel particles and fuel compacts. Therefore it is necessary to establish quality control and assurance methods such as inspection items, sampling method. Basically sampling method is applied to certificate fuel quality. An Inspection lot is a product group of a uniform quality. Namely a raw uranium powder lot is an assemblage of same enriched uranium material which is mixed by solution. A kernel lot and a coated particle lot are arranged by mixing plural fabrication batches uniformly in each process stage. A compact lot means a product assemblage baked at the same time. The lots of raw uranium powder is twelve and those of kernel, coated particle, and compact are about one hundred. The arrangement method of inspection lot is shown in **Fig. 3**.

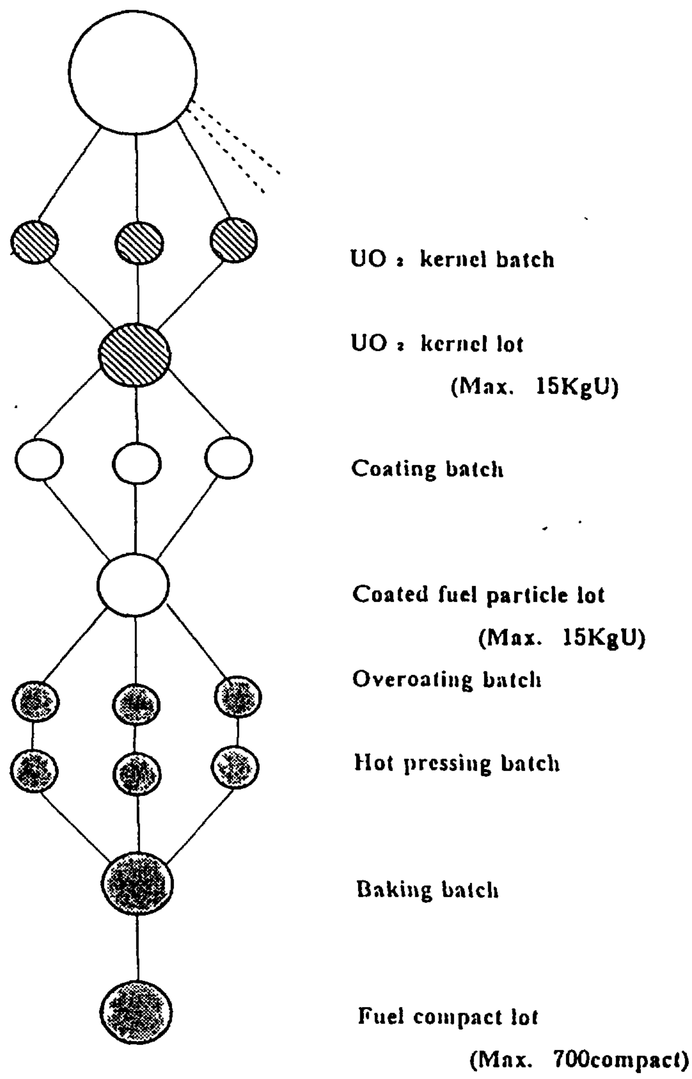


Fig. 3 Lot arrangement of fuel production

Table 3. Inspection item and method

Item	Method
1. UO_2 Kernel	
i) ^{235}U Enrichment	Mass spectro analysis & γ-ray spectro analysis
ii) Impurities	Emission spectro analysis
iii) Diameter	Particle size analysis
iv) Sphericity	Particle size analysis
v) Density	Mercury substitution method
vi) O/U ratio	Oxidation &weighing method
2 Coated Particle	
i) Diameter	Particle size analyzer
ii) Appearance	Visual observation
iii) Cross section	Ceramography
iv) Layer thickness	X-ray radiograph
v) Layer density	Solvent substitution method & Sink-float method
vi) OPTAF (2nd & 4th layer)	Polarization photometer

Item	Method
3. Fuel Compact	
i) Appearance & Marking certification	Visual observation
ii) Dimension	Micrometer
iii) Density (matrix)	Weighing & calculation
iv) U content	γ-ray analysis
v) Failure particle fraction	
Trough-coating-layer	Electric dismounting & nitric acid leaching method
SiC Layer	Burn & leach method
vi) Compressive strength	Strength tester
vii) Impurities	Emission spectro Analysis
viii) O/U ratio	Oxidation & weighing method
4 Fuel Rod	
i) U content	Calculation
ii) Dimension	Caliper
iii) Appearance & Marking certification	Visual observation
5 Fuel assembly	
i) ^{235}U enrichment	Certification of marking
ii) Appearance	Visual observation
iii) U content	Calculation

The inspection items and method for HTTR fuel are shown in **Table 3**. The inspection equipment is set to measure automatically as much as possible to obtain the reliability and rationalization of inspection. Diameter and sphericity of coated particles, the thickness of coating layers, sizes of fuel compacts, and uranium contents of fuel compact are measured automatically. The coating layer thickness is measured by analyzing the image of X-ray radiograph of coated particles.

A computer integrated system for quality assurance enables tracing a large number of inspection data.

The inspections by government (Science and Technology Agency) have been carried out for every process stage of 12 kinds of uranium enrichment. In total over thirty inspections will be carried on during this fabrication of initial HTTR fuel.

4. Quality

The standard deviations of diameter and density of UO_2 kernels, thickness and density of coating layer and diameter of coated particles are shown **Table 4** and **5** and **Fig. 4**. The results are small in comparison with the specifications as shown in table 1. Sphericities of UO_2 kernels and coated particles are also excellent and the difference of mean value of the two is

Table 4. Properties of UO2 kernel

Item	Average Value	Standard deviation
Diameter (μ m)	609	10.5
Density (gr/cm^3)	10.8	0.04
O/U ratio	2.00	——
Sphericity	1.05	——

Table 5. Properties of coated fuel particle

Item	Average Value	Standard deviation
Diameter (μ m)	937	10.8
Layer Thickness (μ m)		
1st layer	59.1	7 4
2nd layer	30.9	2.9
3rd layer	29 8	1 1
4th layer	46 4	4 2
Layer Density (gr/cm^3)		
1st layer	1 16	0.03
2nd layer	1 87	0 02
3rd layer	3 20	0 005
4th layer	1.85	0.04
OPTAF		——
2nd layer	1.00	——
4th layer	1.00	
Failure Fraction		——
Through SiC layer	5.4x10^{-6}	

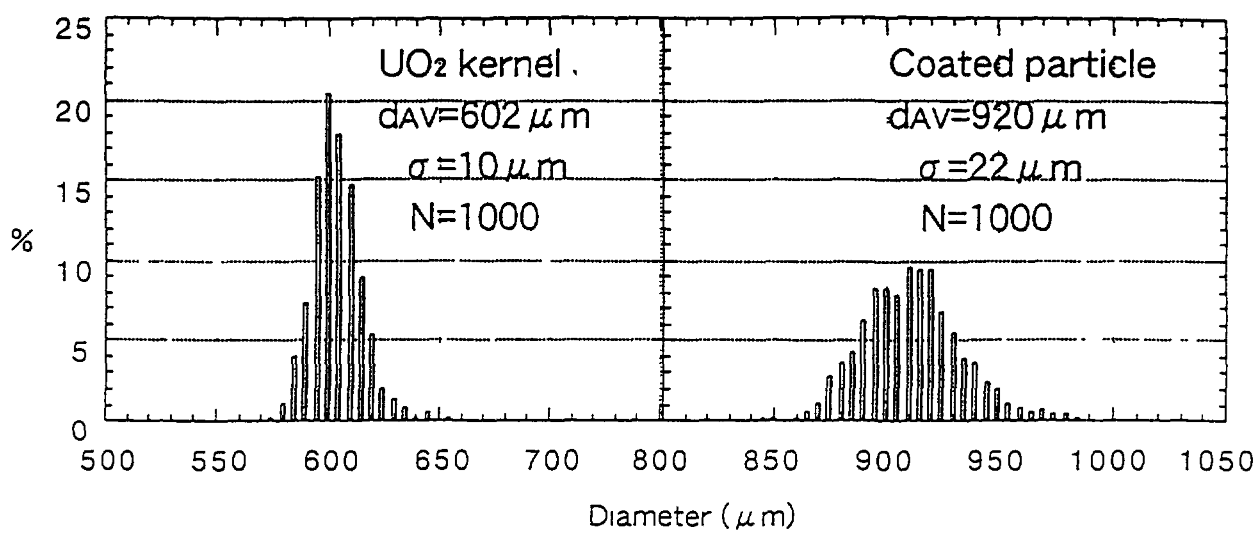


Fig. 4 Typical result of measuring diameterof fuel particle

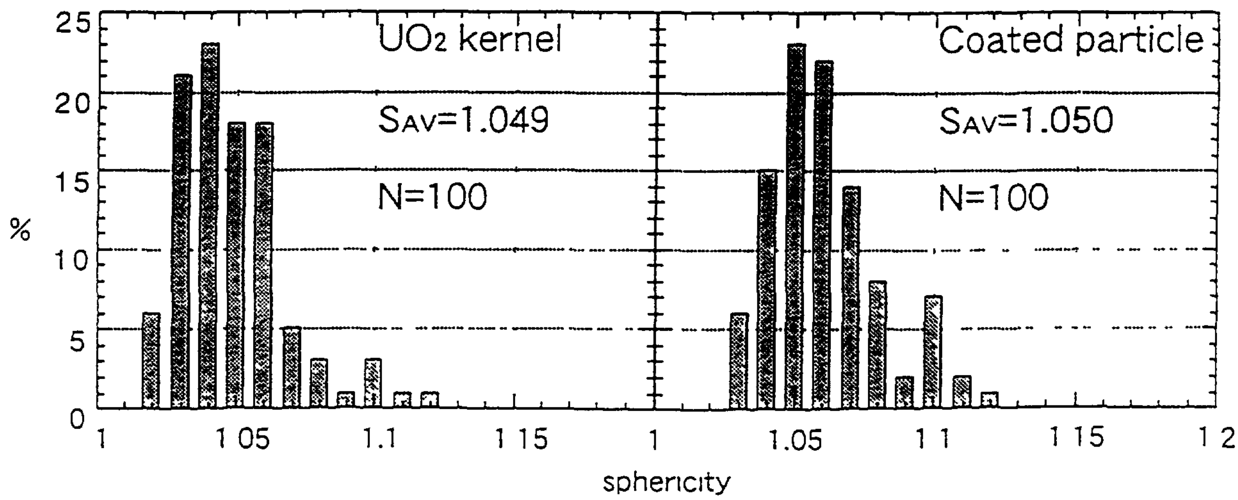


Fig. 5 Typical result of measuring sphericity of fuel particle

very small as shown in **Fig. 5**. Average Failure fractions of SiC layer of coated particles is 5.4×10^{-6} as shown in table 5. Therefore, the coated particle fabrication process is judged to be well controlled.

The inspection results of dimensions of fuel compact satisfy the allowances of specifications, ± 0.1mm.

As for the failure fraction of fuel compact, the numbers of the through-coatings defect particles in fuel compact are 0 or 1 for all fabricated fuel compacts as shown in **Fig. 6**, and the average failure fraction is 4×10^{-6}. The inspection data measured by burn & leach method are shown in **Fig.7**. There are some fuel compacts containing plural defect particles. However, the number of fuel compacts containing defect particles is decreasing. Although the average fraction in the latter half of 1995 is 1.2×10^{-4}, one in the second quarter of 1996 is improved to 5.1×10^{-5}. About 85% of fuel compacts fabricated in the second quarter of 1996 contain 0 or 1 defect particle as shown in **Fig.8**., and average failure fractions in compact lots are ranging between 1.5×10^{-6} and 1.8×10^{-4}.

We have made extensive efforts to stabilize the quality of fuel compact. The reason that causes the SiC-defect in a compact is considered as follows. Since packing fraction of the coated particle is relatively high, 30 vol.%, in the fuel compact and pressing to form compact is carried on under hot condition, a few coated particles could be contacted. We consider that the SiC layer of these particles results in failure during the compact fabrication process. Therefore to prevent damaging the coating layer during the compact fabrication process, we set up proper overcoating condition and pressing condition such as temperature and speed. In the R&D work before initial fuel fabrication, [3] temperature and speed are determined to be as low as possible to prevent contacts of coated particles. Furthermore, during the fabrication

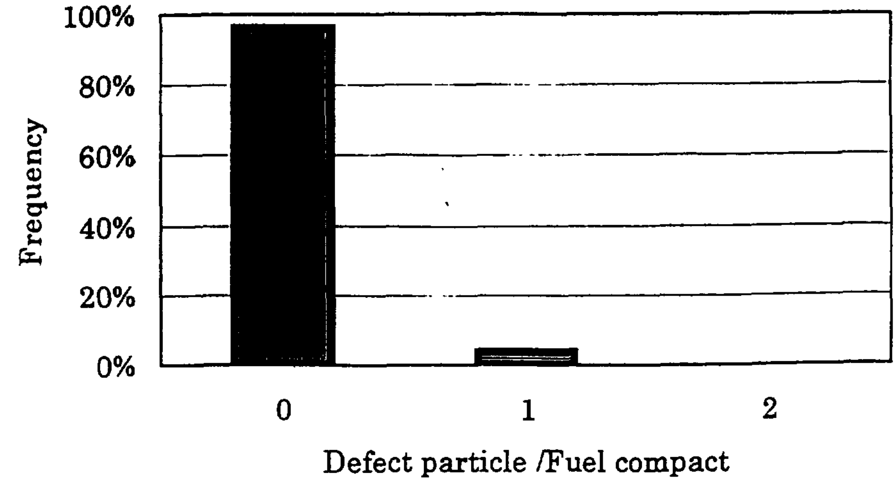


Fig.6 Failure fraction through coating layers

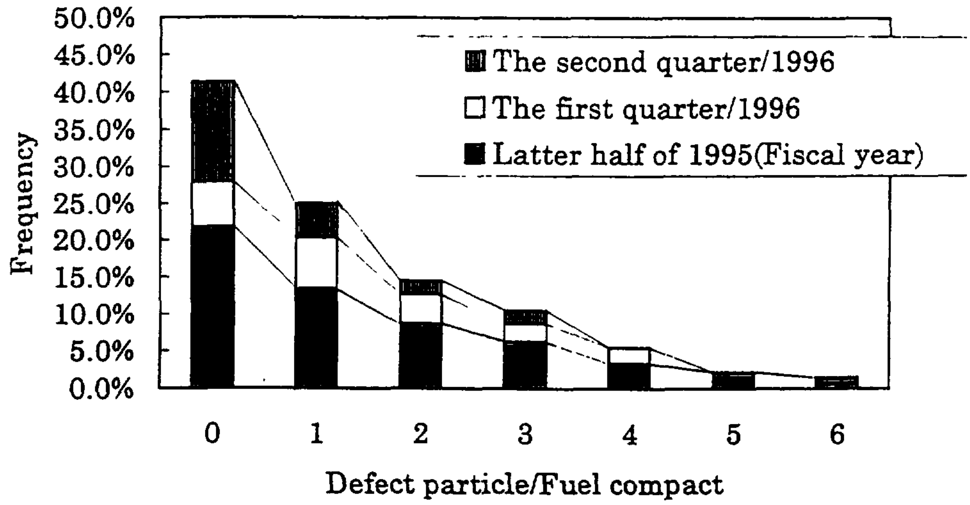


Fig.7 Failure fraction by burn & leach method

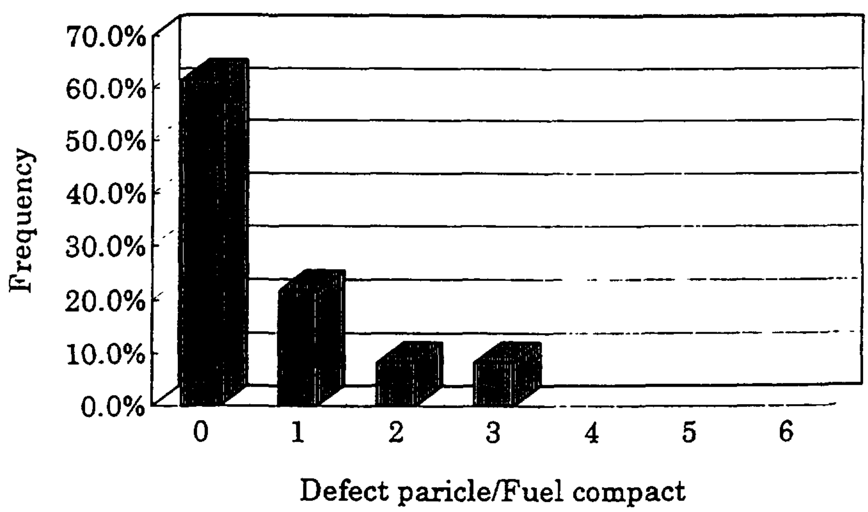


Fig.8 Failure fraction by burn & leach method
(The second quarter/1996)

of initial HTTR fuel, we found out that about 0.1% overcoated particles contained two coated particles. Since these coated particles could easily be contact during pressing, we have removed them from overcoated particle batches. These efforts enable decreasing the failure fraction by burn&leach method.

5. Conclusion

The fabrication of the initial core fuel for HTTR was started in June 1995 and finished about a half of the total amount at Sep. 1996. The fuel fabrication will be completed at the end of 1997. The failure fraction measuring by burn & leach method is 1.2×10^{-4} at the beginning of the fabrication, but it is improved to 5.1×10^{-5} now. The fuel qualities at present are satisfactory, however, we will carry out further efforts to attain the higher quality of fuel.

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TECHNICAL DEVELOPMENTS IN GAS COOLED REACTOR DESIGN

XA9846695

THE EOR SYSTEM IN DURI: COMPARISON BETWEEN CONVENTIONAL AND NON-CONVENTIONAL SYSTEMS

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Abstract

EOR system in Duri. Comparison between conventional and non-conventional systems. One of oil field location in Indonesia, which produces about 25% or more than 300 bpd, located in Duri, Sumatera island. The oil depth location is between 75 until 200 meter. Because the Duri oil is classified as heavy oil, only 7.5% of oil could be pumped out by using conventionally system. By using of EOR system, in this case uses steam injection, it is hoped that the amount of oil pumped could be reach until 65% of oil resources. It is interesting to study again the using of High Temperature Reactor for Duri steam flood (DSF) project after their feasibility study and after the DSF project has been used oil burning for the steam production. The Duri oil field area is divided into 12 regions. The DSF project has finished 8 regions. The other regions are still in project progress. In this study is discussed the EOR seven point system, the comparison between the present and prediction of thermal power required for DSF project and the DSF by using HTRs. It is calculated that the present thermal power required is less than ½ of the last feasibility study by using of (4x200 MWth.) HTRs.

INTRODUCTION

The Duri oil field is located approximately in the center of the province of Riau, which covers the northern central section of the Sumatera island, and has the distance of about 170 km from Pekanbaru. It is the biger oil field in Indonesia. The bigest one is the Minas oil field, which is located about 150 km from Duri. The oil resources of both field are difference. The oil of Minas is categorized as light oil and the oil of Duri is heavy oil. It makes the oil pumpe systems of both region are diffrence. The first oil production in the Duri field was in May 1958 by using the primary recovery system. The oil pumping was successfully, but only 7.5 % oil could be pumped out from the reservoir[1].

To increase the oil production some techniques had been done, i.e.:

1. The water flood techniques.
   It was done in year 1960 until 1963. It was predicted that 16% of oil reservoir could be pumped out.

2. The cyclic steaming or huff-puff technique.
   It was done in year 1967. The steam was injected into the production wells and than the wells were closed. After some days the production wells were opened and the oil was pumped out. The same technique was done again if the oil reservoir temperature and also its oil result decreased. By this technique can be pumped out the heavy oil until 20% from its reservoir.

3. The (continue) steam flood technique.
This technique had been used since 1975. The steam flood technique uses continue steam injection into the injection wells. This techique promises that 65% oil production from the heavy oil reservoir can be reached.

Before the DSF project was done in some Duri oil field regions the steam injection system was investigated by using the HTR's. The comparison between using oil and nuclear heat sources is interesting for HTR studies especially for non electrical applications.

THE STEAM INJECTION SYSTEM

The Duri oil field area is about 76.5 km^2, which is divided into 12 development regions and each has an average area 6 km^2 (Fig.1). Now the 8^{th} region is done by the DSF project. Steam injection system uses *7 point pattern*, which 6 points are located at the hexagonal corners as the production wells. One point is located at the middle of the hexagonal and its functions as the steam injection well. The distance between points at the hexagonal corners is 135 meter[2] (Fig. 2).

At present the DSF project is the bigest steam flood plant in the world. Until 8^{th} region the DSF project needs about 340,000 ton water per day. The steam is produced in the Boilers and the Waste Heat Recovery (WHR) machines. The boilers use oil as fuel and the WHR machines use the heat release from the gas turbines. The DSF project has 5 WHRs. Each WHR can process 1,675 ton water per day. The steam is transported through 75 cm diameter pipes. The using of 5 WHRs reduces the total steam cost.

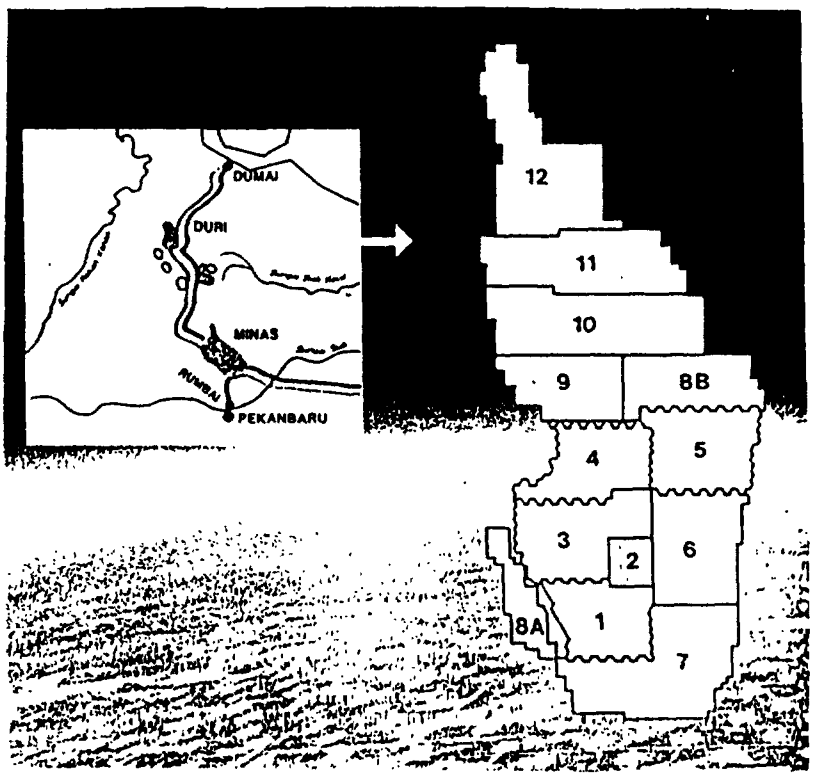

Fig.1 Duri oil field area

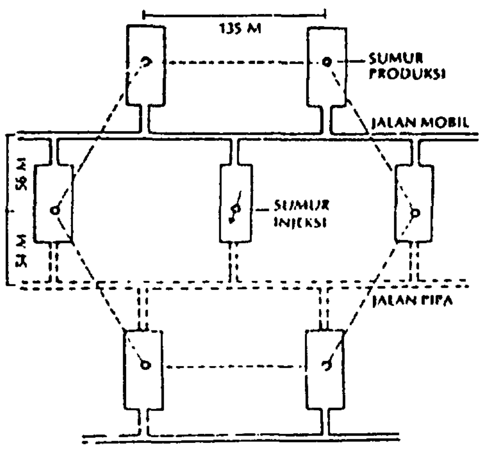

Fig.2 Seven point pattern

POWER REQUIRED

The thermal power is required for the steam production and electrical generation. The total power is predicted for all area of the Duri oil field.

The DSF Project.

Until the 8th region is replaced 367 steam generators with the power required is 6.5 MW thermal. The electrical generation is 22 MWe or about 70 MW thermal. It means that the total power required is about 76 MW thermal. The amount of oil production is 367,000 bpd or about 30% of the total Indonesia oil production. The heat and electricity are produced by using oil in the amount of 64,000 bpd.

The total length of the steam pipe is 3,219 km and the total of injection and production wells is 4,501. It means that each hexagonal well system needs in average about 5 km pipe length. The steam pressure at the steam generator outlet is 65 bar and at the inlet of the injections well is 28 bar or reduce by factor 0.43.

Based on the real power need until the 8th region development, it is predicted that the DSF project will be finished for all 12 region developments and required the thermal power for 114 MW. At that time the production rate is 550,000 bpd.

HTR as power source

Using HTRs for the DSF project has investigated in year 1985/86. Because the application of HTRs is feasible if the oil price is more than 24 US$/barrel[3], and in fact that the oil price until now is less than the calculation. It makes the using HTRs in the DSF project is not feasible. The study, which was done by China and uses the present technology, said that the HTR could be build for EOR if the oil price is more than 18 US$/barrel[4].

In this case the steam quality is better, has a factor 4, than the present technology. But the pressure drop factor[6] is bad (> 0.9) because the distance between the heat source and the injection wells. If 2 HTRs are replaced into the region 3 and 10 (Fig. 1) with the nominal power 200 MWth each, the present production capacity can be reached.

The difference result between this calculation and the old calculation [5] is based on the scenario of the 7 point pattern. This pattern is used at the beginning of 1990. Using this pattern makes that the number of the steam injection wells is less in factor 0.6. This difference is a meaningfull.

CONCLUSION

- Except the oil price variable the using of HTRs for Duri steam flood is interesting, especially the number of the HTRs can be reduced. It is caused by the using of 7 point pattern.
- The pipe length variable is significant to influence the pressure drop. In this case the present system has an advantage, because the steam generators can be build no far from the steam injection wells.
- The using HTRs heat for Natuna CO_2 conversion plant is different as in DSF project. The influence of the pipe length for Natuna is less than in DSF area, especially if the CO_2 conversion plant is adjacent with the reactor building.

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RESULTS OF COST ESTIMATES FOR THE EXPLOITATION OF THE NATUNA GAS FIELD IN INDONESIA USING THE HTR: A NEW MOMENTUM FOR COMMERCIALIZATION

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Abstract

The results of a simplified method for cost estimations are presented. The main result is: There exists a potential of economical competitiveness for the exploitation of the gas of the Natuna Gas Field in Indonesia. The potential consists in the production of Energy Alcohol from the gas of the Natuna Gas Field, making use of nuclear energy in the form of high temperature heat and electricity from the High Temperature reactor, HTR. The reason is that Energy Alcohol is a readily marketable product as a substitute for gasoline and diesel, with their relatively high market values. This is not so for methane, because of its lower market value, being used mainly in the heat market only. In the conversion process some - or a little more - of the carbondioxide CO_2, the main constituant of the gas of the Natuna Gas Field, is used; the competitiveness of the product decreases with the increasing consumption of CO_2.

The results in more detail are: Four processes have been investigated, these are:

Process 1: „Steam plus carbondioxde"-reforming for reformer gas plus electrolytic hydrogen, and synthesis of methanol; remark: 3 variants a; b; and c are considered.
Process 2: „Carbondioxide"-reforming for reformer gas with separation of CO_2, plus electrolytic hydrogen for synthesis of methane (so-called enrichment of CH_4), and synthesis of methanol from the reformer gas.
Process 3: Addition of electroylic hydrogen to Natuna Gas for the synthesis of methane, as an alternative to process 2, if the objective is to produce as much as possible methane.
Process 4: Separation of methane (as product) and CO_2 plus electrolytic hydrogen for synthesis of methanol.

The estimated cost figures C for methanol (100 %, no methane), process 1, variant a; b; and c, are C = 4.9; 7.5; and 15.1 \$/GJ for high specific investments, resp. are C = 4.1; 5.5; and 9.4 \$/GJ for low specific investments (50 % of high S for HTR and electrolyser). The variants a; b; and c make use of increasing amounts of Natuna CO_2, measured by the percentage of Natuna CO_2 of 15; 25 and 100 %. Process 1, variants a and b, are for high specific investments S (and - of course - for low S) economically competitive in comparison to e.g. lower break-even prices of e.g. 8 \$/GJ, but most of the Natuna-CO_2 is emitted. Process 1, variant c, which makes use of all Natuna-CO_2 is economically competitive only for low specific investments S to mean or high break-even prices, of e.g. 10 or 12 \$/GJ. The estimated cost figures C for processes 2; 3 and 4 lie in the high region being economically of less interest.

The discussed projects for the exploitation are large, this offers the possibility of mass-production of HTR-Moduls; representing an encouraging momentum for commercializing the HTR.

1. The Natuna Project and Possible Alternatives

1.1. In Summary: The Natuna Gas Field in Indonesia is discussed for exploitation to enlarge the countries export capabilities. The disadvantage of the Natuna gas in comparison to other gas fields is its high content of carbondioxide CO_2. In conventional technology with re-injection of CO_2 the estimated supply costs are high. Therefore the idea was developed to use nuclear energy for the exploitation of the gas from the Natuna Gas Field, including the carbondioxide.

1.2 In detail on the background of the Natuna Project and possible alternatives:

1.2.1. Indonesia is an oil and gas producing country. Its energy base is confronted with increasing inland consumption and decreasing oil resources, lit. AHIMSA-1995; SASMOJO-SUBKI-LASMAN -1995. The available resources in natural gas, including the Natuna Gas Field, are therefore becoming important.

1.2.2. Indonesia's Pertamina in 1994 was producing as much as 24 to 26 million tons/year of Liquified Natural Gas, LNG, from 6 production units Arun, Aceh, on the northern tip of Sumatra, and 6 plants Bontang, East Kalimantan. Most of the LNG was shipped to Japan, with the remained going to South Korea and Taiwan, lit. IPE-1995, p. 196.

1.2.3. The Natuna Gas Field in Indonesia was discovered in 1973, 225 km from Natuna Island in the South China Sea. This gas field contains approximately 1270 billion cubic meters of recoverable hydrocarbon reserves, making it one of the largest gas fields in the world (in terms of gas in place and recoverable hydrocarbons), lit.: GREENHOUSE ISSUES-1996. But the gas consists with 66.5 % of carbondioxide CO_2. The energy content is equivalent to about 1.5 TWy (TWy = Terra Watt Year), or for 30 years 50 GWt, equiv. to 15 large LWRs.

1.2.4. Pertamina, the National Oil Company of Indonesia, and Esso Exploration and Production Natuna, Inc., an affiliate of Exxon Corp., are planning the development of the Natuna Gas Field under a Production Sharing Contract, PSC, awarded in 1980. Pertamina, the world's largest supplier of LNG has major natural gas and LNG operations. Exxon Corp. has worldwide experience in developing and operating large energy projects, lit. PERTAMINA-ESSO-1993, p. 1. The estimated supply costs for new LNG plants under consideration for the Natuna gas field are 4.27 to 5.20 $/MBTU ($/MBTU = US $ pro Million British Thermal Units), lit.: IPE-1995, p. 197, table 4, equivalent to 4.05 to 4.93 $/GJ, (at 12 % discount rate), and is comparatively expensive. Parts of the additional cost are produced by the proposal of the re-injection of carbondioxide CO_2 into the ground.

1.2.5. On that background the idea is developed to use nuclear energy, and possibly other cheap energies, to make use of the Natuna- CO_2, to contribute to solutions of the countries energy problems, lit. RUSLI-1995-1; SANTOSO-BARNERT-1995.

2. Process Variants, Process Layout and Design

2.1. In summary: The conditions of the feasible products, the unit operations of reforming, and the objective of economical competitiveness make that 6 process variants can be identified. These are four processes, labeled as 1; 2; 3; and 4; with process 1 in three variants, labeled as 1a; 1b; and 1c. The processes 1c; 2; 3; and 4 make use of all Natuna-CO_2, whereas process 1a respectively 1b uses only 15 respectively 25 % of the Natuna-CO_2. The processes in short are:
1: H_2O plus CO_2-REF and ELY H_2 for methanol
2: CO_2-REF and ELY H_2 for methane (enrichtment) and methanol
3: NA and ELY H_2 for methane
4: SEP for methane and ELY H_2 for methanol.

2.2. In detail: On the objectives, the overview, and the approach:

2.2.1. The objectives of a process layout and design are :
1) the product of the process should be economically competitive,
2) the process should be environmentally benign to the largest extend, and
3) modern feasible technology should be used.

The objective 2 means that as much as possible of Natuna-CO_2 should be used in the process and being converted into a marketable product.

2.2.2. The conditions of the feasible products, the unit operations for reforming and the objective of economical competitiveness make that 6 process variants, consisting of 4 processes, labeled as 1; 2; 3; and 4; with process 1 in three variants labeled 1a; 1b; and 1c can be identified. The feasible products are methane, the main constituent of Liquified Natural Gas, LNG, and methanol - a liquid - as the main constituent of Energy Alcohol, as a substitute for diesel and gasoline.The unit operations for reforming are the „steam plus carbondioxide"-reforming, H_2O plus CO_2-REF, as it has been tested in the Research Centre Jülich GmbH, KFA, in 1988, lit. BARNERT-SINGH-1996, and the „carbondioxide"-reforming, CO_2-REF, - in contrary to „steam"-reforming, (often used in industry) -, as being tested since a few years at the Weizmann Institute of Science, Israel, lit. EPSTEIN-1995. The objective of economical competitiveness, being hard to be met, makes that the variants a and b for process 1 are taken into consideration. In these process variants only 15 and 25 % of the Natuna-CO_2 is converted into the marketable product methanol. The amount of used Natuna-CO_2 is measured by the ratio $CO_2/CH4$ in the feed R for the variants a; b; and c with R = 0.4; 0.67; and 2.67. The latter value is that of the Natuna gas. The processes in short description are:

Process 1: Steam plus carbondioxide-reforming, H_2O plus CO_2-REF, for reformer gas, heated by HTR-helium, with addition of electrolytic hydrogen, being produced from nuclear electricity of the HTR, and synthesis of methanol, in mixture with other alcohols. In the variants a and b the processes begin with a separation of most of the CO_2 from the Natuna gas. The separated CO_2 is emited.

Process 2: Carbondioxide-reforming, CO_2-REF., heated by HTR-helium, for reformer gas, separation of the remaining carbondioxide and addition of electrolytic hydrogen, being produced from HTR electricity, into the separated streams for the syntheses of methane and methanol. The amount of the product methane from the synthesis is larger than the amount of methane in the feed to the reformer, this is called the enrichment of methane, lit.: ARBIE-1996, lit.: EPSTEIN-1995.

Process 3: Addition of electrolytic hydrogen, produced by nuclear electricity from the HTR, to the cleaned Natuna gas for the synthesis of methane. This process is taken into consideration as an alternative for process 2, if the objective is to produce as much as possible methane.

Process 4: Separation of methane as the product from the Natuna gas and addition of electrolytic hydrogen, produced from electricity from the HTR, to the remaining carbondioxide for the production of methanol.

Remark: The processes 1 and 2 use the unit operation of reforming.The required high temperature heat is supplied by the HTR. The processes 3 and 4 do not use the unit operation of reforming. The unit operation electrolysis in its conventional form needs only electricity, which can be supplied by any nuclear reactor, e.g. the Light Water Reactor; but there are more modern forms: the steam electrolysis and the so called hot electrolysis, which need high temperature heat, and therefore the HTR.

2.2.3. The approach for the process layout and design, as well as for the cost estimation, is concentrating on the essentials by simplification for the constituents of the Natuna gas, for the mol-balances, for the chemical reactions, and for the energy balances. Thereby a better understanding is being produced. The simplifications are as follows:

A) The calculation is done for an input of 1.2×10^9 SCFD (SCFD = Standard Cubic Food per Day), being equivalent to 17 548.87 mol/s with the important constituents of CH_4, $x = 25.1$ %, equivalent 4 405 mol/s, and higher alcanes, C_{2+}, $x = 1$ %, equivalent 175 mol/s, and with CO_2, $x = 66.5$ %, equivalent 11 670 mol/s. These mole fluxes are put together and described by „simple" numbers with CH_4: 4 500 mol/s and CO_2: 12 000 mol/s, together NA = 16 500 mol/s.

B) The reaction turnovers of the unit operations reforming, electrolysis, methanol-synthesis and methanation are in all cases taken to be 100 %. In reality this can only be approached by respective feedbacks of separated non-reacted mol streams. For the compensation of this simplification in the cost estimation an additional cost charge of 20 % is being introduced.

C) The energy requirements of the unit operations are calculated for the 100 % reaction turnovers. This means that the heat recuperation, e.g. in the reforming is 100 %, which - in reality - cannot be achieved. This simplification is partly compensated by neglecting the rejected heat from the syntheses and - is considered partly by the additional cost charge of 20 % in the cost estimation.

2.3. In detail on one example: process 1, variant b:

2.3.1. Process 1, variant b, fig. 1 and 2, consists in principle (gas cleaning and feedbacks neglected) of a separation, SEP, a steam plus carbondioxide-reforming, H_2O plus CO_2-REF, reaction R1, the electrolysis of water, reaction R2, and the methanol-synthesis, MES, reaction R3. It converts 4 500 mol/s CH_4 and 3 000 mol/s CO_2, fig. 1, table, line 2, into 7 500 mol/s CH_3OH, being equivalent to 4.79 GWt. In the partitioning of the heat vector the conventional approach is applied, fig. 2: for the reforming 200 K are applied, the temperature span is 950 to 750 °C, and the remaining temperature span of 500 K is used for electricity and finally hydrogen production, via the electrolysis of 4 500 mol/s H_2O. The characterising figures of process 1, variant b, are efficiency e = 64 %, yield y = 133 %, CO_2-emisisons at the production site p resp. in total q are p = 1.9 resp. q = 3.4 kmol/GJ.

2.3.2. The processes 1a; 1c; 2; 3; and 4 are not described in detail in this report (an overview is given in paragraph 2.2.2.). Detail can be taken from lit. BARNERT-1996-12.

2.4. In extension to variants of processes 2; 3; and 4:

2.4.1. In principle there are possibilities for variants with less CO_2 conversion (and thereby more CO_2 emission), similar to process 1.

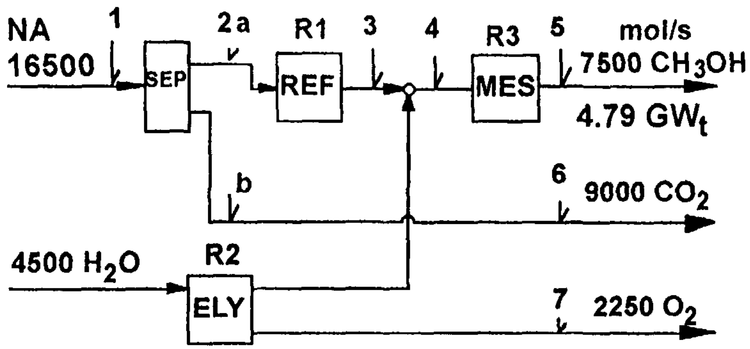
2.4.2. In process 2 the reduction of CO_2 conversion can be achieved by omitting the methanation, but than the „CO_2 reforming“ (without steam) with the new catalyst (R1 in fig. 10) is not needed. This leads to process 1. In process 3 and 4 the reduction of CO_2 conversion finally leads to product costs of the conventional Natuna project, which are considered to be relatively high.

2.4.3. The proposed approach to cost estimations of this report is a simplified one. It does not describe the cost situation of the conventional Natuna project. In more detailed analyses this must be included accordingly.

3. Six Process Variants, Cost Analysis

3.1. In summary: The estimated cost figures C for the product methanol (without a second product, no methane), process 1, variants a; b; and c are increasing with the variants due to the

1b. Methanol middle via SEP, H_2O+CO_2 Ref, Ely H_2

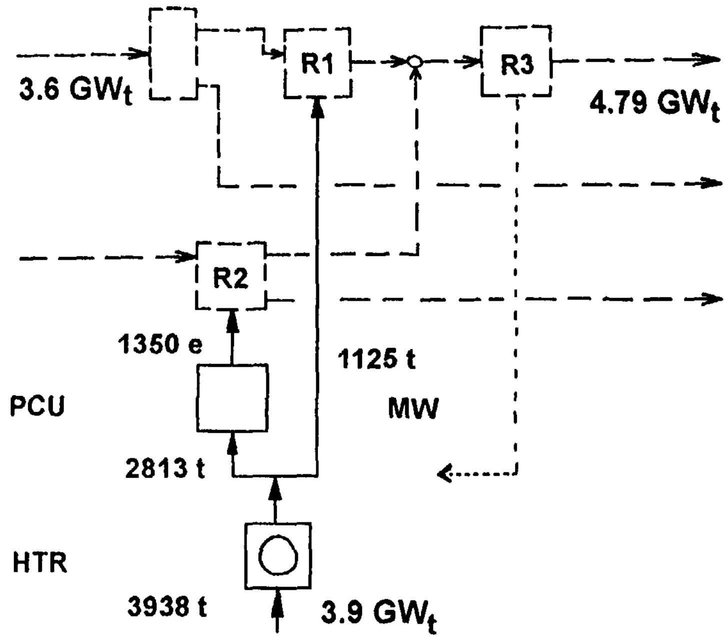


	n; Δn; kmol/s								
		2 a;b	R1 b	3	R2	4	R3 a	R3 b	6 5 7
1 CH_4	4.5	4.5	- 4.5						
2 CO_2	12	3/9		3		3		- 3	9
3 H_2O			- 4.5		- 4.5				
4 H_2			+13.5	13.5	+4.5	18	- 9	- 9	
5 CO			+ 4.5	4.5		4.5	- 4.5	-	
6 CH_3OH							+4.5	+3	7.5
7 O_2					+2.25				2.25
Σ	16.5								

	$\frac{\Delta n}{\text{kmol/s}}$		$\frac{\Delta H}{\text{kJ/mol}}$	$\frac{Q;E}{MW}$	
R1a	0	$(CH_4 + CO_2 = 2CO + 2H_2)$	0	0	+1125 (t)
b	4.5	$(CH_4 + H_2O_2\ \ CO + 3H_2)$	+ 250	+ 1125	(850-600°C)
R2	4.5	$(H_2O = H_2 + \frac{1}{2}O_2)$	+ 300	+ 1350 (e)	
R3a	4.5	$(2H_2 + CO = CH_3OH)$	- 90.7	- 408	- 556 (t)
b	3	$(3H_2 + CO_2 = CH_3OH + H_2O)$	- 49.5	- 148	(350°C)

Fig 1

1b. Methanol middle HTR Energy



R1, HTR, t : 1125 MW_t → (950 — 750)°C ⇒ 200 K

(x2.5)

2813 MW_t ← (750 — 250)°C ⇐ 500 K

⇓

R2, HTR, e : 1350 MW_e (e = 0.48) → 4500 mol/s H_2O

$$e = \frac{4.79}{3.6 + 3.9} = 64\ \% \ ; \qquad p = 1.9 \frac{\text{kmol}}{\text{GJ}}$$

$$y = \frac{4.79}{3.6} = 133\ \% \ ; \qquad q = 3.4 \frac{\text{kmol}}{\text{GJ}}$$

Fig 2

increasing application of Natuna-CO_2 from C = 4.9 to 15.1 \$/GJ (\$/GJ = US Dollar per Giga-Joule, lower heating value, throughout this report) for high specific investments S, resp. 4.1 to 9.4 \$/GJ for low S, fig. 14, left hand side results (same as fig. 1). The estimated cost figures C for processes applying 100 % Natuna-CO_2, - these are the processes 1c; 2; 3; and 4 -, producing methanol (1c), mixtures of methanol and methane (2 and 4) and methane (3) have in a band of max. 12 % similar costs C = 15.1 to C = 16.9 \$/GJ for high specific investments S, resp. C = 9.2 to 10.2 \$/GJ for low S, fig. 14, right hand side results (same as fig. 1).

3.2. In detail on the method of cost calculation and on assumptions, as well as on the results, fig. 3, in particular for process 1, variant b, fig. 4:

3.2.1. The cost calculation is done in the form of tables, fig. 15 to 20 for high specific investments S „High S" and for low specific investmens, „Low S", using the following equations:

$$X = (PT)_F X_F/(PT) + Ia/(PT), \text{ with } I = nS \text{ or } PS, \text{ and } Y = aI, \text{ and } C = 1.2X.$$

The symbols, used in these equations and in the tables have the following meanings and units, respectively values:

C	estimated cost of products, unit: \$/GJ, lower heating value, throughout this report.
X	calculated cost value in table, without correction factor for losses, unit: \$/GJ.
1.2	correction factor for compensation of the assumption that the reaction turnover is over 100 %, considering the losses, unit: 1.
Y	yearly money flow, turnover, unit: Mio \$/y.
a	annuity, a = 12 %/y, as in lit. IPE-1995, p. 195, table 4, right hand side.
I	investment, unit: Mio \$.
n	characteristic mol-stream for costing, unit: mol/s.
P	characteristic power for costing, thermal or electric, unit: MWt, or MWe.
S	specific investment value with adequate unit: 1 000 \$/(mol/s), or \$/kWt, or \$/kWe.
T	yearly hours of operation, availability, T = 8 000 h/y.
$(PT)_F$	yearly amount of thermal energy, fuel, unit: 10^6 GJ/y, for Natuna gas (lower heating value) and for thermal energy of HTR, unit: 10^6 MWht/y and yearly amount of electrical energy, fuel, unit: 10^6 MWhe/y.
(PT)	yearly amount of thermal energy of product, unit: 10^6 GJ/y.
X_F	price of Natuna gas, assumed to be 1 \$/GJ, and price of nuclear fuel, assumed to be 1 \$/GJ for thermal energy, equivalent to 7.2 \$/MWh for electrical energy, with the assumption that the efficiency is 50 %.

3.2.2. The results of the calculation are „estimated costs" only, because of the following facts: the method of calculation is rather simple (e.g. operational costs are not very detailed), the assumption of 100 % reaction turnover is compensated via an over-all correction factor of 1.2 only, the method of calculation of investment with a characteristic capacitive measure and its specific equivalent is simple, because the gas cleaning of the Natuna gas is not included, and because the cost of the Natuna gas with X_F = 1 \$/GJ is a first assumption only.

Cost Estimation NATUNA +HTR

Methanol (Liquid) + Methane (Gas)

4 Processes ,	Process 1 in 3 Variants					
Process Variant	1 a	1 b	1 c	2	3	4
Methanol (liq) ; %	100			51	0	68
Methane (g) ; %	0			49	100	32
Percentage NA-CO_2 ; %	15	25	100	100		
Ratio CO_2/CH_4 ; %	0.4	0.67	2.67	2.67		
	Estimated Cost C ; $ / GJ					
Variation of S for HTR+ELY High S	4.9	7.5	15.1	16.5	16.9	15.2
Variation of S for HTR+ELY Low S	4.1	5.5	9.4	10.2	10.1	9.2

Process:

1 : Methanol min. , middle, max.

2 : Methane (Enrichment) plus Methanol

3 : Methane

4 : Methane (Separation) plus Methanol

High S : 1500 $/ kWe for HTR; 1000 $/ kWe for ELY   Low S : 50 % of High S

Fig 3

1b. Methanol middle Costs

High S

	n P (mol/s; $MW_{t,e}$)	S (10^3\$/(mol/s); \$/$kW_{t,e}$)	I (Mio $)	Y (Mio $/y)	X ($/GJ)	X/X_Σ (%)
1 SEP	16500	20	330			
2 R1, REF	4500	110	495			
3 HTR, t	1125	500	563			
4 R2, ELY, e	1350	1000	1350	405	2.9	39
5 HTR, e	1350	1500	2025			
6 R3, MES	7500	90	675			
7 Inv.	a = 12%/y		5438	625.6		
T= 8000 h/y; t:; e:	P (GW; GW)	PT (10^6GJ/y; 10^6MWh/y)	X_F ($/GJ; $/MWh)	Y_F (Mio$/y)		(H2 " 46.5)
8 NA t	3.6	103.7	1	103.7		
9 R1 HTR, t	1.125	32.4	1	32.4		
10 R2 HTR,e	1.35	10.8	7.2	77.8	0.56	7.5
11 Prod. , t	4.79	138.0		866.5	6.28	
12 Cost C	C = X_{11} x 1.2 (for losses)				**7.53**	

Fig 4

3.2.3. The cost calculation has been done for two sets of specific costs S for the High Temperature Reactor, HTR, as well as for the electrolyser, ELY, to produce an overview on the influence of the coupling of these cost intensive technologies. In the first set of cost calculations, the values of S = 1 500 \$/kWe for HTR for electricity production (and 500 \$/kWt for heat production), and 1 000 \$/kWe for ELY, - called: high specific investment S and labeled „High S“ -, have been taken, fig. 3 and 4, for the example.. In a second set of calculations low specific investments have been taken, - called: low specific investment S and labeled „Low S“, which are 50 % of the values of „High S. The other specific investments for the processing of the gases and the syntheses are not changed and remain the same, e.g. for SEP: $S = 20 \times 10^3$ \$/(mol/s), in all 12 cases.

3.2.4. The estimated cost figure C for methanol (only methanol as product, no methane), process 1, variants a; b; and c are C = 4.9; 7.5; and 15.1 \$/GJ (\$/GJ = US \$ per Giga-Joule, lower heating value throughout this report) for high specific investments S, fig. 3 and fig. 4 for the example. The variants a; b; and c apply to increasing amounts of Natuna-CO_2, measured as the fraction of the Natuna-CO_2 with P = 15; 25; and 100 %, resp. measured by the ratio CO_2/CH_4 in the feed with R = 0.4; 0.67; and 2.67 (the latter is Natuna gas).

3.2.5. The estimated cost figure C for processes producing methane and methanol (process 2 and 4) and 100 % methane (process 3 no methanol) for the processes 2; 3 and 4 are C = 16.5; 16.9 and 15.2 \$/GJ for high specific investments S, fig. 3. In these processes - as well as in the process variant 1c - 100 % of the Natuna-CO_2 is used in the conversion process. It is of interest that these cost figures are higher than the cost figure of process 1, variant c with 15.1 \$/GJ by about 1 % up to 12 %. One reason is the higher needs of hydrogen for the conversion processes.

3.2.6. The comparison of the six process variants shows that the hydrogen production on the base of nuclear energy via electrolysis has a big influence on the total cost, in particular for the processes with a 100 % application of Natuna-CO_2. The cost part fractions for the nuclear electrolytic hydrogen can be calculated from the investments on the HTR (for electricity production), the electrolysis and the fuel for the HTR for electricity production. The result for the fraction of the nuclear electrolytic hydrogen of the total cost for the process are 46.5 %. Only for process 1, variants a and b, the fraction is low resp. not too large. From the view point of the mol-balances it is clear: the H_2 is needed for the conversion of CO_2 into the products.

3.2.7. The estimated cost figure for methanol (the only product, no methane), process 1, variants a; b; and c for the low specific investments S (50 % of High S) are 4.1; 5.5; and 9.4 \$/GJ, fig. 3. Remark: the low specific investments S are rather low values, and may be achieved only in future mass production of these technologies, High Temperature Reactor, HTR, and electrolysis, ELY.

Cost Estimation NATUNA + HTR

Cost vs. Fraction Methanol / Methane

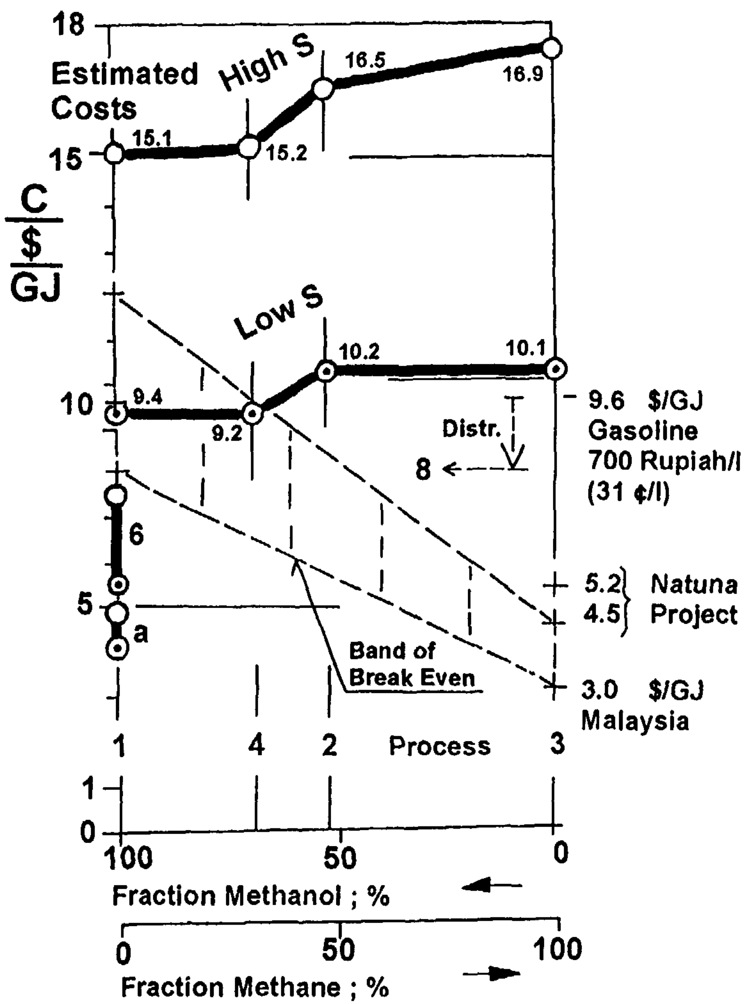


Fig 5

Cost Estimation NATUNA + HTR

Cost vs. Ratio CO_2/CH_4 for Process 1

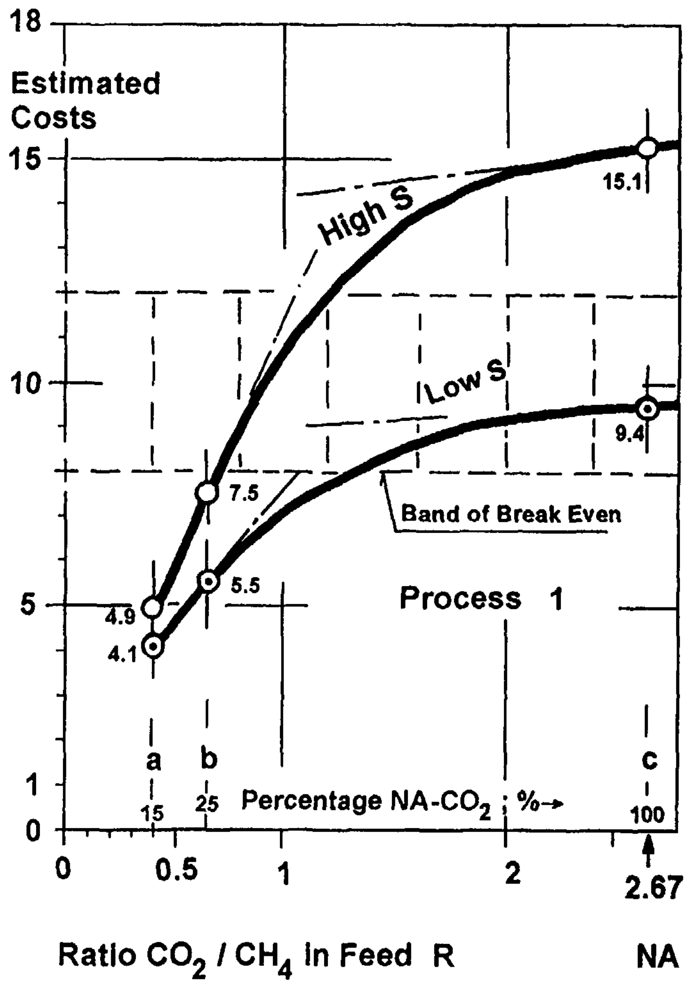


Fig 6

3.2.8. The estimated cost figures C for processes producing methane and methanol (process 2 and 4) and 100 % methane (process 3, no methanol) for the processes 2; 3; and 4 for low specific investments S are C = 10.2; 10.1; and 9.2 \$/GJ, fig.3. Remark: also for the low specific investments S the fraction of costs for hydrogen are still high, e.g. process 3 fraction of nuclear electrolytic hydrogen X (H_2)/C = 78 %.

3.2.9. The estimated cost figures C for processes producing methane and methanol (process 2 and 4) are mean values with the assumption that both products methane and methanol have the same value in the end user market. This is - of course - not the case and therefore needs particular consideration. An example is given for process 4 in fig.5: The mean case of C = 15.2 \$/GJ are to be split, if methane is sold for 3 \$/GJ, than methanol would cost about 21 \$/GJ, according to $PT \cdot C = PT_1C_1 + PT_2C_2$, or $C = F_1C_1 + F_2C_2$ (F: Fraction in fig.5), which represents a straight line in fig.5.

3.2.10. The values for the specific investment S for the chemival conversion unit operation: gas separation, SEP; reforming (of the two kinds), REF; methanol synthesis, MES; and methanation, MET, fig. 4, upper part, have been taken from respective evaluations in the Project „Prototype Plant Nuclear Process Heat“ (PNP) from lit. POTENTIALSTUDY-1987. The values of the specific investment S for the High Temperature Reactor HTR with S = 1 500 \$/kWe and for the unit operation electrolysis; ELY with S = 1 000 \$/kWe; are typcial expected values of the future. They are taken as relatively high, called high specific investmens S, labeled „High S“. For nuclear energy in general, see chapter 5.

4. Estimation on Economical Competitiveness, Six Process Variants

4.1 In summary: There exists a potential of economical competitiveness for the exploitation of the gas from the Natuna Gas Field in Indonesia, it consists in the production of energy alcohol, mainly methanol - a liquid -, as a substitute for gasoline and diesel, from the methane CH_4 gas and from some or a little bit more carbondioxide CO_2 gas from the Natuna gas field, making use of nuclear energy in the form of high temperature heat and electricity from the High Temperature Reactor, HTR. The reason ist that methanol, as a substitute for diesel and for gasoline, is a readily marketable product, competing in the market for motor fuel with a relatively high value. This is not so for methane because this is used only as a heating fuel with a relatively low value.

4.2. In detail on the situation of competition and on the results with respect to the cost figures for economical competitiveness, fig. 5 and 6:

4.2.1. The situation of competition of products from the exploitation of the Natuna gas allows the differentiation into the two rather different commodities of secondary energy carriers, and this is „motor fuel“, in particular diesel and gasoline, and „heating fuel“, here mainly natural gas. For the description of the competition two values are taken for both commodities. For motor fuel the reasoning is: the price of gasoline in Indonesia at the gas station is 700 Rupiah liter, lit. RUSLI-1995, equivalent to 9.6 \$/GJ, minus distribution costs of assumed 1.6 \$/GJ resulting in 8 \$/GJ as a lower point of the edge of the band of break-even. The price of diesel in the European Community in 1990 is 15.3 \$/GJ, lit. EU-1993, p. 34, and in BARNERT-1995-7 p. 19, minus distribution of assumed 3.3 \$/GJ, resulting in 12 \$/GJ as an upper point of the edge of the band of break-even. The reasoning for methane as a heating fuel to be transported in the form of LNG: It is the estimated supply costs for new LNG plants under consideration from lit. IPE-1995, p. 197, table 4, with the costs for the project Malaysia LNG-3 of about 3 \$/GJ as a lower point of the edge of the band of break-even, and the costs of the discussed Natuna-project of about 4.5 \$/GJ as an upper point of the edge of the band of break-even. The situation of competition of the two competitors Motor Fuel and Heating Fuel in dependence from the fraction (Mixture) of the two products methanol and methane is described by a linear interpolation of the respective lower and upper points, forming the band of break-even, depicted in fig. 5 and 6 with dashed lines.

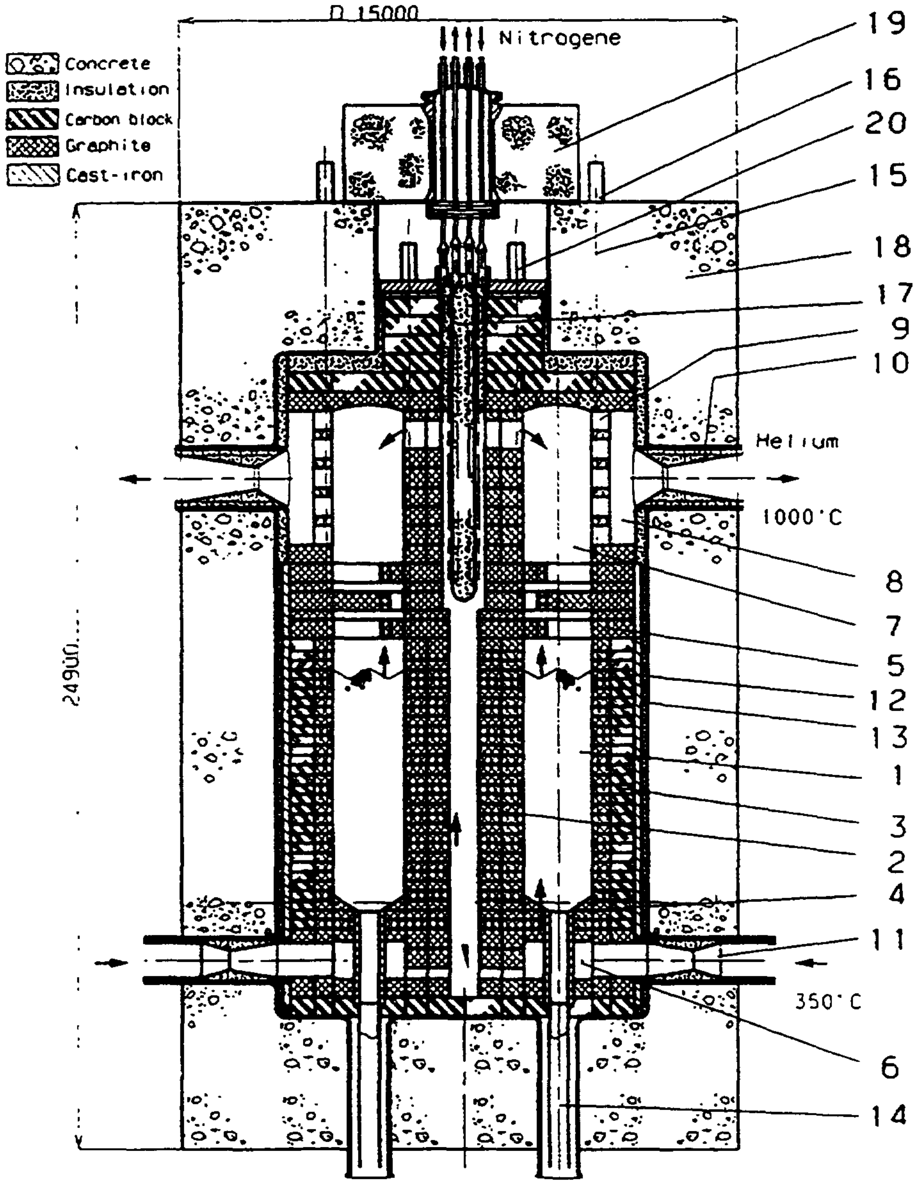
4.2.2. Process 1, variants a and b, both producing methanol, are for high specific investments S with costs of C = 4.9 and 7.5 \$/GJ, and - of course - for low S with costs of C = 4.1 and 5.5 \$/GJ economically competitive in comparison to the lower edge of break-even with a price of e.g. 8 \$/GJ, fig. 5 and 6. But for these variants most of the Natuna CO_2 is after separation emitted to the atmosphere. Process 1, variant c, which makes use of all Natuna-CO_2 (and therefore has zero emission of carbondioxide CO_2 at the production site) is economically competitve only to mean or even high break-even prices of e.g. 10 to12 \$/GJ at low specific values S, fig. 5 and 6.

4.2.3. From the processes 2; 3; and 4, producing mixtures of methane and methanol and pure methane (process 3) only process 4 with low specific investments S may be economically competitive with C = 9.2 \$/GJ against the upper edge of the band of break-even of e.g. 9.6 \$/GJ (linear between 12 and 4,5 for motor fuel and heating fuel), fig. 5. All other processes 2; 3; and 4 for high S, and processes 2; and 3 for low S, are not economically competitive, fig. 5.

4.2.4. The scientific objective to make use of the Natuna CO_2 to an extend, which should be as large as possible, can be fulfilled by process 1, with low specific investments S only with costs of 9.4 \$/GJ against a middle break-even price of e.g. 10 \$/GJ. But the scientific objective is in conflict with the objective to produce products, which should be economically competitve for a lower break-even price of e.g. 8 \$/GJ, fig.5. The economical limit to the ratio CO_2/CH_4 for process 1 with high specific investments is about 0.7 to 1.2 for the lower to higher edges of the break-even band resp. with low S is about R = 1.3 for the lower edge of the break-even band; for low S there is no economical limit for R from the upper edge, that is the Natuna gas can be used in its ratio of R = 2.67 still being economically competitive fig.6. Of course - going to the limits - reduces the profits.

High Temperature Reactor, HTR

Concept of an Advanced HTR (AHTR) for Process Heat Application & Electricity Production



1 Annular core
2 Central-column
3 Side reflector
4 Bottom reflector
5 Top reflector
6 Cold gas chamber
7 Hot gas chamber
8 Hot gas annular chamber
9 Hot gas channels
10 Hot gas venturi jet
11 Cold gas venturi jet
12 Thermal schield
13 Thermal insulation
14 Sphere discharge tube
15 Control rod
16 Control rod drive
17 After heat cooler
18 Prestressed concrete reactor vessel (PCRV)
19 Vessel closure
20 Controlrod resp. KLAK-System

Advanced High Temperature Reactor (AHTR-500)

Fig 7

5. In extension to HTR technology and economics:

5..1. In summary: The technical and industrial feasibility of the production and application of high temperature heat from the HTR has been demonstrated. The economical attractiveness of nuclear energy still exists, if capital cost are lowered. The hugh size of discussed projects for the exploitation of the Natuna Gas Field offers the possibility of mass-production of HTR-Moduls, an encouraging momentum in nuclear energy application in comparsion to the historical development in electricity protection. The demonstration of the coupling needs to be done. A technical answer to the historical cost increases of nuclear energy could be: catastrophe-free nuclear energy technology and simplification, lit.: BARNERT-1995-11.

5..2. The main driving force for economical attractiveness of nuclear energy is the fact that nuclear fuel is by a factor of 3 to 4 cheaper than fossil alternatives. The price of nuclear fuel in FRG is about 3.7 US \$ (90)/MW_t, equivalent to 1.03 US \$(90)/GJt, calculated with an efficiency of 31 % for LWR-fuel. The over all trend fo the development of the nuclear fuel prices in FRG is a reduction of 25 %, this is an encouraging positive fact for nuclear energy from the historical development.

5..3. For the reason of the historical increase of the capital cost the utilities in the US and in Europe have formulated goals (limits) for the capital costs of future nuclear plants. The European Utility Requirements, formulated the goal for the capital cost C = 1 100 ECU/kW_e (ECU = European Currency Unit), equivalent to 1 447 US \$ (95)kW_e (1,8681 DM/1 ECU x 1 US \$/1,42 DM, Oct. 1995).

5..4. A new proposal for the concept of an advanced HTR for process heat application and electricity production summarizes modern features of catastrophe-free design characteristics (self-acting stabilization of power and fuel temperature by negative feedbacks, self-acting removal of after-heat by modularization of the core, self-acting protection against corrosion by coating of graphite surfaces, and self-acting stabilization of pressure boundary by prestressed concrete reactor pressure vessel) with increased in- and outlet temperatures, fig. 7, lit. DRECKER-SINGH-BARNERT-1992.

5..5. The huge size of the Natuna Gas Field and the size of the discussed production lines (in this report the calculation has been done for an input of 1.2 x 10^9 SCFD, see paragraph 3.2.2.A) make that much nuclear energy is needed. An example indicates that: For process 1, variant 6, the needed HTR thermal power is 3 900 MWt, see fig. 7, which represents about 20 HTR-modules, á 200 MW_t. This offers the possibility of series production (for more production lines; mass-production) of HTR-Modules, lit. BARNERT-1996, representing an encouraging momentum in nuclear energy application in comparison to the historical development in electricity generation.

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REACTOR PHYSICS CALCULATIONS ON THE DUTCH SMALL HTR CONCEPT

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Abstract

As part of the activities within the framework of the development of INCOGEN, a "Dutch" conceptual design of a small HTR, the ECN reactor physics code system has been extended with the capability to perform combined neutronics and thermal hydraulics steady-state, burnup and transient core calculations on pebble-bed type HTRs, by joining the general purpose reactor code PANTHER and the HTR thermal hydraulics code THERMIX/DIREKT in the PANTHERMIX code combination. The validation of the ECN code system for HTR applications is still in progress, but some promising first calculation results on unit cell and whole core geometries are presented, which indicate that the extended ECN code system is quite suitable for performing the pebble-bed HTR core calculations, required in the INCOGEN core design and optimization process.

1. INTRODUCTION

In 1993 an HTR project was initiated at ECN within a wider national Dutch nuclear development programme. The most important activity at ECN comprises the first efforts towards a "Dutch" conceptual design of a small HTR. The plan is to compose and optimize a modular HTR design, best suited to Dutch development and market circumstances, built out of components of existing design, always keeping in mind the costs of development and market introduction.

This paper is primarily devoted to the validation of the ECN reactor physics code system OCTOPUS for the reactor physics analyses of the first selection of a small 40 MWth HTR concept. This first concept, INCOGEN, which stands for *I*nherently safe *N*uclear *COGEN*eration [1], is primarily based on the Jülich PAP20 reactor [2]. Emphasis is put on the calculation of safety related physics parameters, like reactivity coefficients and dynamic behaviour during transients.

The ECN reactor physics analysis code system OCTOPUS [3] covers the whole range from nuclear model calculations upto whole-core calculations. Within this code package different calculational approaches can be used to solve the same problem, e.g. deterministic versus Monte Carlo. In the OCTOPUS code system several codes and code systems, including NJOY, SCALE-4, KENO-V, DORT/TORT, WIMS, MCNP4A and PANTHER are interconnected, and also connected to the point-depletion codes ORIGEN-S and FISPACT-4. In Section 2.1 a brief description will be given of the ECN reactor physics code system OCTOPUS in general and with respect to the application in HTR reactor physics calculations.

Recently the capabilities of the ECN reactor physics code system have been extended with the possibility to perform pebble-bed HTR core physics calculations. Both steady-state and burnup core physics and reactivity transient calculations for High Temperature Reactors are being performed by connecting the general purpose modular reactor code PANTHER [4,5] and the HTR thermal hydraulics module THERMIX/DIREKT [6] from KFA Jülich in a code combination *PANTHERMIX*. Some features of this code combination are presented in Section 2.2.

Until a couple of years ago no HTR reactor physics calculations had been performed with the ECN code system. Consequently, benchmarking of our code system with computational and experimental benchmarks was required. Therefore ECN has been participating in the IAEA Coordinated Research Programme on "Validation of Safety Related Calculations for Low-Enriched HTGR's". In this programme institutions from Japan, China, USA, Russia, Germany, Switzerland, France and The Netherlands are collaborating in benchmark calculations for the HTR-PROTEUS experiment in Villigen, Switzerland, which deals with the well-known spherical fuel elements with a diameter of 6 cm. In Section 3 some results will be presented of the validation of the ECN code system for these steady-state HTR applications. An intercomparison is made between the HTR-PROTEUS benchmark calculation results of the deterministic code packages WIMS (versions E-5 and 6) and SCALE-4, and the Monte Carlo code MCNP4A.

The feasibility of the PANTHERMIX code combination for steady-state, burnup and transient calculations is being demonstrated in a benchmark exercise on the PAP20 pebble-bed reactor, which is carried out together with KFA Jülich, comparing the ECN code package, in particular the PANTHERMIX code combination, with the KFA HTR code package. The benchmark exercise is still in progress, but in Section 4 some results will be presented of steady state and reactivity transient calculations on the PAP20 reactor concept. The final benchmark comparison results will be reported at a later stage.

Presently the core physics calculations for the "Dutch" small 40 MW reactor concept INCOGEN are still in progress, but in Section 5 some first results will be presented from the whole-core calculations with the coupled neutronics-thermal-hydraulics code combination PANTHERMIX for the first version of INCOGEN. Also some results will be presented of MCNP shielding calculations on the INCOGEN conceptual design.

2. OCTOPUS - THE ECN REACTOR PHYSICS CODE SYSTEM

2.1. General aspects

The ECN reactor physics analysis code system covers the whole range from basic nuclear data libraries and nuclear model calculations, through (pin-)cell and assembly calculations upto 3-D combined neutronic and thermal hydraulic whole-core calculations. The code system includes internationally recognized codes as NJOY (Los Alamos National Laboratory; basic nuclear data processing and library generation), SCALE (Oak Ridge National Laboratory; deterministic neutron transport for (pin-) cells and assemblies), WIMS (AEA Technology, U.K.; deterministic neutron transport and diffusion for a variety of core subsystems), MCNP (LANL; general purpose continuous energy Monte Carlo), KENO (ORNL; groupwise Monte Carlo), ORIGEN (ORNL; fuel depletion and activation source terms), FISPACT (AEA Technology, U.K.; fuel depletion and activation) and PANTHER (Nuclear Electric, U.K.; steady-state, burnup and transient core physics) [4,5].

The maintenance, verification, validation and application of the codes is subject to strict quality assurance procedures, which guarantee the tracebility and reproducebility of the results obtained.

A relatively new development at ECN is the OCTOPUS system [3], which interconnects and integrates the codes and code systems mentioned above. This provides flexibility and the possibility of parallel routes through the entire code system, which also enables internal verification of calculation results: the same calculation result can by obtained by taking different routes through the code system, i.e. within the code system different calculational methods, e.g. deterministic and Monte Carlo, can be used to solve the same problem. At the moment all reactor physics codes use nuclear data libraries based on the Joint Evaluated File JEF-2.2, but for special purposes the basic nuclear data libraries can be extended.

The OCTOPUS code system is mainly being applied for criticality calculations, shielding analyses, HFR (High Flux materials testing Reactor) calculations, and cell and assembly calculations, in which reactivity coefficients and burnup effects for PWR, BWR, and Candu geometries are determined. Also sensitivity and uncertainty analyses for reactivity calculations and shielding applications can be carried out in this code system.

For steady-state neutronics HTR calculations initially the deterministic codes SCALE (version 4.1) and WIMS (versions E-5, 6 and 7) and the continuous energy Monte Carlo code MCNP (version 4A) [7] code were selected [8,9]. The SCALE system uses the Nordheim method (in the NITAWL code) for the resolved resonance treatment and the Bondarenko method (in the BONAMI code) for the unresolved resonance treatment. In both methods a Dancoff factor is applied in the treatment of the spatial heterogeneity. One-dimensional neutron transport calculations (in slab, cylindrical or spherical geometry) can be performed with the XSDRN code. However, in the SCALE code system no special options exist to treat the double heterogeneous geometry of HTR fuel (spherical fuel pebbles and cylindrical fuel compacts). Therefore, the calculation for a unit cell is performed in two stages [8,9]. First a "micro cell" (or grain cell) calculation is performed by BONAMI, NITAWL and XSDRN in the fine (172-group) energy group structure. The calculated neutron flux distribution in the micro cell is used to homogenize the fuel grain with the coatings and the matrix. This mixture is then used in a "macro cell" XSDRN reactivity calculation in which the entire fuel pebble (or compact) and moderator pebbles (if required) are modelled. The XSDRN neutron transport calculations can be performed using S_{32} quadrature and P_5 scattering order. The Dancoff factor for the double heterogeneous geometry, required in BONAMI and NITAWL, is calculated by the MCNP code, employing a special option built in by ECN [8,9].

The WIMS code (versions E-5, 6 and 7) contains a special module, PROCOL, which can explicitly treat the double heterogeneity associated with granular fuel in HTR fuel elements, both in the resonance shielding treatment by the subgroup method and the subsequent multi-group neutron transport calculation [10]. PROCOL was designed for spherical fuel grains (with possible coatings) embedded in an annular structure of matrix material, which is the correct geometry for the fuel compacts in MHTGR-like reactors [9]. As PROCOL is not capable of treating grains in spheres, an equivalent cylinder approximation was used to model fuel pebbles containing grains. In the equivalent cylinder approximation the radius of the fuel zone containing the grains is chosen in such a way that the average chord length of the spherical fuel zone is preserved, thereby preserving the escape probability in the fuel zone. In order to check this approach, MCNP calculations were performed for both models. A good agreement between equivalent cylinder and spherical unit cell is observed [8,9]. The deviation in k_∞ is within the statistical uncertainty (~ 0.1 %), whereas the deviation in spectral indices is less than 3 %.

For whole-core calculations the SNAP diffusion module is available in WIMS, which can treat e.g. 2-D R-Z and 3-D Hex-Z geometries. The latter is useful for modelling MHTGR-type geometries (consisting of hexagonal blocks), but it can also be used to approximate cylindrical structures (like pebble-bed HTRs) by specifying a small flat-to-flat distance of the hexagons (in the order of centimetres). The SNAP module initially has been checked for HTR application by comparing 2-group SNAP diffusion calculations and fine-group neutron transport calculations by the WIMS modules PROCOL and THESEUS for the same (cylindrical) geometry, yielding satisfactory agreement [9]. Further validation of SNAP (also in multigroup mode) has been done in the framework of the HTR-PROTEUS and PAP20 benchmark studies, presented in this paper.

The continuous energy Monte Carlo code MCNP (version 4A) offers the possibility to model explicitly all details of a 3-dimensional geometry. E.g. fuel grains (including the coating shells) in a unit cell can be modelled, thereby explicitly treating the double heterogeneity, but it is also possible to model fuel grains in pebbles in a complete (3-D) reactor [7,8]. In view of its extensive geometric and explicit resonance treatment capabilities the MCNP code was selected for reference calculations and also for shielding calculations.

In order to validate the codes mentioned above for steady-state pebble bed HTR applications, calculations have been performed within the framework of the HTR-PROTEUS benchmark [11]. In Section 3 some results will be presented of cell and whole core HTR-PROTEUS calculations by the three code systems.

For the HTR power reactor physics calculations the core physics code PANTHER was selected. The commercially available, general purpose modular reactor code PANTHER [4,5] has been in use at ECN for some time now. The code's capabilities comprise steady-state, burnup and transient combined reactor physics (neutronics) and thermal hydraulics calculations for a number of different reactor types. Its main areas of application concern light water (PWR, BWR) and gas cooled (Magnox, AGR) reactors, for which thermal hydraulic models are included in the code. However, the code does not contain a model for detailed thermal hydraulics (including more-than-one-dimensional heat transfer) of hexagonal prismatic or pebble-bed High Temperature Gas-cooled Reactors. For the purpose of pebble-bed HTR core physics calculations the thermal hydraulic code THERMIX/DIREKT [6] from KFA Jülich has been combined with the PANTHER code, enabling both steady state and burnup core physics and reactivity transient calculations.

2.2. PANTHERMIX - a PANTHER-THERMIX/DIREKT interaction

The combined code system PANTHER-THERMIX/DIREKT (*PANTHERMIX*) [12] consists of the PANTHER code, which calculates (in steady-state and transient mode) the 3-dimensional neutron flux and power distribution for a given temperature distribution, the THERMIX/DIREKT [6] code, which calculates (in steady-state or transient mode) the temperature and flow distribution for a given power distribution, and two interface codes for data transfer and format conversion between the main codes. THERMIX/DIREKT is an improved version of the THERMIX/KONVEK module of the VSOP code [13]. The combined code system is controlled from the PANTHER code, which calls the interfacing codes and THERMIX/DIREKT from within a calculation loop.

It is important to note that for the development of the combined code system PANTHERMIX no changes to the PANTHER code were required (Quality Assurance!) and that only minor changes were required to the THERMIX/DIREKT code. These changes concern only in/output procedures to generate of files containing data (fluid- and solid-state temperature distributions) to be transferred to PANTHER through one of the interfacing codes. The transfer of data (power distribution) from PANTHER to THERMIX/DIREKT is also performed by an interface code, which generates part of the THERMIX/DIREKT data input from the power distribution information contained in a standard formatted, generated by PANTHER.

In order to check the suitability of the PANTHER neutronics modelling for HTR applications an intercomparison was carried out, within the framework of an MHTGR benchmark exercise [9], between the WIMS module SNAP (in 2-D R-Z and 3-D Hex-Z mode) and PANTHER (in 3-D Hex-Z mode), both using 2-group nuclear constants generated by the WIMS code package, employing the PROCOL module. A very close agreement (within 1 pcm) was found for k_{eff} (in Hex-Z geometry). Also the agreement between the calculated k_{eff} for Hex-Z and R-Z geometries in SNAP (preserving material volumes) was found to be quite close [9], from which it may be concluded that a cylindrical structure (such as a pebble-bed HTR) can be approximated, for neutronics purposes, by a Hex-Z geometry, provided that the hexagons are not too large.

This is very important for the development of the PANTHERMIX code combination, in view of the fact that PANTHER has to approximate the 2-D R-Z geometrical structure employed in THERMIX/DIREKT by a 3-D Hex-Z model with small ("mini") hexagons. This is done as follows. A radial mesh in the THERMIX/DIREKT geometry specification can be associated with a set of hexagons in the PANTHER geometry description, of which the total area is equal to the area of the THERMIX/DIREKT radial mesh (see FIG. 1 and 2). For radial meshes and hexagon sets associated in this way the power distribution and temperature distribution information is transferred between the codes. The linear power densities for a set of hexagons associated with a particular radial mesh, as calculated by PANTHER, are averaged and transformed into a power density for the associated radial mesh, as required by THERMIX/DIREKT, thereby preserving the total mesh power. The fluid and solid-state temperatures for a radial mesh, as calculated by THERMIX/DIREKT, are transferred to all associated hexagons.

The procedure described above is followed for each axial mesh. For the axial direction no special treatment is necessary as the axial geometry can be modelled in the same way in both PANTHER and THERMIX/DIREKT.

PANTHERMIX requires a nuclear database containing - presently - 2-group nuclear data for all reactor materials, depending on irradiation, fuel temperature, xenon density, etc. The WIMS code (version 7) was selected to generate these databases, in view of the presence of the special PROCOL module and also because of the possibility to perform burnup calculations in a convenient way within the WIMS7 code system. The WIMS7 code system also provides the 6-group delayed neutron data (yields β_i), required for transient calculations by PANTHER. In the near future a multigroup (up to 12 neutron energy groups) version of PANTHER will be operational.

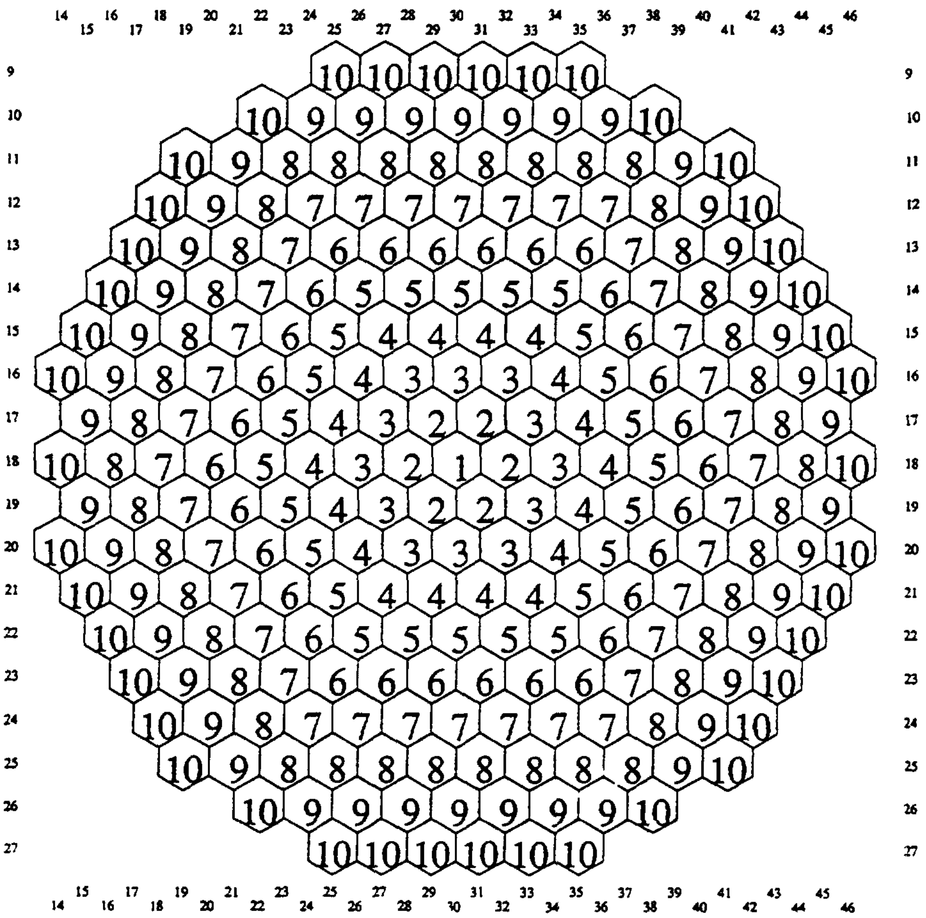

FIG. 1 "Mini" hexagon structure in PANTHER. The numbers indicate the associated radial mesh in FIG. 2.

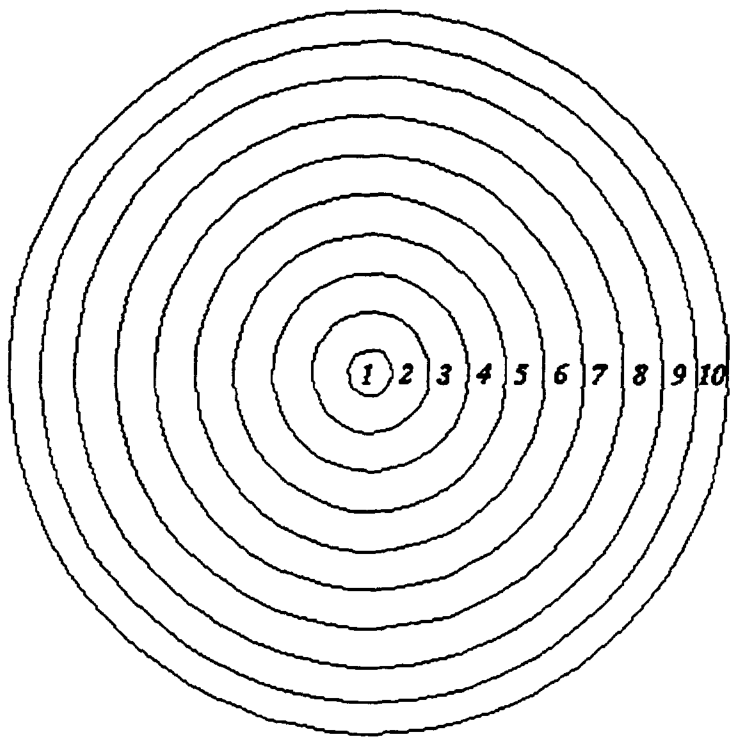

FIG. 2 Radial mesh structure in THERMIX/DIREKT, corresponding to the "mini" hexagon structure shown in FIG. 1.

In the PANTHERMIX code combination the core height of the pebble-bed HTR model (i.e. the position of the boundary between the core and the gas space) can be varied continuously by making use of the control rod treatment method of the PANTHER code. In the PANTHER part of PANTHERMIX this treatment uses "unrodded" material (the fuel pebbles) and "rodded" materials (the gas space above the core). The boundary between the "rodded" and "unrodded" material is marked by the position of the "control rod" tip. Also within-node "rod tip" positions are possible, in which case the "rodded" and "unrodded" material properties (viz. 2-group nuclear constants) for that particular node are mixed by axial flux-volume weighting. In the THERMIX/DIREKT part of PANTHERMIX the boundary between the top of the core and the gas space must coincide with an axial mesh boundary. PANTHERMIX automatically selects the top layer of the core in the THERMIX/DIREKT model in such a way that the actual upper boundary of the core, in the PANTHER model, is contained in the core top layer of the THERMIX/DIREKT model, thereby also avoiding direct energy production in the gas space.

As PANTHER uses diffusion theory, calculational stability problems can be expected when using the actual diffusion coefficients (in the order of metres) in the gas spaces. Gerwin and Scherer have provided a recipe [14] to calculate the effective radial and axial diffusion coefficients in a cylindrical gas (or vacuum) space. This recipe is succesfully applied in our pebble-bed HTR PANTHER calculations. However, the calculated k_{eff} and flux shapes are not sensitive to changes in the diffusion coefficients in the gas space, as long as these coefficients are still large.

The PANTHERMIX code combination can be used for both steady-state and transient pebble-bed HTR calculations. For some transient scenarios (e.g. LOCA - Loss Of Coolant Accident) the reactor is subcritical for an extended period of time until recriticality (as a consequence of xenon decay and decreasing temperatures) occurs. In order to prevent numerical instability and precision problems as the power keeps on decreasing during that period, a fixed source is modelled as a seventh delayed neutron group with a very small yield ($\beta_7 = 1.0 \times 10^{-9}$) and a small decay constant ($\lambda = 1.0 \times 10^{-5}$ s^{-1}; $T_{1/2} = 19.3$ h). In the cases studied this puts a lower limit on the reactor power of about 0.5 W.

The decay heat is modelled in PANTHER according to [15], employing 23 decay heat groups.

Presently a benchmark exercise is being carried out, together with KFA Jülich, on the PAP20 pebble-bed HTR, comparing PANTHERMIX with the KFA HTR code package. In Section 4 some initial results will be presented of steady-state, burnup and transient calculations on the PAP20 reactor, demonstrating the feasibility of the PANTHERMIX code combination for pebble-bed HTR applications.

3. PROTEUS BENCHMARK EXERCISE

3.1 Introduction

In the framework of the IAEA Co-ordinated Research Programme (CRP) on "Validation of Safety Related Physics Calculations for Low-Enriched (LEU) HTGRs" calculational benchmarks are performed on the basis of LEU-HTR pebble-bed critical experiments carried out in the PROTEUS facility at PSI, Switzerland. ECN joined this CRP early 1994 with the aim to validate the available reactor physics codes and nuclear data for application to this type of reactors. Of special interest is the treatment of the double heterogeneity of the fuel particles and the spherical fuel elements of these pebble bed core configurations. Also of interest is the proper calculation of the safety related physics parameters, like the effect of water ingress and control rod worth.

3.2. Problem description

The PROTEUS benchmark consists of a series of 6 graphite reflected 16.7 % enriched uranium pebble bed systems, named LEUPRO-1 to LEUPRO-6 [11]. There are 3 different types of geometrical arrangements of the pebble bed lattice (with packing fractions 0.74, 0.60 and 0.62, respectively) and 2 different moderator to fuel pebble ratios (1/2 and 2/1, respectively). The pebble diameter is 6 cm. Each fuel pebble contains about 9400 coated particles (in total about 1 g ^{235}U) in a graphite matrix. The diameter of the fuel zone is 5 cm. The coated particles (TRISO type) consist of a fuel grain with a diameter of 500 μm and 4 layers of different coatings, with a total diameter of 915 μm.

In the PROTEUS facility the cylindrical core, which is filled with fuel and moderator pebbles, has a diameter of 125 cm, and is surrounded by radial and axial graphite reflectors. The control and shutdown rods are situated in the radial reflector. The core is filled until criticality is reached.

The benchmark exercise comprises unit-cell calculations and whole-core calculations for the 6 different pebble-bed configurations. Unit cell calculations have been performed with WIMS (versions E-5 and 6) for LEUPRO-1 till -6, and with SCALE-4 and MCNP4A for LEUPRO-1 and -2 only (hexagonal closed-packed configuration with M/F ratio of 1/2). Whole-core calculations have been performed with WIMS (versions E-5 and 6) for LEUPRO-1 till -6 and with MCNP4A for LEUPRO-1. Emphasis is put on the intercomparison of the results obtained by the deterministic codes and the reference Monte Carlo calculations. For intercomparison of the codes all the ECN codes will make use of JEF-2.2 based nuclear data libraries.

3.3. Unit cell calculations

In the unit cell the fuel zone is represented by a sphere of 5 cm diameter surrounded by a 5 mm thick graphite shell, which together represent the fuel pebble. Around this fuel pebble sphere a spherical shell with outer diameter of 7.592 cm (for the LEUPRO-1 unit cell) represents the homogenized mixture of moderator pebbles and lattice void.

The unit cel calculations were performed for the single heterogeneous model and the double heterogeneous model. In the single heterogeneous model the coated particles and the matrix material in the fuel pebbles are homogenized over the fuel zone of 5 cm diameter. Of special interest are the cell calculations of the double heterogeneous model, in which the fuel grains are explicitly modelled inside the fuel zone of the fuel pebbles.

TABLE I, II and III show the final results of the double heterogeneous unit cell calculations for LEUPRO-1 and -2 (which are representative for the entire series of LEUPRO cases) obtained with WIMS (using the 69-group "1986" library and the 172-group, JEF-2.2 based "1993" "XMAS" library), SCALE-4 and MCNP4A. Also for each item to be determined the benchmark average [16] is presented. Good agreement of

TABLE I Unit cell calculation results for LEUPRO-1.

	WIMS6 (172 gr.)	SCALE (172 gr.)	MCNP	BM average [16]
$k_\infty(B^2 = 0)$	1.7274	1.727	1.729	1.7184
$k_\infty(B^2 = B_{cr}^2)$	1.6775	1.678	1.681	1.6721
B_{cr}^2 (cm^{-2})	7.91 -10^{-4}	-	-	7.73 -10^{-4}
M^2 (cm^2)	857	-	-	874
ρ^{28}	7.68	7.68	7.57	7.65
δ^{25}	0.110	0.109	0.109	0.112
δ^{28}	1.85 -10^{-3}	1.85 -10^{-3}	1.85 -10^{-3}	1.647 -10^{-3}
C^*	0.193	0.193	0.191	0.1921

TABLE II Unit cell calculation results for LEUPRO-1 with 20 vol. % water.

	WIMS6 (172 gr.)	SCALE (172 gr.)	MCNP	BM average [16]
$k_\infty(B^2 = 0)$	1.598	1.597	1.600	1.5971
$k_\infty(B^2 = B_{cr}^2)$	1.582	1.582	1.573	1.5791
B_{cr}^2 (cm^{-2})	-	-	-	1.325 -10^{-3}
M^2 (cm^2)	-	-	-	438
ρ^{28}	3.24	3.25	3.33	3.28
δ^{25}	0.0460	0.0457	0.0456	0.0467
δ^{28}	1.55 -10^{-3}	1.54 -10^{-3}	1.53 -10^{-3}	1.379 -10^{-3}
C^*	0.0971	0.0976	0.0992	0.0988

TABLE III Unit cell calculation results for LEUPRO-2.

	WIMS6 (172 gr.)	SCALE (172 gr.)	MCNP	BM average [16]
$k_\infty(B^2 = 0)$	1.7566	1.756	1.756	1.7508
$k_\infty(B^2 = B_{cr}^2)$	1.7198	1.720	1.722	1.7154
B_{cr}^2 (cm^{-2})	6.57 -10^{-4}	-	-	6.40 -10^{-4}
M^2 (cm^2)	1096	-	-	1117
ρ^{28}	4.17	4.17	4.21	4.17
δ^{25}	0.0583	0.0578	0.0591	0.0593
δ^{28}	1.178 -10^{-3}	1.200 -10^{-3}	1.230 -10^{-3}	0.958 -10^{-3}
C^*	0.119	0.119	0.120	0.1188

k-values and spectral indices for double heterogeneous unit cell calculations of the pebble bed HTR type reactor configuration is observed between the deterministic and Monte Carlo types of codes. These results are also in reasonable agreement with the results of other participants in the PROTEUS benchmark. More details are given in [8,9,16,17,18].

3.4. Whole reactor calculations

Whole-core calculations have been performed with WIMS (version E-5 with the 69 group "1986" library and version 6 with the 172 group "1994" "XMAS" library) for LEUPRO-1 till -6 and with MCNP for LEUPRO-1 only.

The simplified two-dimensional R-Z geometry (see FIG. 3) was used in the whole-core calculations by both WIMS and MCNP. In this model the core cavity is filled with a

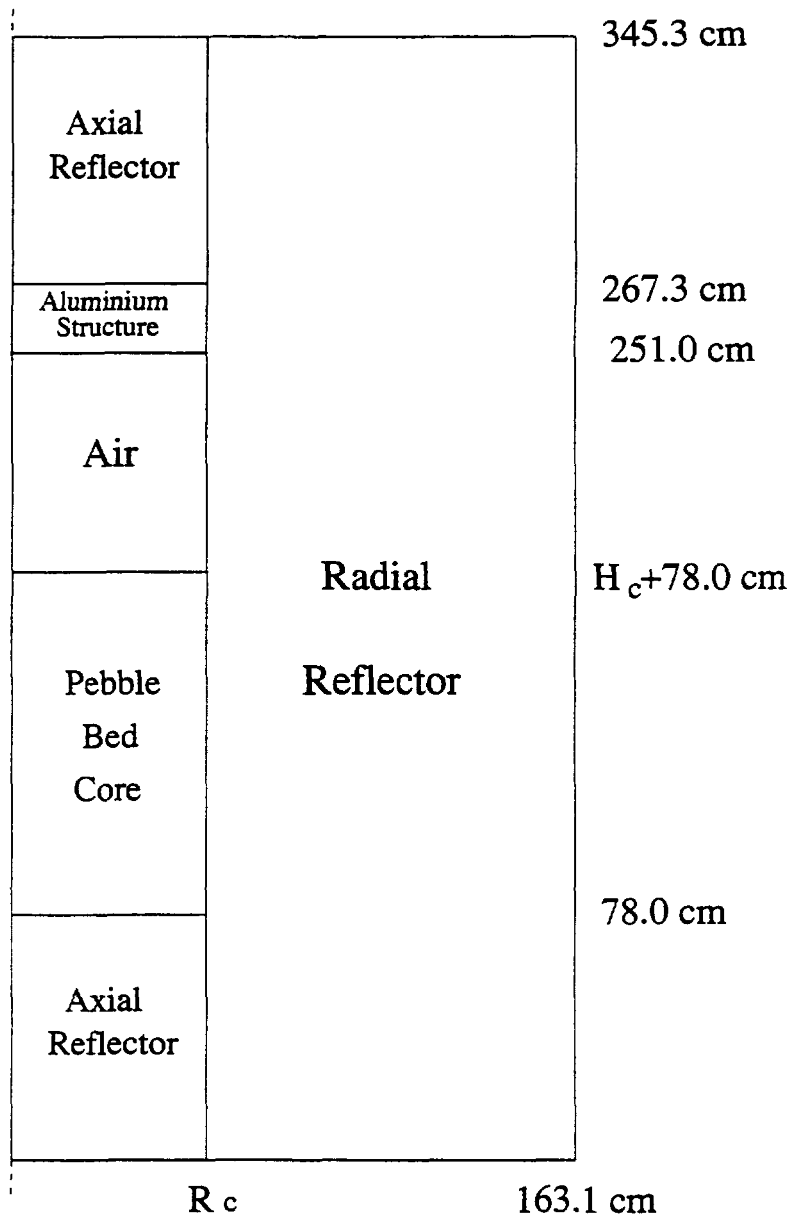

FIG. 3 R-Z model of HTR-PROTEUS.

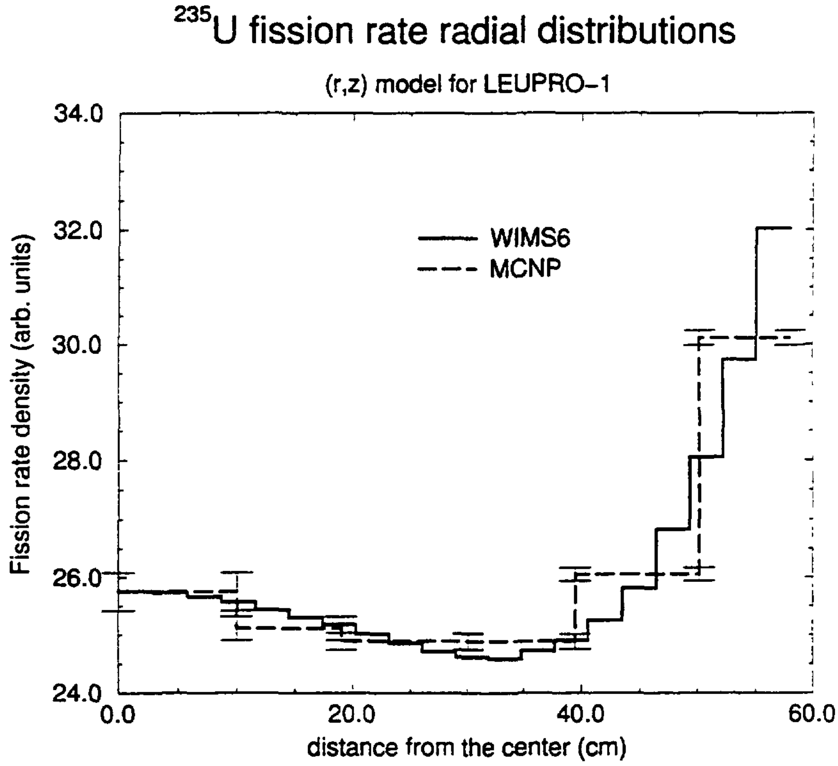

FIG. 4 Comparison of WIMS and MCNP calculation results: radial distribution of the ^{235}U fission rate for LEUPRO-1.

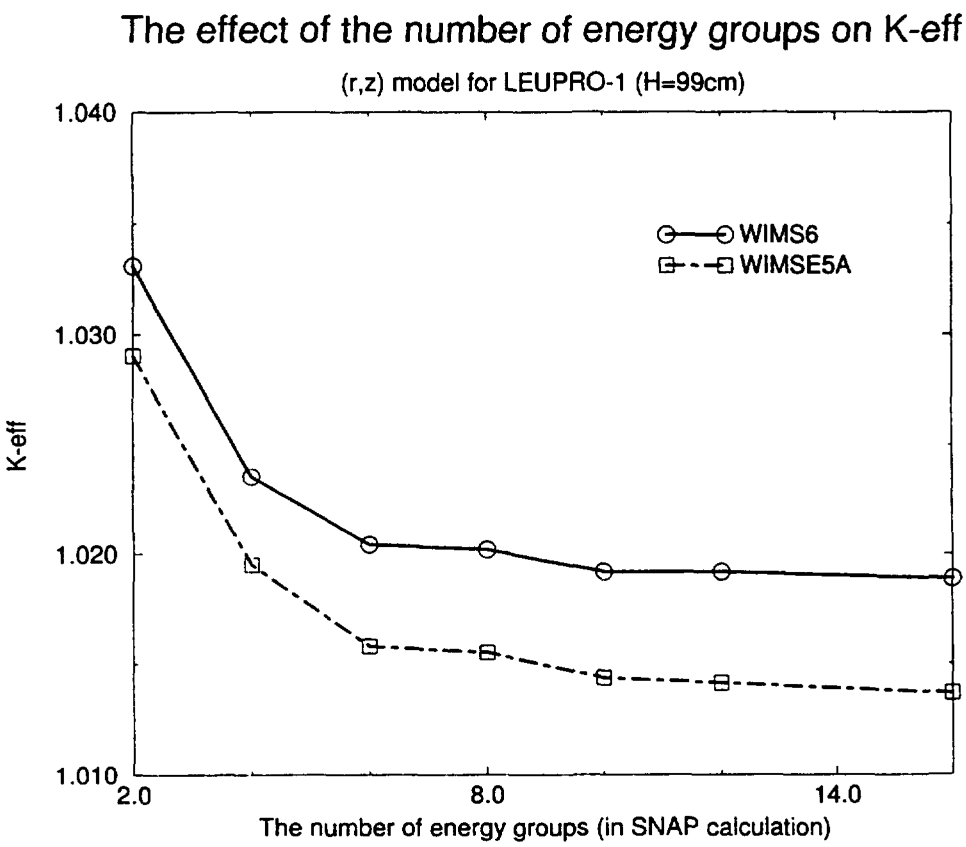

FIG. 5 Influence of the number of energy groups on the calculated k_{eff}, as calculated by WIMS/SNAP in 2-D R-Z mode for LEUPRO-1.

mixture of fuel and moderator pebbles. The core is surrounded by radial and axial reflectors. Between the pebble bed core and the upper axial reflector, there is a void space and an aluminium structure which is 16.3 cm thick.

In WIMS the whole core calculations have been performed with the diffusion module SNAP (in 2-D R-Z geometry) with a small number of energy groups (varying from 2 to 16 groups). The few-group cross sections for all materials and zones were generated by condensing fine energy groups after one-dimensional radial and axial calculations with the WIMS modules PROCOL and THESEUS, respectively. In MCNP the LEUPRO-1 core (in the R-Z geometry as of FIG. 3) with all individual fuel and moderator elements was explicitly modelled in closed packed hexagonal geometry.

TABLE IV, V and VI show the results of whole core calculations for LEUPRO-1 and -2 (which are representative for the entire series of LEUPRO cases), obtained with WIMS/SNAP (10 energy groups) and MCNP (for LEUPRO-1 only). For each item to be determined also the average outcome of the calculations of all benchmark participants is presented [16]. For LEUPRO-1 (with and without water ingress) a good agreement is found between the results obtained with WIMS (version 6) and those obtained with MCNP, with both codes using JEF-2.2 based nuclear data libraries. Also a reasonable agreement is found with the results of the other benchmark participants. However, the k_∞- and k_{eff}-values, as calculated by the various participants, agree much better with each other than the spectral indices, which may be associated to compensating errors in the eigenvalue calculations. The values of δ^{28} and - to a lesser extent - δ^{25} are particularly discrepant between the various participants [16], which calls for a further investigation.

The good agreement between the WIMS and MCNP results is demonstrated once more in FIG. 4, which shows the ^{235}U fission rate distribution for LEUPRO-1 in radial direction, as calculated by WIMS and MCNP.

In the analysis also the influence of different mesh sizes, geometry descriptions and number of energy groups have been examined [18]. In FIG. 5 the influence is shown of the number of energy groups in the 2-D R-Z SNAP calculation on the value of k_{eff} for the LEUPRO-1 case. For an increasing number of energy groups a lower value of k_{eff} is found. So, a 2-group calculation, as is performed e.g. with the PANTHER code, is expected to yield a conservative value for k_{eff}.

3.5 Conclusions

Calculations have been performed, by the WIMS, SCALE and MCNP codes, for unit cell and whole reactor geometries of the LEUPRO-1 till -6 cases of the HTR-PROTEUS benchmark. For the unit cell calculations a particularly good agreement was found between the results of the codes in use at ECN and also generally a good agreement was found with the results of the other benchmark participants. Also for the whole reactor calculations a good agreement was found between the results of the WIMS and MCNP codes.

The results obtained with the WIMS, SCALE and MCNP codes for the unit cell calculations, and those obtained with WIMS and MCNP for the whole core PROTEUS calculations indicate that the codes in use at ECN are quite suitable for (steady-state neutronics only) pebble-bed HTR applications. For further investigations, in particular the INCOGEN core design and optimization, the WIMS code (version 7) was selected for steady-state calculations (without thermal hydraulics feedback) and nuclear database generation for combined neutronics/thermal hydraulics calculations by PANTHERMIX. For selected cases, the WIMS (and PANTHER) calculation results will be checked against a detailed geometry/continuous energy reference calculation by the MCNP code.

Presently, the MCNP code, and associated cross section library, is being validated further for pebble-bed HTR applications by calculating the PROTEUS Core 5 experiment. First results indicate that MCNP reproduces very well the experimental value of k_{eff}.

4. PAP BENCHMARK

4.1. Introduction

PAP20 is a conceptual design of KFA Jülich for a small simplified pebble bed reactor of 20 MW (thermal) where new fuel elements are added little by little (i.e. peu à peu) [2]. No fuel is removed during the life of the core. At *E*nd *O*f *L*ife (EOL) all fuel elements are unloaded in one step. The initial core height is about 1 m. At the end of the core life (about 18 years) the core height is about 4 m. This fuelling mode avoids the handling of irradiated elements over the core life and devices for fuel unloading during reactor operation are superfluous.

In view of its simplicity (no fuel unloading during operation, no fuel recycling) the PAP20 was selected to serve as benchmark for the validation of the PANTHERMIX code combination for steady-state, burnup and transient pebble-bed HTR core calculations. Presently this benchmark study is being carried out, together with KFA Jülich, comparing the code system is use at KFA (viz. VSOP [13]) with PANTHERMIX, and also with WIMS/SNAP and MCNP, for selected steady-state cases.

The benchmark exercise is still in progress, but some first results of steady-state, burnup and transient calculations on the PAP20 by the PANTHERMIX code combination will be presented.

4.2. Benchmark description

In FIG. 6 the - simplified - PAP20 geometry is presented, together with the geometry of the first conceptual design of the INGOGEN reactor (to be discussed in Section 5), which is almost identical to the PAP20, except for the B_4C-layer in the bottom reflector. In TABLE VII the main parameters of both PAP20 and INCOGEN (first concept) are listed. In both reactors the fuel is contained in the well-known 6 cm fuel pebbles. Compared to the original PAP20 model some radial dimensions have been slightly changed, in order to facilitate "mini-hexagon" modelling in the PANTHER part of PANTHERMIX. Further details of the (modified) PAP20 reactor will be given in the benchmark report (to be published).

The PAP20 benchmark exercise comprises initial (*B*eginning *O*f *L*ife - BOL) steady-state, burnup (fill up) and transient calculations. Presently, only the definition of the initial steady-state part has been agreed upon by ECN and KFA. The definition of the burnup and transient cases is in progress. However, some preliminary calculations, based on proposed specifications, have already been performed. The final specifications of all benchmark cases will be included in the forthcoming benchmark report.

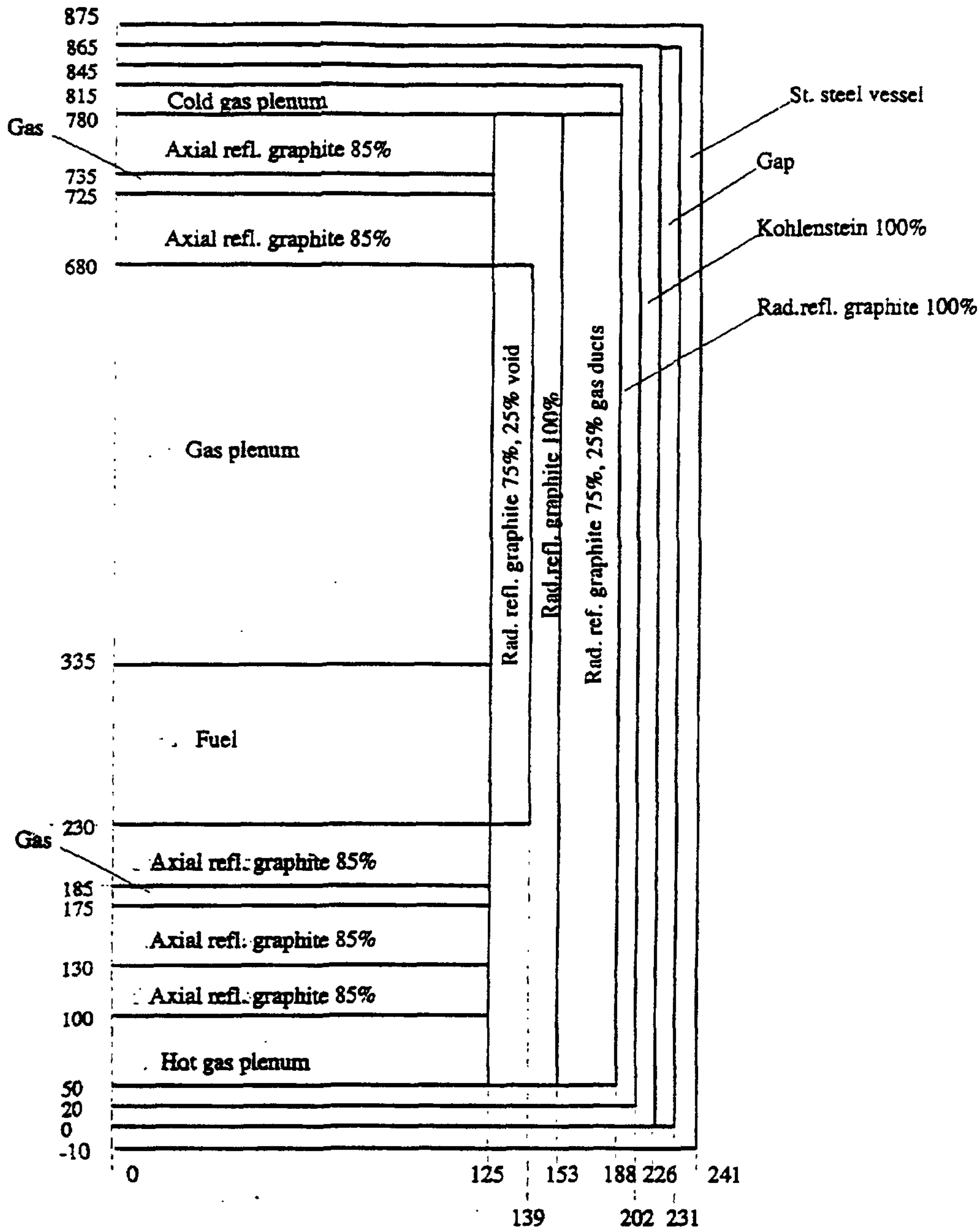

FIG. 6 Geometry of the (modified) PAP20 and (the first concept of) INCOGEN. The bottom reflector of INCOGEN contains B_4C.

The four cases of the initial (BOL) steady-state part of the benchmark exercise are the following:

1. CZP - Cold Zero Power
Uniform temperature throughout reactor: 20 °C (293 K)
Helium temperature: 20 °C (293 K)
Adiabatic boundary condition (no heat transfer to ambient)

2. HZP1 - Hot Zero Power 1 ("low" temperature)
Uniform temperature throughout reactor: 550 °C (823 K)
Helium inlet temperature: 550 °C (823 K)
Helium mass flow: 18.7 kg/s
Adiabatic boundary condition (no heat transfer to ambient)

3. HZP2 - Hot Zero Power 2 ("high" temperature)
Uniform temperature throughout reactor: 750 °C (1023 K)
Helium inlet temperature: 750 °C (1023 K)
Helium mass flow: 18.7 kg/s
Adiabatic boundary condition (no heat transfer to ambient)

4. HFP - Hot Full Power (nominal conditions; see TABLE VII)
Power: $P_{nom} = 20$ MW
Xe-equilibrium
Helium inlet temperature: 550 °C (823 K)
Helium outlet temperature: ~750 °C (1023 K)
Helium pressure: 50 bar
Nominal helium mass flow: 18.7 kg/s
Heat transfer coefficient to ambient: $\alpha = 10^{-3}$ W/cm^{2}K or $\alpha = 10^{-6}$ W/cm^{2}K (approx. adiabatic)
Search on helium mass flow to reach criticality

4.3. Calculation method

At ECN, the PAP20 benchmark core calculations are generally being performed by the PANTHERMIX code combination in 2 energy groups. In the PANTHER part of the PAP20 model, the 2-D R-Z structure is approximated by a 3-D Hex-Z structure, consisting of "mini-hexagons" with a flat-to-flat distance of 14.96 cm (i.e. an area of 193.92 cm^2). As mentioned above, the original radial mesh dimensions of the PAP20 have been slightly changed to ensure equivalence of corresponding volumes between the Hex-Z and R-Z descriptions of the geometry.

Reference calculations for the initial steady-state cases with negligible thermal hydraulic feedback (i.e. with approximately uniform temperature throughout the geometry) have also been performed by WIMS/SNAP (R-Z, multigroup diffusion theory) and MCNP (R-Z, continuous energy Monte Carlo).

For PANTHER calculations a nuclear database is required, which contains 2-group nuclear constants for all reactor materials, depending upon e.g. (local) irradiation (burnup) and (local) temperature. The nuclear database for the HTR applications presented here was generated by WIMS (version 7) burnup and branching calculations in a 1-D cylindrical and in a 1-D slab pseudo reactor, respectively, using the 69-group WIMS "1995" library, based on JEF-2.2.

4.4. Initial steady-state calculations

One of the items to be determined for the initial steady-state cases is the critical core height. In PANTHERMIX the critical core height search can be performed by actually adjusting the core height continuously (employing the control rod insertion mechanism of PANTHER) until criticality is reached. For the codes WIMS/SNAP and MCNP the critical core height search is performed by calculating k_{eff} voor different core heights H_c. As $1/k_{eff}$ is, to a very good approximation, a linear function of $1/H_c^2$, the critical value of H_c can be easily obtained.

TABLE IV Whole core calculation results for LEUPRO-1.

	WIMS6 (JEF-2.2)	WIMSE-5 ("1986")	MCNP (JEF-2.2)	BM average
$k_{eff}(H = 99$ cm)	1.01911	1.01437	1.0279	1.0211
H_{cr} (cm)	94.0	95.2	93.0	93.54
Prod./abs. (core)	1.76871	1.76998	-	1.7589
Core centre:				
ρ^{28}	7.8457	7.90513	7.563	7.68
δ^{25}	0.11974	0.11173	0.1130	0.1139
δ^{28}	1.89519 -10^{-3}	1.92133 -10^{-3}	1.917 -10^{-3}	1.75 -10^{-3}
C^{*}	0.19597	0.19536	0.1895	0.1921
Core average:				
ρ^{28}	4.7281	4.7974	-	4.94
δ^{25}	6.7099 -10^{-2}	6.7449 -10^{-2}	-	7.36 -10^{-2}
δ^{28}	1.4411 -10^{-3}	1.4261 -10^{-3}	-	1.341 -10^{-3}
C^{*}	0.13043	0.13747	-	0.1357

TABLE V Whole core calculation results for LEUPRO-1 with 20 vol. % water.

	WIMS6 (JEF-2.2)	MCNP (JEF-2.2)	BM average [16]
$k_{eff}(H = 99$ cm)	1.10800	1.1103	1.1122
H_{cr} (cm)	71.7	69.5	70.10
Prod./abs. (core)	1.60065	-	1.5979
Core centre			
ρ^{28}	3.2368	3.488	3.16
δ^{25}	4.5850 -10^{-2}	4.644 -10^{-2}	4.62 -10^{-2}
δ^{28}	1,5462 -10^{-3}	1.561 -10^{-3}	1.445 -10^{-3}
C^{*}	0.097113	0.1030	0.0955
Core average			
ρ^{28}	2.5826	2.640	2.66
δ^{25}	3.6537 -10^{-2}	3.705 -10^{-2}	3.74 -10^{-2}
δ^{28}	1.2854 -10^{-3}	1.398 -10^{-3}	1.163 -10^{-3}
C^{*}	0.082641	0.08398	0.0847

TABLE VI Whole core calculation results for LEUPRO-2.

	WIMS6 (JEF-2.2)	WIMSE-5 ("1986")	BM average [16]
k_{eff}(H = 99 cm)	0.9918	0.9862	0.9941
H_{cr} (cm)	141.7	144.3	140.90
Prod./abs. (core)	1.76233	1.76511	1.7573
Core centre			
ρ^{28}	4.2243	4.2627	4.16
δ^{25}	5.8785 -10^{-2}	5.8942 -10^{-2}	5.91 -10^{-2}
δ^{28}	1.1937 -10^{-3}	1.2144 -10^{-3}	0.991 -10^{-3}
C^*	0.11962	0.11930	0.1179
Core average			
ρ^{28}	2.9015	2.9518	2.99
δ^{25}	4.0340 -10^{-2}	4.0798 -10^{-2}	4.40 -10^{-2}
δ^{28}	0.9917 -10^{-3}	1.0468 -10^{-3}	0.863 -10^{-3}
C^*	0.09028	0.090597	0.0920

TABLE VII Main (nominal) parameters of PAP20 and INCOGEN

	PAP20	INCOGEN
Power (MWth)	20	40
He inlet temperature (°C)	550	493
He outlet temperature (°C)	750	~800
He pressure (bar)	50	23.2
He mass flow (kg/s)	18.7	25.0
Core diameter (m)	2.50	2.50
Core cavity height (m)	4.00	4.00
Fuel	UO_2 10 % enriched (startup) 19.75 % enriched (fuelling)	UO_2 10 % enriched (startup) 19.75 % enriched (fuelling)
UO_2 density (g/cm^3)	10.5	10.5
HM loading per pebble (g)	12	12

TABLE VIII PAP20 critical core height (in cm)

Benchmark case	CZP	HZP1	HZP2	HFP
2-gr. PANTHERMIX	81.0	95.0	101.0	112.5
2-gr. SNAP	82.2	96.4	102.0	-
4-gr. SNAP	81.7	-	-	-
6-gr. SNAP	82.2	-	-	-
MCNP	82.0	-	-	-
4-gr. VSOP	86	94	99	110

TABLE IX PAP20 steady-state HFP results

	Non-adiabatic ($\alpha = 10^{-3}$ W/cm^{2}K)		Adiabatic ($\alpha = 10^{-6}$ W/cm^{2}K)	
	PANTHERMIX	VSOP	PANTHERMIX	VSOP
He inlet temp. (°C)	550	550	550	550
He outlet temp. (°C)	751	750	750	750
He mass flow (kg/s)	18.7	18.7	19.2	19.25
Power (MW)	20	20	20	20
Heat removal by He flow (MW)	19.5	-	20	-
Heat leakage (MW)	0.5	-	0	-
Max. power density (MW/m^3)	5.86	5.55	5.86	5.55
Max. fuel temp (°C)	843	911	840	908
Max. gas temp (°C)	769	762	767	761
k_{eff}	0.99671	1.00061	0.99670	1.00130

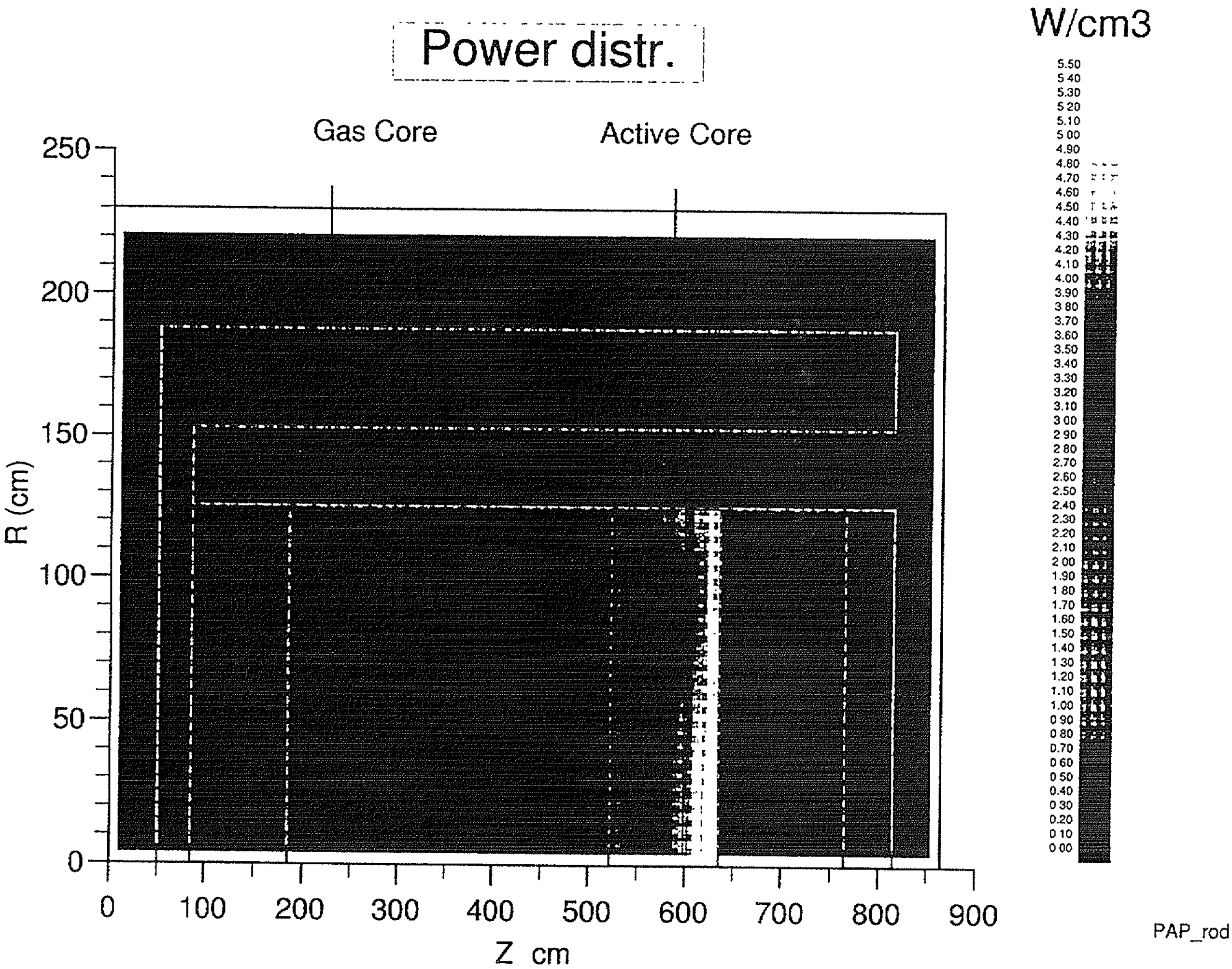

FIG. 7 PAP20 HFP power distribution at BOL with xenon-equilibrium (R denotes the radial -horizontal- distance to the core centre line).

In TABLE VIII the critical core heights are listed for the four BOL benchmark cases, as calculated by PANTHERMIX, WIMS/SNAP, MCNP and also by VSOP (KFA Jülich [19]). Note the very good correspondence, for the CZP case, between the results of 2-group PANTHERMIX, 2-, 4- and 6-group WIMS/SNAP and continuous energy MCNP calculations. Also note the good correspondence, for both HZP cases, between the results of 2-group PANTHERMIX and WIMS/SNAP calculations. Comparison of the results of 2-group PANTHERMIX calculations with those of VSOP shows that for the CZP case the critical core height, as predicted by PANTHERMIX is about 5 cm less than the value predicted by VSOP, whereas for the "hot" (HZP and HFP) cases the critical core height, as predicted by PANTHERMIX is approx. 2 cm higher than the value predicted by VSOP. Further investigation of these discrepancies seems justified, but it should also be noted that all ECN calculations are based on JEF-2.2 cross section data, contrary to the VSOP calculations by KFA.

In TABLE IX some further results for the HFP case are presented. Note the generally good correspondence between the results obtained by PANTHERMIX and those obtained by VSOP. However, one item, viz. the maximum fuel temperature, seems to be rather discrepant between PANTHERMIX and VSOP. The maximum fuel temperatures, as predicted by VSOP, are 68 °C higher than the corresponding values obtained by PANTHERMIX. This is connected to a difference in thermal hydraulic modelling. KFA has specified a heterogeneous pebble model in the THERMIX part of VSOP, whereas ECN presently uses a homogeneous pebble model in the THERMIX part of PANTHERMIX. So, the VSOP calculations yield the centre temperature of the pebbles, whereas the present PANTHERMIX calculations yield the average pebble temperature. However, the THERMIX part of PANTHERMIX is also capable of modelling heterogeneous pebbles. So, for consistency reasons, the heterogenous pebble model will be incorporated in the benchmark specifications and consequently the final benchmark calculation results will be obtained employing a heterogeneous pebble model.

In FIG. 7, 8 and 9, respectively, the power distribution, the solid state temperature distribution and the gas flow distribution are shown for the HFP case with $\alpha = 10^{-3}$ W/cm^{2}K. Note that for the BOL HFP situation the maximum power density and maximum fuel temperature occurs at the bottom of the core. Also note that the gas flow in the core and in the gas space above the core is only in the vertical (downward) direction, the average mass flow density being 3.81 $\cdot 10^{-4}$ kg s^{-1}cm^{-2}.

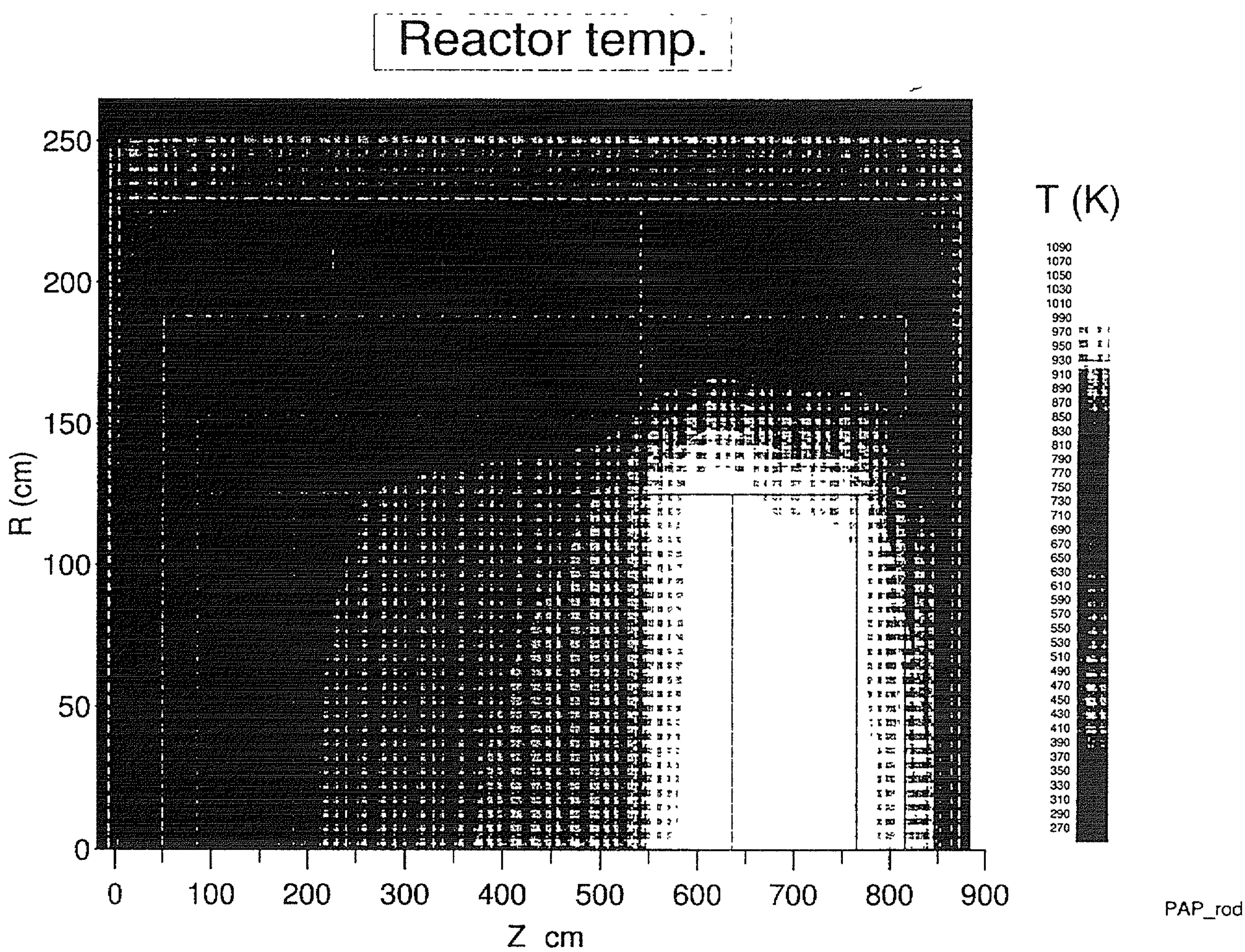

FIG. 8 PAP20 HFP solid state temperature distribution at BOL with xenon-equilibrium (R denotes the radial -horizontal- distance to the core centre line).

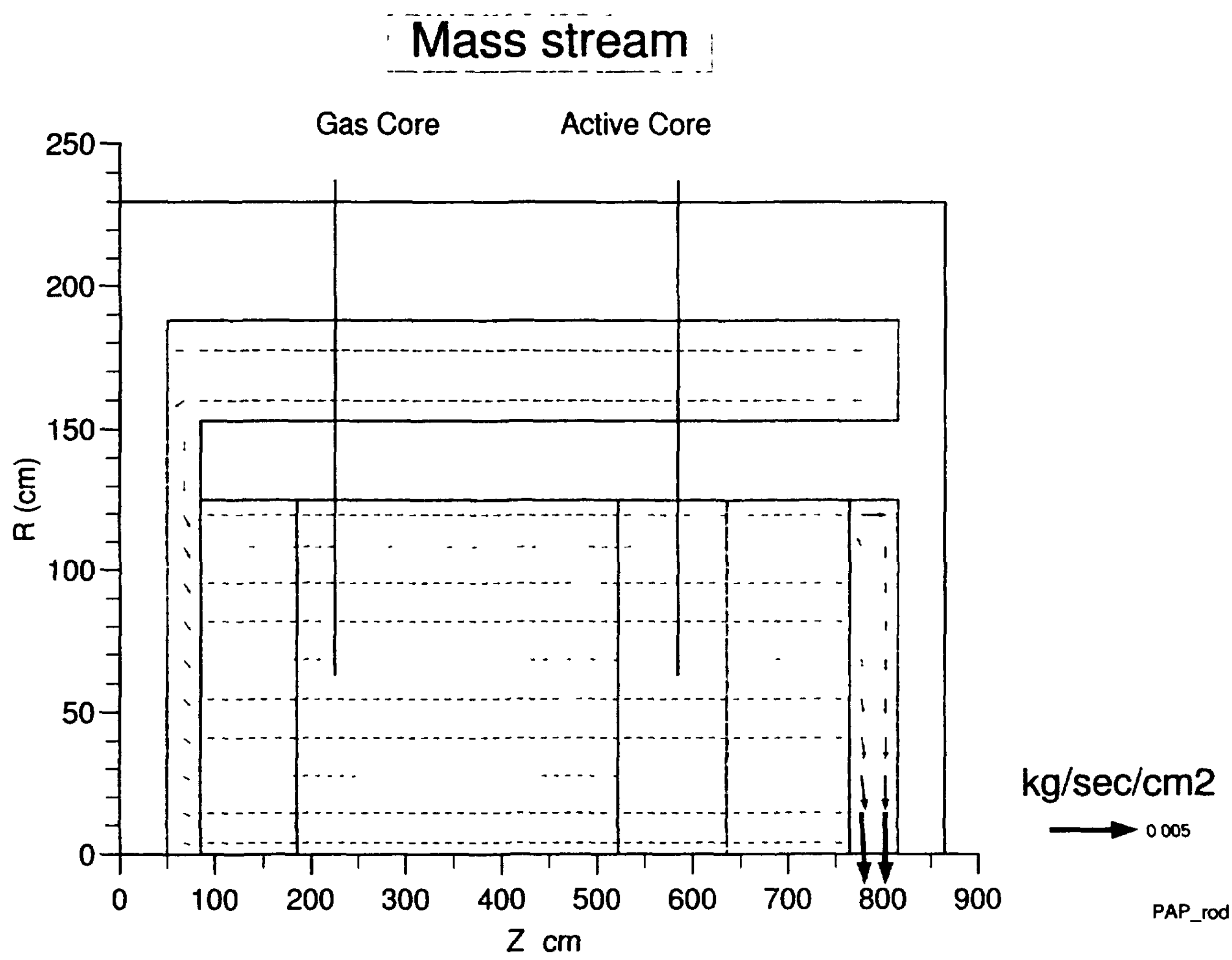

FIG. 9 PAP20 HFP gas flow distribution (mass flow density distribution) at BOL with xenon-equilibrium (R denotes the radial -horizontal- distance to the core centre line).

4.5. Burnup calculations

As mentioned above, the specification of the burnup cases of the PAP20 benchmark is still in progress. However, some preliminary calculations have been performed, on the basis of proposed specifications. In these calculations 10 % enriched fuel is added instead of 19.75 %. For the final benchmark calculations the 19.75 % enriched fuel will be used.

The core height in the PANTHER part of PANTHERMIX can be continuously varied. This offers the possibility of performing the burnup/fuelling calculations in two different ways:

1. Continuous fuelling at constant reactor power, hereby maintaining criticality throughout core life.

2. Batch refuelling at constant reactor power. Each time a complete axial mesh is filled with fresh (10 % enriched; or, following the benchmark specifications, 19.75 %) fuel. So, just after refuelling the reactor is supercritical. This is the method employed in VSOP.

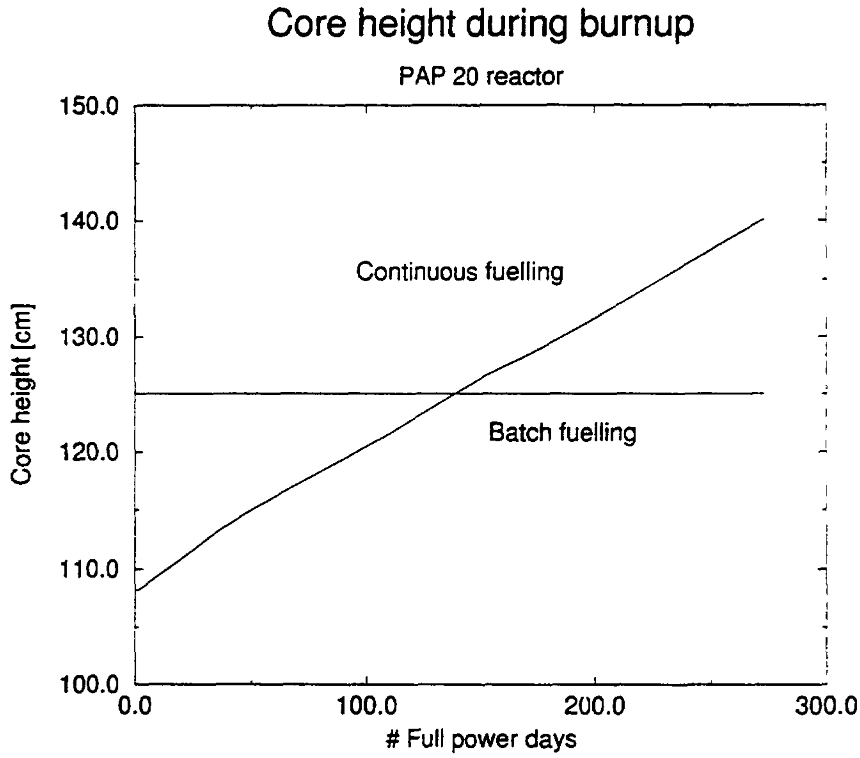

FIG. 10 PAP20 core height during burnup.

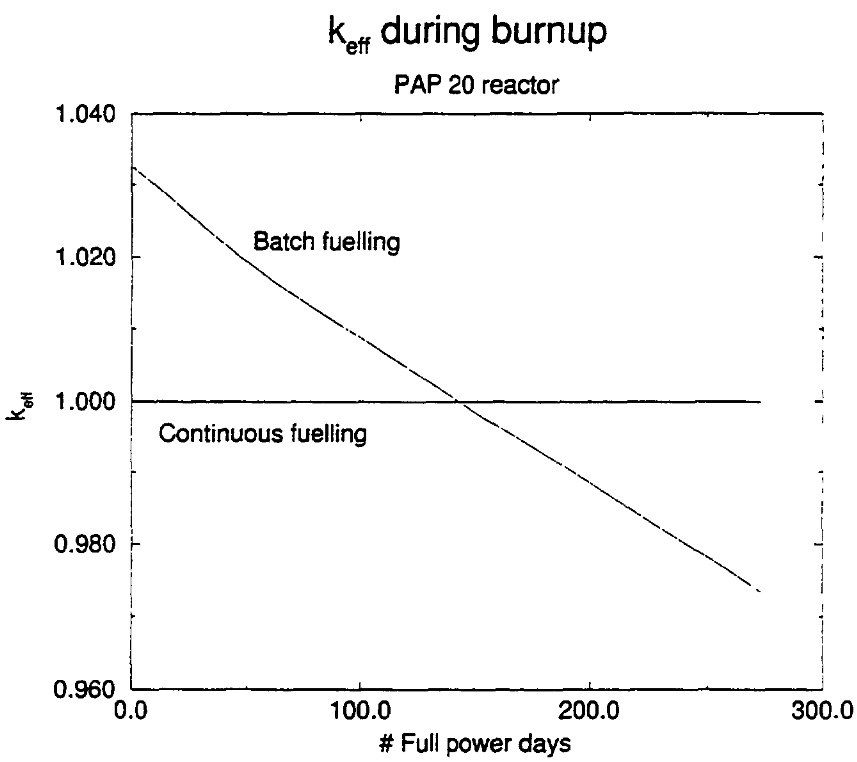

FIG. 11 PAP20 k_{eff} during burnup.

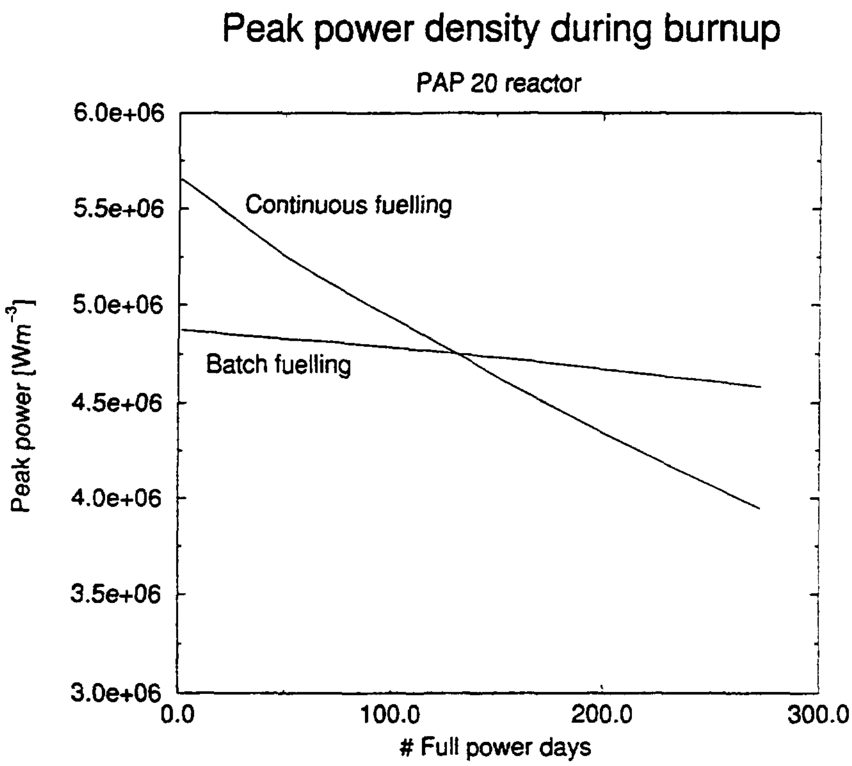

FIG. 12 PAP20 peak power density during burnup.

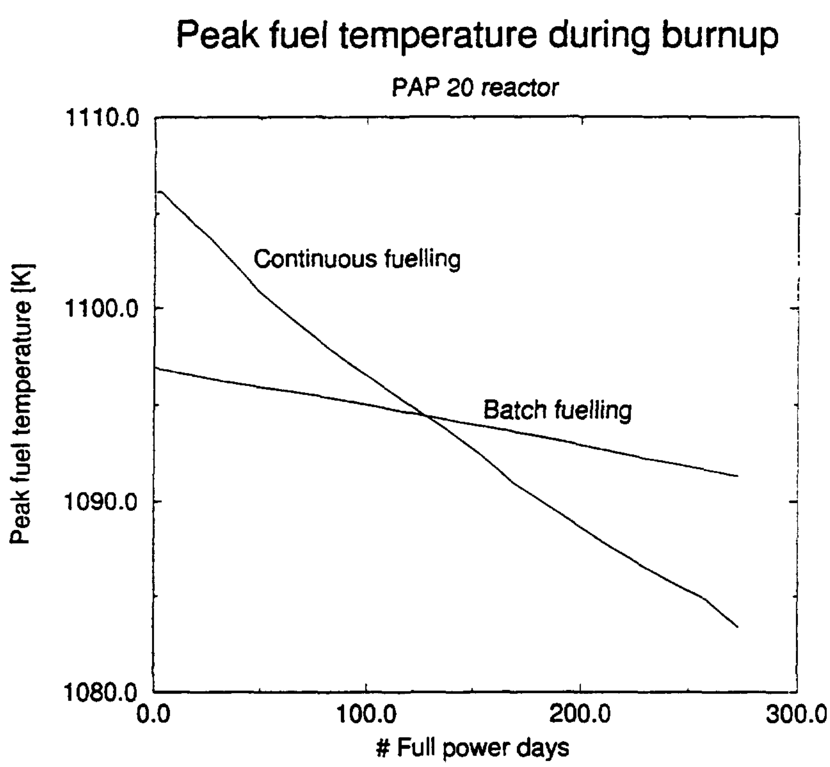

FIG. 13 PAP20 peak fuel temperature during burnup.

The two methods of calculation are compared in FIG. 10, 11, 12 and 13, which show the core height, k_{eff}, peak power density and the peak fuel temperature, respectively, as a function of the number of full power days. Note that, for the "batch" fuelling case, only the effect of a single batch is shown.

Comparing the two methods of calculation, it is observed that, for equal numbers of full power days, and approximately equal average core height, the continuous fuelling, which closer approximates the actual fuelling scheme of the reactor, yields higher values for peak power and peak fuel temperature.

In the forthcoming benchmark report the final specifications and calculation results of the burnup cases will be presented. Also a comparison of ECN and KFA results will be included.

4.6. Transient calculations

To demonstrate further the capabilities of PANTHERMIX, two transient calculations were performed on the basis of proposed benchmark specifications for a *L*oss *O*f *F*low *A*ccident (LOFA) and a *L*oss *O*f *C*oolant *A*ccident (LOCA). The initial state is BOL at nominal (HFP) conditions (see TABLE VII) with Xe-equilibrium. The following scenarios have been calculated:

LOFA - Loss Of Flow Accident
Linear helium mass flow decrease from 18.7 kg/s to 0.01 kg/s in 30 s.
Helium pressure of 50 bar is maintained.

LOCA - Loss Of Coolant Accident
Linear helium mass flow decrease from 18.7 kg/s to 0.01 kg/s in 10 s.
Linear helium pressure decrease from 50 bar to 1 bar in 10 s.

In FIG. 14 and 15 the power and the maximum fuel temperature as function of time are shown for the LOFA case. The maximum fuel temperature initially rises quickly from 833 °C (1106 K) to 887 °C (1160 K), after which it decreases (at a rate of approximately 0.35 K/min) to 719 °C (992 K), at t = 489 min, at which recriticality occurs for the first time due to xenon decay and - mainly - to temperature decrease. The first power peak (of 0.5 MW) occurs at t = 500 min, whereas the second power peak occurs at t = 540 min. After the first recriticality, the time-averaged power and time-averaged maximum fuel temperature slowly increase. The simulation ends at t = 1650 min, at which, however, neither the power nor the maximum fuel temperature have reached their final values. The final calculation, to be presented in the forthcoming benchmark report, will proceed until the final values have been reached within acceptable limits.

FIG. 16 shows the flow pattern at t = 400 min, at which only decay heat is being produced, as the reactor is still subcritical. At the moment of the "snapshot" almost no gas circulation seems to occur, except in the gas space above the core, where a natural convection flow pattern has developed, which goes upward along the core centre line and downward at the radial reflector. This natural convection pattern largely takes care of the heat transfer from the top of the core to the reflector. Although hardly visible in the picture, in the core also some natural convection flow is present. Heat transfer from the core to the ambient therefore takes place by radiation, natural convection within and above the core, and conduction.

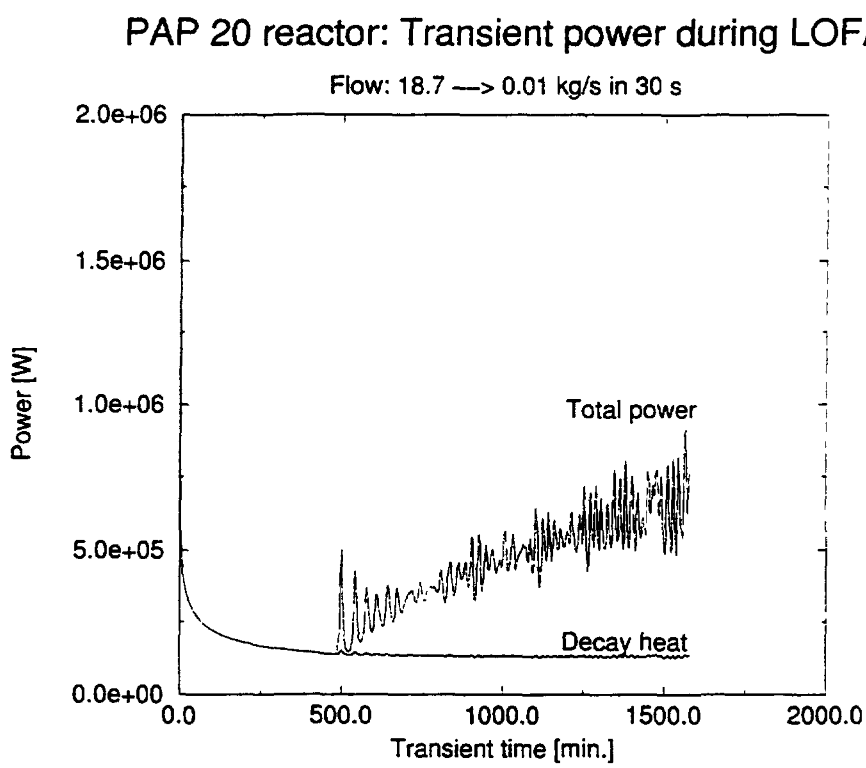

FIG. 14 PAP20 LOFA: power as function of time.

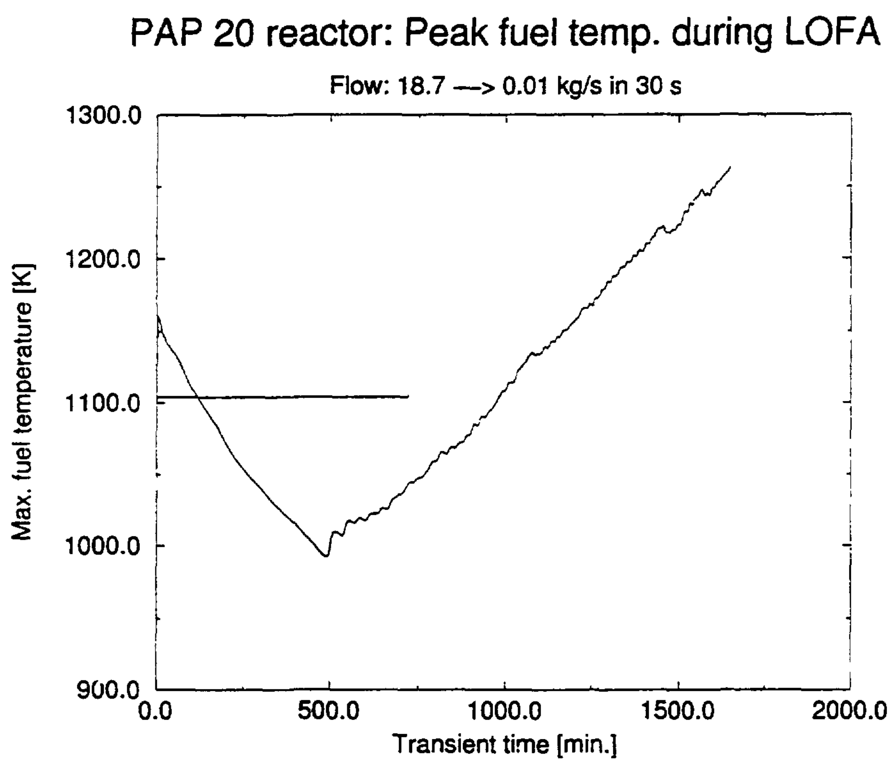

FIG. 15 PAP20 LOFA: maximum fuel temperature as function of time.

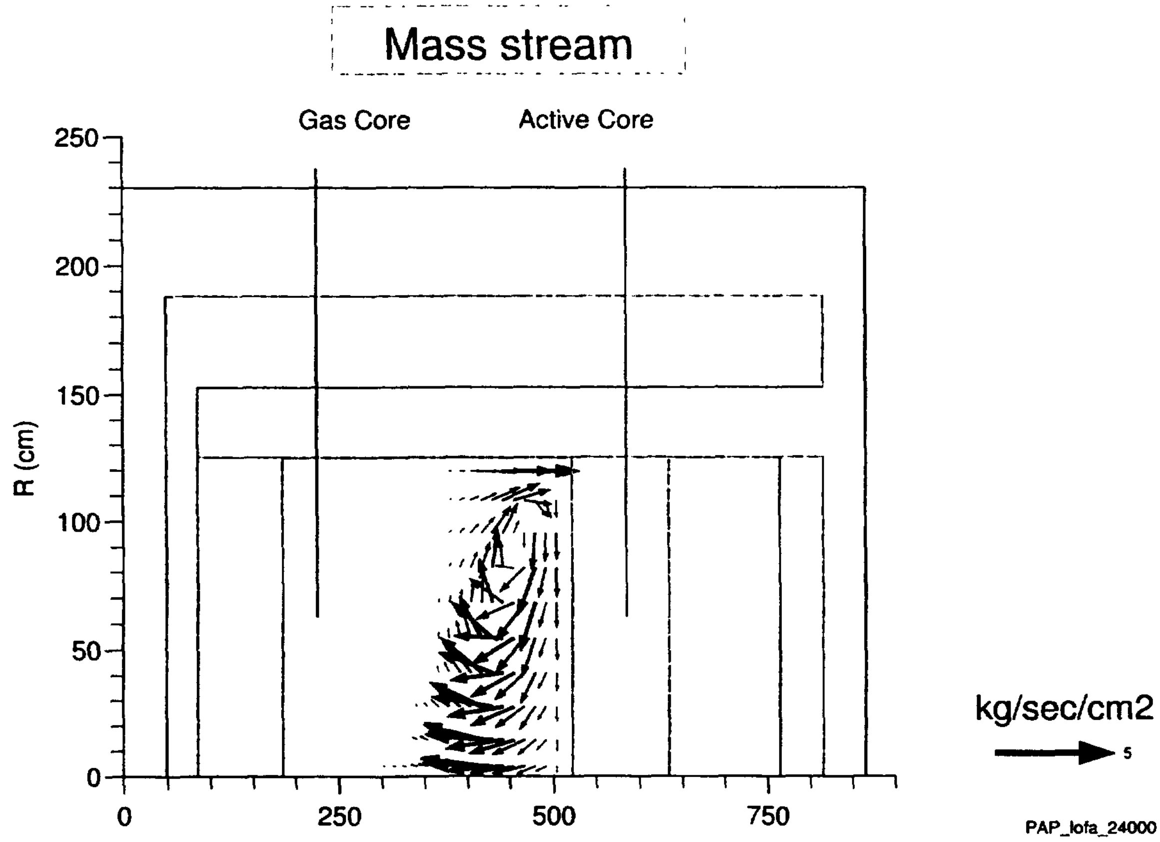

FIG. 16 PAP20 LOFA: flow distribution at t = 400 min (R denotes the radial -horizontal-distance to the core centre line).

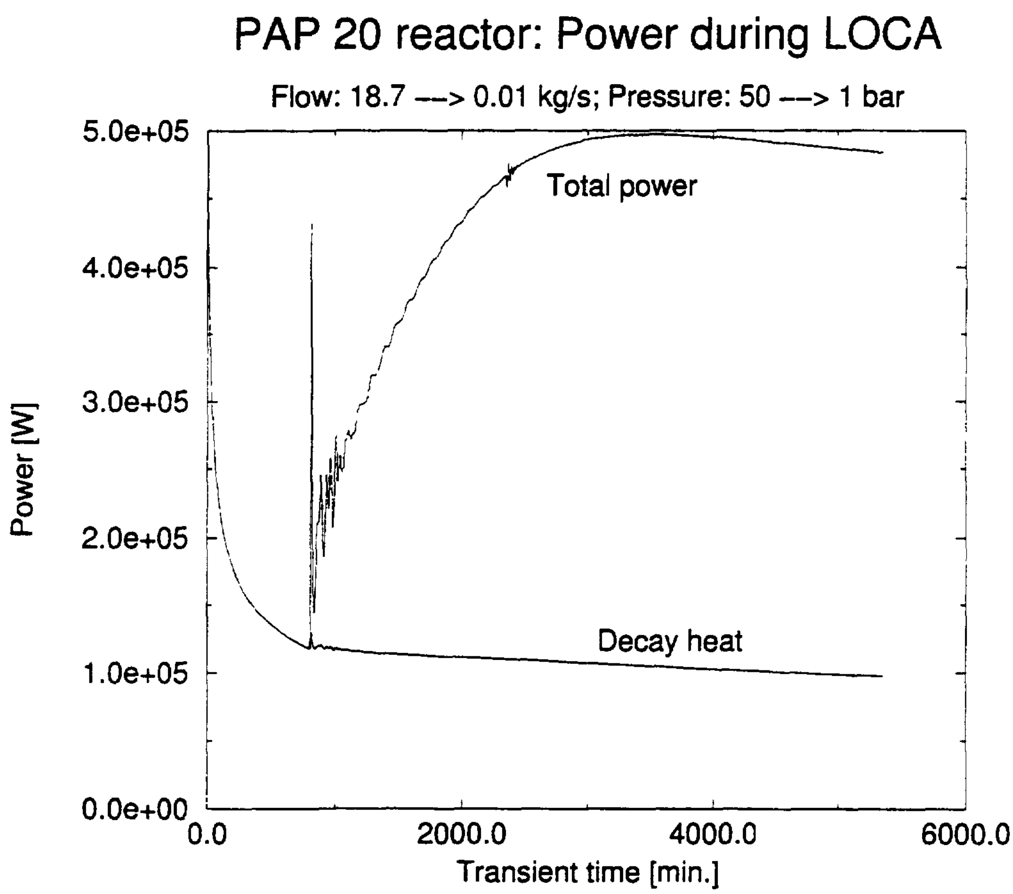

FIG. 17 PAP20 LOCA: power as function of time.

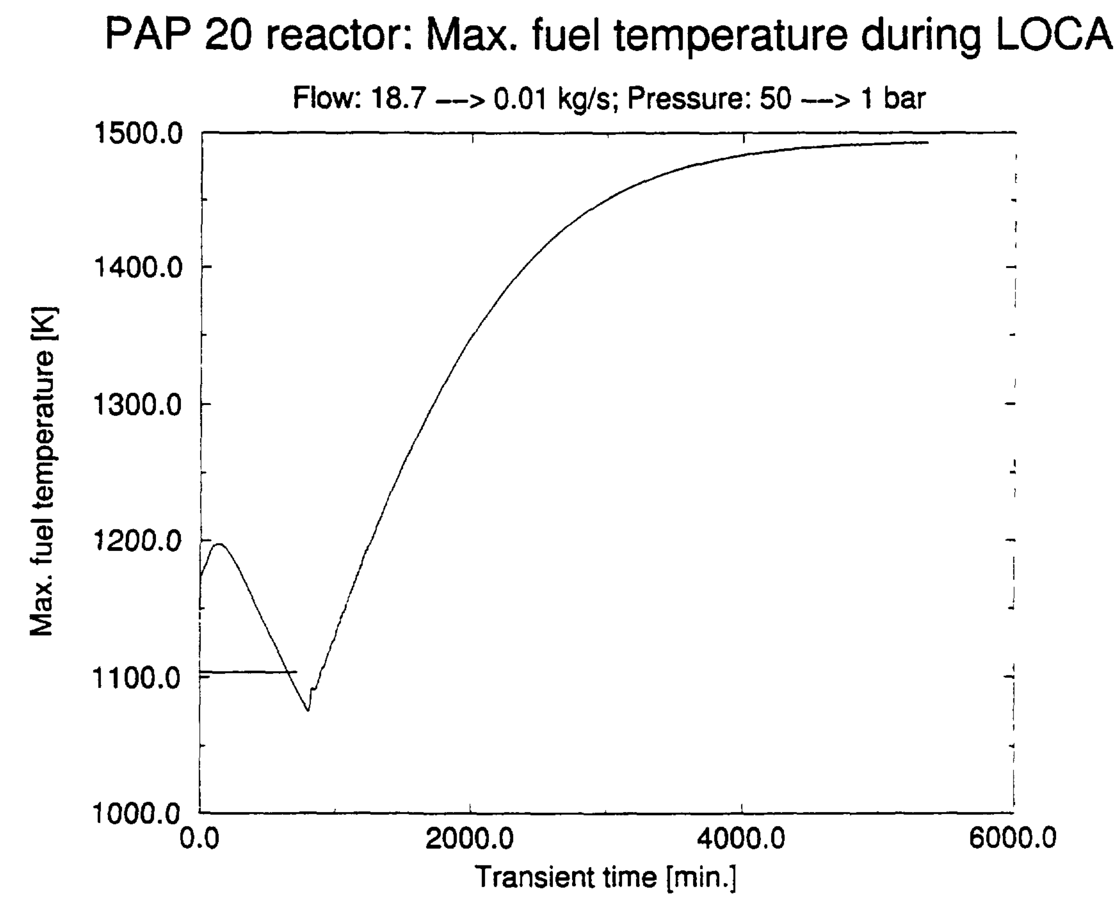

FIG. 18 PAP20 LOCA: maximum fuel temperature as function of time.

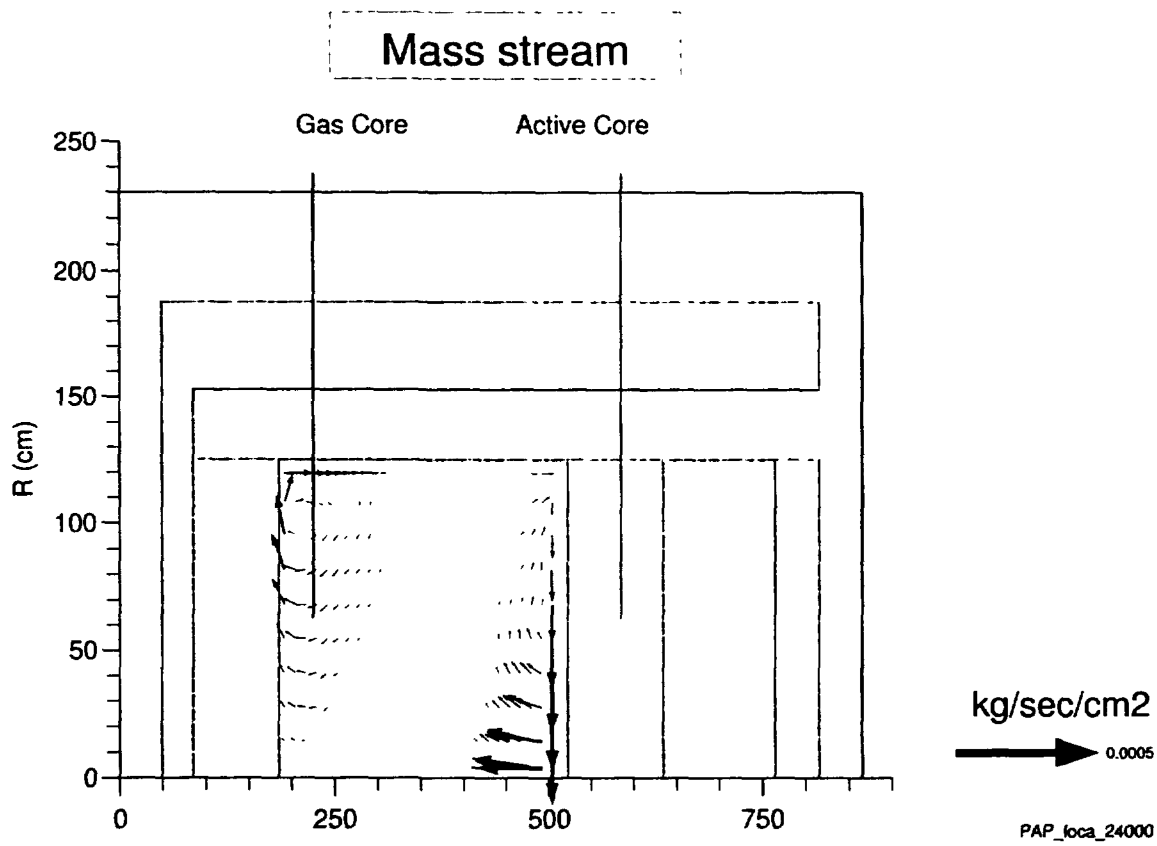

FIG. 19 PAP20 LOCA: flow distribution at t = 400 min (R denotes the radial -horizontal- distance to the core centre line).

For the LOCA case the power and maximum fuel temperature as function of time are shown in FIG. 17 and 18, respectively. Initially, the maximum fuel temperature (which initially is 1106 K) rises to about 1200 K at t = 150 min, after which the maximum fuel temperature decreases. At t = 620 min the maximum fuel temperature is again 1106 K. As a result of this behaviour of the temperature, recriticality is expected to occur about 300 min later than in the PAP20 LOFA case. Recriticality, caused by temperature decrease and xenon decay, occurs, for the first time, at t = 800 min. At t = 3500 min a maximum power of 500 kW is reached, after which the power decreases slowly but monotonically. At the end of the simulation, at t = 5350 min, the peak fuel temperature has reached 1493 K. Although at this point in time the temperature is still increasing, it is expected that the final peak fuel temperature will not be much higher than 1500 K, in view of the small and decreasing rate of increase, and the decreasing power.

FIG. 19 shows the flow pattern at t = 400 min. At the time of the "snapshot" only a little gas flow circulation occurs and the heat transfer from the core to the ambient mainly takes place by radiation and conduction.

Although the PAP20 benchmark exercise is still in progress, it already seems that the PANTHERMIX code combination is quite suitable for performing steady-state, burnup and transient pebble-bed HTR core calculations. Final conclusions will be presented in the forthcoming benchmark report, which will also contain the final specifications and results of the transient calculations, together with a comparison of the ECN and KFA calculation results.

5. INCOGEN - THE "DUTCH" SMALL HTR CONCEPT

5.1. Introduction

The ultimate goal of the validation of the ECN reactor physics code system OCTOPUS (with PANTHERMIX) for pebble-bed HTR applications is to acquire and reinforce the capability to perform the core design and optimization for the "Dutch" small HTR concept INCOGEN.

The first concept of INCOGEN is based on the PAP20 design of KFA. In fact, the geometry is almost identical to that of the PAP20, except for the borated graphite layer in the bottom reflector, as was shown in FIG. 6. But also the nominal power is twice that of the PAP20, which causes changes in the nominal values of helium temperatures, pressures and mass flow. In TABLE VII the main parameters have been listed of both the original PAP20 design and the first concept of INCOGEN.

5.2 First calculation results

Within the framework of the INCOGEN core design and optimization activities, steady-state, burnup and transient calculations are under performance, which are very similar to the calculations carried out in the PAP20 benchmark. As mentioned before, also these calculations are in progress, each calculation following the corresponding PAP20 calculation, taking advantage of the experience gained with the latter.

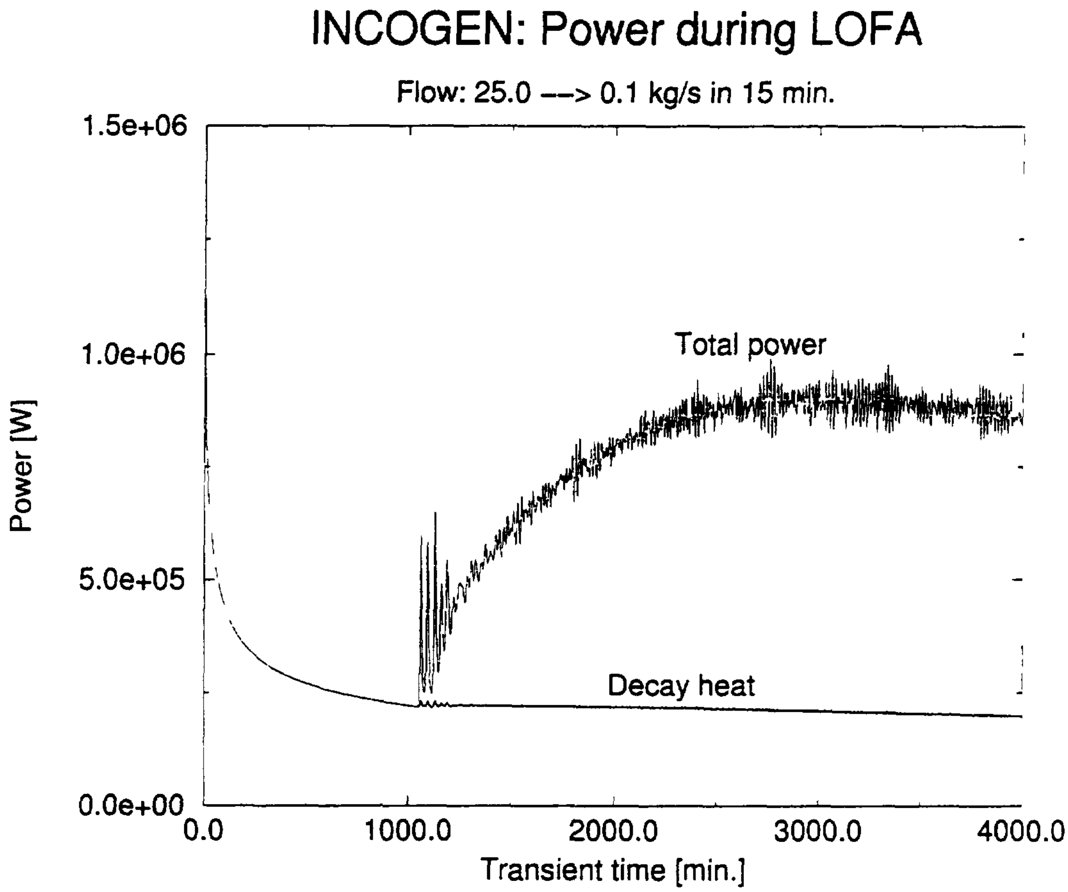

FIG. 20 INCOGEN LOFA: power as function of time.

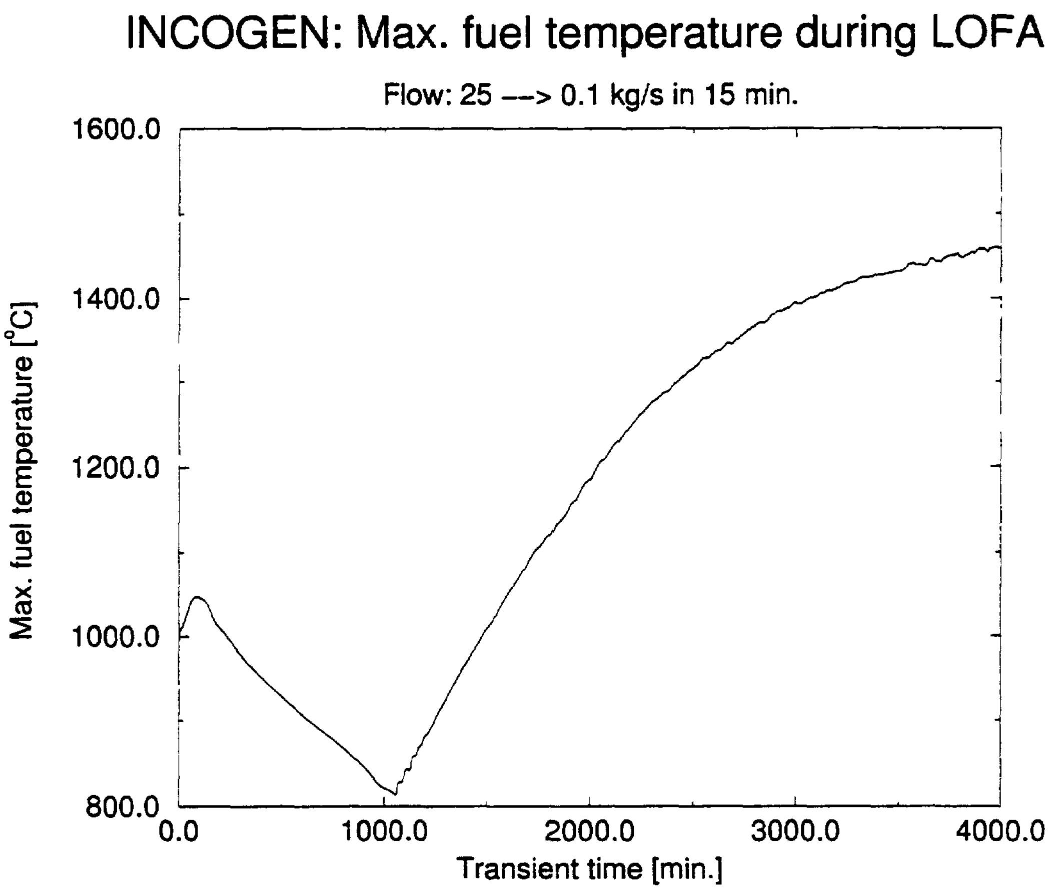

FIG. 21 INCOGEN LOFA: maximum fuel temperature as function of time.

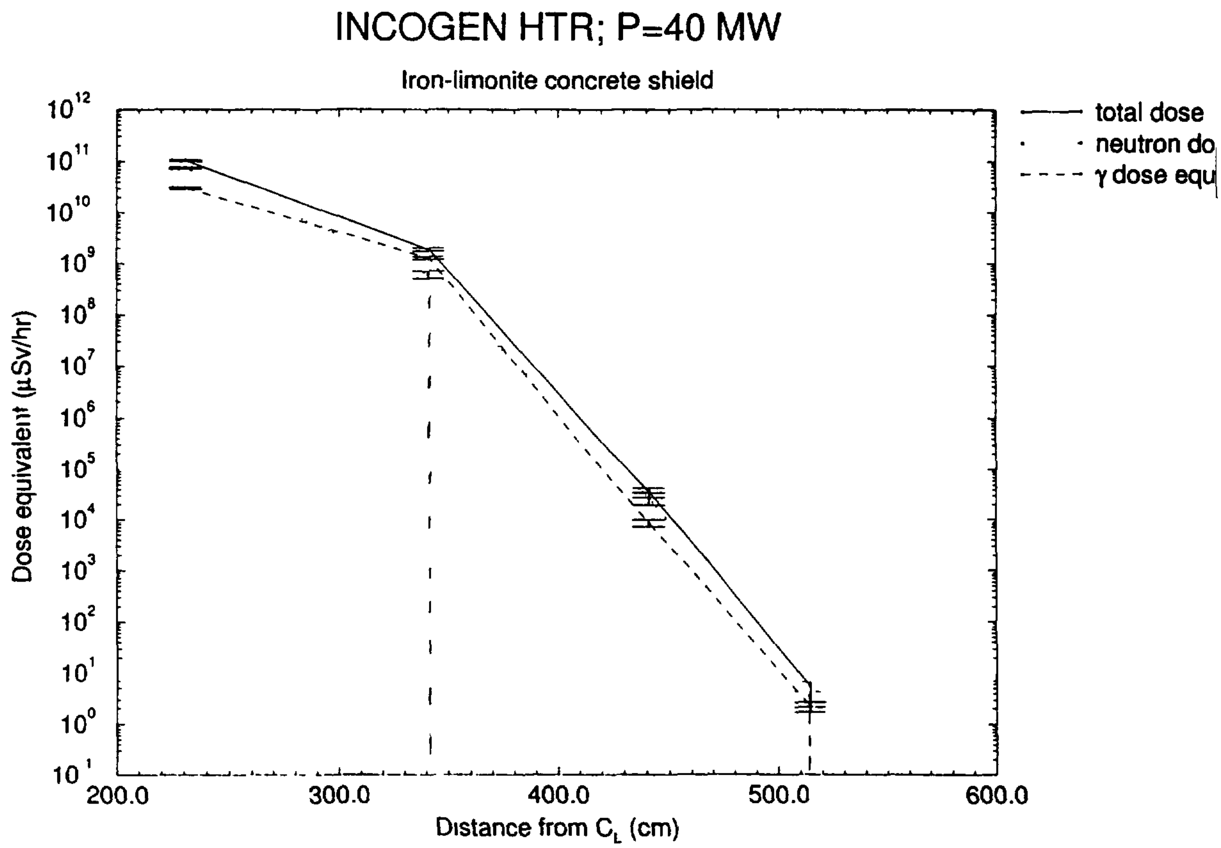

FIG. 22 MCNP shielding calculation for INCOGEN: iron-limonite concrete shield.

In FIG. 20 and 21 the power and maximum fuel temperature, respectively, are shown as function of time, as obtained from a LOFA calculation, very similar to the one presented in Section 4 for the PAP20 reactor. The initial state is the HFP state at nominal conditions for the INCOGEN reactor, as specified in TABLE VII. In this case the helium mass flow is reduced from 25 kg/s to 0.1 kg/s in 15 min. The maximum fuel temperature initially rises relatively quickly from 923 °C (1196 K) to 982 °C (1255 K) at t = 63 min, after which it decreases (at a rate of approximately 0.146 K/min) to 842 °C (1115 K) at t = 1025 min, at which recriticality occurs for the first time, due to xenon decay and - mainly - temperature decrease. Note that the time until recriticality is more than twice that in the case of the PAP20 LOFA. This is connected to the fact that the helium pressure (and therefore the helium inventory) in INCOGEN is about half of that in the PAP20, which leads to a slower temperature decrease. The first power peak occurs at t = 1031 min, whereas the second power peak occurs at t = 1063 min. After the first recriticality, the time-averaged power and time-averaged maximum fuel temperature slowly increase. The power finally stabilizes around 0.85 MW, which is the heat leakage to the ambient. At the end of the simulation (t = 4000 min) the maximum fuel temperature is about 1460 °C (1733 K). Altjough at this point in time the peak fuel temperature is still increasing, the rate of increase is already small and decreasing. It is therefore expected that the final peak fuel temperature will be below 1600 °C.

As mentioned before, the MCNP code is used for both detailed reference calculations and for shielding applications. An example of the latter is presented in FIG. 22, which shows the neutron, gamma and total dose rates, as function of radial distance from the centreline core. In this case the reactor is surrounded by a limonite (iron ore) heavy concrete shield.

From this calculation it could be concluded that a limonite concrete shield of 175 cm thickness sufficiently reduces the neutron and gamma radiation. However, in axial direction large variations in dose rates occur. An improvement may be obtained by borating the outer part of the reflector, hereby reducing the capture gamma sources in the vessel and concrete (7.5 MeV gammas caused by (n,γ)-reactions in iron). This also reduces radiation embrittlement in the vessel-steel. Additional studies are being performed.

6. CONCLUSIONS

Within the framework of the development of INCOGEN, a "Dutch" conceptual design of a small HTR, the ECN reactor physics code system OCTOPUS has been extended with the capability to perform combined neutronics and thermal hydraulics steady-state, burnup and transient core calculations for pebble-bed HTR type reactors. This extension was obtained by connecting the general purpose modular reactor code PANTHER and the HTR thermal hydraulics code THERMIX/DIREKT in the PANTHERMIX code combination.

In the process of validating its reactor physics code system for HTR applications, ECN is taking part in two benchmark exercises, viz. HTR-PROTEUS (steady-state; neutronics only) and PAP20 (power reactor; steady-state, burnup and transient; combined neutronics and thermal hydraulics).

The results obtained by the WIMS, SCALE and MCNP codes for the LEUPRO unit cell calculations, and with WIMS and MCNP for the whole core LEUPRO calculations, indicate that the codes in use at ECN are quite suitable for (steady-state neutronics only) pebble-bed HTR applications. Presently, the MCNP code, with associated JEF-2.2 based cross section library, is being validated further by calculating the PROTEUS Core 5 experiment. First results indicate that MCNP reproduces very well the experimental value of k_{eff}.

Presently the PANTHERMIX code combination is being compared with the KFA code system in the PAP20 benchmark exercise, which comprises initial steady-state, burnup and transient combined neutronics and thermal hydraulics calculations. Although the benchmark exercise is still in progress, the first results are promising and indicate that the ECN reactor physics code system, extended with PANTHERMIX, is quite suitable for performing the pebble-bed HTR core physics calculations required in the INCOGEN core design and optimization process.

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APPENDIX A - SPECTRAL INDICES

ρ^{28} = ratio epithermal-to-thermal captures in ^{238}U

δ^{25} = ration epithermal-to-thermal fissions in ^{235}U

δ^{28} = ratio ^{238}U fissions to ^{235}U fissions

C^{*} = ratio ^{238}U captures to ^{235}U fissions

The thermal break is at 2.1 eV.

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OTHER PAPERS SUBMITTED

PRELIMINARY STUDY ON THE DEVELOPMENT OF AN IN-SERVICE INSPECTION TECHNIQUE FOR GRAPHITE COMPONENTS IN THE HTTR

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Abstract

Development of a nondestructive measuring method on a residual strain and/or stress accumulating in the graphite component is one of major research subjects in the HTTR. For the purpose of the development of the method deformation characteristic under a small magnitude of a pressure load was measured using graphite specimens on which several levels of residual strains were produced. The pressure load was applied using a microhardness testing machine. In the experiment indentation load to depth characteristic and a microhardness were measured as a function of the residual strain.

In this paper we describe the deformation characteristic influenced by the residual strain, and discuss the applicability of the micro-indentation method to the estimation of the residual strain arising in graphite components as an inservice inspection technique.

1. Introduction

The High Temperature engineering Test Reactor (HTTR) which is under construction at Japan Atomic Energy Research Institute, is a graphite-moderated and helium gas-cooled reactor with a maximum helium coolant temperature of 950 °C at the reactor outlet[1]. The core internals are made of mainly graphite material which is superior in thermal resistivity. The arrangement of graphite components is shown in Fig.1. Core graphite components are the replaceable structures connected by the dowel pin and socket elements, and core support components are the permanent structures connected by the key and keyway elements.

The graphite component is subjected to high temperature and high temperature gradient in neutron irradiation environment to achieve a high temperature coolant gas. Although no significant thermally-induced creep of the graphite occurs less than about 2000°C, neutron irradiation-induced creep occurs at significant rates at the temperature range of the graphite component in the HTTR, which is about 500°C to 1200°C. Moreover, neutron irradiation causes dimensional change in the graphite component, which is a function of the neutron fluence and the irradiation temperature. These deformations accumulate gradually in the graphite component with the reactor operation. Especially for the fuel graphite block the accumulating residual strain and/or stress becomes high enough to limit the lifetime of it. Thereby, an evaluation of the residual strain and/or stress by a nondestructive measurement is important to confirm the structural integrity of graphite components.

Some methods on the measurement of the residual strain have been developed; e.g. the cutting method[2] in which the released deformation is measured by cutting at the accumulating residual strain area, the X-ray method and so on. However these methods are not applicable in this case, since the residual strain and/or stress has to be measured nondestructively and to be measured on an arbitrary surface by a remote control technique.

It is well known that the graphite and carbon materials have a nonlinear relationship in the stress-strain behavior. If the tensile or compressive load applies to the component in which the residual strain accumulates, the deformation characteristic, e.g. the deformation and/or strain under a tensile or compressive load, would be thought to change with the residual strain owing to the decrease of the deformation resistivity with increasing residual strain; the tangent on the stress-strain curve is gradually decrease with increasing the strain.

The deformation characteristic was measured, therefore, under small magnitude of pressure load so as not to make a damage to the component. The microhardness testing machine was used in the test. The indentation depth under a constant indentation load and the microhardness were measured to clarify the deformation resistance with the residual strain. The applicability of the micro-indentation method was studied from the preliminary test result.

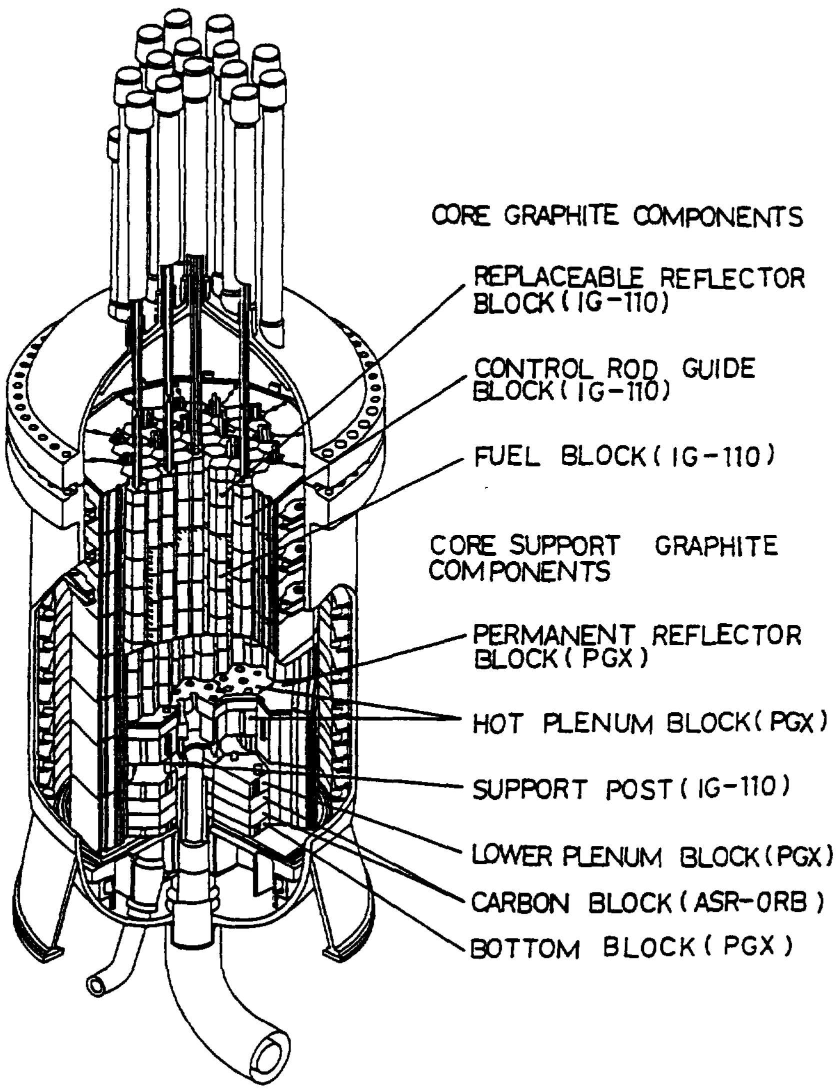


Fig. 1 Arrangement of core internals in the HTTR.

2. Residual stress in the graphite component

Figure 2 shows the horizontal view of the HTTR core. The column A, B and C shows the fuel region columns. In the fuel block at the column C the higher residual stress arises due to the higher temperature difference within the component; the irradiation-induced creep strain becomes higher owing to higher thermal stress as well as the higher irradiation-induced dimensional change (irradiation strain) owing to higher temperature difference within the block. The residual strain and/or stress was estimated by the VIENUS code[3], visco-elastic stress analysis code, which treats the steady state and transient creep strains besides irradiation and thermal strains as shown in Fig.3.

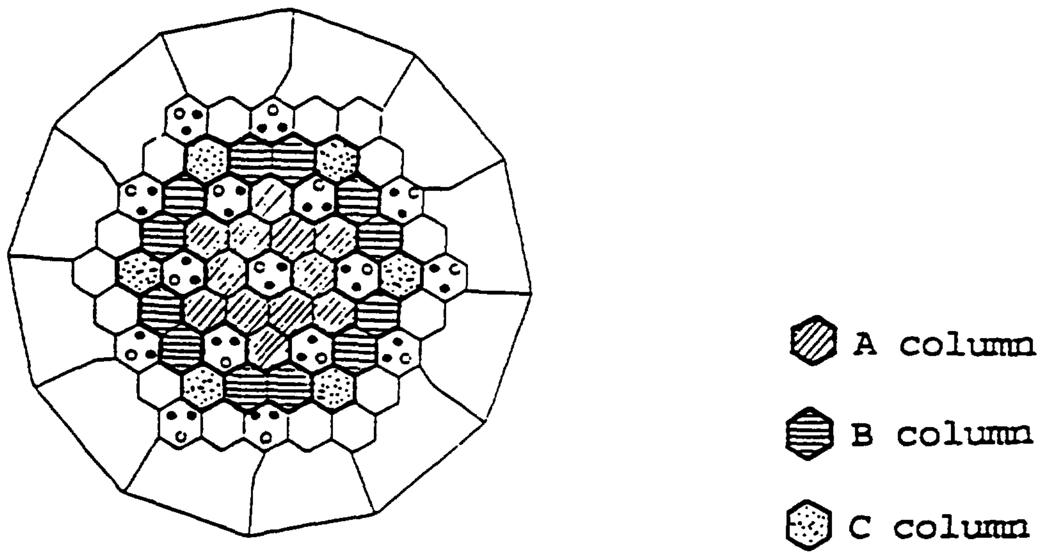


Fig. 2 Horizontal view of the core graphite blocks in the HTTR.

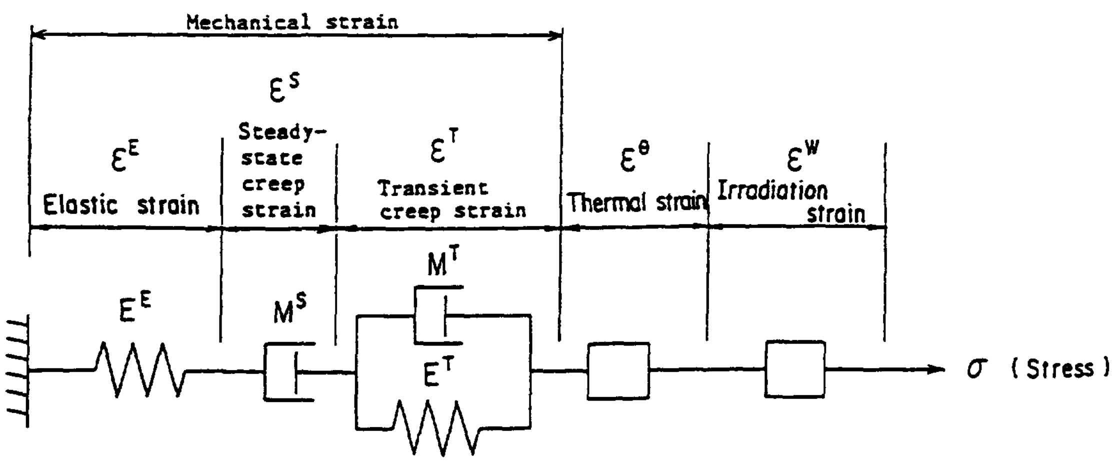


Fig. 3 Analytical model of graphite under neutron irradiation.

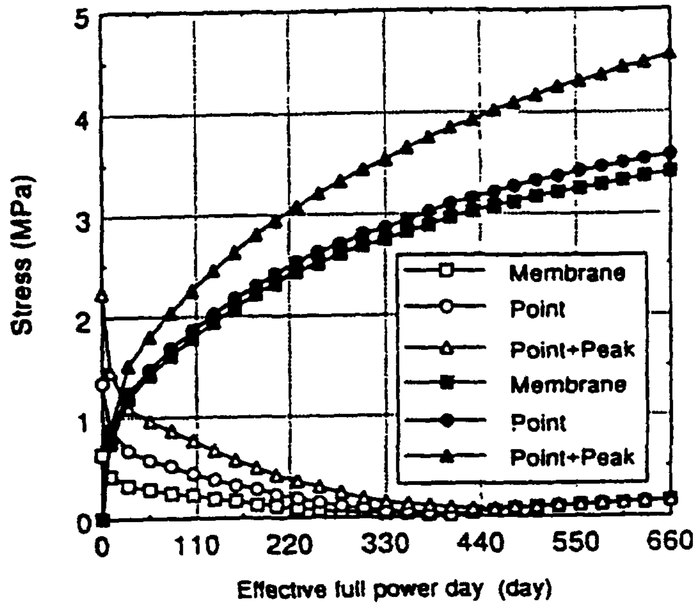
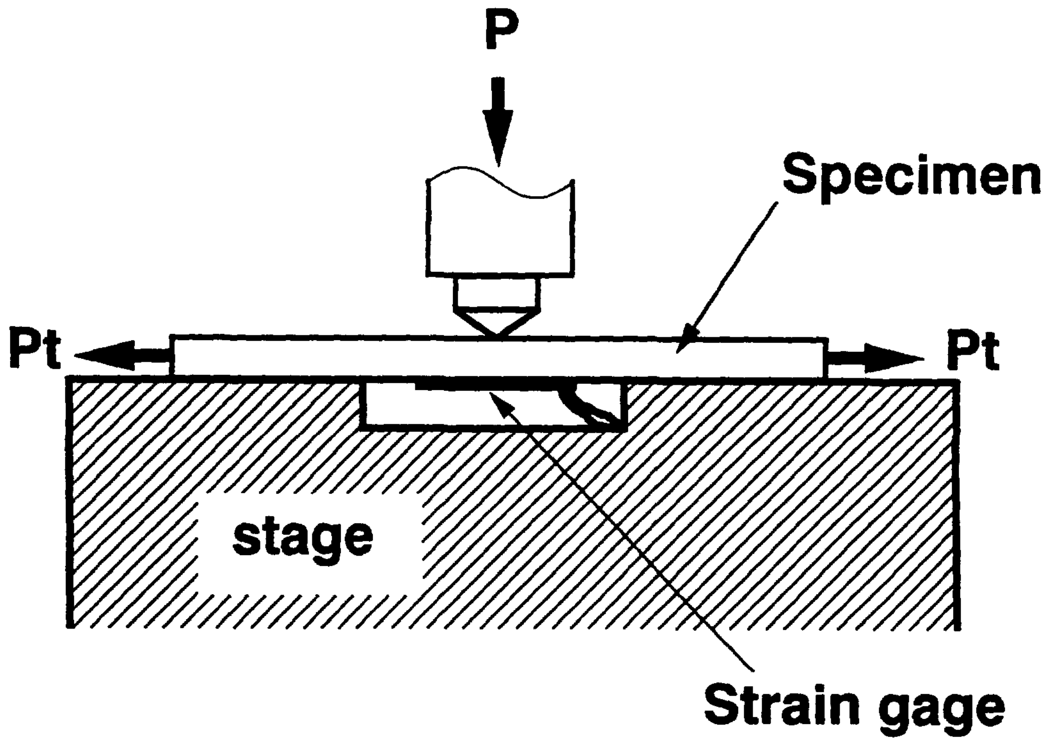


Fig. 4 Operational stress and residual stress of fuel graphite block at the C column in Fig.2 calculated by the VIENUS code.



P : Indentation load

Pt : Tensile load

Fig. 5 Measuring method of the deformation characteristic using microhardness testing machine.

The calculation result by the VIENUS code for the fuel block at C column is shown in Fig.4[4]. The operational stress which is mainly caused by the thermal stress in the fuel block is gradually reduced with the plant operation by the irradiation-induced creep deformation. However the residual stress arises gradually with plant operation due to the accumulating creep and irradiation strains in the graphite block. For the safety operation it is important, therefore, to estimate the residual strain and/or stress state in the graphite component as an inservice inspection.

3. Experiment

For the purpose of the estimation of the residual strain and/or stress arising in the graphite component with plant operation deformation characteristic was measured using the microhardness testing machine.

3.1 Experimental method

As shown in Fig.5 the residual strain arising in the graphite component was imitated by the strain under the tensile load. The magnitude of the imitated strain was measured by the strain gage attached on the other side of the specimen. The pressure load was applied by the microhardness testing machine using trigonal pyramid type indentor on the specimen in which a specified residual strain was produced. The pressure load was 1N in the test. The indentation load to depth characteristic and the microhardness were measured at different residual strain levels.

3.2 Test specimen

The specimen was made of IG-11 graphite, fine-grained isotropic nuclear-grade graphite for the fuel block in the HTTR reactor. The typical mechanical properties are listed in table I. The specimen configuration is shown in Fig.6. The parallel section is 7.5 mm in width and 10 mm in length with 5 mm thickness. The indentor was pressed at the parallel section of the specimen.

3.3 Experimental result

Obtained typical indentation load to depth curve was shown in Fig.7. We can see from this figure that the deformation resistivity decreases with increasing the residual strain. The mean indentation depth at 1N of the indentation load was 16.7μm and 17.6μm at 0% and 0.2% (that corresponds to about 40% to 50% of the tensile fracture strain) residual strains, respectively.

The hardness, Dh, for the trigonal pyramid type indentor was given by the equation

$$Dh=3.858x10^{3}P/\delta^{2} \qquad (1)$$

where P is the indentation load (N) and δ is the indentation depth(μm). From this equation, the change in the hardness is sensitive to the estimation of the residual strain compared with the change in the indentation depth. Hence the change in the hardness as a function of the residual strain was investigated.

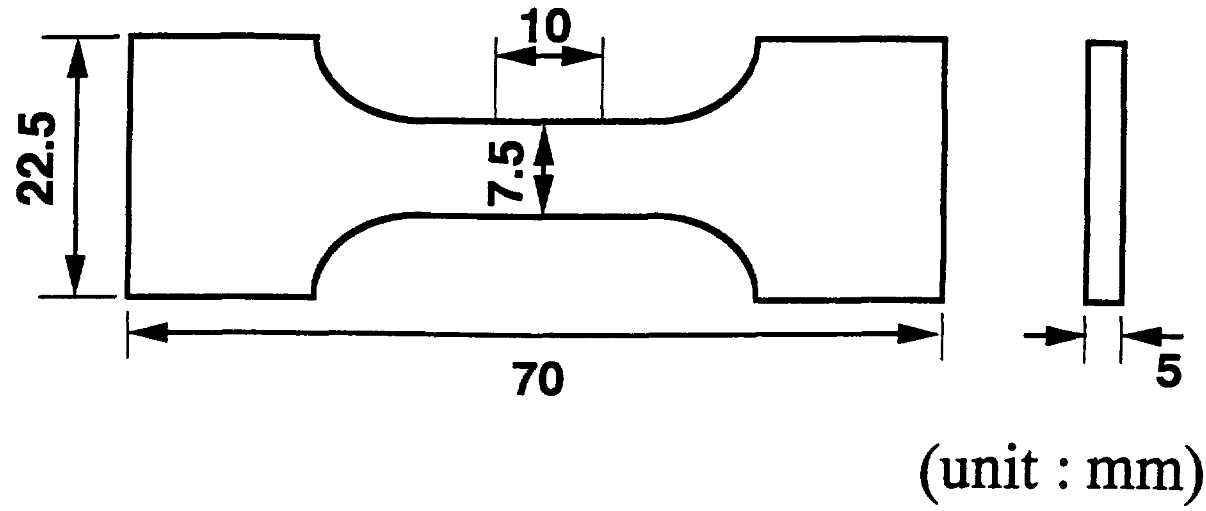


Fig. 6 Configuration of the test specimen.

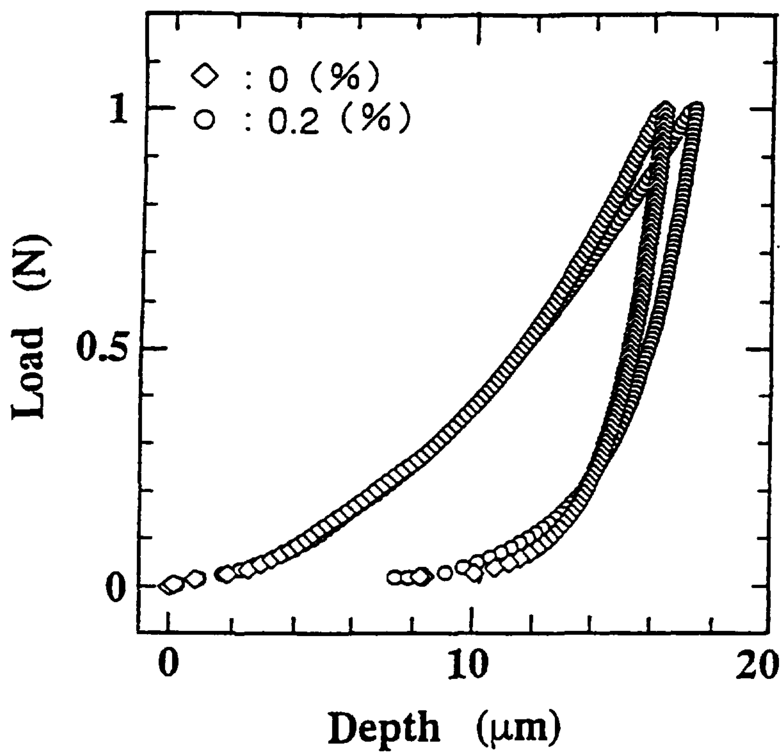


Fig. 7 Relationship between indentation load and depth.

The hardness calculated by equation (1) was plotted in Fig.8 at several imitated residual strain levels. In this figure the hardness was normalized by the mean hardness at 0% residual strain. From this figure we can see that the hardness was gradually decreased with increasing the residual strain owing to the decreasing of the deformation resistivity.

The relationship between the hardness and the residual strain was estimated as following manner. The deformation resistivity, d_r, is defined by the indentation load and depth as

$$P \propto d_r \cdot \delta \qquad (2)$$

For the constant indentation load condition equation (2) is

$$d_r \cdot \delta = C_1 \ (\text{constant}) \qquad (3)$$

If the deformation resistivity is assumed to be proportional to the tangent on the stress-strain curve, equation (3) is rewritten as

$$(d\sigma/d\varepsilon) \cdot \delta = C_2 \ (\text{constant}) \qquad (4)$$

For the IG-110 graphite the stress-strain relationship was reported[5] as

$$\varepsilon = 1.003 \text{x} 10^{-4} \sigma + 2.039 \text{x} 10^{-6} \sigma^2 \qquad (5)$$

From equations (4) and (5) the normalized indentation depth is given as

$$\delta = (1+40.3\varepsilon)^{1/2} \qquad (6)$$

where the normalization was performed by the mean indentation depth at 0% of the residual strain. Therefore, from equation (1) the normalized hardness is estimated as

$$Dh = (1+40.3\varepsilon)^{-1} \qquad (7)$$

where the normalization was performed by the mean hardness at 0% of the residual strain.

The estimated line from equation (7) was also plotted in Fig.8. It is found from this figure that the change of the hardness as a function of residual strain could be estimated by the model in which the deformation resistivity is assumed to be proportional to the tangent of the stress-strain curve. From the result of the experiment it can be said that the residual strain and/or stress accumulating in the graphite component could be estimated by the indentation method.

4. Application of Indentation Method as Inservice Inspection

From the obtained hardness data the evaluation line to the accumulating residual strain in the graphite component is proposed as the line with 99% confidence level in Fig.9. The evaluation line is calculated using the average of the hardness at each strain level; in the calculation it is assumed that the difference between the average hardness at each strain level and the mean line obeys the normal distribution function. The residual strain is, therefore, estimated by the average of the hardness obtained in the inservice inspection using the evaluation line in Fig.9. From the obtained preliminary test data which are measured at indentation load of 1N using trigonal pyramid the minimum detectable level of the residual strain was around 0.05%, which corresponds to about 1/10 of the tensile fracture strains.

The minimum detectable level of the residual strain is influenced by the variation of the measured hardness; the smaller variation of the hardness data leads to the smaller minimum detectable residual strain. As shown in Photo 1, which is the micro structure of the IG-110 graphite, we can see many pores and grains. Since the variation of the obtained indentation data would be caused by these micro structure, it is necessary to optimum the indentation testing condition such as the indentation load, shape of the indentor and so forth as well as to verify the environmental effect on the change in the hardness in order to detect residual strain and/or stress precisely.

5. Conclusion

To develop the estimation method of the residual strain and/or stress accumulating in the graphite component with plant operation a deformation characteristic was measured using the graphite specimen with the imitated residual strain. From the experiment the obtained conclusions are summarized as follows;

Table I Typical mechanical properties of IG-11 graphite.

Tensile strength (MPa)	Compressive strength (MPa)	Bending strength (MPa)	Young's modulus (GPa)	Poisson's ratio
25.3	76.8	36.8	8.3	0.14

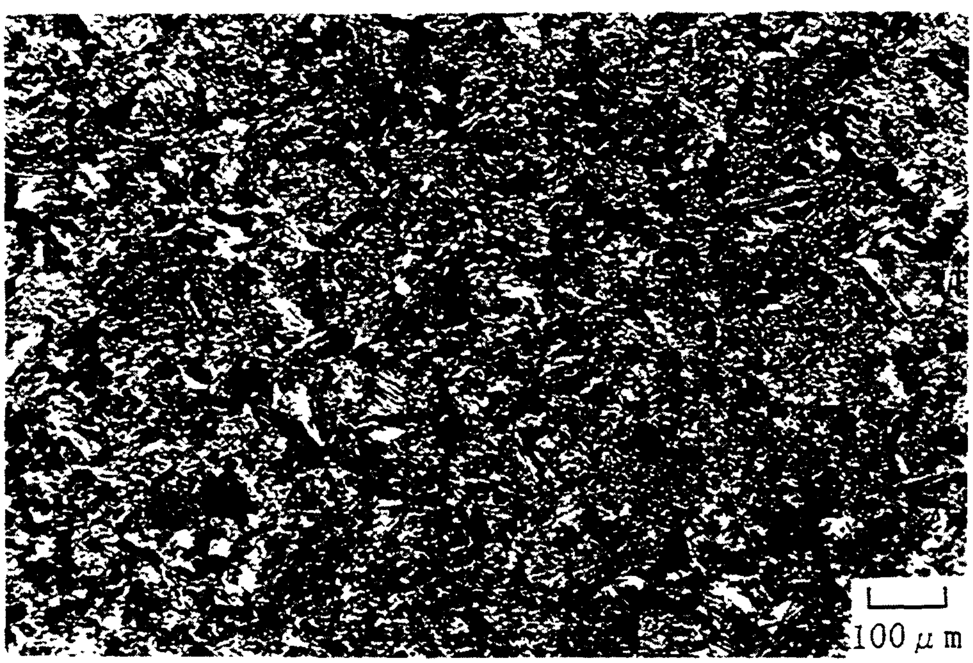


Photo. 1 Microstructure of the IG-11 graphite.

(1) Under the constant indentation load indentation depth increased with increasing the residual strain; i.e. the deformation resistivity decreased with increasing the residual strain.
(2) The hardness decreased with increasing residual strain. The change of the hardness could be estimated by the model in which the deformation resistivity assumed to be proportional to the tangent on the stress–strain curve.
(3) The indentation method has an applicability as an evaluation method for the inservice inspection of the graphite component.

Further investigation on the testing condition would be necessary to optimum the measuring method such as indentation load, indentor shape etc. as well as to clarify the environmental effect on the hardness such as an irradiation damage, an oxidation damage and so on in order to achieve more precisely estimation of the residual strain and/or stress in the graphite component.

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POST-IRRADIATION HEATING TESTS OF ZrC COATED FUEL PARTICLES

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Abstract

Postirradiation heating tests of the ZrC–Triso coated UO_2 particles were performed at 1600°C for 4500 h and at 1800°C for 3000 h to study the release behavior of fission products. The burnup and irradiation temperature of the heated particles were 1.5%FIMA and 900°C, respectively. During heating of the particles, the radioactivity in flowing helium was monitored with an ionization chamber to detect fission gas release. The radioactivity in graphite components of a cold–wall furnace was measured by γ–ray spectrometry to identify and evaluate the released metallic fission products. The fission gas release monitoring and the X–ray microradiography revealed that no pressure vessel failure occurred during the tests. Diffusion coefficients of ^{137}Cs and ^{106}Ru in the ZrC coating layer were evaluated from the release curves based on a diffusion model. The ZrC–Triso coated fuel particles showed better cesium retention than the standard Triso–coated fuel particles. In spite of better cesium retention, the ZrC layer showed a less effective barrier to ruthenium than the SiC layer.

1. INTRODUCTION

In the current designs of the high temperature gas–cooled reactor (HTGR), the Triso–coated fuel particles are to be used [1–3]. The Triso–coated fuel particle consists of a microspherical kernel of oxide or oxycarbide fuel and coating layers of porous pyrolytic carbon (PyC), inner dense PyC (IPyC), silicon carbide (SiC) and outer dense PyC (OPyC). The function of these coating layers is to retain fission products within the particle. The SiC coating layer gives mechanical strength to the particle and acts as a barrier to the diffusion of metallic fission products which diffuse easily through the IPyC coating layer.

Although SiC has excellent properties, it gradually loses mechanical integrity at very high temperatures, especially above 1700°C, by thermal dissociation and transformation of β–SiC to α–SiC [4–6]. The fuel temperatures are, therefore, limited to well below 1700°C during the design–basis accidents in the current HTGR designs [1–3].

Zirconium carbide (ZrC), which is known as a refractory and chemically stable compound, is a candidate to replace the SiC coating layer of the Triso–coated fuel particles; the resulting particles are termed ZrC–Triso coated fuel particles. The results of early irradiation experiments of the ZrC–coated fuel particles were encouraging [7–9]. It has been demonstrated that the ZrC–Triso coated fuel particles have much higher temperature stability than the normal SiC–Triso coated fuel particles [9,10]. In addition, the ZrC coating layers have higher resistance to chemical attack by fission products such as palladium than the SiC coating layers [11,12]. However, the data on retention of fission products by the ZrC coating layer are limited [13–15], which could allow no conclusive evaluation of the fission product retentiveness of the ZrC coating layer.

The present paper describes the results of postirradiation heating tests of the ZrC–Triso coated fuel particles at 1600°C for 4500 h and at 1800°C for 3000 h, which were performed to study the fission product retentiveness of the ZrC coating layer.

2. EXPERIMENTAL

2.1. Samples

The ZrC–Triso coated fuel particles heated in the present experiments were sampled from an irradiated fuel compact after electrolytic deconsolidation. The fuel kernel was UO_2, and the ZrC coating layer was deposited by the bromide process [16]. TABLE I shows characteristics of the particles.

The fuel compact was irradiated in a gas-swept capsule in the Japan Materials Testing Reactor (JMTR) [10]. The irradiation conditions of the fuel compact are listed in TABLE II. The release-to-birth ratio of ^{88}Kr ($T_{1/2}$=2.84 h) during the irradiation was from 2×10^{-7} to 4×10^{-7} at about 900°C, which indicated that no failed particle was contained in the fuel compact.

TABLE I. CHARACTERISTICS OF COATED FUEL PARTICLES

	Diameter or Thickness (μm)	Density (Mg/m^3)
UO_2 kernel	608	10.6
Buffer layer	64	1.11
IPyC layer	26	1.84
ZrC layer	31	n.d.[a]
OPyC layer	55	1.95

[a] not determined.

TABLE II. IRRADIATION CONDITIONS OF FUEL COMPACT

Duration	79.9 EFPD
Temperature	900 °C
Burnup	1.5 %FIMA
Fluence	1.2×10^{25} m^{-2} (E>29 fJ)
R/B(^{88}Kr)	2×10^{-7} to 4×10^{-7}

2.2. Postirradiation Heating

The equipment used for the postirradiation heating was installed in a hot cell. A schematic representation of the furnace is shown in FIG. 1. The furnace is composed of a graphite heater, a graphite sample holder, graphite holder disks and carbon insulators within a stainless steel vessel. The coated fuel particles were placed individually in the holes of the graphite disks. Total 100 particles loaded in two disks were heated in flowing helium. The temperature was read and controlled with an optical pyrometer from the top of the furnace.

The heating tests at 1600°C for 4500 h and at 1800°C for 3000 h were divided into seven and eight time steps, respectively, as shown in TABLE III. At the end of each time step, the graphite components and the carbon insulators were replaced by new ones and the coated particles were examined by X-ray microradiography to confirm the integrity of the coatings. The coated particles were heated repeatedly for 4500 h and 3000 h, respectively.

During the heating tests, the fission gas release was monitored by an ionization chamber; the activity in flowing helium was due mostly to ^{85}Kr ($T_{1/2}$=10.73 y). The graphite components and the carbon insulators, which were removed from the furnace at the end of each time step, were measured by γ-ray spectrometry to identify and evaluate the released metallic fission products.

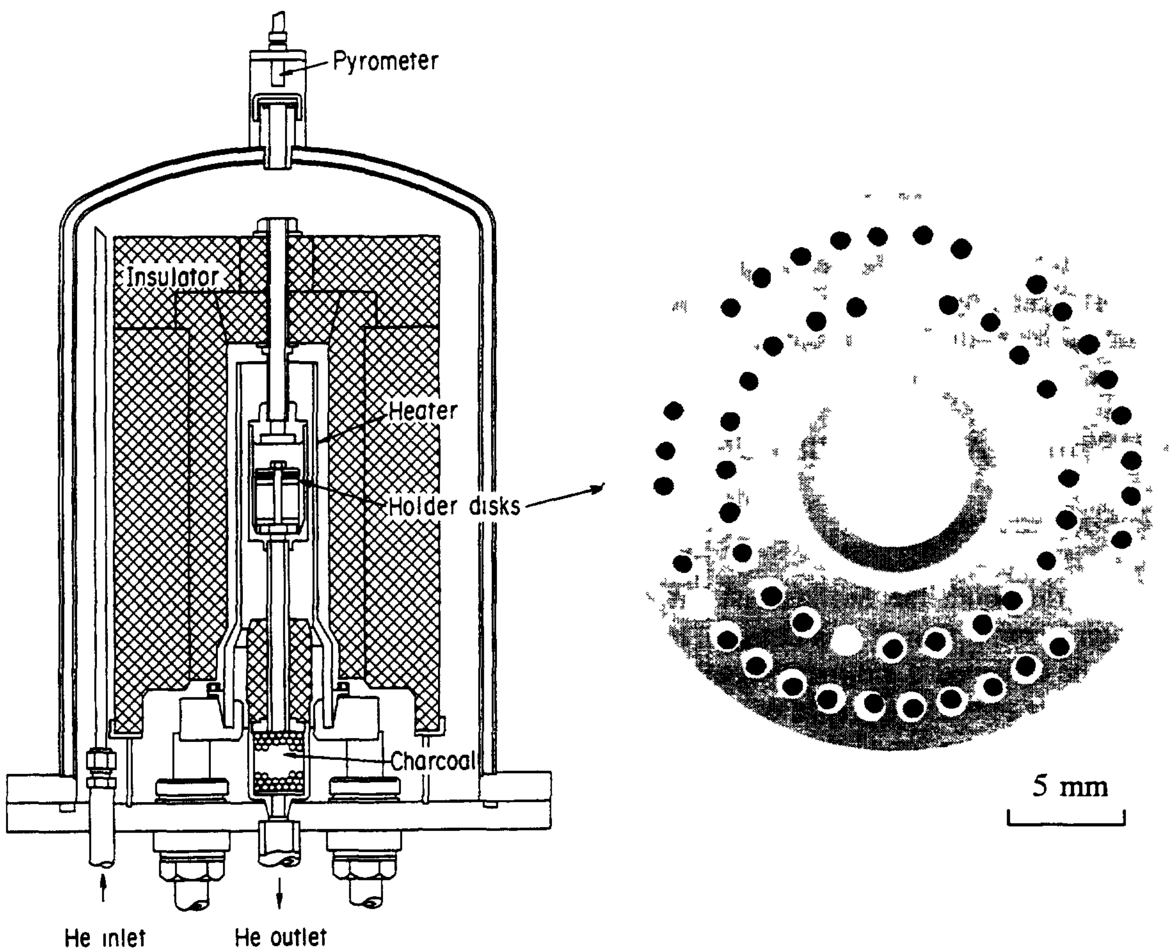

FIG. 1. Schematic representation of the furnace and X-ray microradiograph of the holder disk for postirradiation heating.

TABLE III. HEATING CONDITIONS OF SAMPLES

1600°C Heating Test			1800°C Heating Test		
Step	Time (h)	Total time (h)	Step	Time (h)	Total time (h)
1	100	100	1	24	24
2	300	400	2	46	70
3	500	900	3	80	150
4	700	1600	4	200	350
5	900	2500	5	550	900
6	1000	3500	6	700	1600
7	1000	4500	7	900	2500
			8	500	3000

The fission product release data were treated as the fractional release:

$$F = R / I, \qquad (1)$$

where F is the fractional release of a nuclide, R the released amount of the nuclide measured by γ-ray analysis, and I the inventory of the nuclide measured before the heating test.

After the heating tests, all the coated fuel particles were examined by X-ray microradiography. About fifty particles were polished to the equator of the particle and observed by optical microscopy.

3. RESULTS AND DISCUSSION

3.1. Coating Integrity

During both the heating tests at 1600°C for 4500 h and at 1800°C for 3000 h, no through-coating failure was detected by the ^{85}Kr release monitoring. A burst release of ^{85}Kr would be observed if a through-coating failure, or a pressure vessel failure, occurs in a short time. X-ray microradiographs of the coated particles after the heating tests revealed no OPyC failure, which confirmed the results of the gas release monitoring.

Typical polished cross sections of the ZrC-Triso coated fuel particles after the heating test at 1600°C for 4500 h are shown in FIG. 2. No failure was observed in the ZrC and OPyC coating layers. In some particles the IPyC coating layers were cracked in the radial direction. No palladium attack and thermal degradation of ZrC were observed.

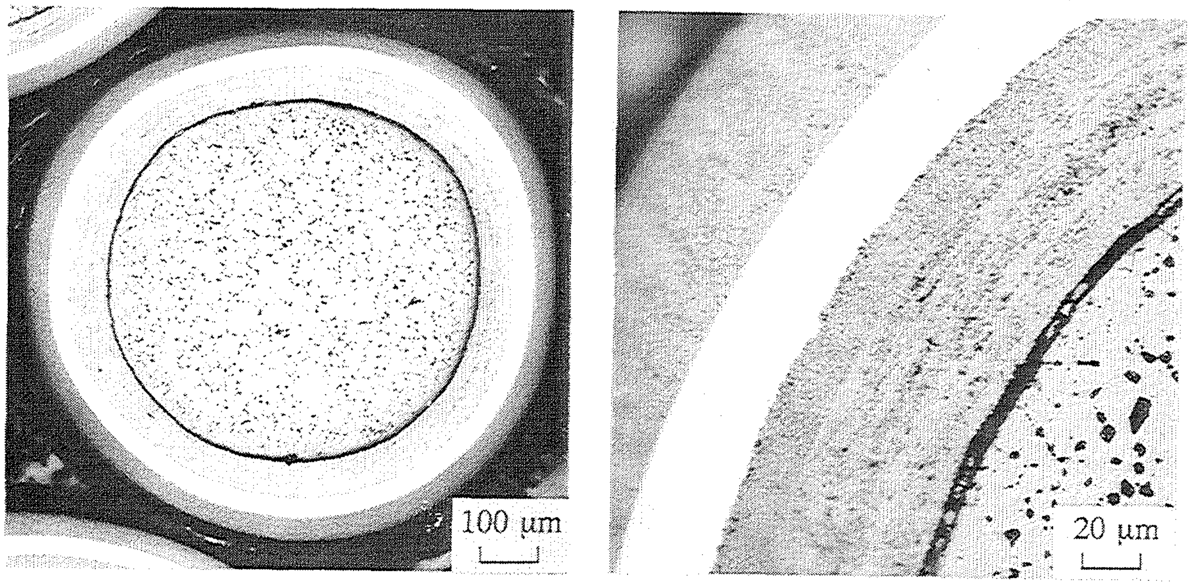

FIG. 2. Optical micrographs of polished cross section of ZrC–coated fuel particles after heating at 1600°C for 4500 h.

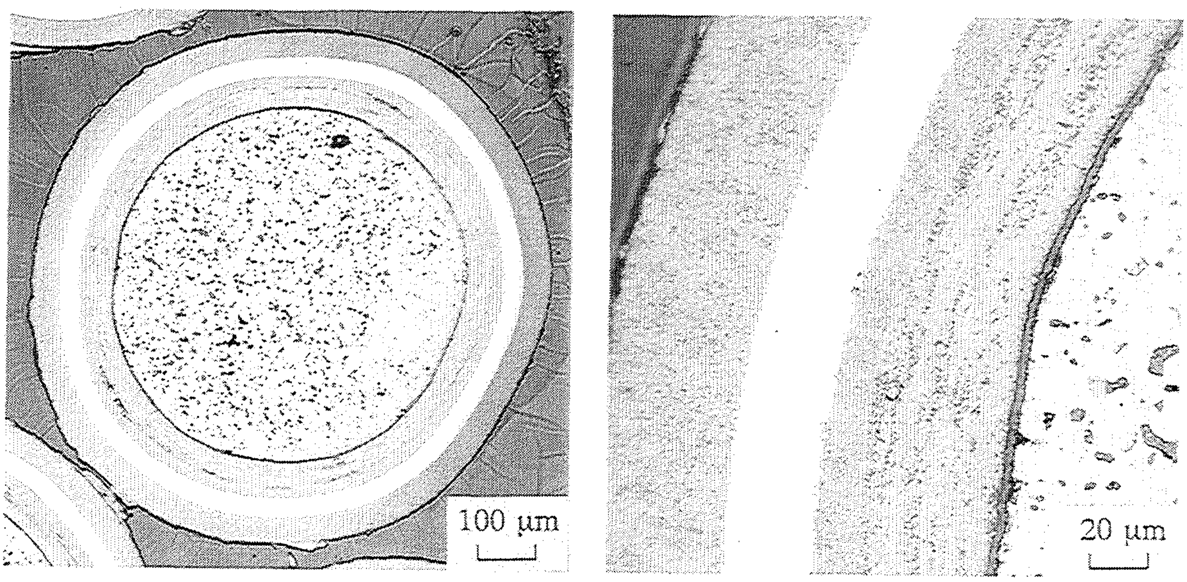

FIG. 3. Optical micrographs of polished cross section of ZrC–coated fuel particles after heating at 1800°C for 3000 h.

On the polished cross sections of the particles heated at 1800°C for 3000 h, no failure of the ZrC and OPyC coating layers was also observed, as shown in FIG. 3. However, some degradation of the ZrC coating layer seemed to have occurred. The inner and outer surfaces of most of the particles heated at 1800°C were not smooth, which was in contrast with the case of the particles heated at 1600°C. Further experiments are needed to clarify the cause of the degradation of the ZrC coating layer.

3.2. Release of Cesium

The measured fractional releases of ^{137}Cs at 1600°C for 4500 h and at 1800°C for 3000 h are shown in FIG. 4 as a function of heating time, where the broken curves are smooth fits to either data points. The fractional release of ^{137}Cs was below 1×10^{-3} even after heating at 1600°C for 4500 h or at 1800°C for 3000 h. The retention of ^{137}Cs in the ZrC-Triso coated particles was much better than that in the standard SiC-Triso coated particles.

Four additional curves are presented in FIG. 4 to estimate the effective diffusion coefficient of ^{137}Cs in the ZrC coating layer, $D_{ZrC}(^{137}Cs)$. These curves were drawn based on a diffusion model, where a fuel kernel with a single coating layer was assumed. This assumption is valid, since the ZrC coating layer is a far better diffusion barrier than the PyC coating layers. Further details of the analytical model were described elsewhere [15].

It could be obtained from the figure that $D_{ZrC}(^{137}Cs)$ at 1600°C lies between 1×10^{-18} and 5×10^{-18} m²/s, and at 1800°C between 2×10^{-18} and 1×10^{-17} m²/s.

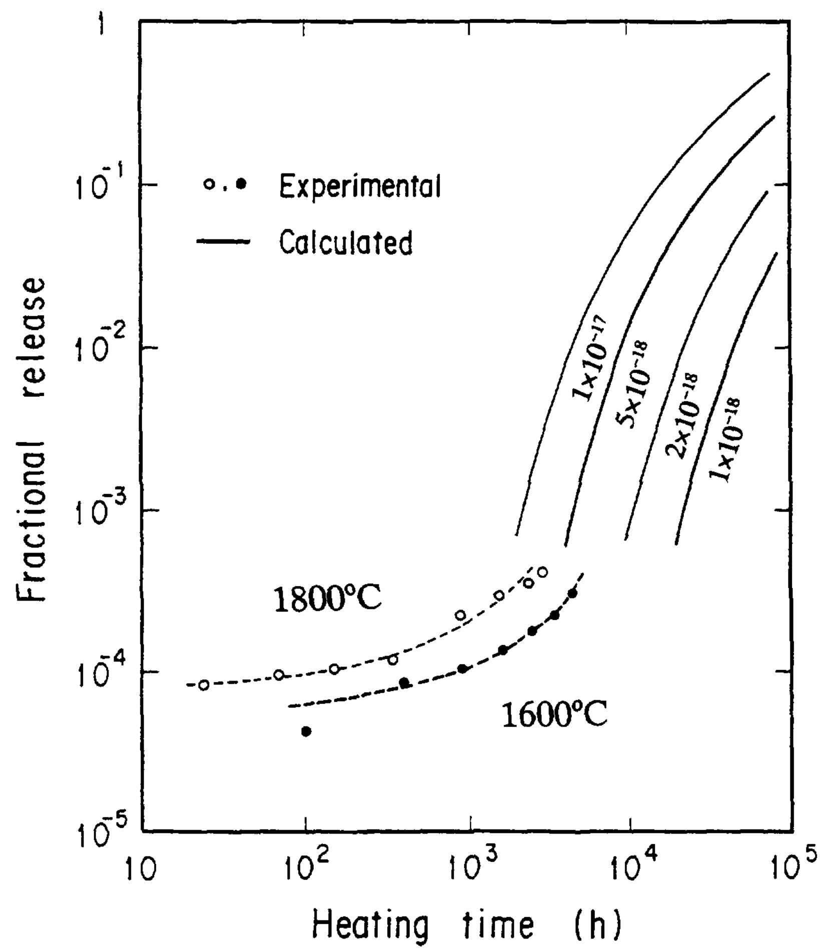

FIG. 4. Fractional releases of ^{137}Cs from ZrC-coated fuel particles at 1600°C and 1800°C as a function of heating time. Solid lines are diffusive release curves calculated by a model of diffusion in one-layer coating.

It is apparent in the figure that the shape of the measured fractional release curves were different from that of the calculated diffusive release curves. The rather flat parts of the measured curves at very low fractional release were attributed to the release of contaminated ^{137}Cs in the OPyC layer, which was not considered in the calculation. In both the heating tests, the diffusive release through the ZrC layer just started near the end of the heating tests. The fractional release curves of this kind were often observed in several postirradiation heating experiments [17–19].

The present values for $D_{ZrC}(^{137}Cs)$ are compared with the reported effective diffusion coefficients of ^{137}Cs in the SiC coating layer, $D_{SiC}(^{137}Cs)$ [17,20–22] in FIG. 5. At 1600°C the present $D_{ZrC}(^{137}Cs)$ was more than one order of magnitude as low as $D_{SiC}(^{137}Cs)$, whereas at 1800°C $D_{ZrC}(^{137}Cs)$ was more than two orders of magnitude as low as $D_{SiC}(^{137}Cs)$.

Although $D_{ZrC}(^{137}Cs)$ data at higher temperatures than 1800°C are needed, these results encourage us to use the ZrC–Triso coated fuel particles at higher temperatures than the SiC–Triso coated fuel particles from the viewpoint of the retention of the key fission product.

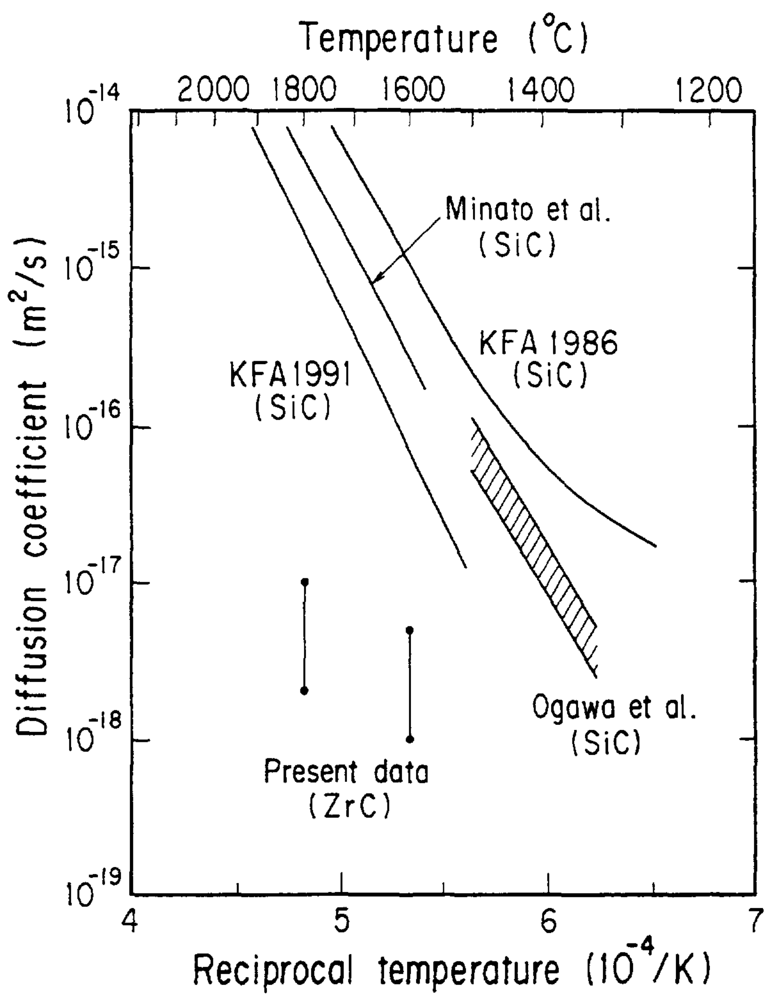

FIG. 5. Comparison of diffusion coefficients of ^{137}Cs in ZrC with those in SiC as a function of temperature. Data from literature are Minato et al. [17], Ogawa et al. [20], Moormann & Verfondern [21], and Verfondern & Müller [22].

3.3. Release of Ruthenium

The measured fractional releases of ^{106}Ru at 1600°C for 4500 h and at 1800°C for 3000 h are shown in FIG. 6 as a function of heating time. The calculated diffusive release curves are also shown in the figure, which were fitted to the data points of either heating tests. The diffusion model used was the same as that described above.

The release behavior of ^{106}Ru was determined by the transport both in the UO_2 kernel and in the ZrC coating layer, which was in contrast with the ^{137}Cs release behavior controlled by the transport in the ZrC coating layer. For the best fit the diffusion coefficients of $D_{ZrC}(^{106}Ru)=3\times10^{-16}$ m²/s at 1600°C and $D_{ZrC}(^{106}Ru)=5\times10^{-15}$ m²/s at 1800°C were obtained.

The release of ^{106}Ru was not reported in the postirradiation heating tests of the SiC–Triso coated fuel particles at 1600°C as long as 500 h and at 1800°C as long as 200 h [17–19]. The present fractional releases of ^{106}Ru were more than 1×10^{-1} at 1600°C for 4500 h and about 9×10^{-1} at 1800°C for 3000 h. These values were very large compared with the case of the SiC–Triso coated particles, suggesting that SiC is a better diffusion barrier to ruthenium than ZrC. Ogawa et al. [13] previously pointed out that the diffusion coefficient of ruthenium in ZrC is somewhat larger than that in SiC and discussed the fast diffusion of ruthenium by means of the atomic radius ratio of Ru/Zr.

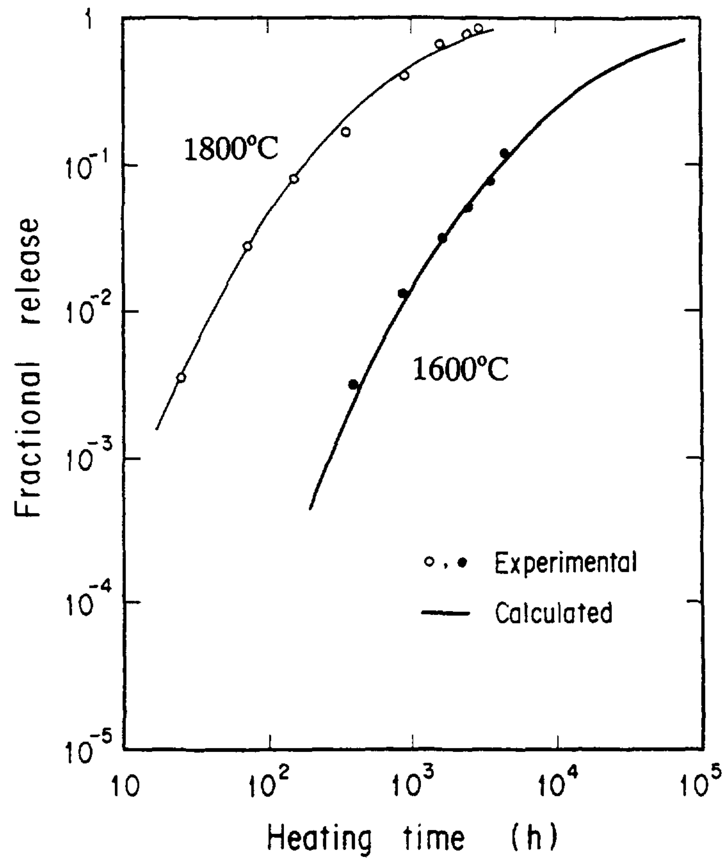

FIG. 6. Fractional releases of ^{106}Ru from ZrC-coated fuel particles at 1600°C and 1800°C as a function of heating time. Solid lines are diffusive release curves calculated by a model of diffusion in a fuel kernel and one-layer coating.

3.3. Release of Other Fission Products

Besides ^{137}Cs and ^{106}Ru, the release of fission products of ^{134}Cs, ^{144}Ce, ^{154}Eu and ^{155}Eu were detected by γ-ray spectrometry of the graphite components. These fission products must have been released through the coating layers. Although the release data of ^{137}Cs and ^{106}Ru could be treated quantitatively, the accuracy of the measured values for ^{134}Cs, ^{144}Ce, ^{154}Eu and ^{155}Eu was not enough to evaluate the effective diffusion coefficients since the counts by γ-ray spectrometry were small.

The fission products with relatively short half lives could not be detected since the present experiment was carried out about 9 years after the end of irradiation of the ZrC-Triso coated fuel particles.

The radionuclide ^{110m}Ag, whose half life is 250.4 d, is known as one of the most releasable radionuclides from the SiC-Triso coated fuel particles. Although the release of ^{110m}Ag was not detected in the present experiment, it cannot be concluded that the ZrC coating layer has excellent ^{110m}Ag retentiveness. In the present study, the ratio of inventories of ^{110m}Ag to ^{106}Ru was less than 1×10^{-3} just before the heating tests, meaning that the inventory of ^{110m}Ag was too small to be detected even if ^{110m}Ag was released.

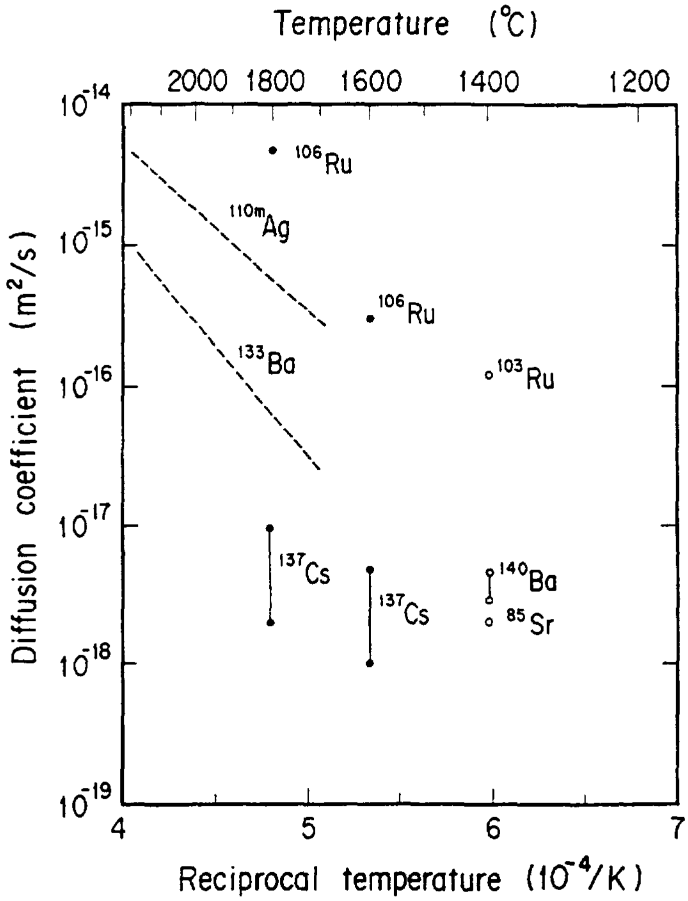

FIG. 7. Diffusion coefficients of fission products in ZrC as a function of temperature. Data of solid circles are from the present experiments, open circles from Ogawa & Ikawa [13], and broken lines from Chernikov[14].

The diffusion coefficients in ZrC obtained in the present experiment and those in literature [13,14] are shown in FIG. 7. The number of the data is limited and further experiments are, therefore, needed.

4. CONCLUSIONS

Postirradiation heating tests of the ZrC–Triso coated UO_2 particles were performed at 1600°C for 4500 h and at 1800°C for 3000 h to study the release behavior of fission products. The following was concluded:

(1) The fission gas release monitoring and the X-ray microradiography revealed that no pressure vessel failure occurred during the heating tests.

(2) The ZrC–Triso coated fuel particles showed better cesium retention than the standard Triso-coated fuel particles.

(3) In spite of better cesium retention, the ZrC layer showed a less effective barrier to ruthenium than the SiC layer.

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