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Design and development of gas cooled reactors with closed cycle gas turbines

Proceedings of a Technical Committee meeting held in Beijing, China, 30 October – 2 November 1995

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FOREWORD

It has long been recognized that substantial gains in the generation of electricity from nuclear fission can be obtained through the direct coupling of a gas turbine to a high temperature helium cooled reactor. This advanced nuclear power plant is unique in its use of the Brayton cycle to achieve a net electrical efficiency approaching 50% combined with the attendant features of low initial capital costs due to plant simplification, public acceptance from the safety attributes of the high temperature gas cooled reactor (HTGR), and reduced radioactive wastes.

The Technical Committee Meeting (TCM) and Workshop on the Design and Development of Gas Cooled Reactors with Closed Cycle Gas Turbines was convened within the frame of the International Working Group on Gas Cooled Reactors as part of the IAEA nuclear power technology development programme.

Technological advances over the past fifteen years in the design of turbomachinery, recuperators and magnetic bearings provide the potential for a quantum improvement in nuclear power generation economics through the use of the HTGR with a closed cycle gas turbine. Enhanced international co-operation among national gas cooled reactor programmes in these common technology areas could facilitate the development of this nuclear power concept thereby achieving safety, environmental and economic benefits with overall reduced development costs. This TCM and Workshop was convened to provide the opportunity to review and examine the status of design activities and technology development in national HTGR programmes with specific emphasis on the closed cycle gas turbine, and to identify pathways which take advantage of the opportunity for international co-operation in the development of this concept.

EDITORIAL NOTE

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SUMMARY

The Technical Committee Meeting (TCM) and Workshop on Design and Development of Gas Cooled Reactors with Closed Cycle Gas Turbines was held in Beijing, People's Republic of China, from 30 October to 2 November 1995. 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 Institute of Nuclear Energy Technology (INET), Tsinghua University. The meeting was organized to review the status and technology development activities for high temperature gas cooled reactors with closed cycle gas turbines (HTGR-GT), and to identify pathways of opportunity for international co-operation in the development of common systems and components. It was attended by twenty-three participants and observers from eight countries (China, France, Germany, Japan, the Netherlands, the Russian Federation, South Africa and the United States of America). Sixteen papers were presented by the participants on behalf of their respective countries. Professor Z. Wu, Director of INET, chaired the meeting. Each presentation was followed by general discussion within the area covered by the paper.

Tours of the INET research facilities and the High Temperature Reactor (HTR-10) construction site followed the meeting and workshop.

Recent advances in turbomachinery, magnetic bearings and heat exchanger technology provide the potential for significant improvement in nuclear power generation economics through use of the HTGR with closed cycle gas turbines. This TCM was used as a forum to share the recent advances in this technology and to explore those technical areas where international co-operation would be beneficial to the HTGR-GT. Areas of investigation included materials development, component fabrication, qualification of the coated fuel particles and fission product behaviour in the power conversion system, national HTGR development and testing programmes including gas turbine related systems and components.

Summaries of national and international gas cooled reactor development programmes included presentations from representatives of Japan, China, the Russian Federation, the Netherlands, France, the United States of America and Germany.

The Japanese High Temperature Engineering Test Reactor (HTTR) is currently under construction at the Japan Atomic Energy Research Institute (JAERI), Oarai Research Establishment site. This is a 30 MWth reactor with a maximum outlet temperature of 950°C. Construction of the HTTR began in March, 1991, with initial reactor criticality scheduled for late 1997. The reactor building and containment vessel, including the main components such as the reactor vessel, intermediate heat exchanger, hot gas piping and the reactor support structure are now installed. The HTTR project is intended to establish and upgrade the technology basis necessary for advanced HTGR development. The heat utilization systems planned for demonstration with this reactor are currently under consideration as part of the IAEA Co-ordinated Research Programme on the Design and Evaluation of Heat Utilization Systems for the HTTR. The gas turbine and CO_2 and steam reforming of methane are considered as first priority heat utilization system candidates. Safety demonstration tests on the HTTR are also planned to confirm the inherent safety features of the HTGR.

Construction continues on the HTR-10 at the INET site outside of Beijing, China. This pebble-bed helium cooled test reactor is being supported by the government of China. The construction permit for this plant was issued by the national safety authority at the end of 1994

and construction began in June, 1995. First criticality is scheduled for 1999. Most of the main components, including the spherical fuel elements, are to be manufactured in China. Development of the HTGR in China is being undertaken as both a source of electrical generation as a supplement to their water cooled reactors and as a universal heat source to provide process heat for various industrial applications. There will be two phases of high temperature heat utilization from the HTR-10. The first phase will require a reactor outlet temperature of 700°C. with a conventional steam cycle turbine in the secondary loop. The second phase will utilize a core outlet temperature of 950°C. primarily to investigate a steam cycle/gas turbine combined cycle system where the gas turbine and the steam cycle are independently parallel in the secondary side of the plant.

Russia has had a HTGR development programme spanning over 30 years. Detailed designs of the VGR-50, VG-400 and VGM have been completed. A feasibility study for a 215 MWth (termed VGMP) plant was performed in 1991 and 1992. Presently, development work within the frame of international co-operation with General Atomics of the USA is taking place on the gas turbine modular helium reactor (GT-MHR). This plant would have a thermal power rating of 550-600 MWth with a net electrical efficiency of approximately 46%. The Russian gas cooled reactor strategy includes a pebble bed modular HTGR with a thermal power level of about 200MW aimed at process heat production, and a prismatic core modular HTGR with a gas turbine for electrical generation. In the relatively long history of HTGR development, Russia has carried out significant development work on HTGR components including the high temperature heat exchanger, steam generator, circulator, fuel, graphite and structural materials.

Research and development activities on the HTGR began in the Netherlands in 1993. Their activities now concentrate on the development of a small pebble bed HTGR for combined heat and power production with a closed cycle gas turbine. The concept is based on the peu-a-peu design from Forschungszentrum Jülich GmbH (KFA), with an overall design philosophy of achieving significant plant simplicity. An independent neutronics and thermal hydraulics computer code system is being developed with international co-operation in performing benchmark calculations. The objectives of the Netherlands' HTGR activities include restoring social support for their nuclear programme, achieving commercial viability of the HTGR and to make the market introduction of the HTGR economically feasible with limited development and first-of-a-kind costs.

France has built and operated eight natural uranium graphite moderated CO_2 cooled reactors and exported one to Spain. All of these reactors are now out of service and their present gas cooled reactor related activities focus on study and development of techniques of plant decommissioning.

The gas cooled reactor activities in the USA have focused on the conceptual design of the GT-MHR. Significant progress has been made on this concept since its introduction in 1993. However, as part of a broad based decrease in government funding, the United States Department of Energy (DOE) terminated this programme in July, 1995. The GT-MHR programme work performed to date includes the evaluation of the technical issues centering on the power conversion components such as the turbo machine, plate-fin recuperator, magnetic bearings and helium seals. A key event in the United States GT-MHR programme was the review by Electric Power Research Institute utility representatives in June, 1995, which focused on the power conversion system. The review team concluded, in part, that the GT-MHR is a bold and innovative use of emerging technologies that have the potential for much improved efficiency, simplification and reduction of costs and wastes. This review team also acknowledged that the current United States GT-MHR conceptual design needs substantial detailed design, experimental verification and licensing effort before it can be viewed with confidence as a breakthrough technology that would provide a technically and economically viable future option for potential utility customers. The utility representatives found no show-stoppers that would prevent the GT-MHR from eventually meeting regulatory safety goals and achieving the technical and economic goals set by the DOE and the designers.

In Germany, the gas cooled reactor programme at KFA, Jülich, is evaluating the technical line of gas turbine combined cycle units aimed at achieving higher efficiency of electrical production. A comparative study of efficiency potential has shown that an electrical generation efficiency of 54.4% is achievable with a gas turbine inlet temperature of 1050°C. The mechanism of fission product release from spherical HTGR fuel elements has been reviewed to evaluate the technical feasibility of increasing the coolant outlet temperature of pebble bed reactors. Good operating experiences at the AVR plant with a mean outlet temperature of 950°C, together with other experimental results and proposals to increase the retention capability of the HTGR fuel elements, is regarded by German experts as reasons for possible realization of reactor outlet temperatures approaching 1050°C.

Both China and Japan are investigating conceptual designs featuring the use of gas turbines. In China, the focus is on a 200 MWth pebble bed MHTGR with an indirect gas turbine cycle utilizing helium as the primary coolant and nitrogen as the coolant to the turbine. This plant would have a core outlet temperature of 900°C and is expected to have an electrical generation efficiency of about 48%. The Japanese are investigating the direct cycle helium gas turbine for both a practical unit of 450 MWth and an experimental unit of 1200 MWth. Comparisons have been made of the single shaft helium gas turbine type employing an axial-flow compressor and a twin shaft type employing the centrifugal compressor. The single shaft unit has the advantages of better structure and control design, but the twin shaft unit has higher efficiency. Investigation now centers on the safety features and startup characteristics of each design.

Testing of the power conversion components as an integral system utilizing a non-nuclear heat source is seen as a prerequisite prior to connection to the reactor. The thermodynamic performance of the power conversion system can be demonstrated and verified to full temperature and speed conditions at considerably lower pressures then the plant design pressure. A proposal from Japan utilizes an electrical heater as the primary energy supply. Due to the high efficiency of the power conversion system, an outside power supply of only approximately 10% of the thermal plant rating would be required for a test of this nature.

Fission product behaviour for the direct cycle gas turbine HTGR is a very important aspect in considering the design and maintenance of this plant. The turbine blades are the first component encountered by the hot primary coolant helium upon leaving the reactor. Silver, with its tendency to plate out at high temperatures, and iodine-131 and caesium-137 with plate out at lower temperatures, are necessary design considerations in order to minimize the radiological aspects on this plant. As the behaviour of silver is not as well known as that of the noble gases, iodine and cesium, this has been an area under investigation by Japan in their evaluation of the GT-MHR. Also, an irradiation test program carried out by France in their COMEDIE BD1 loop included securing data on fission product release from a fuel element, on plate out and on fission product liftoff. After steady state irradiation, the loop was subjected to a series of in-situ blowdowns at shear ratios ranging between 0.7 and 5.6. Evaluation of the results with regard to fission product profiles show that the plate out profiles depend on the fission product chemistry and that the depressurization phases led to significant desorption of iodine-131, but had almost no effect on other fission products such as silver-110m, caesium-134 and -137, and tellurium-132.

The HTR-10 is the first high temperature gas cooled reactor to be licensed and constructed in China. The purpose of this reactor project is to test and demonstrate the technology and safety features of the advanced modular high temperature design. The licensing process for this first HTGR in China represented challenges to both the regulator and the designer. The maximum fuel temperature under the accident condition of complete loss of coolant was limited to values considerably lower than the safety limit set for the fuel element. Conversely, the reactor incorporates many advanced design features in the area of passive and inherent safety, and it is presently a worldwide issue as to how to properly treat these safety features in the licensing review process. Some of the main safety issues addressed in the licensing procedure included fuel element behaviour, source term, classification of systems and components, and containment design. The HTR-10 was licensed as a test reactor rather than a power reactor. The licensing experiences in China for this reactor could be of significant reference value in future HTGR licensing efforts worldwide.

The closed cycle gas turbine design generally incorporates a plate fin type recuperator in the power conversion system. The normal means of connecting the plates and fins is by brazing which may not have long term reliability. Diffusion welding of the plates to the fins is currently being developed for these recuperators. Tensile and creep strength in the diffusion welds, especially in high temperature applications, has shown to be superior to brazing. Early testing of the diffusion welds also has indicated high reliability.

Because of the strong relevance of the IAEA's HTGR programme to current developmental activities associated with the gas turbine, the IAEA Nuclear Power Technology Development Section initiated a task to assimilate the design and operational experiences of high temperature helium driven turbomachinery testing previously carried out in Germany. The information assembled as the result of this effort was reviewed by the author at the TCM and is included herein.

A comprehensive programme was initiated in 1968 in Germany for the research and development of a Brayton (closed) cycle power conversion system. The programme was to ultimately use a HTGR for electrical generation with helium as the working fluid. This programme continued until 1982 and involved two experimental facilities. The first was an experimental co-generation power plant (district heating and electrical generation) constructed and operated by the utility, Energieversorgung Oberhausen (EVO). This consisted of a fossil fired heater, helium turbines, compressors and related equipment. The second facility was the High Temperature Helium Test Plant (HHV) for developing helium turbomachinery and components at KFA, Jülich. The heat source for the HHV was derived from an electric motor driven helium compressor. A broad experimental development base concerning helium turbines, compressors, hot-gas duct, high temperature materials, recuperators, fuel elements, graphite, etc., was performed within the frame of this programme in order to assure the feasibility of this technology. Positive and negative experiences were gained from both facilities. The dynamical performance of the HHV turbomachine was patently excellent, whereas the EVO machine at first showed insufficient dynamical behaviour. This behaviour was significantly improved with shaft and bearing modifications. However, the power deficit of the turbomachine could not be overcome without significant rebuilding or exchange of the machine. Modifications to the HHV corrected oil ingress and excessive leakage problems. The positive experiences achieved with both facilities after overcoming initial deficiencies included excellent performance of the gas and oil seals, hot-gas ducting, turbomachine cooling, operation of the helium purification system and of the instrumentation and regulation systems.

The workshop following the TCM focused on identifying pathways for international co-operation in the development of common areas of the HTGR-GT including plant safety, power conversion system and component design and fuel and fission product behaviour. Included was a summary of the current technological and economic investigation by Eskom, the electrical utility of South Africa, into the possible deployment of small (≈100 MWe) closed cycle helium driven gas turbines to augment their electrical system capability. Under investigation is the utilization of the pebble bed modular reactor coupled to a three shaft closed cycle gas turbine power conversion system. The initial technical/economic evaluation is to be completed by the end of 1996. A decision will then be made of whether to proceed into the detailed design phase and the subsequent ordering of long lead-time components and fuel. Also addressed was a review of the role of the IAEA in the international gas cooled reactor programme. The discussions centered around the general need to co-ordinate this programme worldwide with future emphasis in the HTGR-GT areas of defining minimum safety related requirements and design basis accidents, developing a new co-ordinated research programme on power conversion system components, documenting former HTR programmes from countries such as Germany, Switzerland, USA, etc., and establishing a user/utility/vendor association for the promotion of the high temperature reactor.

SUMMARY OF NATIONAL AND INTERNATIONAL ACTIVITIES IN GAS COOLED REACTORS

Session 1

CONSTRUCTION OF THE HTTR AND ITS TESTING PROGRAM FOR ADVANCED HTGR DEVELOPMENT

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Abstract

Concerning about global warming due to emission of greenhouse effect gas like CO_2, it is essentially important to make efforts to obtain more reliable and stable energy supply by extended use of nuclear energy including high temperature heat from nuclear reactors, because it can supply a large amount of energy and its plants emit only little amount of CO_2 during their lifetime. Hence, efforts are to be continuously devoted to establish and upgrade technologies of High Temperature Gas-cooled Reactor (HTGR) which can supply high-temperature heat with high thermal efficiency as well as high heat-utilizing efficiency. It is also expected that making basic researches at high temperature using HTGR will contribute to innovative basic research in future. Then, the construction of High Temperature engineering Test Reactor (HTTR), which is an HTGR with a maximum helium coolant temperature of 950℃ at the reactor outlet, was decided by the Japanese Atomic Energy Commission (JAEC) in 1987 and is now under way by the Japan Atomic Energy Research Institute (JAERI).

The construction of the HTTR started in March 1991, with first criticality in 1997 to be followed after commissioning testing. At present the HTTR reactor building and its containment vessel have been constructed and its main components, such as a reactor pressure vessel, an intermediate heat exchanger, hot gas pipings and core support structures, have been installed in the containment vessel. Fuel fabrication has been started as well.

The project is intended to establish and upgrade the technology basis necessary for advanced HTGR developments. Some heat utilization system is planned to be connected to the HTTR and demonstrated at the former stage of the second core. At present, steam-reforming of methane is the first candidate. Also the HTGR with Gas-Turbine has been studied for assessment of technical viability.

Besides the demonstration of the heat utilization system, the JAERI plans to carry out safety demonstration tests to confirm the salient inherent safety features of the HTGR. In addition material and fuel irradiation tests as upgrading HTGR technologies after attaining rated power will be conducted. Preliminary tests on selected research subjects such as composite material and ZrC coated fuel developments, have been carried out at high temperature and under irradiation.

1. Introduction

A High Temperature Gas-cooled Reactor (HTGR) can supply high temperature heat as high as 1000℃, which has a potential of obtaining high thermal efficiency as well as high heat-utilizing efficiency. It also has excellent features such as high inherent safety, easy operation and high fuel burnup. From the view point of the global environmental protection and diversification of energy usage, non-electrical application of nuclear energy, hydrogen production for example, is very important. Therefore, in order to establish and upgrade the technological basis for HTGRs and also to use as a tool of basic researches for high temperature and neutron irradiation, the Japan Atomic Energy Research Institute (JAERI) has been constructing a 30MWt High Temperature engineering Test Reactor (HTTR) at the Oarai Research Establishment. The first criticality is scheduled in 1997. This report describes present status of the HTTR construction and its testing program for advanced HTGR developments.

2. Present status of HTTR construction

The HTTR plant is composed of a reactor building, a spent fuel storage building, a machinery building and so on. The reactor building is 48m x 50m in size with two floors on the ground and three under ground. Major components such as the reactor pressure vessel, primary cooling system components etc. are installed in the containment vessel. Air cooling towers for the cooling system are located on the roof of the reactor building. The construction of reactor building started in March 1991 and has been almost completed. The external view of the reactor building is shown in Photo 1. The major specification of the HTTR are shown in Table 1.

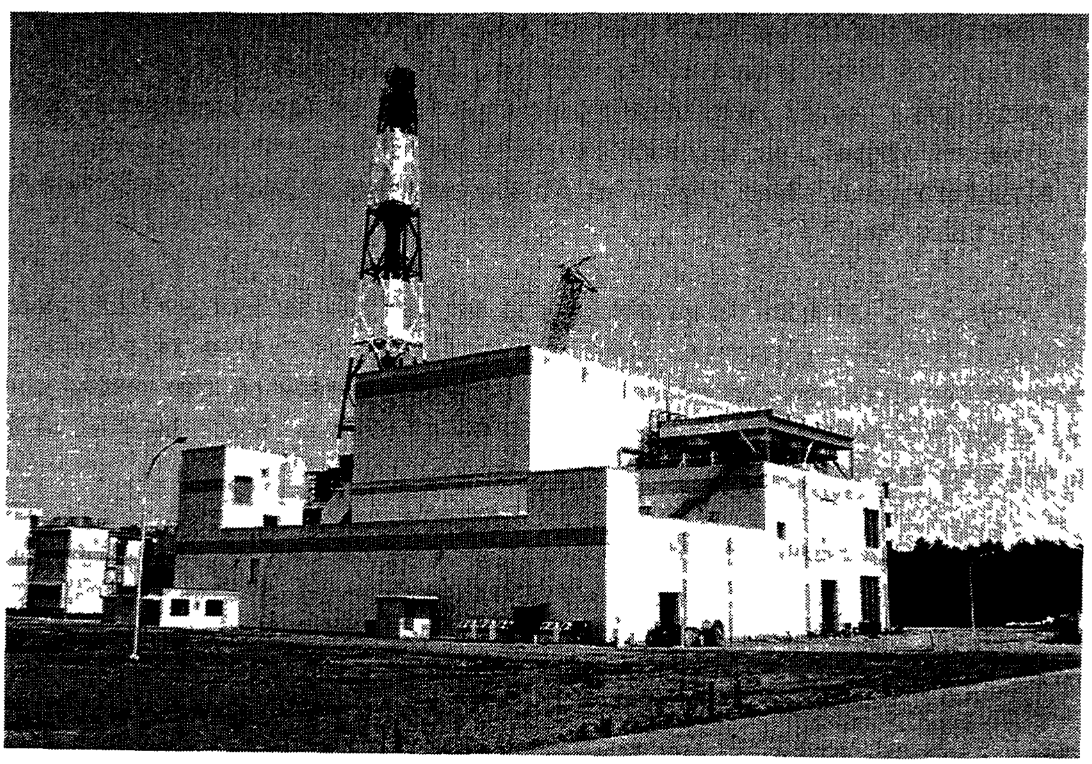

Photo 1. External view of the reactor building

TABLE 1. MAJOR SPECIFICATION OF HTTR

Thermal power	30 MW
Outlet coolant temperature	850°C/950°C
Inlet coolant temperature	395°C
Fuel	Low enriched UO_2
Fuel element type	Prismatic block
Direction of coolant flow	Downward-flow
Pressure vessel	Steel
Number of main cooling loop	1
Heat removal	IHX and PWC (parallel loaded)
Primary coolant pressure	4 MPa
Containment type	Steel containment
Plant lifetime	20 years

The block type fuel is adopted in the HTTR considering the advantages of fuel zoning, control of coolant flow rate in each column, easy insertion of CRs, irradiation flexibility in the core. Core components and reactor internals have been installed from May to August 1995. The installed core arrangement of a top replaceable reflector region is shown in Photo 2. In order to verify the seal performance between permanent reflector blocks, the air leakage test had been carried out to measure leakage flow rate. The measured value was found to be less than the assumed limit in the core thermal hydraulic design. The core is cooled by helium gas of 4MPa flowing downward. The core has 30 fuel columns and 7 CR guide columns, which is surrounded by replaceable reflector blocks.

The reactor cooling system is composed of the MCS, ACS and VCS as schematically shown in Fig. 1. The MCS is operated in normal operation condition to remove heat from the core and send it into the environment. The ACS and VCS have incorporated safety features. The ACS is initiated to operate in case of a reactor scram. Besides one out of two components of VCS has sufficient capacity to remove residual heat, the ACS is provided to cool down the core and core support structure. A helically coiled intermediate heat exchanger (IHX) whose heat-resistant material is Hastelloy-XR developed by the JAERI has been installed in September 1994. Nuclear heat application tests using the HTTR, are planned to be carried out, and accordingly a heat utilization system will be connected to the IHX. The fuel fabrication started in June 1995 and will complete in 1997.

The fuel fabrication facility has two production lines of the kernel production and coating processes, and one fuel compact production line.

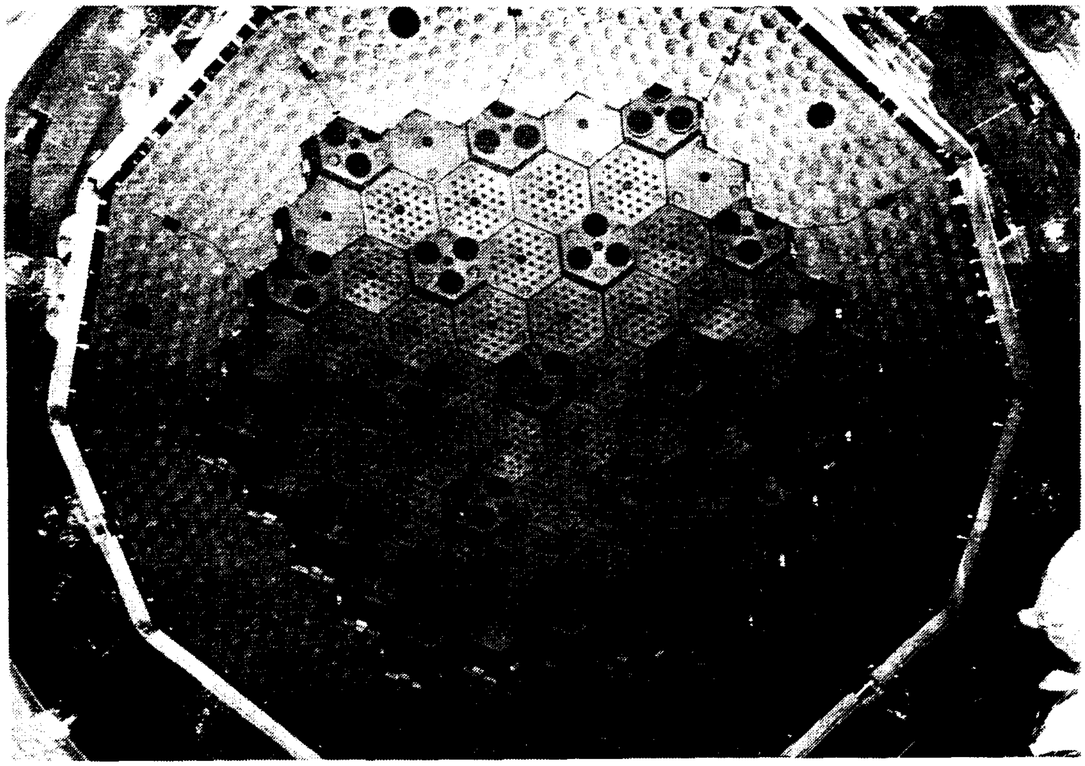

Photo 2. Installed core arrangement of top replaceable reflector region

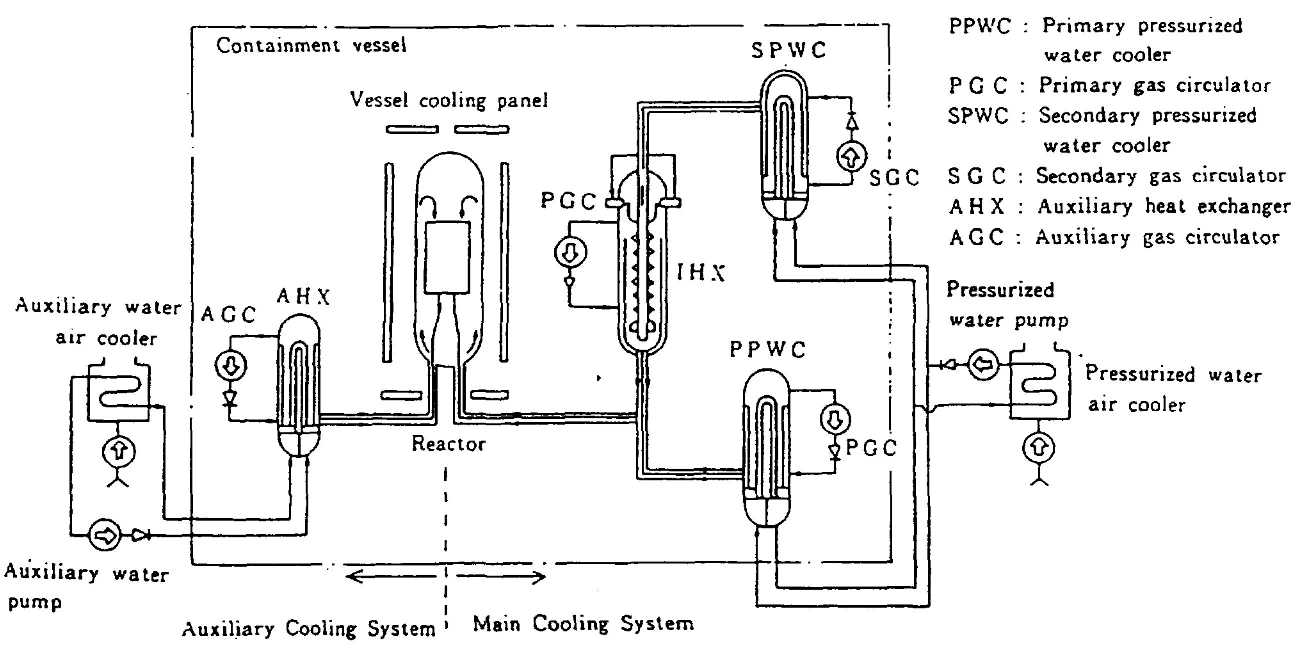

Fig. 1 Cooling system of HTTR

A functional test operation of the reactor cooling system will be performed from May 1996 to September 1997. Fuel will be loaded into the core around in September 1997 and the first criticality is expected in December 1997.

3. HTTR Testing Program for Advanced HTGR Development

The HTTR project is intended to establish and upgrade the technology basis necessary for advanced HTGR developments. Based on the discussion about future prospects of advanced HTGRs for the preservation of the global environment and sustainable development, two types of future promising HTGRs have been selected to be developed.[1] One is the medium-sized modular and severe accident free HTGR to enable the siting in industrial complex with process heat utilization of hydrogen production, conversion of fossil fuels such as coal gasification or liquefaction and/or gas turbine power generation. The other is a small innovative HTGR of a super safe and/or maintenance-free HTGR sited in isolated islands or the basement of some buildings. The design studies and R&D works on the innovative HTGRs have been started in order to make the best of testing program using the HTTR. Around in 1998 the results of these studies will be subject to the review of the Japanese Atomic Energy Commission (JAEC) at the revision of long-term program for research, development and utilization of nuclear energy.

Some heat utilization system is planned to be connected to the HTTR and demonstrated at the former stage of the second core. At present, steam-reforming of methane is the first candidate. Also the HTGR with Gas-Turbine has been studied for assessment of technical viability.

Besides the demonstration of the heat utilization system, the JAERI plans to carry out safety demonstration tests to confirm the salient inherent safety features of the HTGR. In addition material and fuel irradiation tests as upgrading HTGR technologies will be conducted after attaining rated power. Preliminary tests on selected research subjects such as composite material and ZrC coated fuel developments, have been carried out at high temperature and under irradiation.

3.1 Nuclear heat utilization tests

The HTTR is designed to transfer the thermal energy of 10MW to secondary helium at temperature of 905℃ and pressure of 4.1MPa through an intermediate heat exchanger (IHX).

Steam reforming of methane for production of hydrogen is adopted for the HTTR heat utilization system, because of the following reasons.

1) Hydrogen would be an ideal energy carrier if the utilization technologies converted into useful forms of energy (heat, electricity and fuels) as well as those for the storage, transport and safe handling of hydrogen would have been established successfully.

2) The demonstration test of the coupling of the steam reforming system to the HTTR will contribute to the technical solution of all other hydrogen production system because basic system arrangement and endothermic chemical reactions for hydrogen production are similar.

3) Steam reforming system consists of components of matured technologies in the non-nuclear field.

4) Steam reforming is an economical hydrogen production process commercialized at present in the non-nuclear field.

Design works of the HTTR steam reforming system have been started since 1990 aiming at demonstration test of the HTTR at the beginning of the 21st century.

Design of main system arrangement, of key components such as a steam reformer and a steam generator and of control method, especially start-up procedure and development of computer code for transient thermal-hydraulics analysis have been conducted. The steam generator is designed to have a safety function as an interface.

In order to achieve hydrogen production performance competitive to that for fossil fueled plants, measures have been taken to enhance helium side heat transfer rate, to use following bayonet type of tubes and to optimize process gas composition.

1) The orifice baffles with wire nets attached on the outside of steam reformer tubes are installed.

2) The bayonet type of reformer tube is adopted to utilize the outlet process gas at the outlet of catalyst layer at the temperature of approximately 830℃.

3) The temperature of process gas is lowered from 450℃ to 600℃, to increase the heat input preventing carbon deposit. This design achieves heat input of 3.6MW from helium gas to the steam reformer and 1.2MW from outlet process gas of catalyst layer to inlet process gas. Total heat input of 4.8MW are utilized to heat up the process gas and to give steam reforming process heat.

The main system arrangement is shown in Fig. 2. The high temperature secondary helium gas flows at first into the steam reformer and then into a superheater and a steam generator. The superheated steam is provided mainly to the steam reformer.

The heat utilization efficiency of 78% using steam reformer is possible to be achieved, competitive to non-nuclear process (80%-85%).

3.2 Assessment of HTGR with Gas-Turbine technologies

Study Group of the HTGR with Gas-Turbine Power System (HTGR-GT) of the Japan Society of Mechanical Engineers has been set up since November 1993 to review and assess the HTGR-GT technologies and identify technical areas to be solved with their development. This study group has held a series of the meeting and will release a

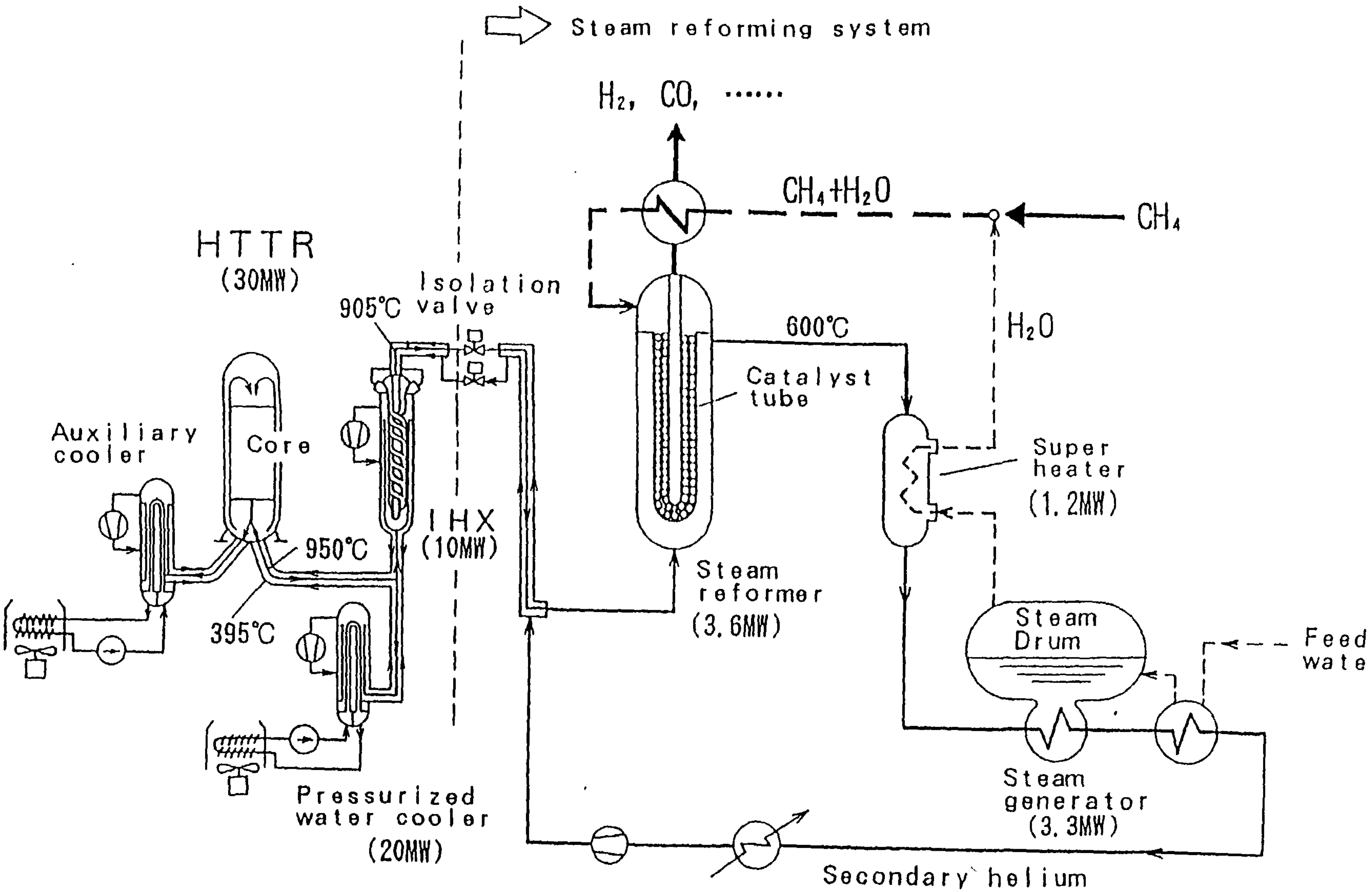

Fig. 2 Steam reforming system connected to the HTTR

report on its activities by the end of February 1996. During this period, voluntary survey and R&D works evaluating related technologies have been conducted by researchers and engineers of JAERI as well as of nuclear industries. Corresponding to these activities, JAERI is now planning to launch a new feasibility study on the HTGR-GT from 1996 to 2000, which is subject to appropriated funds.

3.3 Safety demonstration tests

The following safety demonstration tests are planned in the HTTR to verity inherent safety features of HTGRs.

1) Abnormal control rod withdrawal tests and
2) Coolant flow reduction tests

For these tests the code as a simulator has been developed for performing analyses of HTTR core transients and accidents.

3.4 Material and fuel irradiation tests as upgrading HTGR technologies

To develop advanced HTGRs (reactor outlet coolant temperature about 1100℃, power density 3-6w/cm^3 and fuel burnup about 100GWd/t are targeted respectively), overall R&D program on upgrading HTGR technologies has been drafted in March 1995. However there still exists strong arguments among ourselves what type of advanced HTGRs should be mainly developed for the future. Long-Term program for research, development and utilization of nuclear energy revised by the JAEC in June 1994 says that since the technologies of HTGR with high temperature heat supply should be developed for the broader use of nuclear energy, the construction of the HTTR and its testing program shall be planned and executed for establishing and upgrading the HTGR technology basis and to conduct various innovative and basic researches on high temperature technologies. This program does not directly address future deployment plan of advanced HTGRs following the HTTR. So we started the design studies and R&D works on advanced HTGRs in order to incorporate the results in the long-term program revised by the JAEC around in 1998.

At present the first priority is dedicated to the medium-sized modular and severe accident free HTGR with high temperature heat. In order to upgrade core performance achieving higher power density, higher burnup of the core and higher allowable temperature design limit of the fuel, ZrC coated fuel and c/c composite material for control rod sleeves have been pre-tested.

4. Concluding remarks

The HTTR is a high temperature gas cooled test reactor which has various aims and operational modes. The construction of the HTTR has progressed smoothly and its first criticality is foreseen in December 1997.

The various tests by the HTTR will make a great contribution to confirm salient characteristics of HTGRs including reliable supply of high temperature heat as high as 950℃ and high inherent safety and the application of high temperature heat from HTGRs to various fields will also contribute to relax global environmental problems. Furthermore, the HTTR has a unique and superior capability for carrying out high temperature irradiation tests not only for development of advanced HTGRs but also for basic researches such as new semi-conductors, super-conductors, composite material developments and tritium production and continuous recovery testing of fusion reactor blanket materials. The HTTR is highly expected to contribute so much to promoting the international cooperation in these fields.

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PROGRESS OF THE HTR-10 PROJECT

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Abstract

This paper briefly introduces the main technical features and the design specifications of the HTR-10 Present status and main progress of the license applications, the design and manufacture of the main components and the engineering experiments as well as the construction of the HTR-10 are summarised

1. General Introduction

1 1 Background

Considering the utilization of nuclear energy in next century, China has paid great attention to the development of advanced reactors which have good safety features, economic competitiveness and uranium resource availability The high temperature gas-cooled reactor was chosen as one of the advanced reactor types for future development and covered in the National High Technology R&D Programme in 1986 The Institute of Nuclear Energy Technology (INET) of Tsinghua University was appointed as the leading institute to organize and carry out the key technology development, the conceptial design and the feasibility study of HTGR in 1986-1990, so called the period of " Seventh Five-Year Plan" The conceptual design and the feasibility study report of the HTR-10 was completed in 1991 and then examined by the Expert Committee of the Energy Technology Area of the National High Technology R&D Programme and approved by the State Science and Technology Commission (SSTC) Finally, The 10MW high temperature gas-cooled test reactor (HTR-10) project was approved by the State Council in March 1992

INET is responsible for design, license applications, construction and operation of this test reactor Now the HTR-10 is being constructed in the site of INET which is located in the North-west of Beijing city and has erected other two test reactors, e g the 2MW swimming pool type experimental reactor and the 5MW nuclear heating test reactor

1 2 The Objective of the HTR-10

The construction of the HTR-10 is the first step of the HTGR development strategy in China The objective of the HTR-10 is to verify and demonstrate the technigue and safety features of Modular HTGR and to establish an experimental base for developing the nuclear process heat applications The specific aims of the HTR-10 have been defined as follows:

(1) To acquire the experience of HTGR design, construction and operation
(2) To carry out the irradiation tests for fuel elements
(3) To verify the inherent safety features of the Modular HTGR
(4) To demontrate the electricity/ heat co-generation and gas/steam turbine combined cycle
(5) To develop the high temperature process heat utilizations

1.3 The Design Specification

The HTR-10 is a pebble bed type high temperature gas-cooled reactor, it uses the spherical fuel elements with ceramic coated fuel particles. The reactor core which has diameter of 1.8, mean height of 1.97m and the volume of 5.0m^3 is surrounded by the graphite reflectors. 27,000 fuel elements are loaded in the core. The fuel elements use the low enrichment uranium and the design mean burnup is 80,000 MWd/t. The pressure of the primary helium circuit is 3. 0MPa. In the first phase, the HTR-10 will be operated at the core outlet temperature of 700℃ and inlet temperature of 250℃. At the secondary circuit, a steam turbine cycle for electricity and heat co-generation is designed. The steam generator produces the steam at temperature of 440℃ and pressure of 4.0MPa to provide a standard turbine-generator unit. The main design date of the HTR-10 are shown in Table 1.

Table 1. The main design data of the HTR-10

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

In the second phase, the HTR-10 will be operated at the core outlet temperature of 900℃ and inlet temperature of 300℃. A gas turbine (GT) and steam turbine (ST) combined cycle for electricity generation is preliminarily designed. The intermediate heat exchanger (IHX) with therml power of 5MW provides the high temperature nitrogen gas of 850℃ for the GT cycle. The steam generator (SG) with rest thermal power of 5MW produces the steam at temperature of 435℃ for the ST cycle. The main design data of the second operation phase are shown in Table2.

Table 2. The main design data of GT-ST Combined Cycle

· Reactor	
Thermal power	10MW
Primary helium pressure	3.0MPa
Core outlet temp.	900℃
Core inlet temp.	300℃
Primary pressure	3.0MPa
· IHX	
Thermal power	5MW
Temp.of primary helium side	900℃ / 600℃
Temp.of secondary nitrogen side	850℃ / 483℃
Scondary pressure	3.2MPa
· SG	
Thermal power	5MW
Temp.of primary helium side	600℃ / 287℃
Temp.of secondary water side	435℃ / 104℃
Secondary pressure	3.43 / 4.20MPa

· Power output

Power of GT cycle	1.94MWe
Power of ST cycle	1.36MWe

1.4 The Main Technical Features

Comparing with the previous pebble bed HTGR, the particular technical features of HTR-10 are as following:

(1) The core residual heat is designed to be dissipated by a passive heat transfer system.

(2) The two reactor shut down systems which are consisted of 10 control rods and 7 small absorber ball holes are all positioned in the side reflector. The in core control rods are not needed.

(3) The pneumatic pulse singulizer for discharging the spherical fuel elements from reactor core is used in the fuel handling system.

(4) The reactor and the steam generator are arranged side by side. The pressure boundary of the primary circuit is consisted by the reactor vessel, the steam generator vessel and the connected vessel (hot gas duct vessel)

(5) The integrated steam generator and intermediate heat exchanger are designed. The SG is a once through, modular small helical tube type. The IHX can provide the high temperature nitrogen or helium of 850℃ to 900℃ in the secondary circuit for the gas-trubine cycle or process heat utilization testing.

(6) 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.

(7) The digital protection system is used in HTR-10.

2. Progress of the HTR-10 Project

2.1 The Safety Review and Licensing Application

For the application of the construction permission, the following procedures should be passed.

(1) INET had compiled the Environmental Impact Report (EIR) of the HTR-10 and submitted it to the National Euvironmental Protection Administration(NEPA) in the mid of 1992, the report was reviewed by a expert committee, then the NEPA approved the EIR of the HTR-10 in November 1992. It is one of the necessary basis for the application of the reactor site.

(2) The Siting and Seismic Report (SSR) of the HTR-10 was submitted to the National Nuclear Safety Administration (NNSA). After examination the reactor site was approved in December 1992.

(3) The Preliminary Safety Analysis Report (PSAR) had been completed and submitted to the NNSA for the application of the construction permission in December 1993. The activities of the licensing procedure lasted for one year. The NNSA formally issued the construction permission of the HTR-10 in December 1994.

2.2 The Design of the HTR-10

(1) For the design and licensing reguirement, the INET had prepared the technical documents which are the design criteria of the HTR-10 and the format and content of the safety analysis report of the HTR-10. These two documents were examined and approved by the NNSA in August 1992 and March 1993 respectively.

(2) The basic design and the budget estimate of the HTR-10 was carried out in the mid of 1994 and then examined and approved by both the State Education Commission (SEC) and the State Science and Technology Commission (SSTC) in the end of 1994.

(3) The detailed design of the components, systems and buildings is being carried out by the INET in Cooperation with the Southwest Center of Reactor Engineering Research and Design (SWCR) for the helium purification system and other helium auxiliary systems, and the Beijing Institute of Nuclear Engineering (BINE)© for the steam electricity conversion system and the turbine generator building. For the detailed design of the main components e.g. the reactor pressure vessel, the steam generator and the helium circulator, the design engineers of the INET have close contacted and discussed with the engineers of the manufacturers to modify and improve the design drawing and the technical specifications. The detailed design of the HTR-10 is planed to complete in the next year.

2.3 The Engineering Experiments

A engineering experiments program for the HTR-10 key technique has been performed in INET for years. The main aims of the engineering experiments are to verify the design characteristics of the components and systems, to demonstrate the relevant features and to obtain the operation experience in the simulated conditions.

The various experimental facilities have been set and tested or are being established. The key engineering experiments are as following:

- The high temperature helium test loop and the relevant helium technology.
- The fuel handling system test.
- The control rod driving apparatus test.
- The small absorber ball simulating system.
- The hot gas duct test facility.
- The stability test of the steam generator model.
- The helium flow temperature mixing.

The test components of the fuel handling system and the small absorber ball system, the prototype of the coutrol rod driving apparatus and the test section of the hot gas duct are designed in 1: 1 scale. It is planed to perform the experiments at the operation temperature and helium atmosphere conditions. The experiments of the fuel handling system and small absorber ball simulating system at ambient temperature had been carried out.

2.4 The Manufacture of the Main Components

The main components of the HTR-10 such as the reactor pressure vessel, the steam generator vessel and its internal parts, the helium circulator and the core matallic internal are fabricated by the domestic factories which have the ability and experience of manufactring PWR's components. The graphite of the core internal and part of the safety grade helium valves will be imported from the foreign suppliers.

The reactor pressure vessel is a safety grade I component. It is a cylindrical vessel which has height of 11.4m, diameter of 4.2m and total weight of 142tons. It is fabricated by the Shanghai Boiler works.

The steam generator vessel as part of the pressure boundary of the primary circuit is also a safety grade I component. It has height of 11.2m, diameter of 2.5m and total weight of 70tons. The once through type steam generator is consisted by 30 small helical heating tubes. The diameter of the heating tube is $\phi 18\times 2mm/\phi 18\times 3mm$ and the effective length is 35m. The helical tube unit is 115mm in diameter. This component (vessel and internals) is fabricated by the Shanghai Power Station Auxiliary Equipment Works.

The helium circulator is a vertical single-stage centrifugal one, the impeller is at the end of the shaft. The circulator has the same axle with its drive motor and fixed in the circulator pressure vessel which is the top part of the steam generator vessel. The helium circulator is fabricated by the Shanghai Blower Works.

The core metallic internal which consistes of the metallic core vessel, biological shielding structure, bottom support plate and enhanced plate, top pressing plate will be manufactured by the Shanghai No.1 Machine Works.

The graphite bricks of core reflector will be supplied by the Toyo Tanco Co. Ltd. Japan, the final machining is planed to be done in the works of INET. The carbon bricks of the reflector will be domestically fabricated.

The components of the fuel handling system, the helium purification system and other auxiliary systems will be also domestically made.

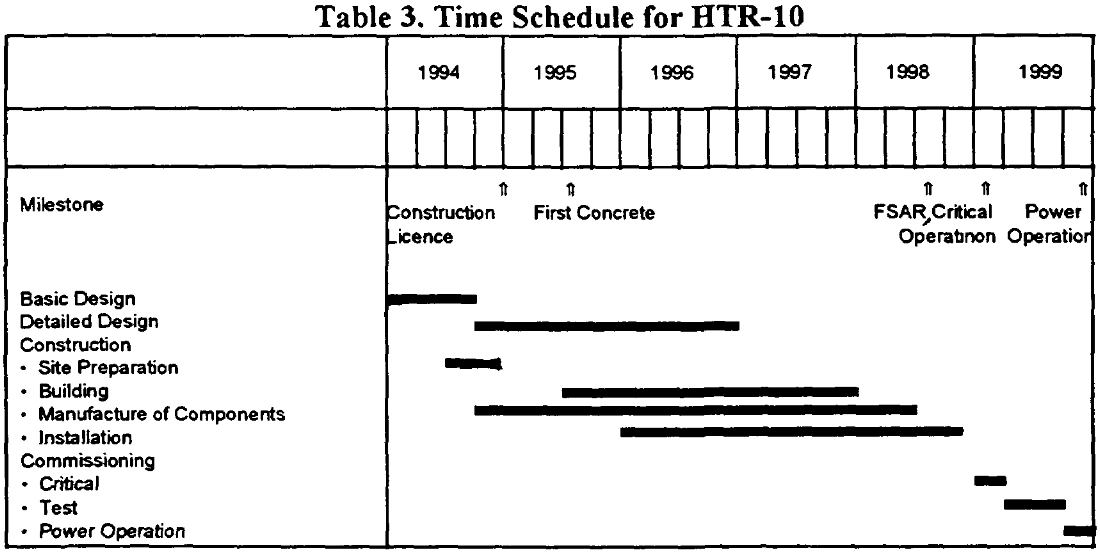
2.5 The Building Construction

The HTR-10 test plant includes a reactor building, a turbine generator building with two cooling towers and a ventilation centre with a stack. The buildings are to be arranged and constructed in the area of $100\times 130m^2$.

The building construction and installation are contracted by the Engineering Company No.23 of the China National Nuclear Corporation (CNNC) The excavation ground was completed in the end of 1994, the constraction of the HTR-10 was formally started in June 1995, the basement of the reactor building was poured in september 1995.

3. The Time Schedule

Table 3. Time Schedule for HTR-10



The time schedule of HTR-10 construction is shown in Table3. The reactor building construction will be lasted two and half year and scheduled to complete in the end of 1997. In parallel, the manufacture of the components and installation of the main components, equipments and auxiliary systems will be close followed the progress of the building construction and scheduled to complete in the end of 1998. The first criticality of the reactor is planed to be reached in the beginning of 1999

STATE OF HTGR DEVELOPMENT IN RUSSIA

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Abstract

Detailed designs of VGR50, VG400 and VGM reactors were developed in Russia. The most important result of the activity was formation of contacts between different organizations and creation of technology basis for HTGRs. At present it is assumed that advantages of HTGRs as compared to other types of reactors can be more completely demonstrated if module reactors with thermal power of about 200 MW and pebble bed core will be aimed only at process heat production, and for electricity production through gas-turbine cycle will be used module reactors with thermal power of about 600 MW and a core from prismatic blocks. To this effect, feasibility study of VGMP reactor was carried out and development of GT-MHR with gas-turbine cycle is under way.

1. Brief information on HTGR development history in Russia.

Development of HTGRs was initiated in Russia about 30 years ago. During this period designs of VGR-50, VG-400, VGM and VGMP reactor plant have been worked out. General information on the projects is given in Table 1.

VGR 50 - helium cooled reactor with pebble bed core was intended for electricity production and radiation of polyethylene tubes. To this effect circulation of spherical fuel elements around a closed path was provided.

TABLE 1.

Name of the project	Dates of beginning and completion of the project	General Designer of the reactor	Phase of the development
VGR 50	1963-1985	Research Institute of Machinery for Atomic Industry, Moscow	detailed design
VG 400	1974-1987	OKBM Nizhny Novgorod	detailed design
VGM	1986-1991	OKBM	detailed design
VGMP	1991-1992	OKBM	feasibility study

Parameters of VGR 50 reactor are shown in Table 2.

TABLE 2.

Parameter	Magnitude
Thermal power, MWt	136
Electrical power, MWt	50
Helium temperature, ^{o}C	
reactor input	260
reactor output	810
Helium flow rate, kg/s	54
Helium pressure, MPa	4

For a number of reasons activity on the design was ended in the middle of 80-th.

VG 400 reactor plant was intended for both electricity production and process heat. The heat is transferred to a methane steam reformer and hydrogen gained is used for ammonia production. The reactor plant parameters are given in Table 3. The key components of the reactor plant are housed in the prestressed concrete reactor vessel (Fig 1).

Parameters of VG400 reactor are shown in Table 3

TABLE 3.

Parameter	Magnitude
Thermal power, MWt	1060
Electrical power, MWt	300
Number of loops	4
Helium temperature, ^{o}C	
reactor inlet	350
reactor outlet	950
Helium temperature at steam generator inlet, ^{o}C	750
Helium flow rate, kg/s	340
Helium pressure, MPa	5

On the preliminary phase of the design development two variants of the core were analyzed: on the basis of pebble bed and prismatic fuel blocks. As u result of calculational, design and engineering analysis the pebble bed core was chosen for further development. The pebble bed core option for this reactor was made taking account of the following considerations:

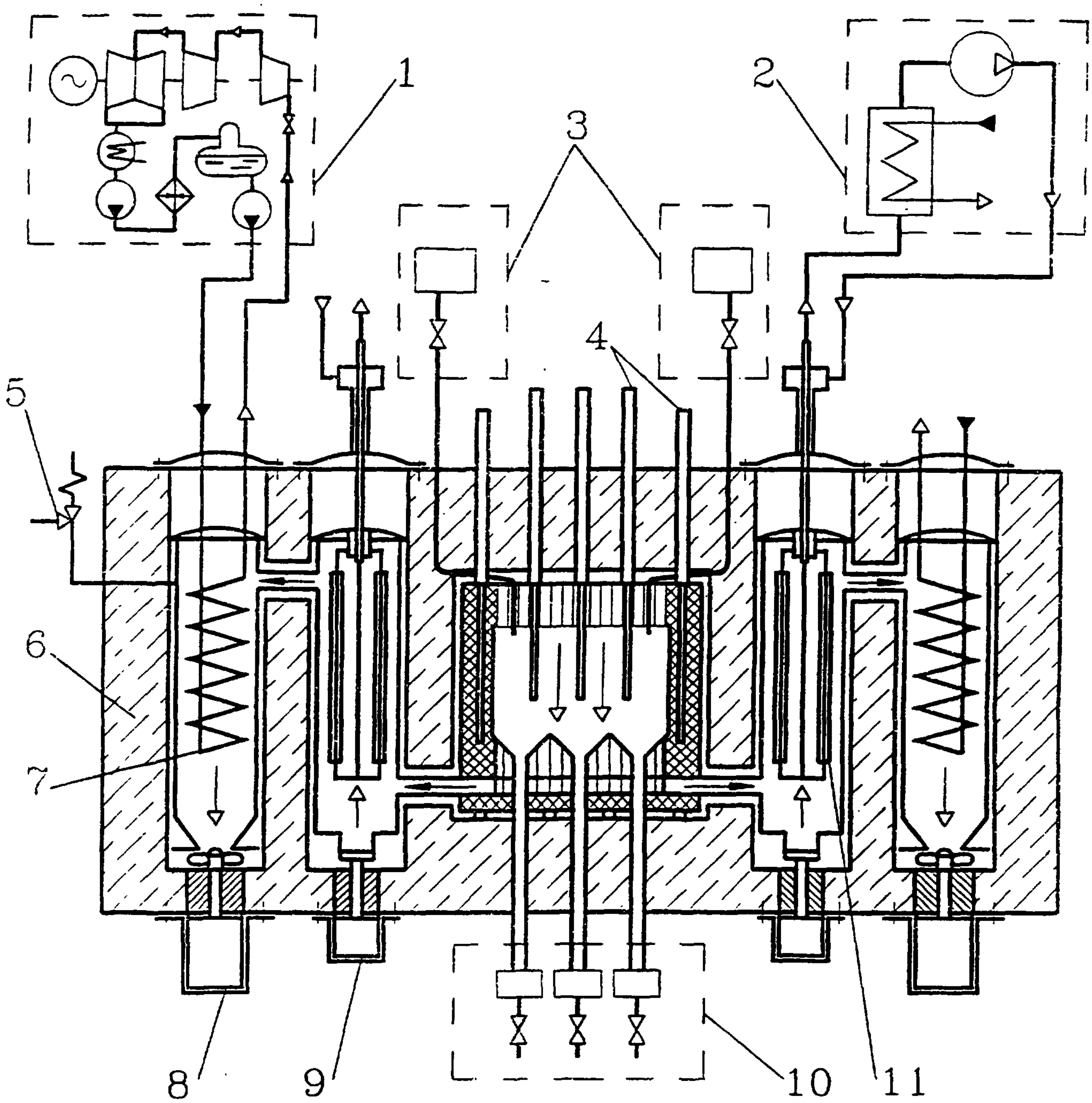


Fig.1. VG-400 reactor plant

1 -	steam turbine plant	6 -	PCRV
2 -	intermediate circuit helium	7 -	steam generator
		8 -	gas circulator
3 -	fuel loading system	9 -	bypass valve
4 -	CPS rod drives	10 -	fuel unloading system
5 -	relief valve	11 -	intermediate heat exchanger

- more simple technology of fuel elements manufacture and possibility of their full scale testing in experimental reactors;

- use of more simple mechanisms for the core refueling;

- possibility of the core refueling with the reactor running.

VG 400 reactor with closed gas-turbine cycle was analyzed as well.

VGM modular reactor was designed to validate main technical decisions connected with production of high temperature process heat. VGM reactor operation experience would promote later utilization of HTGRs in the Russian Industry.

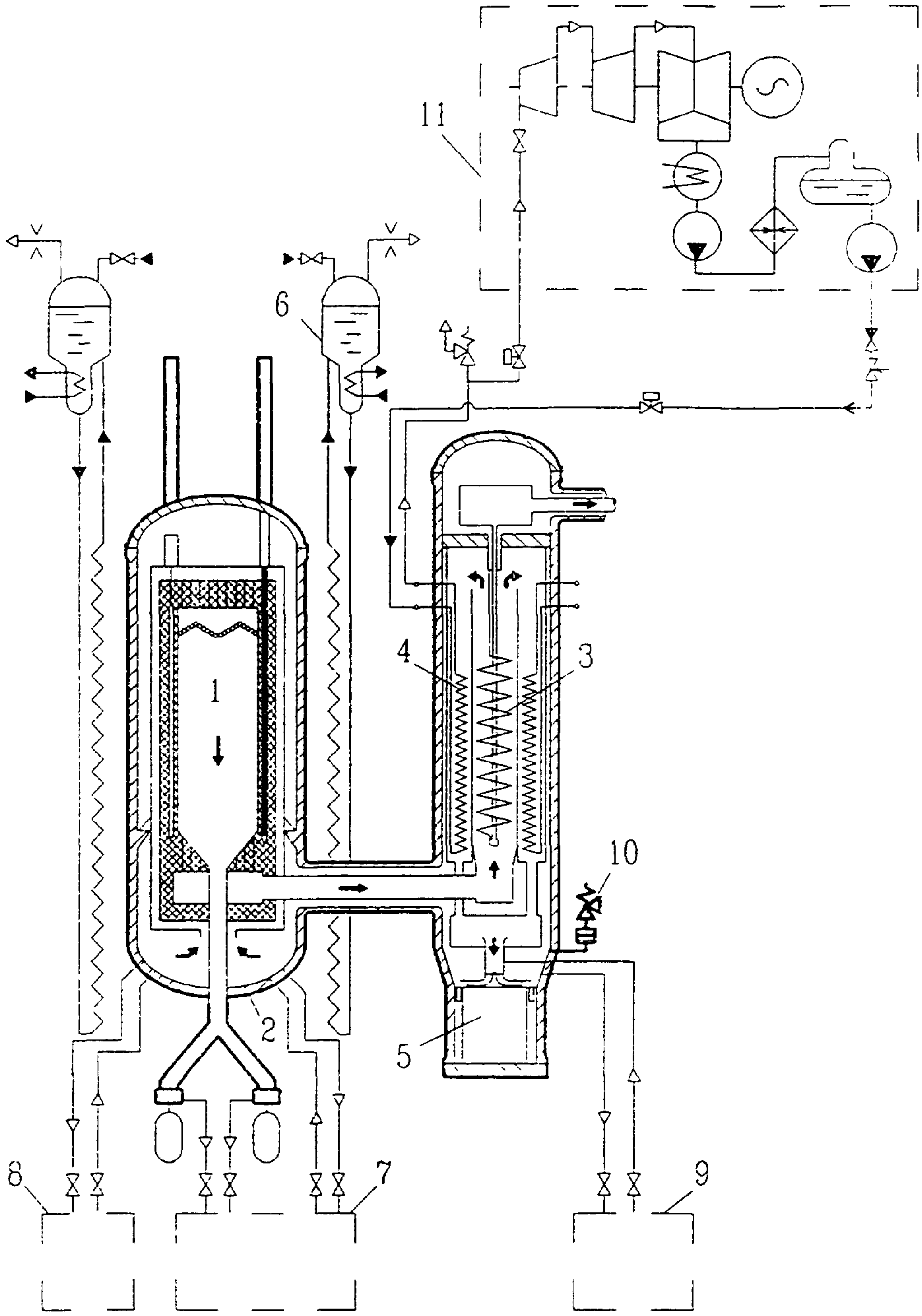
On the basis of VGM reactor the development of VGMP reactor for production of middle temperature heat was initiated.

Main parameters of the reactor plants are given in Table 4.

TABLE 4.

	Magnitude	
	VGM	VGMP
Thermal power, MWt	200	215
Helium temperature, ^{o}C		
reactor inlet	300	300
reactor outlet	750...950	750
Helium flow rate, kg/s	59...85	91,5
Helium pressure, MPa	5	6
Number of loops	1 main and 1 auxiliary	1 main

The reactor components are arranged in steel vessels (Fig 2 and 3).



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. 2. VGM reactor plant

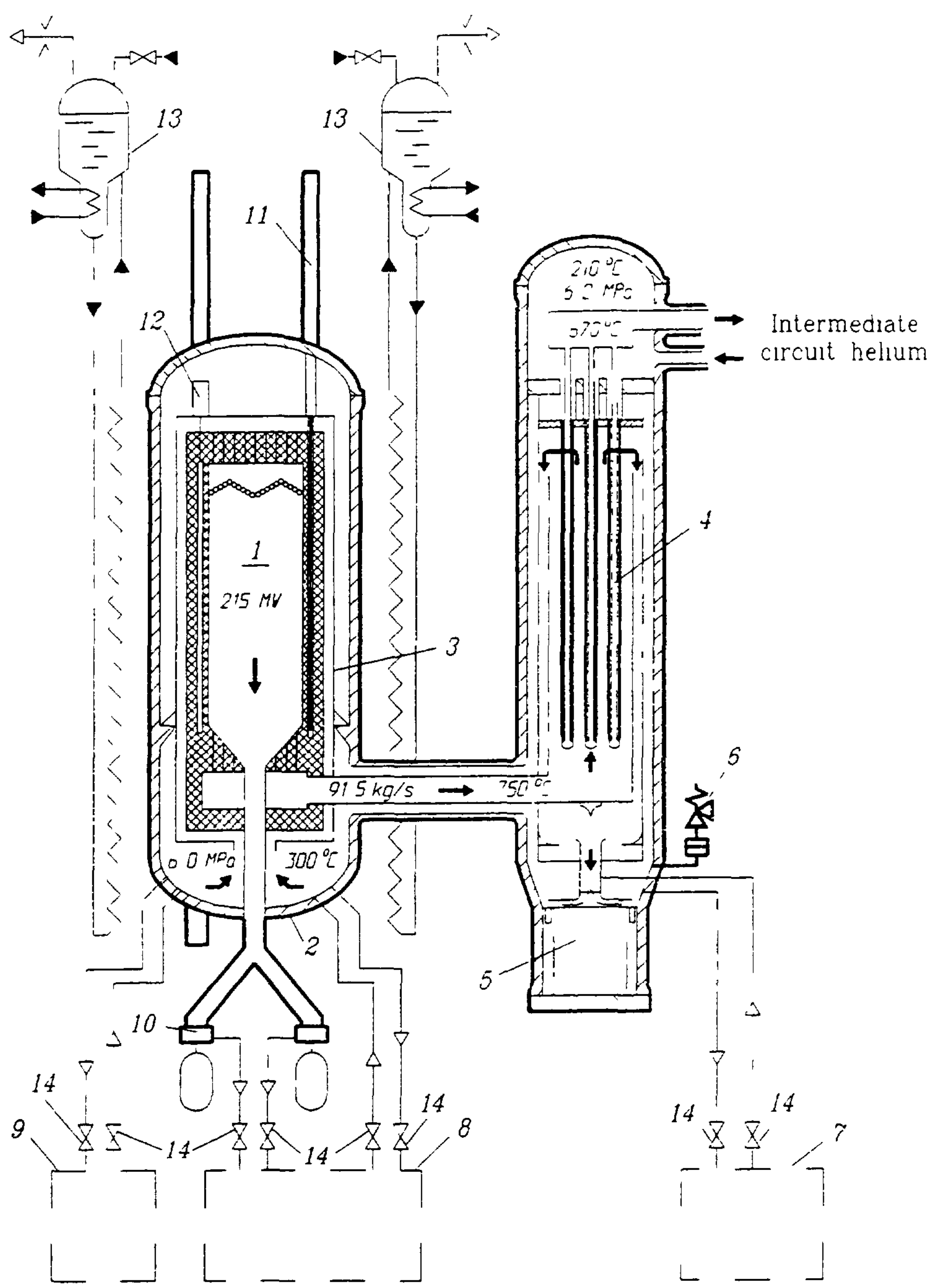


Fig 3 VGM-P RP schematic diagram 1 - reactor core, 2 - vessels block, 3 - core protective housing, 4 - intermediate heat exchanger, 5 - main gas blower with a cut-off valve 6 - relief valve, 7 - helium purification system, 8 - fuel circulation system 9 - small absorbing balls system. 10 - fuel element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

2. State of HTGRs component development.

The great attention was paid to the development of high temperature heat exchanger, steam generator, circulator, fuel, graphite and structural materials. Different designes of steam generator and heat exchanger have been considered, experimental facilities for the investigation of thermohydraulic and vibration characteristics have been created (ST-1312, ST-1565).The 5 MWt steam generator model has been fabricated to be tested at ST-1312 facility. The facility was in trial operation at the heaters power of 30 %of the nominal value. The helium temperature and pressure were 730 ^{o}C and 5 MPa.

A full-scale prototype of the gas circulator has been fabricated for testing at ST-1383 facility. Startup and trial operation of the facility under partial loads have been performed. The further testing has been delayed.

Russian organization has considerable experimental background in coated particle and fuel compact manufacture. A pilot plant with an output of 10^{8} coated particles per year was developed The Chemical Concentrates Plant (Novosibirsk) has process equipment to manufacture fuel particles and fuel elements. The manufacturing process employs the "sol-gel" type. The plant has produced about 10,000 spherical fuel elements. The fuel was irradiated up to temperatures of 1600^{o}C.

The Research Institute of Graphite (Moscow) has been developed technology of RBMK graphite and new GR-1, MPG graphites which are based on calcinated and noncalcinated coke pressing.

Test specimens of GR-1 and MPG graphites were irradiated under fluencies in the range of 0,5 10^{22}...2,5 10^{22} cm^{-2} and temperatures in the range of 500 ^{o}C...1000 ^{o}C.

The development of structural materials for reactor components was carried out at the Research Institute "Prometey"(S.Petersburg). Perlite type steel for reactor vessels is certificated and can be used up to 350°C. Material of steam generator tubes is stainless steel ЧС-33. The characteristics of the steel are validated at test facilities under temperatures up to 750 °C The most complex operating conditions are concerned to heat exchanger. Chrome-nickel alloy ЧС-57 is a material for heat exchanger tubes. Its strength is validated at temperature of 930°C during 10000 hours.

3. Assessement of status and prospects of HTGRs development in Russia.

The information presented shows that in Russia cosiderable experience of HTGRs development has been accumulated.

The long-term activity on the HTGRs projects led to apperance of cooperation between various organizations capable to carry out all necessary scope of research, development and design works. There are sufficient number of facilities for testing of HTGRs equipment, irradiation of fuel, graphite and structure materials.

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

What are prospects of HTGRs development in Russia under existing conditions? There is weighty background for the statement that HTGRs should play significant role in nuclear energetics of Russia. The objective need for HTGRs is clearly defined by their unique potential. Indeed, temperature level to 350°C is an area of LWRs application, temperature level to 550°C belongs to LMRs, upper level up to 1000°C is within the capacity only of HTGRs. Therefore, the task is efficiently to use the advantage of HTGRs. The following ways are preferable for that.

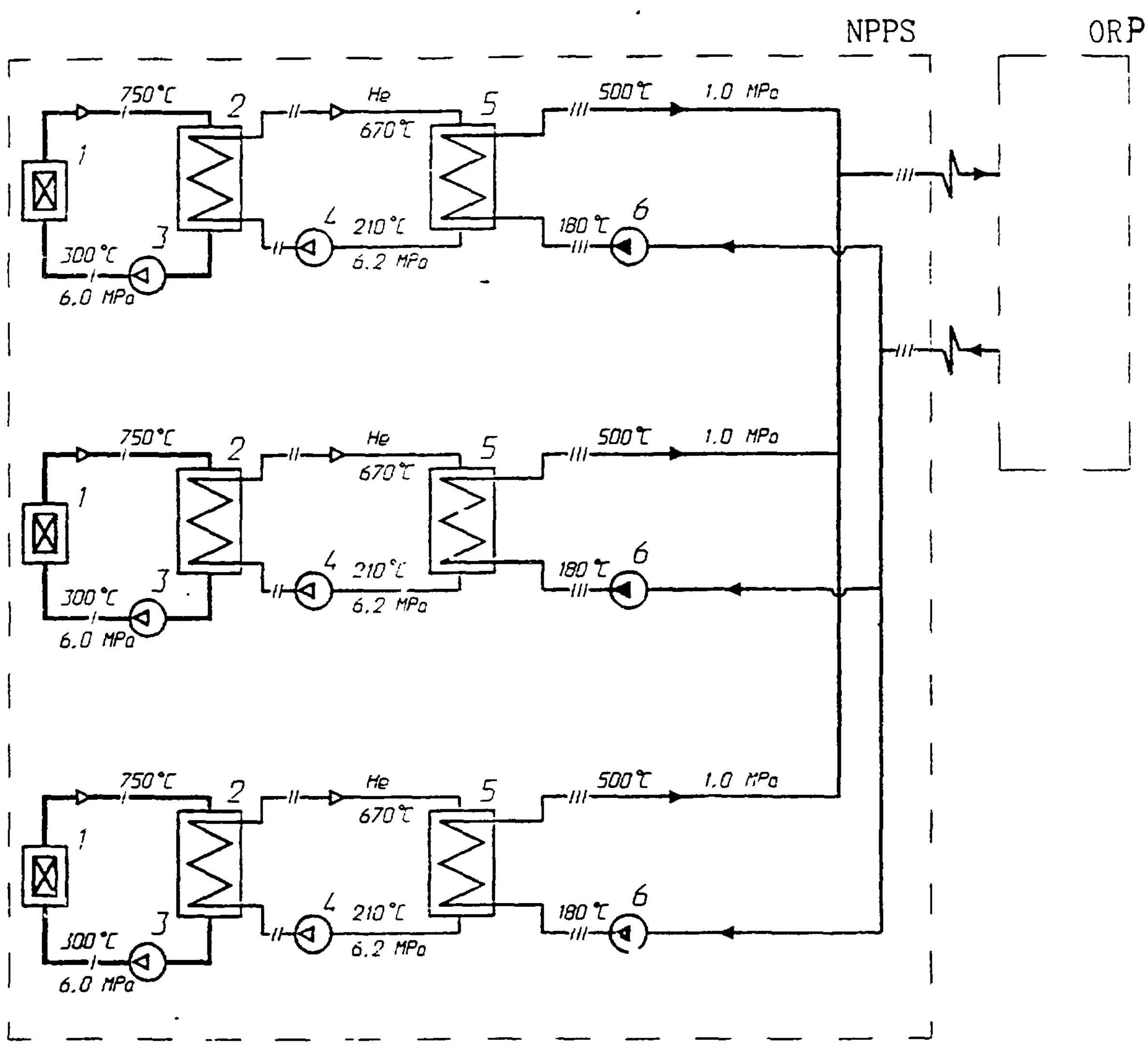
The first is production of high temperature heat for various industrial processes It is known well that consumption of fossil fuels for heating and industrial processes is more than for electricity production. As far as demand in energy on fossil fuels faces all growing ecological, reserve and transport problems, it is inevitable more broad application of nuclear plants as alternative sources of energy.

Basic consumption of heat in industry falls on temperatures up to 550-600^{o}C. This level can be provided by HTGRs with helium temperature of 750^{o}C. Almost all oil refineries and plants for production of petrol and diesel fuel from coal can be served by HTGRs 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 on additional unique feature of HTGRs which allows to orientate them towards producing process heat, 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). Moreover, unlike other types of reactors the residual heat is removed not only under loss of flow but if the coolant escaped from the primary circuit. In addition, maximum fuel temperature doesn't exceed design basis limit of 1600^{o}C, if nominal power of the reactor with pebble bed core is about 200 MW and with prismatic one about 600 MW. In this case the possibility of core meltdown and its relocation is completely excluded taking into account very high temperature of graphite sublimation (>3000^{o}C).

This enables to assure high level of safety and to locate HTGRs not far from process plants what is necessary from economic point of view.

If VG-400 and VGM projects were aimed at combined production of process heat and electricity, VGMP project was completely orientated towards production of process heat. VGMP was desined for a standard oil-refinery plant with taking account of the User's requirements. The feasibility study showed that VGMP can provide process heat supply for all basic regimes of the oil-refinery plant. The nuclear plant includes three VGMP reactors. The heat is transferred through intermediate circuits to a process one (Fig 4).



-I- - *Primary circuit*
-II- - *Intermediate circuit*
-III- - *Heating system*

Fig.4. NPPS flow scheme: 1 - reactor; 2 - intermediate heat exchanger; 3 - main gas blower; 4 - intermediate circuit gas blower; 5 - process heat exchanger; 6 - technologic. circulator

VGM reactor is a background for development of VGMP, what enables to use scientific and technical potential has been created and to elaborate the design if necessary, for short period.

The second way is use of HTGRs only for electricity production. It is to be noted however, that connection of HTGRs with conventional steam-water cycle doesn't correspond to their capabilities and can't provide competitiveness of HTGRs with other sources of energy. Realisation of HTGR advantages should look for a combination of HTGRs with new technologies. For that matter, a coupling of HTGRs with gas - turbine cycle is an advanced solution.

Thus, at present in Russia changes in strategy of HTGRs utilasation has formed, namely:

- for production of only process heat, using module type of HTGRs with thermal power of about 200 MW and pebble bed core, instead of combined function: high temperature part (700°C-950°C)for process heat, low temperature part (300°C-700°C) for electricity production through steam-water cycle;

- fór electricity production through gasturbine cycle using module type of HTGRs with thermal power of about 600MW and a core formed from prismatic blocks.

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

4. Activities related to HTGRs with gas turbine cycle.

Initial work on direct gas turbine cycle was performed at OKBM in early of 80-th as applied to VG 400 reactor. The layout and parameters of the reactor are given in Fig.5. The cavities of the reactor vessel are contained the core, two horizontal turbomachine, recuperators and coolers. Residual heat removal under normal and accident conditions is

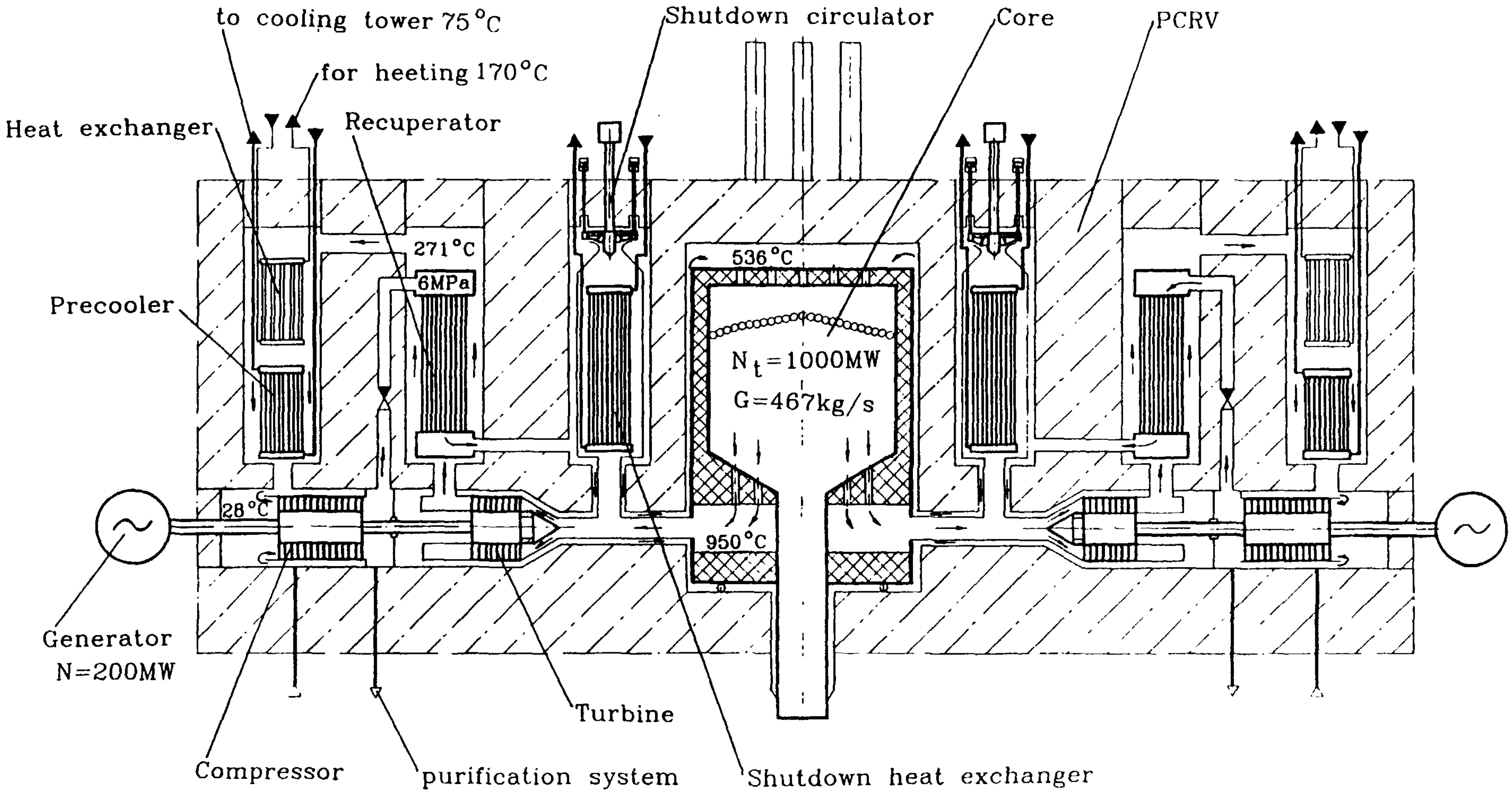


Fig.5 VG 400 with gas turbine cycle

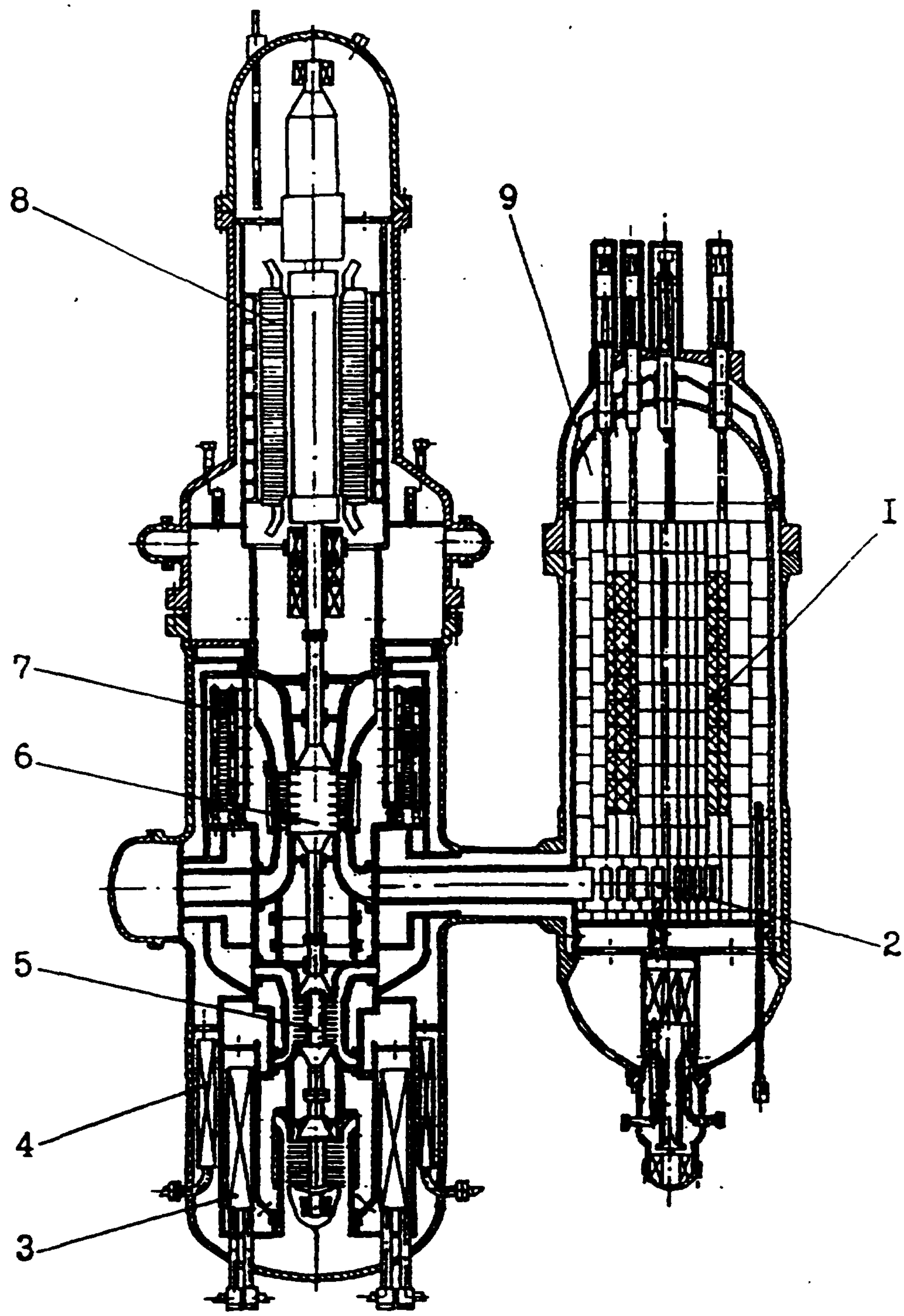


Fig. 6. GT–MHR reactor arrangement
1 – reactor core; 2 – core outlet plenum;
3 – intercooler; 4 – precooler;
5 – compressor; 6 – turbine;
7 – recuperator; 8 – generator;
9 – core inlet plenum.

provided by two independent loops with circulators and coolers. The works on the project couedn't be continued dne to interruption of designing VG 400 reactor.

GT-MHR project is a joint Russian-American design of the nuclear power plant with direct gas turbine cycle.

GT-MHR project is being developed by the group of Russian enterprises headed by OKBM (Nizhny Novgorod). General Atomics (San Diego) will be the leading firm from American side.

Russian GT-MHR project is based on the concept of General Atomics reactor[1].

GT-MHR plant is a coupling of passively safe modular reactor with helium coolant with up-to-date technology developments: large industrial gas turbines, large active magnetic bearings, compact highly-effective heat exchangers, high-strength high temperature steel alloy vessels.

Fig.6 shows the common general view of the reactor.
The three vessels are located in the underground silo.

The reactor vessel houses the annular reactor core, core supports, control rods drives, heat exchanger and gas circulator of the auxiliary loop. The reactor core contains hexagonal graphite fuel columns, which contain fuel (weapon- grade plutonium) encapsulated in ceramics coated particles.

The reactor vessel is surrounded by a reactor cavity cooling system, which provides totally passive decay heat removal. A separate cooling system provides decay heat removal for refuelling activities.

The power conversion vessel contains the turbomashine and three compact heat exchangers. The turbomashine consists of a generator, turbine and two compressor sections, mounted on a single shaft suspended by magnetic bearings.

GT-MHR plant has the following parameters:

Reactor power	550-600 MW(t)
Turbine inlet temperature	850°C
Reactor inlet temperature	490°C
Compressor inlet temperature	26°C
Turbine inlet pressure	7,02 MPa
Overall pressure ratio	2,86
Primary circuit pressure losses	7-8 %
By-pass helium flow rate for cooling and leaks in the sealings	2,5%

Recuperator efficiency	0,95
Generator rotational frequency	50 Hz
Turbine adiabatic efficiency	93%
Low pressure compressor adiabatic efficiency	88%
High pressure compressor adiabatic efficiency	87%
Generator efficiency	98,5%
Plant efficiency	47-48% net

At present the conceptual design of GT-MHR is being carrved out and capabilities for performing tests of fuel and main components are being analyzed.

5. CONLCUSIONS

5.1. Russian organizations accumulated significant experience associated with HTGRs development on the basis of VG400 and VGM projects. Experimental facilities for testing of main components of the reactors have been created.

At present the strategy of HTGRs utilasation has been changed. Module HTGRs with pebble bed core will be aimed only at process heat production, and with a core from prismatic blocks at electricity production through gas-turbine cycle. Ror this purpose a feasibility study of VGMP reactor has been perfomed and conceptual desidn of GT-MHR is under way.

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HIGH TEMPERATURE REACTOR DEVELOPMENT IN THE NETHERLANDS

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Abstract

This year, some clear design choices have been made in the WHITE Reactor development programme. The activities will be concentrated at the development of a small size pebble bed HTR for combined heat and power production with a closed cycle gas turbine. Objective of the development is threefold: 1. restoring social support, 2. commercial viability after market introduction, and 3. to make the market introduction itself feasible, i.e. limited development and first-of-a-kind costs. This design is based on the peu-à-peu design of KFA Jülich and will be optimized. The computer codes necessary for this are being prepared for this work. The dynamic neutronics code PANTHER is being coupled to the thermal hydraulics code THERMIX-DIREKT. For this reactor type, fuel temperatures are maximal in the scenario of depressurization with recriticality. Even for this scenario, fuel temperatures of the 20MWth PAP-GT do not exceed 1300°C, so there should be room for upscaling for economic reasons. On the other hand, it would be convenient to fuel the reactor batchwise instead of continuously, and the use of thorium could be required. These two features may lead to a larger temperature margin. The optimal design must unite these features in the best acceptable way.

To gain expertise in calculations on gas cooled graphite moderated reactors, benchmark calculations are being performed in parallel with international partners.

Parallel to this, special expertise is being built up on HTR fuel and HTR reactor vessels.

1. Introduction

R&D activities dedicated to the helium cooled graphite moderated high temperature reactor (HTR) started in 1993. In 1994 the name WHITE Reactor was given to the programme, standing for Widely applicable HIgh TEmperature Reactor. The programme is mainly executed by ECN, although three partners contribute in the framework of the Programme to Intensify the Nuclear Competence (PINK) of the Dutch Ministry of Economic Affairs. These are the Interfaculty Reactor Institute of Delft, the engineering company Stork NUCON and the utility research institute and engineering company KEMA. Parallel to this, an HTR Technology Assessment Study is being performed by ECN and the University of Utrecht. This study mainly investigates societal aspects of the technology.

In 1994, ECN hosted an IAEA Technical Committee Meeting on Development Status of Modular High Temperature Reactors and their Future Role, and the Dutch activities on HTR were presented on a Workshop organized on the occasion of the TCM [1].

2. Design requirements and basis

The objective of the WHITE Reactor programme is to contribute to HTR development for safe, environmentally friendly and economical energy supply. Three requirements are to be met for the HTR design:

1. restoring social support,
2. commercial viability after market introduction, and
3. to make the market introduction itself feasible, i.e. limited development and first-of-a-kind costs.

To comply with these requirements, some clear design choices have been made in the WHITE Reactor development programme this year. Design efforts are focused on simplicity. To restore public support for nuclear power, designs will have to be very transparant. No emergency core cooling systems, backup shutdown systems or containments should be needed. This limits the per level per unit, so modularization will be needed to comply with larger power level demands. More important, the low power level even has a positive effect on plant economics, because components can be omitted and the safety grade of certain components could be lower than usual for nuclear power plants. A small plant size will also benefit market introduction, because building a small unit can be an "adventure" of limited size. The activities will therefore be concentrated at the development of a small size pebble bed HTR for combined heat and power production with a closed cycle gas turbine. The power level will be no more than 100 MWth, to keep the design as simple as possible and to minimize development and prototype cost. Pebble bed fuel is chosen for two reasons: 1. the possibility of continuous fuelling, and therefore assure a limited overreactivity, and 2. less development work remains to be done on the pebble fuel type than on the prismatic fuel type. The combined heat-and-power application has been chosen because it fits the chosen power range and to cover a market segment additional to large scale base load electricity generation for the nuclear industry.
The design is based on the GHR-20 design by BBC/HRB [2] and on the peu-à-peu design of KFA Jülich [3]. This is an extremely simple pebble bed HTR design. The reactor is being fuelled on-line continuously or in small batches. After a period of several years, the reactor is unloaded off-line. Like all modular HTRs, the reactor employs no emergency core cooling system, since even after the worst core heatup accident (depressurization) the reactor is shut down by it's negative temperature coefficient and the fuel does not attain unacceptably high temperatures, offering ample time for an active shutdown by insertion of rods. But the shut-down by negative temperature coefficient is only temporary, because of cooling of the fuel and decay of xenon. To decrease the safety function of the shutdown system, the reactor is also required to cope with recriticality after core heatup. Calculations performed at KFA Jülich show this is possible with wide margins for 20 and 40 MWth peu-à-peu designs. It is intended to uprate the power level and to optimize the design.

3. Computer codes for core design

Although the pebble bed neutronics/thermal hydraulics code VSOP of KFA Jülich is available through the NEA Databank now, ECN decided to develop it's own code system. The dynamic neutronics code PANTHER has been acquired from AEA, UK. This code is being coupled to the thermal hydraulics code THERMIX-DIREKT, kindly delivered to ECN by KFA Jülich in a cooperation framework.
Much attention is being paid to the generation of nuclear data. For the PANTHER-THERMIX system, neutron cross sections are generated by WIMS-E and SCALE-4 codes. The Monte-Carlo code MCNP is used for this as well, and to check certain reactor calculations.

To gain expertise in calculations on gas cooled graphite moderated reactors, benchmark calculations are being performed with these codes in parallel with international partners on similar systems: the PROTEUS experiment at PSI Villigen, and the 450MWth MHTGR of General Atomics.

4. Fuel and pressure vessel

Parallel to this, special expertise is being built up on HTR fuel and HTR reactor vessels. Experiments are being performed on oxidation and interaction with fission products of SiC as well as the stability of UO_2/UC_2 mixtures. The pressure vessel stress assessment code ISAAC is being equipped with a fracture mechanics and creep assessment feature for high temperatures, to be able to design a licensable HTR pressure vessel in the near future.

5. Future plans

In 1996, static and dynamic calculations are being performed on the 20MWth PAP-GT of KFA Jülich and compared to their calculations with different codes. For this reactor type, fuel temperatures are maximal in the scenario of depressurization with recriticality. Even for this scenario, fuel temperatures do not exceed 1300°C, so there should be room for upscaling for economic reasons. On the other hand, it would be convenient to fuel the reactor batchwise instead of continuously, and the use of thorium could be required. These two features may lead to a larger temperature margin. The optimal design must unite these features in the best acceptable way.
An economic evaluation of the optimized design will be made. Attention will be paid to the possibility of omitting components and decreasing of the safety grade of components.

5. Conclusion

After a familiarization phase, the Dutch HTR program is well under way. First decisions have been made on design features for the HTR conceptual design in The Netherlands. The activities will be concentrated at the development of a small size pebble bed HTR for combined heat and power production with a closed cycle gas turbine. An independent neutronics and thermal hydraulics code system is being developed. An optimized design based on the GHR-20 and the PAP-GT is planned for 1996.

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FRENCH ACTIVITIES ON GAS COOLED REACTORS

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Abstract

The gas cooled reactor programme in France originally consisted of eight Natural Uranium Graphite Gas Cooled Reactors (UNGG). These eight units, which are now permanently shutdown, represented a combined net electrical power of 2,375 MW and a total operational history of 163 years. Studies related to these reactors concern monitoring and dismantling of decommissioned facilities, including the development of methods for dismantling. France has been monitoring the development of HTRs throughout the world since 1979, when it halted its own HTR R&D programme. France actively participates in three CRPs set up by the IAEA.

1 - NATURAL URANIUM - GRAPHITE-GAS COOLED REACTORS

France has built and operated 8 Natural Uranium-Graphite-Gas cooled Reactors (UNGG) and exportated one in Spain which was operated by a French-Spanish company (Hifrensa).
The two first one, Marcoule G2 and G3, was operated by CEA and later by COGEMA. The other, Chinon 1, 2 and 3, Saint Laurent 1 and 2, Bugey 1, was operated by EDF. Today all of them are shutdown. The table 1 gives data for each of them.

The studies related to these reactors concern monitoring and dismantling of decommissioned facilities.
The CHINON A1 reactor was converted into a nuclear museum which is very much visited. The other gas cooled reactors are being dismantled. The MARCOULE G2 and G3 and CHINON A2 reactors have been dismantled to level 2. CHINON A3 have been dismantled to level 1 and is waiting for administrative authorization to undertake works to reach level 2. St-LAURENT A1 and A2 and BUGEY 1 reactors are in the stage of "definitive stop phase". That means all nuclear fuels have been discharged and transfered to the reprocessing plant.
A study is actualy running for a possible decision concerning an immediate dismantlement at level 3 for MARCOULE G2 and G3 reactors. The question is economical and technical (what to do with the graphite and how). For the EDF reactors, this level 3 is scheduled about 50 years after level 2 dismantling. Special attention is given to structure activation studies and evaluation of dose rates at a time when the technical support teams, in particular the neutronics teams, and the operating teams are still at the site or operational. It is effectively of first importance to prepare carefully files required fo dismantling which will occur after a such long time.

Other studies are being conducted to develop methods for dismantling problem - raising structures such as those comprising irradiated steels and graphites. An arc furnace at Marcoule made an industrial demonstration on melting more than 5000 tons of slightly contaminated steel from CO_2 primary circuit. The benefit of this melting is evident concerning the volume of wastes, but also on the contamination level. Only cobalt subsists on the ingots which are use to manufacture waste containers. It have a capacity of 15 tons with possible production of 10,000 tones per year, for size pieces such as diameter 2.2 m, height 1.3 m. This furnace is actualy used for G2-G3 steam generator melting. A fluidized graphite incineration pilot built by FRAMATOME has a capacity of 40 kg/h, whereas the dermo laser system, installed at Marseille in collaboration with the CEA, is currently capable of incinerating 35 kg of graphite per hour.

2. HIGH TEMPERATURE REACTORS

France has been monitoring the development of High Temperature Reactors (HTR) throughout the world since 1979, when it halted its own HTR research and development program. The trend toward low power reactors with specific properties enabling them to easily satisfy tighter safety requirements has led to studies assessing this concept. These projects fall under the scope of a wider program on future reactors. The results of french studies are in good agrement with US and German studies on modular reactors for normal and accidental conditions.

France's interest is demonstrated by its participation in three CRPs set up by IAEA :

- "Validation of safety related physics calculations in low enriched gas-cooled reactors" in which a french expert was involved for the Proteus experimentations.

- "Heat transport and after heat removual for gas-cooled reactors under accident conditions". The CEA is involved in a benchmark which is a code to experiment comparaison between JAERI HTTR experiment and french calculations - These calculations were made with a finite elements method 3D code (TRIO-EF) which is a general code. With it, it is possible to have conduction, convection and radiation heat transfert coupled.
The results of calculation are in good agreement with HTTR experiment (scale 1/4).

- "Experimental work on validation of prodictive methods for fuel and fission product behaviour is gas-cooled reactors". In this area, the CEA undertook an experimental study in a research reactor (SILOE at Grenoble Centre) in the COMEDIE loop on behalf of the DOE. The deposit of fission products generated by particules that were deliberately not tight was studied, along with their migration during depressurization at various level. The test was performed sucessfully in the last quarter of 1992. Until today, the results were not published. As we have got DOE's agreement to release this information, you will be the first to hear a piece of new on this subject. A paper signed by GA, ORNL and CEA will be presented during this IAEA meeting.

In addition, France examines the possibility of participation in a fourth CRP on "Design and Evaluation of Heat Utilisation systems for the HTTR". The final decision is not yet taken.

TABLE 1.

REACTOR	NET ELECTRICAL POWER	DATE OF GRID CONNEXION	DATE OF SHUTDOWN	SERVICE LIFE	REASON FOR SHUTDOWN
MARCOULE G2	40	22/4/59	2/2/80	21	Technical : graphite expansion
MARCOULE G3	40	4/4/60	28/6/84	24	Technical : steel embrittlement
CHINON A1	70	14/6/63	16/4/73	10	Economic
CHINON A 2	210	24/2/65	14/6/85	20	Economic
CHINON A3	480	4/8/66	15/6/90	24	Economic
ST-LAURENT A1	480	14/3/69	18/4/90	21	Economic
ST-LAURENT A2	515	9/8/71	27/5/92	21	Economic
BUGEY 1	540	15/4/72	27/5/94	22	Economic
VANDELLOS 1	480	6/5/72	19/10/89	17	Accident : alternator fire
TOTAL	2855			180	

STATUS OF GT-MHR WITH EMPHASIS ON THE POWER CONVERSION SYSTEM

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Abstract

The conceptual design of the Gas Turbine-Modular Helium Reactor (GT-MHR) has made significant progress in the past year. Evaluation of an external versus internal (submerged) generator and modifications as a result of an internal seal task force were completed. Significant progress was also made on the design of the generator utilizing existing technology. Conceptual design of the turbocompressor was confirmed, including extensive evaluation of the entire turbomachine (turbocompressor and generator) rotor dynamics. Results concluded in a revised configuration for the location of magnetic bearings supporting the entire machine. Integration of the turbomachine with the recuperator, precooler, intercooler and internal ducts and seals progressed to improve maintenance and operation. This resulted in some changes and improvements in the overall arrangement of the power conversion module. The paper also provides a summary of the fuel and safety assessment progress.

1. Introduction

At the IAEA Technical Committee Meeting on Development Status of Modular High Temperature Reactors and Their Future Role held in November, 1994 at ECN, Petten, The Netherlands, the GT-MHR's design status and technical issues were reported. At that time the decision to focus the U.S. program on the direct cycle gas turbine power conversion had been made, the power level of the module had been selected, and design work was identifying the technical issues with the integrated power conversion module. AlliedSignal Aerospace (Tempe, Arizona, USA) had been selected as the turbomachine vendor and presented the paper (Ref. 1) discussing the power conversion system design status and technical issues.

In the past year AlliedSignal selected GEC-Alsthom (Belfort, France) to work on the generator and Mechanical Technology Inc. (Latham, New York, USA) to work on the magnetic and auxiliary catcher bearings. General Atomics, the overall system designer, has integrated this turbomachine team with the AlliedSignal (Torrance, California, USA) efforts on the helium recuperator and the ABB-CE (Windsor, Connecticut and Chattanooga, Tennessee, USA) work on the precooler, intercooler and pressure vessel. Significant progress on the conceptual design of the Power Conversion System (PCS) has been accomplished.

This paper updates the design status of the GT-MHR with emphasis on the PCS. Section 2 reports on the trade study that addressed the question of whether the generator should be submerged in helium within the power conversion vessel or be external to the vessel necessitating a rotating shaft seal. The significant efforts by GEC-Alsthom to utilize their conventional hydrogen-cooled generators for this application also are summarized. Section 3 addresses the coupling of the generator to the turbocompressor and the concomitant changes in the bearing locations. The latest analyses by MTI on the rotor dynamics are presented. Section 4 summarizes the findings of a task force comprised of PCS team members that addressed the demanding requirements for the sliding seals between PCS components to limit helium leakage. Section 5 captures improvements in other power conversion areas such as the precooler and intercooler configuration and the power conversion vessel layout. Section 6 briefly summarizes the status in other areas of the GT-MHR design development including an independent review by EPRI and its member utilities. Finally Section 7 presents the conclusions.

2. Design Progress on the Generator

The generator had received relatively little attention prior to GEC-Alsthom joining the team. Cursory studies by General Electric Schenectady had indicated that the initial concept proposed by GA of a top-mounted, vertical generator submerged in helium, although non-conventional, was feasible (Ref. 2). Early concerns involved the development needs of a large, high speed generator immersed in helium within a pressure vessel versus a more conventional generator located outside the vessel. To address these concerns AlliedSignal took the lead on a trade study to investigate the advantages and disadvantages of an external generator with a rotating shaft seal versus the existing submerged design.

The trade study defined requirements for a rotating shaft seal between the turbocompressor and the generator external pressure vessel. For this configuration the seal leakage must be held to stringent limits since the leakage consists of primary coolant helium with potential radioactive contamination. It was assumed that rotating shaft seal systems were not restricted to fit in the confines of the existing pressure vessel configuration, that is, that the pressure vessel could be altered to accommodate any desired seal configuration. The trade study was initiated by requesting seal vendors to establish the feasibility of a low leakage helium retaining shaft seal.

Both single shaft seals and buffered shaft seal systems were examined. The single seals included labyrinths, dry gas face seals, and dry gas ring seals. The buffered seal systems included combinations of labyrinth seals and either dry gas seals or wet (oil or water) gas seals. It was found that a labyrinth seal by itself would not meet the leakage requirements. Dry gas seals, either alone or in combination with a labyrinth, would entail considerable development effort and risk, since existing designs have a much smaller diameter than that required and do not meet the life requirement. A combination of a wet gas seal with a labyrinth would entail the risk of oil (or other liquid) getting into the primary system, and entail new helium-oil separation systems and possible mixed waste disposal needs. The study concluded that the problems associated with a rotating shaft seal are significant and would necessitate a major extension of existing technology to develop a suitable seal with the required diameter, life, and low leakage necessary for the GT-MHR application.

At this point GEC-Alsthom came on board and aggressively addressed the concerns with a submerged design. They indicated that they do not expect significant problems operating vertically in helium and results from planned tests should confirm their expectations. As the second largest generator manufacturer in the world they modified one of their standard hydrogen cooled generators to operate vertically in helium with magnetic bearings. The key generator requirements and their bases are provided in Table 1.

The resulting generator is a two-pole, 3600 rpm synchronous unit rated at 286 MWe. This selection has two major advantages: (1) permits a direct turbocompressor-to-generator coupling (i.e., freedom from the use of a gearbox or frequency converter), and (2) facilitates the use of the generator as a motor during plant startup. A brushless exciter system with a shaft-mounted exciter alternator and shaft-mounted diodes is used for supplying and controlling the dc field current in the generator rotor. The overall generator conventional frame assembly (which can be removed and replaced as a unit) consists of the generator stator and rotor, exciter stator and rotor, magnetic and auxiliary catcher bearings, casing, and four water coolers. Heat is removed from the electrical equipment by means of conventional heat exchangers. Helium replaces the hydrogen coolant used in conventional generators of this size and is an excellent heat transfer media. Axial flow fans are mounted on the rotor and these circulate helium throughout the generator. Table 2 summarizes

Table 1
Generator Key Requirements and Their Bases

REQUIREMENT	BASIS
1. Net power output (net power): 286 MWe	February 94 overall system point design.
2. Efficiency: Over 98%	Energy balance at the February 94 point design.
3. Power factor: 0.85	Meets American and international market require-ments for grid stability.
4. Output frequency: 60 Hz	American market requirements. For 50 Hz markets, only minor changes are required in generator length and diameter, which can be easily accommodated in existing vessel envelope.
5. Operate as motor during startup, shutdown and reactor refueling	Provide helium circulation without need for nuclear power.
6. Vertical orientation	Compatible witih turbocompressor orientation.
7. Magnetic bearings	Eliminates possibility of oil contamination. Compatible with turbocompressor bearings.
8. Submerged	Eliminates need for rotating shaft seal. Helium provides for excellent heat transfer.
9. Two poles	Two poles allow higher speed (3600 rpm) operation than four poles (1800 rpm). Compatible with GT-MHR turbine speed.
10. 130% design overspeed	Maintain structural integrity in design basic regime.

Table 2
Generator Design Features and Their Bases

DESIGN FEATURE	BASIS
1. Synchronous	Most efficient, most compact.
2. Solid, single forging rotor ("Turborotor")	Turborotor is now the standard design in the specified power range. Turborotor design is stronger than salient pole design and generates lower windage losses.
3. Brushless excitation	Eliminates need for slip rings in the generator cavity. Slip rings can cause reliability problems.
4. Asychronous exciter: Alternating current is used in the stationary exciter armature to induce alter-nating current (AC) in the exciter rotor. It is then rectified in the rotor to feed the generator field winding.	Provides excitation for the generator over the entire speed range, including startup zero speed condition, without brushes.
5. Armature voltage as a function of helium pressure in generator cavity For power operation, 19kV at ≥10 bar For startup or shutdown, 4.2kV at 2 bar For refueling, 1kV at 1 bar	Voltage levels are dictated by electro-magnetic design, are compatible with dielectric properties of helium, and are compatible with GEC/Alsthom technology.
6. Helium to water coolers	Conventional hydrogen design modified for helium.
7. Radial structural supports take stator reactions through flexible suspension	Conventional design.
8. Axial structural supports mounted on turbocompressor	Facilitates alignment.
9. Startup driven by a static frequency converter up to the moment when turbine torque becomes positive	This startup sequence is very close to GEC/Alsthom practice on large gas turbine applications.
10. Layout includes conventional frame, cooling sys-tem, electromagnetic circuit, windings, and flange coupling	Conventional design selections maximize the use of proven technology.

these and other design features of the selected generator and their bases. Figure 1 provides a schematic of the generator within the top section of the power conversion vessel. Except for orientation and a few modifications for use in helium and with magnetic bearings, the GT-MHR generator is almost identical to a standard GEC-Alsthom generator. This can be seen from Figure 2 where the changes to the horizontal hydrogen cooled generator for this application are highlighted.

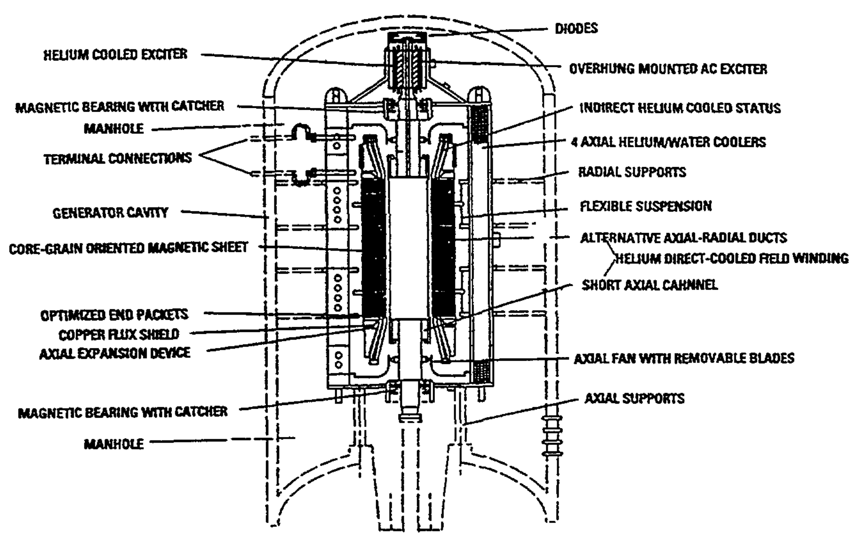

Figure 1. Schematic of GT-MHR Generator

In summary, the generator configuration has received concentrated attention to confirm the initial design approach. A conventional generator has been employed to limit the development and testing needs to a few areas. By utilizing the traditional generator unit within a packaged frame concerns regarding maintenance have been lessened. The vendor is supportive of the maintenance approach that has the generator removed from the vessel periodically (approximately every seven years when the turbocompressor is removed for maintenance) and accessed for visual and hands-on inspection. Startup and operating conditions have been specified to limit the voltage at low helium pressure and dielectric strength. Tests are planned to confirm the behavior of the insulation at low helium pressures and during rapid depressurization from high pressure. A scale-up of existing power penetration technology to higher pressure will be required.

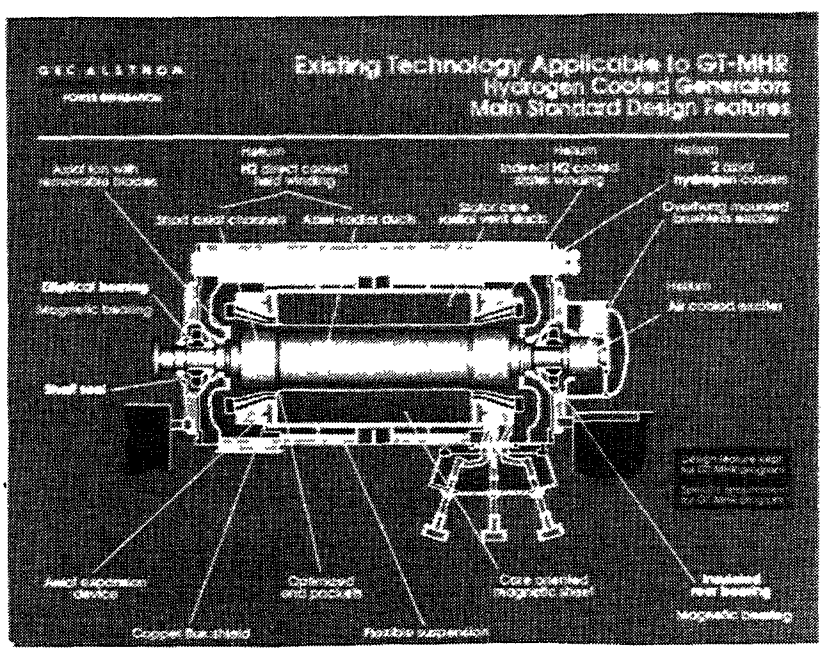

Figure 2. Existing Technology Applicable to GT-MHR

3. Design Progress on Turbocompressor

As opposed to the generator, design progress on the turbocompressor has not resulted in large visible changes to the layout. There has been no change in the arrangement of the turbine relative to the two compressors. Rather effort has focused on changes introduced by adopting a conventional generator and changes resulting from progress on two key technical issues previously identified: rotor dynamics and differential thermal expansion.

The initial turbomachine design which was premised on very limited access to the generator cavity utilized a curvic coupling between the generator and the turbocompressor. The coupling was located in the generator cavity which is separated from the hotter and radioactive area of the turbocompressor by a labyrinth seal. A tie rod to the coupling extended from the top of the generator through the center of the generator rotor. Therefore, in this design access to the coupling was not required. However, the tie bar introduced complex interactions with electronic leads to the exciter, tensioning, and vibration. With the decision to utilize a conventional generator, a conventional coupling was adopted requiring local access into the generator cavity. Providing adequate access was in itself highly desirable and consistent with standard practice by GEC-Alsthom. Continuing with the approach to allow access to the generator cavity, the next step involved relocating the axial thrust bearing from the top of the generator to just below the coupling. With the increased span between the generator and the turbocompresor for access to the coupling, a second radial bearing was added so that there is one on each side of the coupling.

Rotor dynamics and differential thermal expansion were addressed first by observing that the area with the greatest temperature differences was between the hot turbine and the cold high pressure compressor where a radial bearing was cooled. One of the first tasks tackled by MTI was to see if acceptable rotor dynamics could be shown if the bearings in the generator cavity were changed as described above and if this turbine-compressor bearing could be removed.

Figure 3 provides a schematic comparing the new arrangement of the coupling and bearings with the previously reported configuration. In both cases the rotor dynamics challenge involves the relatively heavy generator rotor with an overhung exciter above and a long slender turbo-compressor rotor below. The type of analysis that MTI conducted to determine the rotor bearing characteristics included: critical speed analysis, unbalance response analysis and stability. In order to assist in determining the source of the resonant frequencies, the generator-turbine-rotor train was analyzed for critical speeds and mode shapes by separating the rotating elements. This was accomplished by conducting an independent critical speed analysis of the generator, the turbine, the high pressure compressor, the low pressure compressor, and the combined turbine and high pressure compressor. Given the mode shapes from the individual components, worst case unbalance forces were applied in the generator, at the coupling, and at the low pressure compressor.

The following conclusions were drawn from the rotor dynamics analysis.

- The revised arrangement is a significant improvement over the prior concept.
- The preferred approach to the bearing design is to use highly damped, low stiffness bearings. This significantly eases requirements for support structures.
- Two resonant modes of the bearing-rotor system were encountered in relatively close proximity to the operating speed. The modes are attributed to the generator rotor at the exciter.
- The magnetic bearing near the turbine inlet hot zone was eliminated.
- Unbalance response analysis indicated that shaft amplitudes can be accurately controlled over the operating speed range.

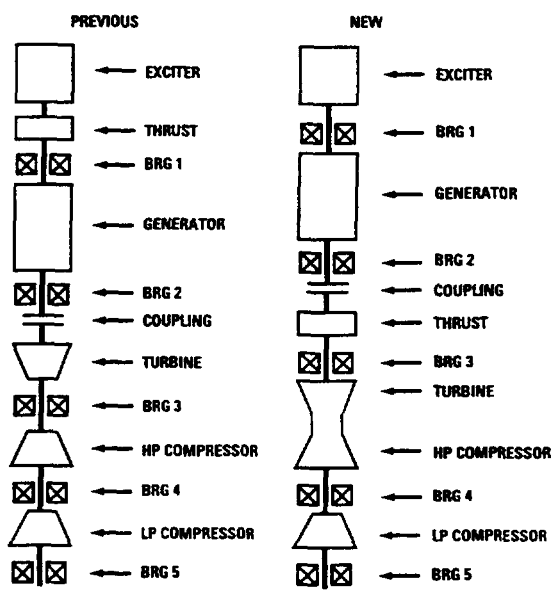

Figure 3. Comparison of Rotor Bearing Schematics

Based on these results considerable progress has been made in understanding the turbomachine rotor dynamics. Furthermore, programmable control of the magnetic bearings has been utilized to provide soft bearings with large damping that is well suited to the vertical rotor configuration. An improved configuration with less differential thermal expansion has utilized a conventional coupling. Future rotor dynamics work will be focused on the possible addition of a magnetic damper bearing at the overhung exciter. A further improvement to reduce the number of bearings is the possible removal of the bearing between the compressors.

4. Design Progress on the Sliding Seals

Seals are extensively used within the PCS at component interfaces to maintain the integrity of the helium flow path and associated operating conditions. These conditions lead to significant pressure and temperature differentials across the seals, as well as to substantial differential motions of the components which each side of the seals contacts.

The initial seal leakage estimates assumed a uniform leakage gap around the perimeter of each seal. The gap was thought to be difficult to attain but was important to plant performance. Much of the leakage was initially anticipated to be associated with the five horizontal turbocompressor seals and the vertical face seal at the turbine inlet. However, seal associated with the recuperator were found to be much more complex and without suitable provision for removal and replacement. The recuperator seal complexity resulted from the use of a single manifold to supply inlet flow to all six recuperator modules. This necessitated baffle plates above, below, and between modules to separate inlet flow from the outlet flow; and a system of interconnecting linear seals integrated with the baffle plates. The precooler and intercooler seals were also found to have significant complexities. The intercooler seal complexity resulted from large diameter duct outlet seals which were not visible during initial installation and were not accessible for removal and replacement, as well as multiple water conduit leaf spring seals, omega seals, and bellows that were not accessible for removal and replacement.

A seals task force comprised of PCS team members was established to address these problems with the objective of developing viable seal concepts that would resolve the initial seal uncertainties. The scope of the task force included those seals which were representative of the most difficult seals to implement within the PCS. The seals addressed by the task force are shown in Figure 4. As a consequence of the activities of the task force which included input from seal vendors, major changes were made in the arrangements of the seals and oftentimes the associated components. These changes are summarized in Table 3.

The recuperator was reconfigured to include a dedicated manifold for each module that is attached to the module during fabrication. A comparison of the recommended design with dedicated manifolds for

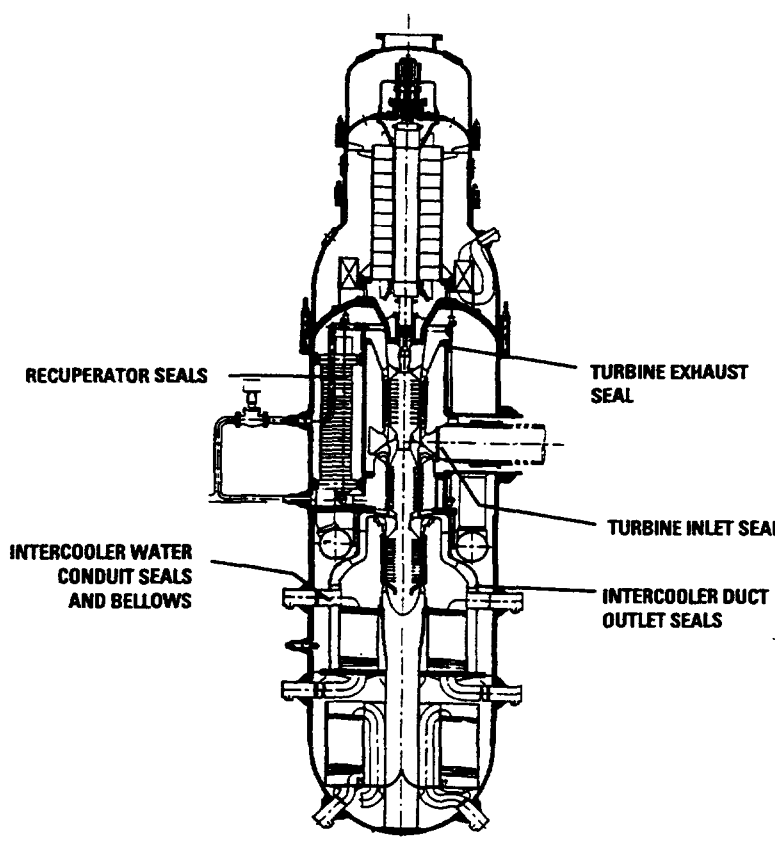

Figure 4. Seals Addressed by Task Force

Table 3
Results of PCS Seals Task Force

LOCATION	INITIAL CONCEPT	TASK FORCE RECOMMENDED CONCEPT	ADVANTAGES
Recuperator LP inlet	Common manifold for all modules. Baffle plates above, below and between modules. Linear seals defining a multiple sided polygonal cylinder.	One manifold per module. Single horizontal top plate above modules. Segmented cylindrical piston rings between module and top plate, and at I.D. at top plate.	Eliminates linear seals and baffle plates between modules Uses same type seal as turbine exhaust
Turbine exhaust (representative of all horizontal turbocompressor seals).	E-seal	Segmented cylindrical piston ring.	Available in large diameters Accommodates relative motions and out of roundness Low development risk
Turbine inlet	C-face seal with pneumatic actuated cylindrical piece at end of hot duct.	Two horizontal and one vertical segmented cylindrical piston ring seals at ends of new tee-shaped adapter.	Eliminates face seal and bellows Accommodates relative motion Uses same type seal as turbine exhaust
Intercooler duct outlet	Two large diameter inaccessible seals near duct inlet requiring blind installation at shrouds defining boundary between intercooler inlet and outlet, and between intercooler outlet and precooler inlet.	Single segmented cylindrical piston ring seal at duct outlet interfacing with turbocompressor. Seal added to turbocompressor seals, similar to exhaust seal and accessible through turbomachine cavity.	Reduced outer seal diameter Inner seal eliminated Seal accessed through turbo-machine cavity Uses same type of seal as turbine exhaust
Intercooler water conduit outlet	Leaf seal at inner duct outlet shroud. Welded omega seal at outer duct outlet shroud. Bellows between outer shroud and tube sheet.	Elimination of all seals through re-routing of water piping.	All seals and bellows eliminated

each recuperated module to the previous single common manifold design is shown in Figure 5. Based on analyses the dedicated manifold is not expected to impact recuperator effectiveness, however this will be confirmed in planned flow distribution tests. The initial baffle plates were eliminated and replaced with a single horizontal top plate. The initial network of linear seals was also eliminated and replaced with segmented piston ring seals at the top plate. The piston ring seals are the same type that are proposed for use at the turbine exhaust and turbine inlet.

The turbine exhaust E seal was replaced with a segmented cylindrical piston ring seal, which became typical of all other seals. The turbine vertical inlet face seal was replaced with two horizontal segmented cylindrical piston ring seals between the turbocompressor and a new adapter. These seals, located above and below the hot duct, are the same type as the turbine exhaust seal. The tee-shaped adapter also interfaces to the hot duct with a segmented piston ring seal that replaces the hot duct bellows that was previously used to accommodate thermal expansion.

The intercooler duct outlet was designed to utilize a single seal at the end of the duct as a replacement for the two seals near the beginning of the duct. The new single seal interfaced with the turbocompressor, and therefore became another turbocompressor seal, which could be a segmented piston ring similar to the turbine exhaust seal. It also had the accessibility features of

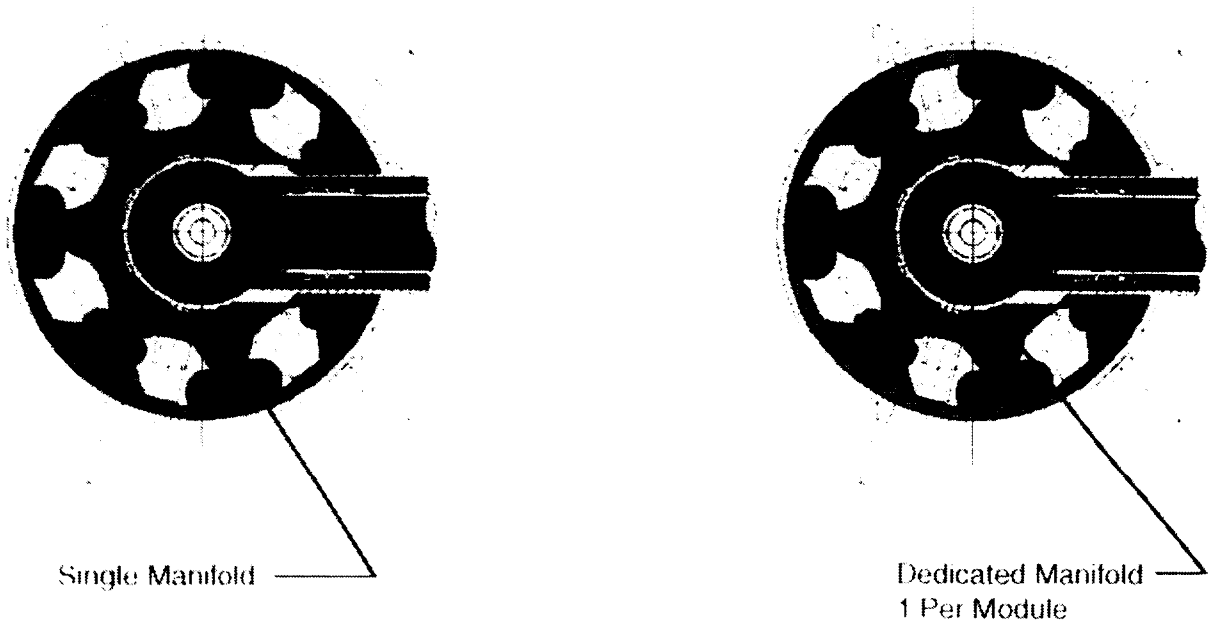

Figure 5. Comparison of Initial Recuperator Manifold with Recommended Design Incorporating Six Dedicated Manifolds (One per Module)

the turbine exhaust seal. An arrangement of the intercooler was also developed which eliminated the water conduit seals and bellows in their entirety.

With the above modifications, the complexity of many different types of unique seals was replaced with one type of accessible seal that has a proven history of use. This is a segmented cylindrical piston ring seal which incorporates both circumferential (or radial) and axial springs to maintain contact between the seal and its mating surfaces to attain a low leakage rate at low differential pressures, and utilizes the pressure differences to further reduce the leakage rate at higher differential pressures. This type of seal is shown in Figure 6, and is available from at least two US sources: Stein Seals and Cook Airtomic.

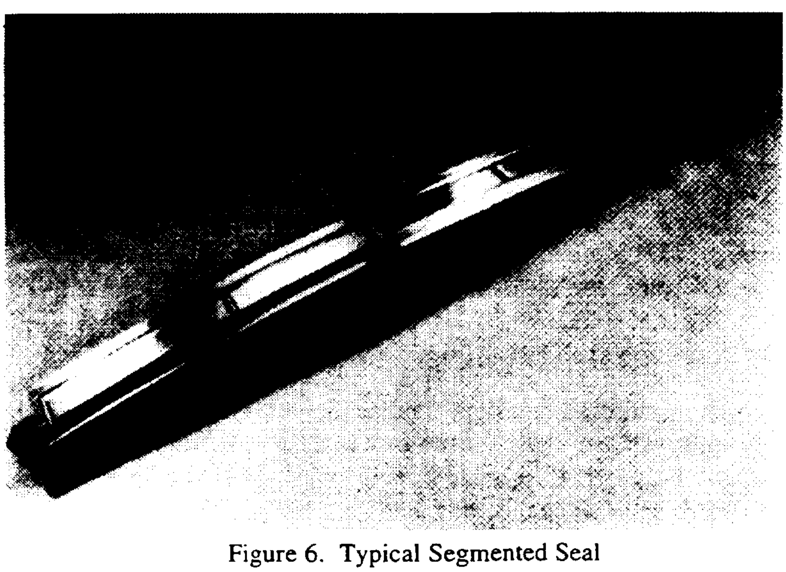

Figure 6. Typical Segmented Seal

With the adoption of the seal concepts proposed by the seals task force, all seals are accessible for removal and replacement when the turbomachine is removed. The seals will utilize a softer material than those interfacing surfaces which are not intended to be replaced over their 60 year design life. The hardness of these latter surfaces will be attained with suitable coatings, of which chromium-carbide is one candidate. A second sealing location on these surfaces is incorporated in the design as a backup should the first location become inadvertently marred. This second surface would be used only after a visual and/or optical examination of the first surface (via the turbomachine or vessel access penetrations) indicated such a need. Provision for removal and/or resurfacing of the "60 year" surface is also provided should the need arise.

Although the task force changes have led to viable seal concepts which are expected to meet the design goals, outstanding issues remain. These will be resolved through further design and development. Testing will be conducted to establish and/or confirm the adequacy of seal and coating materials; to confirm seal design life and leakage in a helium environment under design conditions; and to confirm the effectiveness of the recuperator with the new seal and manifold configuration. The integration of the turbomachine seals with any flanges that are needed for turbomachine assembly will be resolved through further design effort. The actual design of the seals will be performed and proven in confirmatory tests before the seals are incorporated into a prototype PCS, where their performance will ultimately be demonstrated. The methods of detecting excessive inservice seal leakage are being developed.

5. Design Progress of Other Power Conversion Components

Efforts in the bottom half of the power conversion vessel were focused on volume reduction and simplification. A key decision emanating from the seals task force was to route the intercooler water outlet circuit from the top of the intercooler inward radially to the center and then down to the bottom of the intercooler before exiting radially through the power conversion vessel. In this way seals were eliminated at the intersection of the helium shroud and the water conduits in the previous design that exited radially at the top of the intercooler. Prior to this decision some thought had been given to moving the precooler up to the same elevation as the intercooler (a side-by-side concept) to save on vessel length and silo depth. While those benefits had some appeal, the elimination of helium shroud-water conduit seals was judged of greater importance. In addition since the precooler and intercooler helium are at significantly different pressures (due to the low pressure compressor), the supports are of much different thicknesses which added complexity if both were in the same plane at the same elevation. Finally, maintaining the intercooler above the precooler (in an in-line configuration) but with the water outlets routed back down the center provides very similar designs for both finned tube helical coil water heat exchangers.

Vessel length and silo depth were saved by changing the power conversion vessel's bottom head from a spherical design to an elliptic one. This also allowed the precooler to be moved out radially which in turn decreased the precooler's height. The vessel and cooler improvements are best demonstrated in Figure 7 by comparing the overall power conversion modules layout from the 1994 IAEA report and as integrated this year from each team members

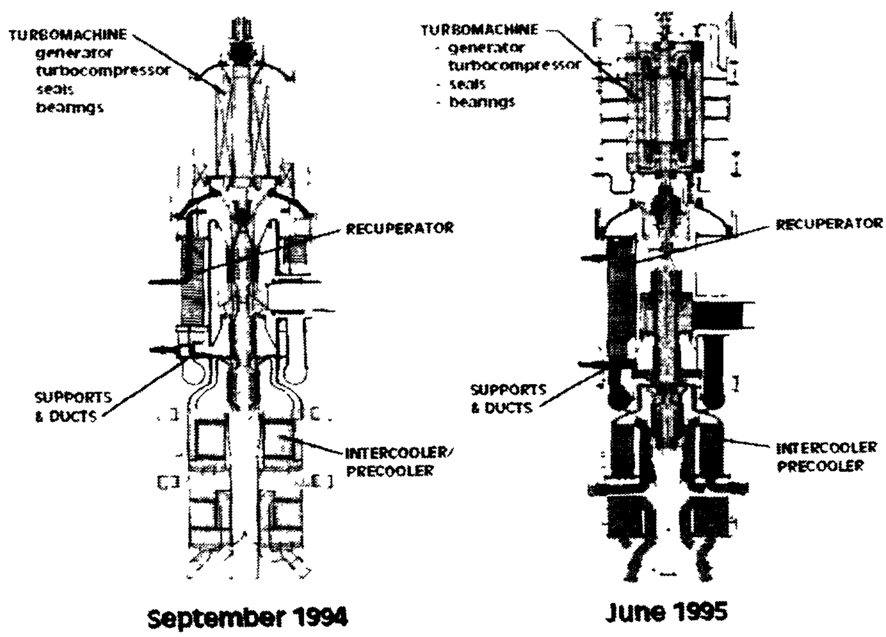

Figure 7. Comparison of Power Conversion Modules

electronic files. Note also the overall increase in vessel height and the change in shape at the top due to the improved generator definition and provisions for man access.

6. Overall GT-MHR Design Status

As described above a major focus of the GT-MHR design and development in the last year was the PCS. However, progress was made in a number of other areas.

Fuel. A plan was formulated to utilize the proven TRISO coating designs developed and demonstrated in both US and Germany prior to the 1987 unsuccessful introduction of coating design and process changes. In the past year effort has been focused on improving the compacting process relative to FSV experience so that particle SiC damage, both of a mechanical and of a chemical nature, is reduced. The SiC and contamination specification of 6×10^{-5} particle defect fraction in compacts was achieved by improvements in controlling particle forces during the compact matrix injection, reducing the matrix impurities and better conrols in the heat treatment process. The next capsule irradiation to confirm compact performance, MHR-1, was planned and designed. Demonstration compacts that met the spec were manufactured and are available for irradiation. The next step in the planned program is to modify the existing full size coater at GA and/or install a German coater recently acquired from HOBEG. Kernels, manufactured at Babcock & Wilcox, would be coated using the modified (or acquired) coater, formed into compacts using the improved process, and will be irradiated in a second capsule, MHR-2, to demonstrate the complete fuel system. Adopting this approach to use existing technology sigificantly reduces risk and development costs.

Spent Fuel Disposal. Options for GT-MHR spent fuel disposal were evaluated. Whole graphite fuel element disposal was preferred over separation of the fuel compacts from the graphite element because it involves less handling, the waste form is well suited for permanent disposal, and the greater volume increases diversion resistance. Significantly it was found that the whole element disposal option does not adversely impact repository land requirements because they are determined by decay heat loads. Transportation of the whole elements can use a standard LWR shipping cask. The SiC coated fuel provides excellent long term retention. It was concluded that the GT-MHR with its TRISO coated fuel in prismatic graphite fuel elements is well suited for a once-through fuel cycle with no additional processing of spent fuel.

Plant Transient and Safety Assessments. The startup strategy reported last year in Ref. 3 was confirmed with input from the generator vendor. The generator is initially used as a motor to provide mechanical shaft power for the turbocompressor. The generator cavity pressure is controlled to a high enough level for a given generator voltage to maintain an acceptable helium dielectric strength. The pressure is gradually increased so that a modestly sized static frequency converter can be used for motoring the generator. As the pressure increases, successively higher generator voltage levels are achievable until synchronization with the grid at 3600 rpm and 19 kV.

More detailed steady state heat balances have resulted in power conversion component requirements that include margins for design evolution. Component designers have also received transient requirements for a number of planned and unplanned events, such as load following, reactor trip, and loss of electric load.

Accident evaluations specific to the GT-MHR confirmed that the passive safety characteristics of the previous steam cycle modular high temperature gas-cooled reactor designs were maintained. Events initiated by one or more turbine blade failures were assessed. It was found that the resulting differential pressure forces across the prismatic core did not exceed the allowable graphite stresses. Since the dominant risk contributor for the steam cycle design were initiated by water ingress from the steam generators, the GT-MHR is expected to have a lower risk profile to the public. References 4 and 5 provide more information on the GT-MHR safety evaluations.

Utility/EPRI Review. A key event in the U.S. GT-MHR program was the review by utility/Electric Power Research Institute representatives held June 20 and 21, 1995 at General Atomics in San Diego. The review was prompted by an Advanced Reactor Corporation report that characterized the GT-MHR as potentially a "breakthrough technology" due to its combination of built-in safety and high plant efficiency. The ARC report proposed that a broader, intensive review be conducted by utility representatives assembled by EPRI.

The objective of the review and subsequent report (Ref. 6) was to provide the reactor designers and other organizations with interest in future electric power generating options with an independent evaluation of the conceptual GT-MHR design from the viewpoint of potential owner/operators. The review focused on the PCS. Two paragraphs from the summary are reproduced below from their report.

> "The review team concludes that the GT-MHR makes bold and innovative use of emerging technologies that have the potential for much improved efficiency, simplification, and reduction of cost and waste. However, the current GT-MHR conceptual design needs substantial development, detailed design, experimental verification, and licensing effort before it can be viewed with confidence as a breakthrough technology that provides a technically and economically viable future option for potential utility customers."
>
> "The review team found no obvious show stoppers that would prevent the GT-MHR from eventually meeting regulatory safety goals and achieving the many technical and economic goals set by DOE and the design team. Meeting these goals is viewed as both a major challenge and a prerequisite to the GT-MHR joining the ALWR in future U.S. nuclear power plant deployment."

The utility design review provides very valuable input to the GT-MHR program. The review team identified a number of needs or R&D "challenges" additional to those identified by the design team. The design team intends to carefully consider and respond to each recommendation. This utility input will provide the initial starting off point as the design progresses.

7. Conclusions

In the last year considerable progress has been made on the GT-MHR program. An experienced capable team has been assembled to address the design and development of the power conversion system. Many key design issues have been addressed and solutions identified. The results of this progress continue to support earlier conclusions that there are no feasibility issues and that continued design and development can indeed lead to a breakthrough technology.

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HTR PLUS MODERN TURBINE TECHNOLOGY FOR HIGHER EFFICIENCIES

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Abstract

The recent efficiency race for natural gas fired power plants with gas-plus steam-turbine-cycle is shortly reviewed. The question 'can the HTR compete with high efficiencies?' is answered: Yes, it can - in principle. The gas-plus steam-turbine cycle, also called combi-cycle, is proposed to be taken into consideration here. A comparative study on the efficiency potential is made; it yields 54.5. % at 1 050 °C gas turbine-inlet temperature. The mechanisms of release versus temperature in the HTR are summarized from the safety report of the HTR MODUL. A short reference is made to the experiences from the HTR-Helium Turbine Project HHT, which was performed in the Federal Republic of Germany in 1968 to 1981.

Keywords:
High Temperature Reactor HTR, modern turbine technology, gas-plus steam-turbine cycle, combi-cycle, efficiency natural gas fired power stations with gas-plus steam-turbine cycle, 3-pressure-steam-turbine cycle, release versus temperature, experiences from HTR-Helium-Turbine Project HHT.

HTR plus Modern Turbine Technology for Higher Efficiencies

1. Efficiency Race Triggered by Natural Gas

1.1. In summary: The decline of the price of fossil energy carriers after the end of the oil price crisis, in particular the low price of natural gas, have triggered an impetuous development in gas turbine cycle technology. An efficiency race has been opened up to achieve higher values of efficiencies for fossil fired power plants, and in particular for natural gas fired power plants. The preferred solution of modern turbine technology is the gas-plus steam-turbine cycle technoloy, also called combi-cycle. A high efficiency value of existing plants is e.g. 52 %; a typical value for the future perspective is 58 %.

1.2. As an appetizer for this chapter two references:

1.2.1. Siemens AG, Bereich Energieerzeugung (KWU): "The development of gas turbines did achieve a new culmination at December 1994. During normal operation at our test site in Berlin the model V84.3A demonstrated performances which are number one in the world:

- An efficiency (of a gas turbine) of 38 %, which leads to an efficiency of a GUD-plant of 58 % (GUD = Gas- und Dampf-Turbine, Trademark) and

- a power (of a gas turbine) of 170 MW, the largest in the class of the 60-Hz-turbines." Lit. SIEMENS-1995.

1.2.2. From a scientific report: "The efficiency can be increased from yesterday's promising 54 % in two steps to values around 60 % at the end of the decade. In each step three parameters are important, these are: higher effiencies by optimalization of the design of the blades, increased gas turbine-inlet temperature and improvements in the steam-turbine process, e.g. with a sub-critical 3-pressure-process including re-heat steps", lit. RIEDLE-1994, S. 39.

1.3. In detail on efficiencies on gas turbines combi-cycles and future prospects of natural gas fired power plants:

1.3.1. In 1994 natural gas-fired power plants with gas turbines with the total power of about 240 GWe were in operation with a total efficiency of 32 %, and natural gas fired power plants with combi-cycles, that is gas-turbine plus steam-turbine cycles,GST, with the total capacity of about 130 GWe were in operation with an efficiency of 49 %, RIEDLE-1994, fig. 1. The bigger number of capacity for gas turbines - in contrary to combi-cycles - is an indication that smaller unit capacities and smaller capital costs are also decisive in the decision for an investment. But there is a trend to make use of the potential for higher efficiencies with the combi-cycles, fig. 1.

1.3.2. Natural gas fired power stations with steam cycles achieve efficiencies between 42 and 47 %, fig. 1. But obviously the gas turbine technology offers, in particular for natural gas based systems, a number of advantages, e.g. low capital cost, short construction time, and last not least high potential for efficiency.

1.3.3. Official measurements at the power station AMBARLI, Turkey, with a gas plus steam-turbine-cycle, GST, called GUD, and constructed by Siemens resulted in an efficiency of 52,5 % at nominal power, fig. 1, and 53,2 % at peak power, lit. SIEMENS-1993. This applies for the first block with the total power of 450 MWe. The second and third block showed efficiencies at nominal power of 52.0 % and 51.9 %. These measured values, fig. 1, fit with the theoretical evaluations, lit. RUKES-1993 and REUTER-1993, and they apply for a gas turbine-inlet temperature of 1050 °C. The recently finished construction of the 205 MWe GUD power station TROMBAY, India, has a measured value of the efficiency at nominal operation of 50,48 %, lit. NAUEN-1995, table 1, at the air temperature of 35 °C, which adjusted to 24 °C means about 51,5 % at a gas turbine-inlet temperature (ISO 2314) of 1037 °C, lit. NAUEN-1995, table 2.

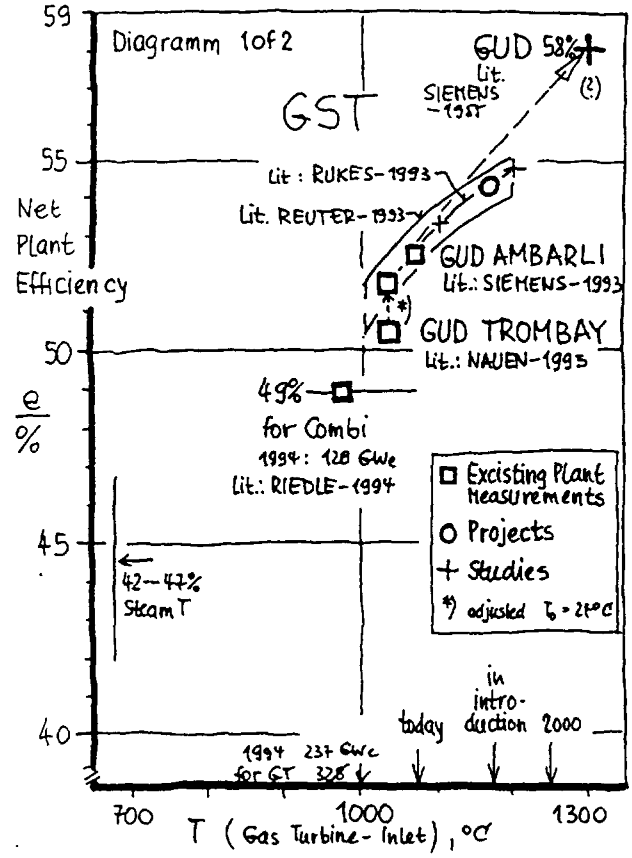


Fig. 1: Efficiency race natural gas, status and trend for the steam turbine cycle, the gas turbine cycle and the gas-plus-steam-turbine cycle: Net plant efficiency versus gas turbine inlet temperature, diagram 1 of 2.

1.3.4. With increasing gas turbine inlet temperature the efficiency increases at about 2 %-points/100 K gas turbine-inlet temperature, so that at around 1200 °C about 55 % can be achieved, fig. 1. Gas turbines with gas turbine-inlet temperatures in the range of 1 100 to 1 200 °C are now in introduction into the market. The perspectives is that at about the year 2000 the gas turbine-inlet temperature could achieve 1 250 °C, lit. RIEDLE-1994.

1.3.5. It should be remarked here that the information about the promising value of efficiency of 58 % in the advertisement of the vendor industry - as usually - does not contain any precise information about the gas turbine-inlet temperature; therefore the respective value in fig. 1 is labled with a question mark. The reason is simple: The gas turbine inlet temperature is the simplest indicator for progress in the gas turbine technology.

1.3.6. Another factor to be considered is - of course - the capital investment.

Table 1: HTR with Gas-plus Steam-Turbine Cycle, 2 Designs

1. Primary circuit		Design A	Design B
Cooling fluid		Helium	Helium
Thermal power	MW	200	200
Reactor inlet temperature	°C	350	364
Reactor outlet temperature	°C	950	1050
Reactor outlet pressure	bar	60	60
Relative pressure losses	%	3,3	0,8
Helium mass flow	kg/s	64,2	58,7
Electric blower power	MW	3,3	0
2. Gas turbine circuit			
Working fluid		Helium	same
Log. temperature difference in IHX	°C	50	-
Relative pressure losses	%	6	3
Turbine inlet temperature	°C	900	1050
Turbine outlet temperature	°C	580	548
Relative cooling massflow	%	3	3
Turbine mass flow	kg/s	64,2	58,7
Turbine pressure ratio		2,42	3,71
Polytrope turbine efficiency	%	90	92
Compressor inlet temperature	°C	97	97
Compressor outlet temperature	°C	291	396
Compressor mass flow	kg/s	66,2	60,4
Compressor pressure ratio		2,58	3,91
Polytrope compressor efficiency	%	90	91
Generator power	MW	39	59,2
3. Steam turbine circuit			
Helium inlet temperature in waste heat boiler	°C	580	548
Helium outlet temperature in waste heat boiler	°C	97	97
HP - High pressure steam\HP	°C/bar	550/60	515/140
LP -Low pressure steam\MD	°C/bar	200/8	300/29,4
HD -Steam mass flow\LP	kg/s	43,8	230/5,6
LP - Steam mass flow	kg/s	6,9	-
Polytrope efficiency of steam turbine	%	85	85
Generator power	MW	57,9	52,3
4. Total plant			
Internal thermal efficiency of combined cycle	%	48,9	55,7
Total power (sum minus mech. losses)	MW	96,8	110
Total efficiency	%	48,4	55,0
Self demand (with blower)	MW	4,2	1,0
Net power	MW	92,7	109
Net efficiency	%	46,3	54,5

Remark:
Design A: Lit.: HAVERMANN-1993, BARNERT-SINGH-1994
Design B: This study

2. HTR with Gas-plus Steam-Turbine, GST

2.1. In summary: The efficiency potential of the gas turbine technology for the conversion of high temperature heat from the HTR into electricity is - in principle - as high as that based on natural gas. Therefore it is proposed here to take the gas-plus-steam-turbine-cycle, GST, into consideration, in particular with a "3-pressure-steam-turbine-cycle". With further improvements, in particular in the gas turbine cycle, and with the assumption that the gas turbine-inlet temperature is 1 050 °C (100 K more than AVR in 1974) the calculated net efficiency is 54.5 %. A particular advantage of the GST versus the gas turbine cycle with recuperation is that the core-inlet temperature is smaller at comparable efficiency conditions.

2.2. In detail on efficiencies from various projects and on the efficiency potential, in comparison to the conventional natural gas based technology:

2.2.1. The measured efficiency of electricity production in the THTR-300 as the ratio of the measured generator power vs. the thermal power is 39,7 % (303 MWe/763 MWt, not the net-

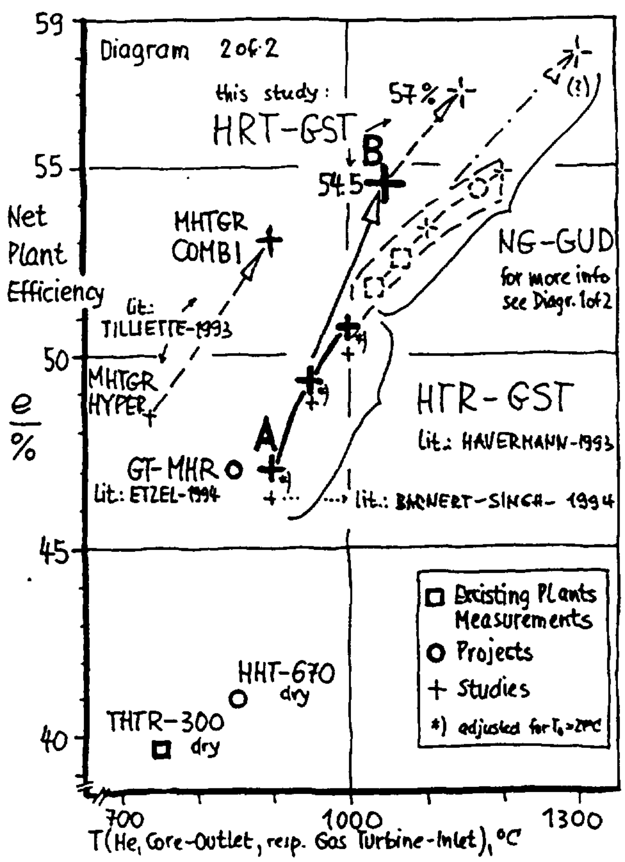


Fig. 2: HTR with gas-plus steam-turbine-cycle, GST, trends in efficiency and comparison to natural gas: Net plant efficiency versus helium temperature at core-outlet, respectively gas turbine-inlet, diagram 2 of 2: The HTR-GST has the same efficiency potential as the natural gas based system.

efficiency), lit. SCHWARZ-1987, p. 8, table 2, measurement made 100 % power. These measured values were close to the calculated values. The THTR-300 had a steam turbine cycle and a dry cooling tower, it was operated at a core-outlet temperature of 750 °C, fig. 2. The operation of the THTR-300 was terminated in 1989, due to financial and political difficulties; rigth now (October 1995) the unloading of the fuel pebbles is finished since more than half a year.

2.2.2. The HHT-670 (HHT = HTR plus Helium Turbine, 670 MWe, projected demonstration plant) had an efficiency of 41 % at a gas turbine-inlet temperature of 850 °C, lit. ARNDT-1979. It should be remarked, that this efficiency value applies for dry cooling. The project HHT was performed from 1968 to 1981 in the Federal Republic of Germany; it was terminated in 1981 because of the decision that the THTR-300 follow up-plant should be an HTR with a steam cycle. More details are given in lit. WEISBRODT-1995, respectively in this workshop.

2.2.3. The efficiency of the GT-MHR (GT-MHR = Gas Turbine-Modular High Temperature Reactor) is 47 % at the gas turbine-inlet temperature of 850 °C, lit. ETZEL-1994, table 1. The gas turbine cycle includes a compact plate fin recuperator. An important design feature in that project is that all part of the gas turbine and the heat exchangers, as well as the generator, are located in the power conversion vessel.

2.2.4. The MHTGR-Combi (MHTGR = Modular High Temperature Gas-Cooled Reactor with Combined Gas Turbine - Hypercritical Steam Turbine Cycle, 238.5 MWe) has an efficiency of 53 % at the gas turbine-inlet temperature of 900 °C, fig. 2, lit. TILLIETTE-1993, fig. 11 on page 14, with an MHTGR of 450 MWt. It is foreseen to be operated with an intermediate heat exchanger, the core outlet temperture is 950 °C and the core inlet temperature 512 °C. The base for this proposal is an MHTGR-HYPER utilising an advanced, hypercritical steam cycle alone with an efficiency of 48.4 % at the steam generator-inlet temperature of 730 °C, fig. 2. The interesting thermodynamical feature of this proposal is that the hypercritical steam turbine cycle allows a strong reduction in exergy losses in the steam generator: It fits very well to the heat cascading between the gas turbine and the steam turbine cycles.

2.2.5. Another possibility for the reduction of exergy losses in the heat cascading between the gas turbine cylce and the steam turbine cycle is the "2 or 3 pressure steam cycle" as used in modern conventional GST-cycles based on natural gas, chapter 1. Therefore it is proposed here to take the gas-plus steam-turbine cycle, GST, into consideration, in particular with a "3-pressure-steam-turbine-cycle".

2.2.6. The HTR-GST, design A, has an efficiency of about 47 % at the gas turbine-inlet temperature of 900 °C, fig. 2, being adjusted for T_0 = 24 °C in this figure, from 46,3 %, table 1, lit. BARNERT-SINGH-1994 and lit. HAVERMANN-1993. The HTR-GST, design A has the following main data: modular HTR-200 MWt, 950 -350 °C (core-outlet,-inlet temperature), intermediate heat exchanger 900-291 °C, gas turbine with 2,42 turbine pressure ratio, steam turbine cycle with 2 pressures and internal thermal efficiency of 48,9 %, table 1, column design A, lit. BARNERT-SINGH-1994 and lit. HAVERMANN-1993. An important design detail of design A (not given in table 1) is the pintch point temperature difference of 8 K. The higher values of efficiencies in fig. 2, about 49 % at 950 °C and about 50,5 % at 1 000 °C, were achieved by the omittance of the intermediate heat exchanger and the increase of the gas turbine inlet temperature to 1 000 °C, lit. HAVERMANN-1993. Here it can be remarked that the tendency of these efficiency values does well fit with the tendency of measured efficiency values of assisting plants based on natural gas, called NG-GUD in fig. 2.

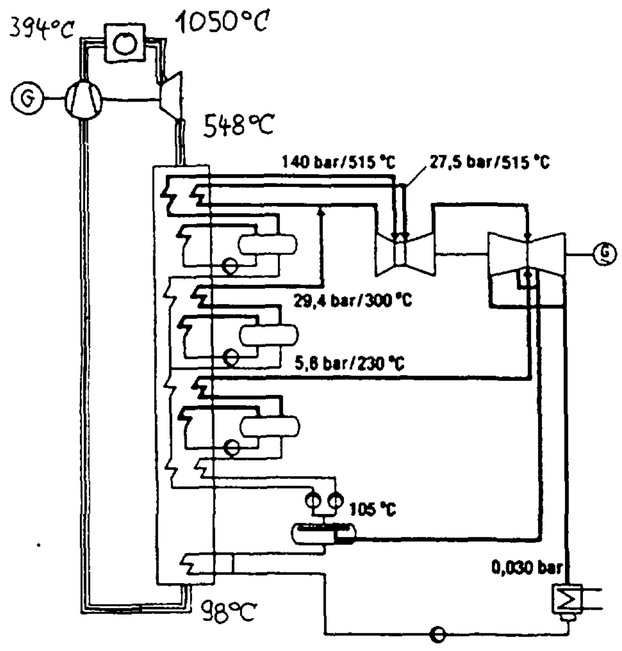
2.3. In detail on HTR with gas-plus steam turbine cycle, GST, in particular with a "3-pressure steam turbine cycle", design B, this study:

2.3.1. The HTR-GST-design B has an efficiency of 54,5 % at the gas turbine-inlet temperature of 1 050 °C, fig. 2, this study. The design B consists of an HTR with 200 MWt, a gas turbine cycle, comparable to the conventional ones and a 3-pressure-steam turbine cycle, taken from conventional design proposals, fig. 3 and fig. 4, lit. RUKESS-1993, p. 28, fig. 6. The calculated design data, table 1, column design B, are made for a gas turbine-inlet temperature of 1 050 °C, and include some improvements of the gas turbine technology as compared to design A. These are: polytrope turbine efficiency: 92 % (instead 90), polytrope compressor efficiency: 91 % (instead 90), relative pressure losses: 3 % (instead 6, because steam generator instead of compact recuperator) and relative cooling mass flow: 3 % (unchanged, in spite of increased gas turbine inlet temperature).

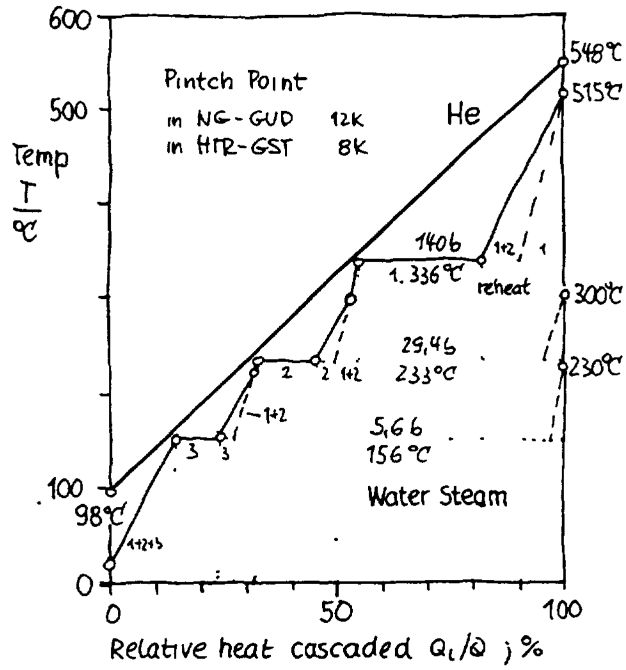
2.3.2. The gas-turbine-cycle of design B is similar to conventional ones, fig. 3. It has the following temperature data: gas turbine-inlet temperature 1 050 °C, and core-inlet temperature 396 °C. It can be remarked, that the low core-inlet temperature is an advantage of the GST-cycle compared to the gas turbine cycle with recuperation.

2.3.3. The 3-pressure-steam-turbine-cycle of design B, fig. 3, has been taken unchanged from a conventional design, because it can be considered to be an optimum design already. The steam generator-inlet temperature is 548 °C, the steam generator outlet temperature is 98 °C. The process also includes some re-heat. A possible disadvantage of the 3-pressure steam turbine cycle in comarison to the 2-pressure steam turbine cycle is the increased value of the 3rd pressure of 140 bar, which is higher than the primary helium pressure and therefore needs to be particularly considered in design base accidents.

2.3.4. An impression on the reduction of exergy losses can be derived from the temperature-heat-diagram, fig. 4. The bundles in the steam generator are arranged in such a way, that the temperature-heat-lines of the steam cycle fit best to that of the helium cycle. The pintch point temperature difference in design B has also not been changed and therefore is 12 K (not



for the 3 Pressure Steam Turbine
Lit RUKES-1993, Fig 6 p 28



for Evaporation Conditions
Lit RUKES-1993, p 28, Fig 6

Fig. 3: Potential HTR-GST with an efficiency of 54.5 %, flow sheet with gas-plus 3-pressure -steam-turbine-cycle, taken from conventional designs, gas turbine cycle adjusted for reactor outlet temperature of 1 050 °C

Fig. 4: Potential HTR-GST with an efficiency of 54.5 %, temperature-heat-diagram for the 3-pressure-steam-turbine for the flow sheet of fig 3· the exergy losses of heat transfer are reduced

mentioned in table 1) For future evaluations this value may be reduced to 8 K, resulting in a further improvement of the efficiency

2.3 5 An even higher efficiency of HTR-GST could be 57 % at a gas turbine-inlet temperature of 1 150 °C, fig 2, dashed arrow and cross; the conditions is - of course - that a future HTR can produce high temperature heat of 1 150C °C.

2 3 6 An important advantage of the "gas-plus steam-turbine cycle" in comparison to the "gas turbine cycle with recuperation" is that the core-inlet temperature is smaller at comparable efficiency conditions. The following comparison can be taken as an example: The GT-MHR with 47 % at 850 °C, fig 2, has an core inlet temperature of 493 °C (919 °F), lit. ETZEL-1994, fig. 7, upper part, the HTR-GST with 54.5 % at 1 050 °C, fig. 2, has an core inlet temperature of 396 °C, fig 3

2.3.7 In addition to these evaluations on efficiencies and other design data, it is of course decisive to take capital investments into consideration.

2 3 8 The gas turbine-inlet temperature of 1 050 °C used in design B of this study is just 100 K higher than the mean core-outlet temperature of the AVR, which was achieved already in 1974

3. HTRs with Increased Gas Outlet Temperature, e.g. 1 050 °C

3 1 In summary The biggest challenge to increase the temperature of the produced heat is put on the HTR by the improving gas turbine technology Therefore future designs of HTR fuel on the basis of the TRISO-coated particle and of new HTR cores may realize increased gas outlet temperatures, e g 1 050 °C Reasons are good experiences of the operation of the AVR, other experimental results and proposals to increase the retention capability of the HTR fuel Another reason may be that contamination of the turbine may not be an important issue as before

3 2 In detail on the retention capability of HTR-fuel in dependence from the temperature

3 2 1 In the THTR-300 the maximum fuel temperature of the fuel pebbles is 1 250 °C as the required design value and - at the same time - as a value for the commercial guarantee, fig 5, lit HKG-1969, Bd 1, S 4 16, tab 4 2 2-1 (THTR-300 Safety Report) For the release it is stated there "For these fuel elements the fraction of release for Xe-133 shall not exceed the value of 3×10^{-5} as the mean over the lifetime and as the mean over the core" The coated particles of the fuel elements of the THTR-300 had a BISO-coating

3 2 2 A good overview on the retention capability of modern fuel pebbles with TRISO coated particles has been prepared for the HTR-MODULE describing the mechanisms of release of fission products, lit SIEMENS-INTERATOM-1988 (HTR-MODUL Safety Report), Bd 1

3 2 2 1 In the lower range of temperature below about 1 200 °C the fraction of brakes of coated particles (expectation value) is alone fabrication induced, it is less than 3×10^{-5}, fig 5 The hereby produced release is rather low and not very much dependent from temperature, lit S 3 2.2 1 -8 and 9

3 2 2 2 At temperatures up to 1 300 °C no brakes of coated particles in material test reactors have been observed, lit S 3 2 1 -7

3 2 2 3 With increasing temperatures above 1 200 to 1 300 ° C some diffusion starts from intact particles An additional fraction of brakes of coated particles in the range of 1 200 to 1600 °C does not need to be considered according to the experiments, lit S 3 2 2 1-9

3 2 2 4 The temperature-induced fraction of brakes of coated particles - as the maximum value - is 5×10^{-5} at 1 600 °C, lit S 3 2 2 1-8

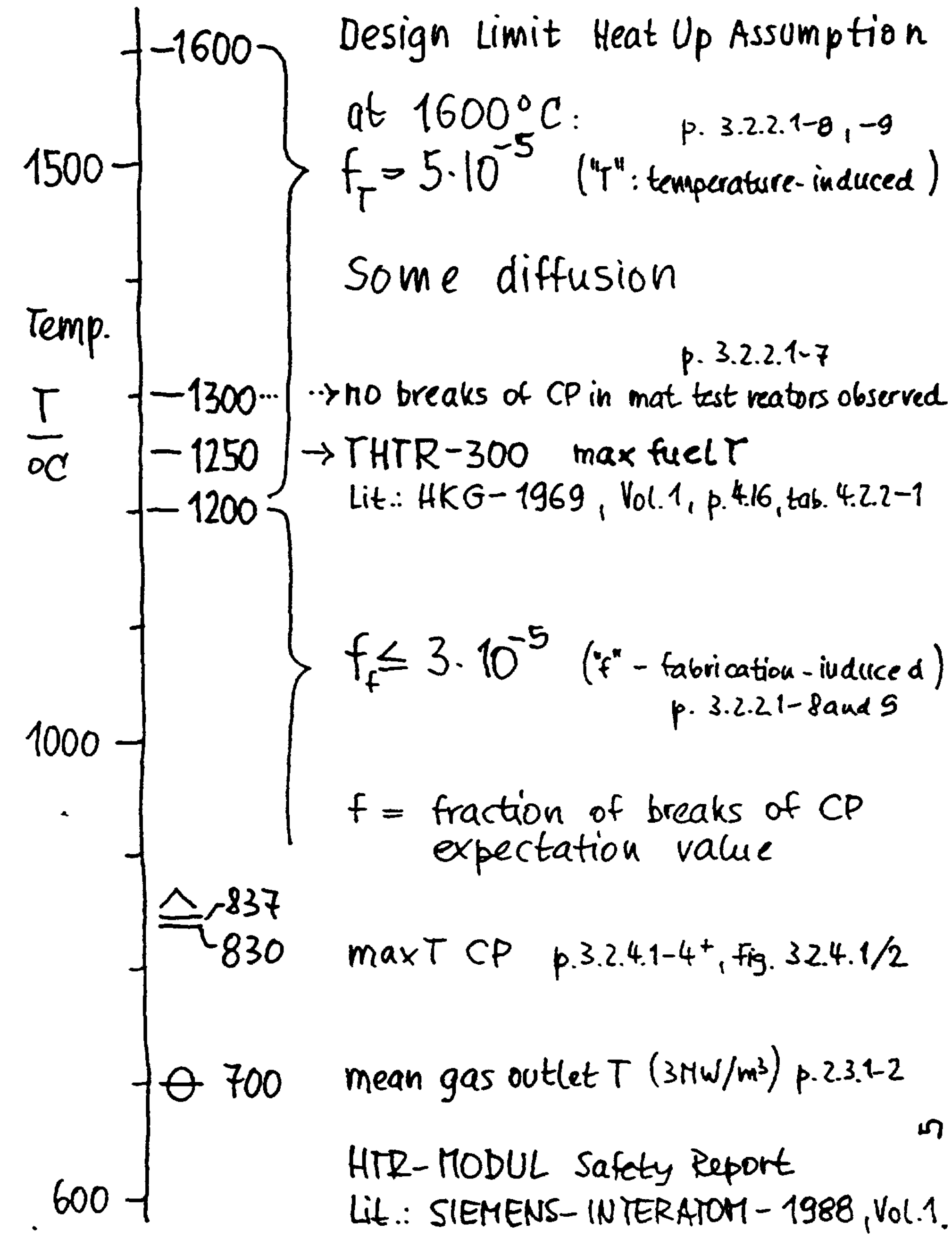


Fig. 5: HTR-MODUL fuel design data, release versus temperature, mechanisms of the release, and THTR-300 fuel data: From these data it is concluded, that the core-outlet temperature can be increased to 1 050 °C for future HTRs with modern gas turbine technology.

3.2.2.5. The design limit for the heat up assumption is 1 600 °C.

3.2.2.6. The various ranges and limits of temperatures are illustrated in fig. 5 in linear scale.

3.2.4. The temperature load conditions for the HTR Modul are: Mean gas-outlet temperature: 700 °C, mean power density 3 MWt/m^3, lit. : S. 2.3.1-2, producing a maximum value of temperature of the coated particles of 830, respectively 837 °C, lit.: S. 3.2.4.1-4$^+$ and fig.

3.2.4.-1/2. These values are also illustrated in fig. 5. As a conclusion: the maximum value for heat transfer in the fuel pebbles, as the difference between the maximum fuel temperature and the mean gas outlet temperature is at a power density of 3 MWt/m^3 about 130 K, respectively 137 K.

3.3. In conclusion for a future core outlet temperature of 1 050 °C:

3.3.1. The perspective for an increased value of the mean core-outlet temperature of the HTR with pebble bed core is the value of 1 050 °C. This follows from the information given in the above chapter 3.2. For this reason the evaluation of the HTR-GST-cycle in chapter 2 has been done with the gas turbine-inlet temperature (being equal to the core-outlet temperature) of 1050 °C.

3.3.2. Among the various applications of the HTR, (as e.g. steam cycle, steam cycle plus district heat or process steam, process heat applications for methane reforming and coal refinement, and others) the biggest challenge to improve the temperature of the produced heat is put on the HTR by the improving gas turbine technology. The reasons is the relatively strong influence on the efficiency, mainly because of its reducing influence on capital costs.

3.3.3. A possible contamination of the blades and other structures of the gas turbine due to the long term release of fission products, e.g. Cs-137, may in future not be an important issue as before, e.g. in the HHT-project, because the technologies for remote handling and inspection as well as other maintenance conditions have improved.

4. Experiences from the Project "HTR with Helium-Turbine, HHT"

4.1. In summary: The project "HTR with Helium-Turbine, HHT" was carried out in the Federal Republic of Germany in 1968 to 1981. It has produced a large number of valuable experiences. The project had been terminated because of the industrial decision to project an HTR with steam turbine plant as the THTR-300-follow-up-plant.

4.2. In some detail on the HHT-project and the high temperature helium test plant HHV:

4.2.1. The project "HTR with Helium-Turbine, HHT", being carried out in the Federal Republic of Germany in 1968 to 1981 in cooperation with the United States and Switzerland" and with the support from Swiss and German utilities had the objective to convert high temperature nuclear heat into electricity, using helium as the working fluid. Within that project to bigger test facilities were design and operated: The "helium turbine co-generation power plant, EVO" (EVO = Energie-Versorgung Oberhausen, Energy Supply at the City of Oberhausen in Germany) and the "High Temperature Helium Test Plant, HHV" (HHV = *Hochtemperatur-Helium-Versuchsanlage, High Temperature Helium Experimental Plant, at* KFA Jülich). A recently produced summary on the technical experiences is given in lit.: WEISBRODT-1995, which is reported also in this workshop.

4.2.2. Within the HHT-project a project study has been performed for a demonstration plant HHT-670 with a net output of 670 MWe, a net plant efficiency of 41 % and with dry cooling. The gas turbine-inlet temperature was 850 °C at a pressure ratio of 2.84 and with a reactor pressure of 70 bar, lit.: ARNDT-1976. Details on the design are given in fig. 6 and on the cycle and turbo-machinery in fig. 7.

4.2.3. An additional look, compared to lit. WEISBRODT-1995, on the High Temperature Helium Test Plant HHV with respect to the turbo-machine, the test loop and the temperature-entropy-diagram is given in fig. 8, lit. WEISKOPF-1970. An interesting feature of the HHV-plant was that heat was brought into the test loop only via the compessor of the turbo-machinery (and not via a heat exchanger). That is illustrated in the temperature entropy-diagram in fig. 8.

5. Summary and Results

5.1. Main result: Nuclear energy, as a source for high temperature heat, e.g. from the High Temperature Reactor, HTR, has - in principle - the same high efficiency potential as the natural gas-based conversion process, both using modern turbine technology. The most modern gas turbine technology is the "gas-plus steam-turbine-cycle, GST", which also can be used in a closed form. A comparative study shows: At 1 050 °C gas turbine-inlet temperature the net efficiency is about 54 %.

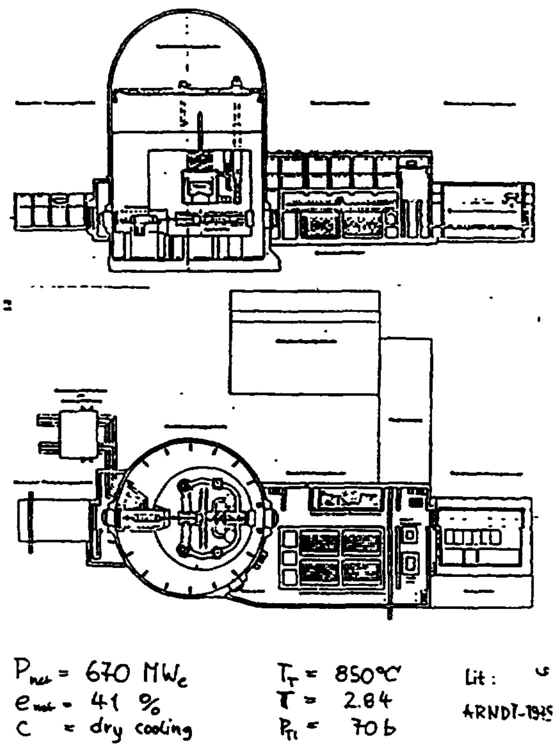


Fig. 6. HTR Heliumturbine Proj. HHT 1968-1981
Project-Studie Demo-Plant HHT 670 1979

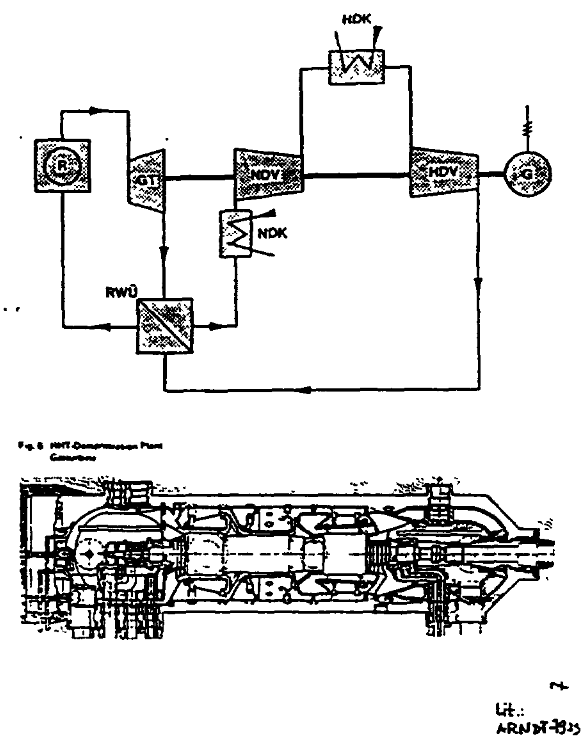


Fig. 7. HTR Heliumturbine Proj. HHT 1968-1981
Flowsheet and Turbomachine

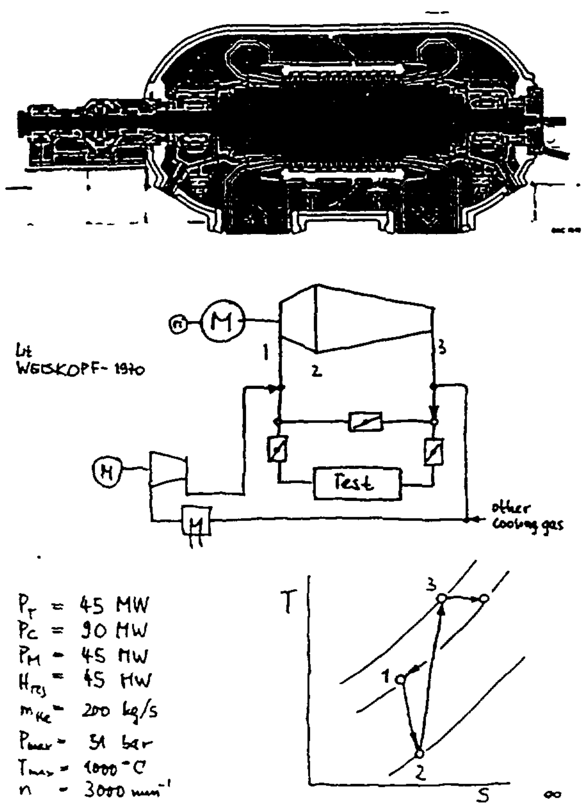


Fig. 8. High Temp Helium Test Plant HHV

5.2. In detail: As a summary of previous chapters:

5.2.1. The decline of the price of fossil energy carriers after the end of the oil price crisis, in particular the low pric of natural gas, have triggered an impetuous development in gas turbine cycle technolgy. An efficiency race has been opened up to achieve higher effiencies for fossil fired power plants, and in particular for natural gas fired power plants. The preferred solution of modern turbine technology is the "gas-plus steam-turbine cycle, GST" also called combi-cycle. A high efficiency value of an existing plant is e.g. 52 %; a typical value for a future perspective is 58 %

5.2.2. The efficiency potential of the gas turbine technology for the conversion of high temperature heat from the HTR into electricity is - in principle - as high as that based on natural gas. Therefore it is proposed here to take the "gas-plus-steam-turbine-cycle, GST", into consideration, in particular with a "3-pressure steam turbine cycle". With further improvements, in particular in the gas turbine cycle, and with the assumption that the gas turbine-inlet temperature is 1 050 °C (100 K more than AVR in 1974) the calculated net efficiency is 54 5 % A particular advantage of the GST-cycle in comparison to the "gas turbine cycle with recuperation" is that the core-inlet temperature is smaller, at comparable efficiency conditions.

5.2.3. The biggest challenge to increase the temperature of the produced heat is put on the HTR by the improving gas turbine technology. Therefore future designs of HTR fuel on the basis of the TRISO-coated particle and of new HTR-cores may realize increased gas outlet temperatures, e.g. 1 050 °C. Reasons are the good experiences of the operation of the AVR, other experimental results and proposals to increase the retention capability of the HTR fuel. Another reason may be that the contamination of the turbine may not be an important issue as before.

5.2.4. The project "HTR with Helium-Turbine, HHT" was carried out in the Federal Republic of Germany in 1968 to 1981. It has produced a large number of valuable experiences. The project had been terminated because of the industrial decision to project an HTR with steam turbine plant as the THTR-300-follow-up-plant.

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DESIGN OF HTGRs WITH CLOSED CYCLE GAS TURBINES

Session 2

DESIGN OF INDIRECT GAS TURBINE CYCLE FOR A MODULAR HIGH TEMPERATURE GAS COOLED REACTOR

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Abstract

This paper describes a design of the indirect gas turbine cycle for the 200MWt pebble bed MHTGR. In the design, the helium out of the Intermediate Heat eXchanger (IHX) is extracted to a small RPV cooling system. The gas flows through a small RPV recuperator and is cooled down, then it is used to cool the RPV. The whole primary circuit is integrated in a pressure vessel. The core inlet/outlet temperatures are 550°C/900°C, which can supply a gas heat source of 500°C/850°C in the secondary side. The heat source could be used to drive a nitrogen gas turbine cycle and a plant busbar electricity generation efficiency of about 48% is estimated. The thermodynamic calculation, preliminary design of the system components, and the important accident analysis are described in this paper.

Keywords: Reactor, Gas Turbine, HTGR

1. Introduction

The Modular High Temperature Gas Cooled Reactors can provide high temperature heat source such as 950 ℃ with keeping the outstanding inherent safety. The electricity production efficiency of the MHTGRs by using Gas Turbine (GT) cycle can reach a remarkable 45 ~ 50%. The direct gas turbine cycles, such as GT-MHR design by MIT and GA[1]. However the possible radioactivity deposition on the turbine blade and thus the increase of maintenance difficulties suggest that indirect gas turbine cycles should be applied firstly before enough experience is achieved to solve the radioactivity deposition problem in the direct cycle.

The major difficulty to limit the higher thermal efficiency of the indirect cycle is the lower core inlet temperature. For the direct cycle, the cold gas leaving the precooler can be extracted to cool the reactor pressure vessel (RPV) and the other steel structures. Therefore the core inlet temperature can be as higher as 500~ 600°C. For the indirect cycles proposed before, such as MGR-GTI proposed by Yan and Lidsky[2], the core inlet temperature is kept as lower as 310°C in order to cool RPV The plant busbar efficiency of MGR-GTI is about 42.1%. In order to achieve higher power plant efficiency, it seems special designs of RPV cooling should be provided for the indirect gas turbine cycle.

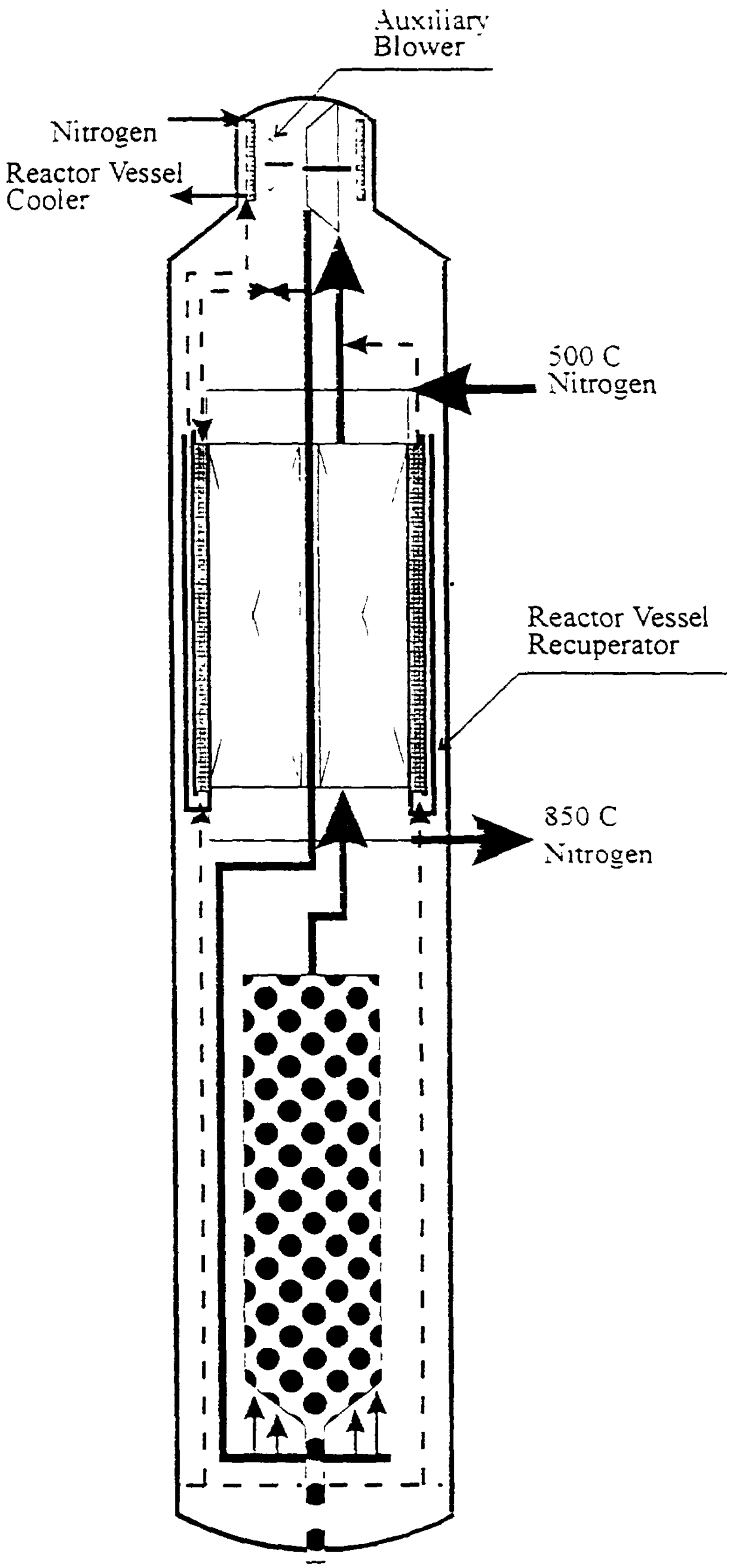

Fig.1. Layout of MHTGR-IGT

This paper provides a preliminary conceptual design of a indirect gas turbine cycle with a special RPV Cooling System (RPCS) The design, named MHTGR-IGT, has the following main features,

- The reactor core, the Intermediate Heat eXchanger (IHX), the RPV Cooling System (RVCS) are contained in a reactor pressure vessel
- High plant efficiency is achieved mainly by increasing core inlet temperature to 550 °C
- Low RPV operation temperature is realized by a special design of RVCS

The thermodynamic analysis, preliminary design of the system components, and the important accident analysis are described in the following sections

2. System Design

As shown in Fig. 1, a 200MWt pebble-bed reactor core is located at the lower position of RPV. The core geometry is same as that of Siemens 200MW HTR-Modular. A straight tube IHX is located at the upper position of RPV and connected with the core through a gas duct. Similar to the AVR, the control rod system is installed at the RPV bottom and all of control rods are inserted upward into the side reflector. The main helium blower and an auxiliary blower for shutdown cooling are located at the RPV top. A RVCS recuperator (RVR) and RVCS cooler (RVC) are installed respectively in annular regions outside IHX and blowers.

The cold helium from the main blower firstly flows downward through vertical channels in the side reflector and comes into a plenum at the bottom reflector. From the bottom plenum, the cold helium flows upward into the pebble-bed core and is heated up from 550℃ to 900 ℃, then the hot helium converges into a plenum at the top reflector and enters the IHX. In IHX, the hot helium flow upward through the IHX shell side and is cooled down by the secondary nitrogen. At last, the 550℃ cold helium flows into the main blower Brief technical data of the integrated MHTGR-IGT are given in Table 1. Detailed information on the design of IHX, RVCS will be given in the following section.

The GT cycle of MHTGR-IGT is similar to the MGR-GT design given by GA. In order to gain high plant efficiency, three stage compression and two stage intercooling are used. The plant busbar efficiency of about 48% is estimated.

Table 1. Brief technical data of MHTGR-IGT

Contents	*Parameters*
Reactor thermal power(MW)	200
Core power density(MWt/m^3)	3.0
Core diameter(m)	3.0
Core average height(m)	9.43
Pressure of the primary loop(MPa)	6.0
Primary coolant	helium
Core inlet temperature(℃)	550
Core outlet temperature(℃)	900
Helium Flow Rate(Kg/s)	110
Working Fluid of GT cycle	Nitrogen (N2)
Pressure in the secondary side of IHX(MPa)	6.0
Inlet temperature of N2 in IHX (℃)	500
Outlet temperature of N2 in IHX(℃)	850
N2 flow rate (Kg/s)	529
RPV height (m)	~ 30
RPV diameter(m)	6.

Table 2. GT cycle of the MHTGR-IGT design

Contents	*Parameters*
Working fluid	Nitrogen(N2)
Flow rate of the working fluid(kg/s)	529
Turbine inlet temperature(℃)	850
Turbine outlet temperature(℃)	525
Precooler inlet temperature(℃)	102
Compressor inlet temperature(℃)	30
Compressor outlet temperature(℃)	77
IHX inlet temperature(℃)	500
Stages of compressors	3

Table 2 *Continued*

Contents	*Parameters*
Stages of intercooling	2
Pressure ratio of the every stage of compression	1 5845
Polytropic efficiency of the turbine(%)	91
Polytropic efficiency of the compressor(%)	91
Plant thermal efficiency(%)	52 5
Plant busbar efficiency	48

3. Component Design

3 1. Intermediate heat exchanger(IHX)

A straight tubular tube-and-shell heat exchanger is selected for the IHX In order to satisfy the compact demand, the heat transfer tubes with small diameter, close pitch, inner spire as well as outer low fin are chosen for the IHX tube bundle Preliminary designs of the IHX are given in Table 3

Table 3 Design Parameters of IHX

Contents	*Parameters*
Heat transfer capability	200. (MWt)
Fluid in shell side	Helium
Fluid in tube side	Nitrogen
Type of construction	Straight tube
Surface geometry	Fin tube with inner spire
Tube diameter and thickness	14× 1. (mm)
Tube pitch	18. (mm)
Number of tubes	45625
Heat transfer area	14239 (m^2)
Height of tube bundle	7 1 (m)
Inner/outer diameter of tube bundle	1100/4390 (mm)
Outer diameter of the IHX shell	4400 (mm)
Height of the IHX	about 10 m

3 2, RPV Cooling System(RVCS)

Calculation by using THERMIX code indicates the maximum temperature of the RPV is smaller than 300℃ if there is no auxiliary RPV cooling system and no leakage flow from the core structure However, in order to avoid thermal shock of leakage flow with high temperature(~ 550℃) to RPV from the reactor graphite structures, a RPV cooling system (RVCS) is designed

The flow diagram of RVCS is shown in Fig 2 The system consists of a RPV recuperator(RVR), a RPV cooler(RVC), orifices and valves As illustrated in Fig 1 and Fig 2, a bypass flow is extracted from the blower outlet and flows downward through the tube side of the RVR, which is installed in the annular region outside the IHX, and is cooled down from 550℃ to 250℃ Then the cold bypass flow enters an annular channel outside the RVR and flows upward into the shell side of a little cooler (RVC) and is cooled form 250℃ to 190℃, the 190℃ cold gas enters the gap between the RPV and core vessel and is heated by the two vessel from 190℃ to 220℃ At last, the 220 ℃ bypass gas flows upward through some vertical pipes and comes into the RVR shell side and returns to the blower inlet The main technical data of the RVCS is given in table 4

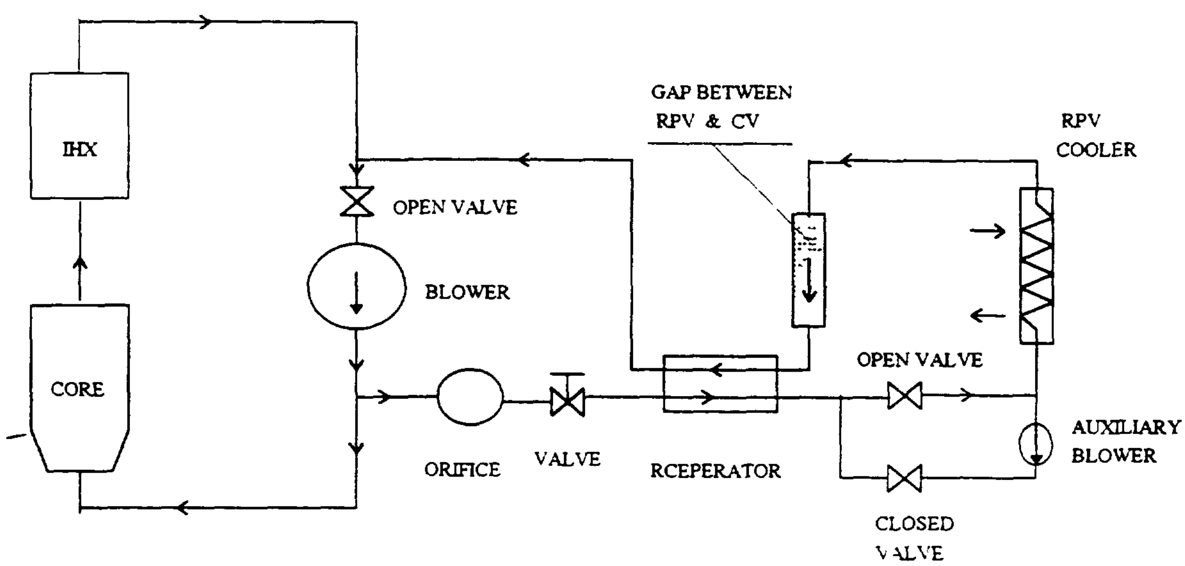

Fig 2 Flow Diagram of the RPV Cooling System

Table 4. Main technical data of the RVCS

Contents	*Parameters*
1, Parameters of RVR	
Thermal capability	6.86 MW
Structure	Spiral tube with tube jacket
Flow rate in shell side or tube side	4.4 (kg/s)
Inlet/outlet temperature in the side	550/250 (℃)
Inlet/outlet temperature in shell side	220/520 (℃)
Heat transfer area	267 (m^2)
Number of assembly	98
Outer diameter of assembly	114 (mm)
Height of assembly	4.44 (m)
Outer/inner diameter of the recuperator	4.92/4.6 (m)
2, Parameters of RVC	
Secondary coolant	Nitrogen
Structure	Spiral tube with tube-jacket
Heat transfer area	24. (m^2)
Number of assembly	33
Outer diameter of assembly	100. (mm)
Assembly height	1.5 (m)
Outer/inner diameter of the cooler	3.8 /3.6 (m)

3.3. Shutdown Cooling System (SCS)

Accompany with the RVCS, a shutdown cooling system(SCS) is proposed in the paper to provide a simple and reliable decay heat removal during normal shutdown period. The system is shown in Fig 3. It consists of an auxiliary blower, a cooler as well as a recuperator. Because the cooler and recuperator are also the parts of the RPV cooling system(RVCS), SCS is simple and its equipments have multi-function. The decay heat removal under the accidental conditions is depends on the passive reactor cavity cooling system. The SCS proposed in this paper therefore is not safety concerned.

In case of normal shutdown, the inlet valve of the main blower is closed and all of equipment in the secondary loop stop at the same time. The auxiliary blower of SCS then starts to drive helium flowing through the core, IHX, RVR shell side, RVC, the RVR tube side, and finally back to the core. In RVC, the decay heat carried by the helium is transferred to the nitrogen in the RVC secondary side. By means of adjusting flow rate or inlet temperature of the nitrogen, the decay heat can be removed out of the core by the SCS.

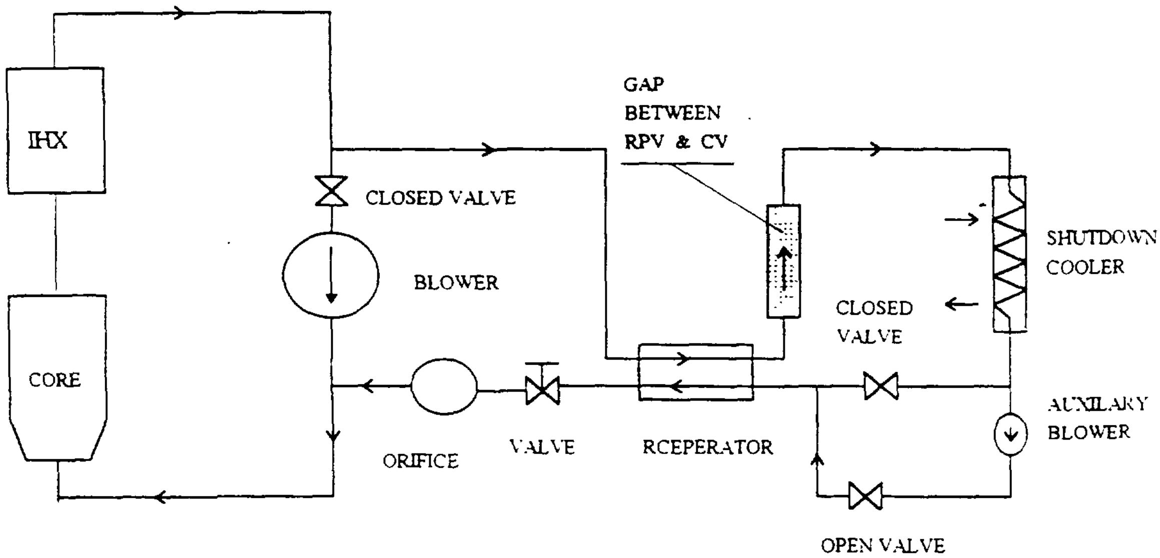

Fig 3. Flow Diagram of the Shutdown Cooling System

4. Preliminary Accident Analysis

Preliminary accident analysis of MHTGR-IGT has been carried out by using THERMIX code. The results of the analysis indicates, under rating operation condition, the maximum temperature of fuel elements is about 980℃. In case of the depressurization with core heat-up, the fuel maximum temperature is 1477℃. In case of the accident of loss of heat sink, the fuel maximum temperature is only 1080℃. It should be emphasized that, due to the integrated structure of the primary loop, the probability of losing primary operation pressure will be greatly reduced.

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CONCEPTUAL DESIGN OF HELIUM GAS TURBINE FOR MHTGR-GT

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Abstract

Conceptual designs of the direct-cycle helium gas turbine for a practical unit (450 MWt) and an experimental unit (1200kWt) of MHTGR were conducted and the results as shown below were obtained. The power conversion vessel for this practical unit can further be down-sized to an outside diameter of 7.4m and a height of 22m as compared with the conventional design examples. Comparison of the conceptual designs of helium gas turbines using single-shaft type employing the axial-flow compressor and twin-shaft type employing the centrifugal compressor shows that the former provides advantages in terms of structure and control designs whereas the latter offers a higher efficiency. In order to determine which of them should be selected, a further study to investigate various aspects of safety features and startup characteristics will be needed. Either of the two types can provide a cycle efficiency of 46 to 48%. The third mode natural frequencies of the twin-shaft type's low-pressure rotational shaft and the single shaft type are below the designed rotational speed, but their vibrational controls are made available using the magnetic bearing system. Elevation of the natural frequency for the twin-shaft type would be possible by altering the arrangements of its shafting configuration. As compared with the earlier conceptual designs, the overall system configuration can be made simpler and more compact; five stages of turbines for the single-shaft type and seven stages of turbines for the twin-shaft type employing one shaft for the low-pressure compressor and the power turbine and; 26 stages of compressors for the axial-flow type with the single shaft system and five stages of compressors for the centrifugal type with the twin-shaft system. An overall system configuration of the flange joint method to preclude leakages from gaps between the elements was developed, using a plate-finned recuperator and intercooler, and a helically-coiled precooler with low fins, and its feasibility is shown. A development program to lead to the commercial MHTGR-GT plant consisting of three phases including the fundamental design of commercial unit, demonstration of the components technologies and design of the demonstration unit, and fabrication and construction of the demonstration unit was also planned.

1. Introduction

As one of the energy sources to provide solutions to the prevailing environmental pollutions (global warming, industry-related pollution) and to dilemma in the energy supply, the high-temperature heat utilization system using the Modular High-temperature Gas-cooled Reactor (MHTGR) and the Gas Turbine Generator System are now under research and development. The gas-cooled reactor used for this power generation system has originally evolved from the research and development works in the U.S.A.[1 et al], showing its salient features of; ① inherent and passive safety characteristics (meltdown-proof), ② availability of high-temperature heat, ③ a high thermal efficiency and ④ extremely low radioactive releases, all of which combine to show a new promising reactor for the next generation nuclear power plant. Particularly, fuel particles of approximately 1mm encapsulated in ceramics, etc., are used as the reactor fuel to provide a containment function of radioactive releases within the fuel particles perse and the gas-cooled reactor is provided with its inherent and passive safety characteristics to maintain the safety of the reactor because the heat generated from the fuel can all be released in terms of the natural heat release alone from the surface of the metallic pressure vessel.

Mitsubishi Heavy Industries, Ltd. has been promoting the research and development of the MHTGR under cooperation with Japan Atomic Energy Research Institute and the other associated organizations over many years. Also, the company is the contractor of the HTTR (High Temperature Test Reactor) under development as a national project in Japan and has undertaken its design and construction works. Furthermore, we have been engaged in research and development works of both types of MHTGRs using the steam cycle and the gas turbine cycle and the energy utilization system involved in heat utilization and electric power generation. Among these research and development activities, this paper presents the conceptual designs of an experimental model unit (heat input 1200kWt) and a utility unit (450 MWt) for the gas turbine generator system including the outline of our study results of the major components used, the fundamental characteristics of these main components as well as the developmental issues. It should be noted that the fundamental concept and configuration of the MHTGR-GT have previously been developed principally by General Atomics.

2. Working Fluids

2.1 Selection of the working fluids

Several candidate working fluids have been investigated as the working fluid for the MHTGR and their physical properties are shown in Table 1 and Fig.1[2]. In the present study, inert helium was selected from these candidate fluids because of its excellent heat transfer property.

Table 1 Physical Properties of Working Fluids

Gas	Molecular Equation	Atomic Number	Molecular Weight	Gas Constant (kJ/kg K)	Thermal Conductivity (W/mK)	Density (kg/m^3)	Specific Heat at 0C (kJ/kgK)		S.H. Ratio
							Cp	Cv	Cp/Cv
Helium	He	1	4.0026	2.0772	0.1462	0.17850	5.19	3.116	1.66
Argon	Ar	1	39.498	0.20813	0.0163	1.783771	0.522	0.312	1.66
Hydrogen	H_2	2	2.0159	4.1244	0.1683	0.089885	14.188	10.06	1.409
Oxygen	O_2	2	31.9988	0.25983	0.0266	1.42900	0.917	0.655	1.399
Nitrogen	N_2	2	28.0134	0.29680	0.0242	1.25046	1.041	0.743	1.400
Air	—	—	28.964	0.28706	0.0242	1.29304	1.006	0.718	1.402
Steam	H_2O	3	18.0153	0.46151	—	—	—	—	—
Carbon Dioxide	CO_2	3	44.01	0.18892	0.0146	1.97700	0.826	0.631	1.301

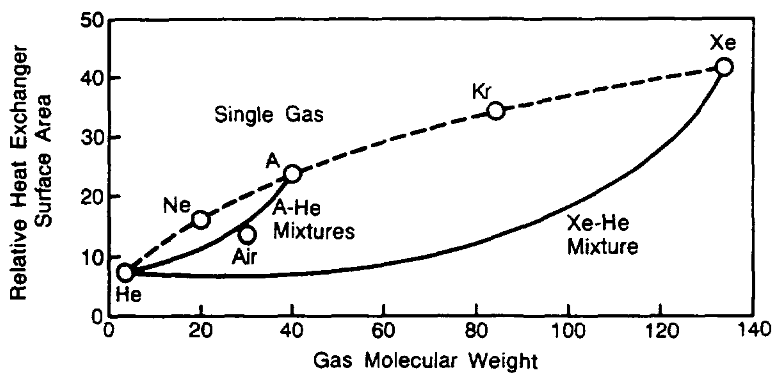


Fig. 1 Relative Heat Exchanger Surface Area vs. Various Gases (By courtesy of Airesearch)

2.2 Features of helium

As compared with the other working fluids listed in Table 1, helium has physical properties; ① a large thermal conductivity, ② a large specific heat, ③ a large gas constant, ④ a small molecular weight and ⑤ a small gas density. In accordance with such physical properties, helium has the features of its working fluid; ① compact design of heat exchangers used, ② heat drop (output) can be large at a small temperature difference, ③ a large velocity is available at a small pressure ratio, ④ it leaks even through a micro pore and ⑤ it has a large volume per unit mass.

In the open cycle gas turbine, its size must be increased when the working fluid has a small gas density whereas in the closed cycle gas turbine, the turbine size can be reduced by elevating the base pressure, utilizing the excellent features of helium. Comparison of helium and air relative to their features of working fluids for the gas turbine is shown in Figs.2-1 to 2-2 (refer to Table 1 together). In the case of a pressure ratio 2.8, the theoretical velocity of the helium gas is approximately 2.62 times that of air and its dimensionless flow rate is 0.165 times as great. Since the output is proportional to the square of the theoretical velocity and to the product of the flow rate, the output from the helium turbine is 1.13 times the case using air under the same working conditions. Because helium has a small molecular weight leading to its leakage through a micro-pore, its leakage is, in general, assumed to be greater than those of the other fluids, but as shown in Fig.2-2, its leak mass flow is approximately one-sevenths of the air leakage.

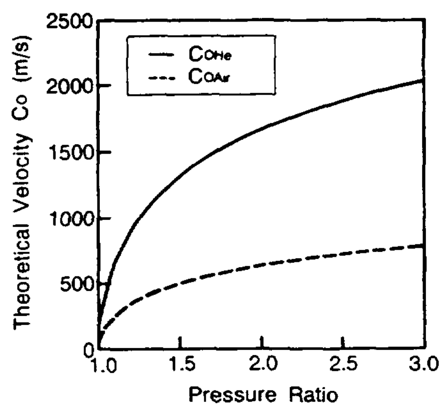


Fig. 2-1 Theoretical Velocity of Helium Gas and Air

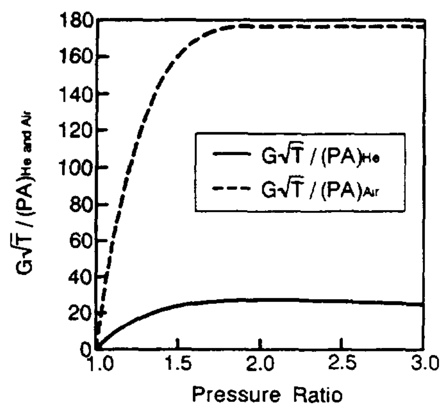


Fig. 2-2 Non-dimensional Flow Rate of Helium and Air

2.3 Features of the turbomachine using helium as its working fluid

In the open cycle gas turbine, the turbine inlet temperature has been set to an elevated point of 1350℃ and the pressure ratio to approximately 30 to improve the thermal efficiency. When this fact is considered, the output from the open cycle gas turbine can be approximately 3.53 times the output from the closed cycle helium gas turbine under the same base pressure condition. Meanwhile, the output from the closed cycle helium turbine can be raised to a level comparable to the output from the open cycle gas turbine by setting the base pressure at more than about 3.53 times the atmospheric pressure.

The base pressure (compressor inlet pressure) of the gas turbine for the MHTGR has been set at approximately 25 ata (2.55 MPa) and its output is approximately seven times the output from the open cycle gas turbine. Although the specific heat of helium is approximately five times that of air, its pressure ratio is well below the level of this difference and the volumetric change of the working fluid is significantly small leading to small changes of its flow paths. Due to this fact, the change of the blading height can be small from the first to the last stages of compressors and turbines.

3. Outlet Temperatures of the Reactor

The MHTGR has a negative reaction property that its reactivity is reduced as the temperature increases. If the internal gas should be released out due to some accidental failures, resulting in a pressure drop and an elevated temperature of the fuel particles, the reactivity would drop, causing the thermal output to be reduced to several percent of the design level. The MHTGR has been designed to provide the so-called inherent and passive safety characteristics; the heat capacity generated during this failure can all be released out of the outer wall of the pressure vessel. The outlet temperature (turbine inlet temperature) of the MHTGR has been set at 850℃ such that the maximum temperature of the fuel during an accident can always be maintained below 1600℃ which can allow the MHTGR. In this paper, study has been performed on the basis of this temperature 850℃. However, the most recent researches show a potential for an elevation of the maximum temperature due to improved fuel particles, re-adjustments of the bypass flows in the reactor and an improvement of the cooling method for the reactor vessel, etc.

4. Cycle Calculations and Component Efficiencies

4.1 Cycle efficiency calculations

The calculated cycle efficiencies are shown in Fig.3. When a recuperater is used, the cycle efficiency increases at the lower side of the pressure ratio as the recuperater exchanging heat capacity increases and as the system pressure loss is reduced. As seen from the theoretical value (component efficiency 100%) of the recuperated cycle, this is because a raised pressure ratio would reduce the exhaust gas temperature resulting in the smaller heat recovery by the recuperater. The efficiency of the recuperated/intercooled cycle is above the theoretical efficiency of the simple cycle in a pressure ratio range below 3.5. Because the efficiency of the

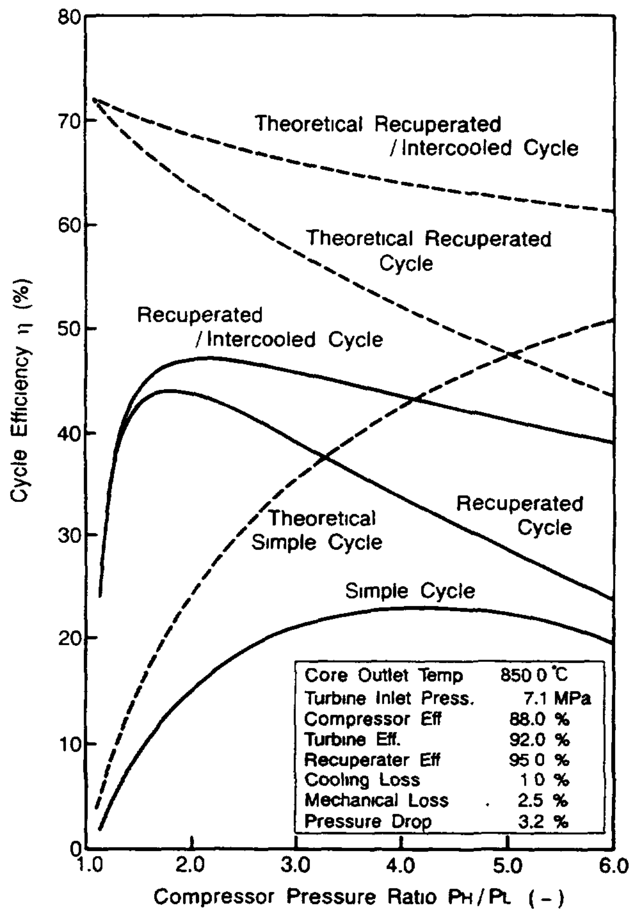


Fig. 3 Cycle Efficiency of Direct Cycle

feasible heat exchanger could be estimated at 95% and the system pressure loss at around 3%, a pressure ratio of 2.8 was selected from Fig.3. Assuming that efficiencies of the components are identical, the efficiency of the simple cycle is approximately 20% and that of the recuperated/intercooled cycle is 46%, respectively at the pressure ratio of 2.8.

4.2 Effects of the component efficiencies and pressure loss on the cycle efficiency

The effects of efficiencies of the main components (turbine, compressor, recuperater, intercooler, etc.) of the gas turbine, the system pressure loss and compressor and turbine inlet temperature on the overall cycle efficiency are shown in Fig.4. If the turbine efficiency, the compressor efficiency or the recuperater efficiency changes by 1%, the cycle efficiency also changes by approximately 0.5%. If the turbine inlet temperature or the pressure loss changes by 1%, the cycle efficiency changes by 0.3% or by 1%, respectively. Because the pressure loss shows the most serious influence on the cycle efficiency, pressure losses through pipings, etc., are needed to be minimized.

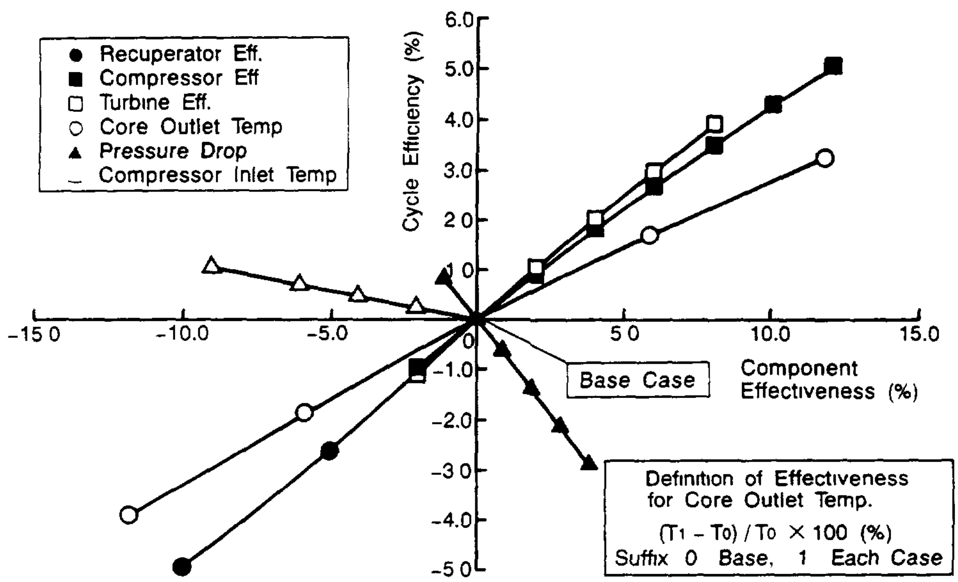


Fig. 4 Effect of Components Effectiveness on Cycle Efficiency (Direct Recuperated / Intercooled Cycle)

5. Type Selection

5.1 Direct cycle and indirect cycle

The direct cycle uses the working fluid in the reactor to directly drive the turbine while the indirect cycle heats another working fluid through heat exchanger to utilize this heated fluid to drive the turbine. The direct cycle system can have a high cycle efficiency realized because its system can be simplified with a high turbine inlet temperature being made available. The indirect cycle system is supposed to show a higher safety reliability because it can minimize a potential for pollution of the turbomachine by radioactive substances due to use of another working fluid through the heat exchangers, and the freedom of its design work can be higher because the working fluid for the turbomachine can be freely chosen. Also, in utilization of the high-temperature, the direct cycle must take the type identical to the indirect cycle type. Both have advantages of their own and the view of the system advantages can vary, depending on what factors one should place emphasis on.

Currently it is shown that the fuel particle has a high reliability and that the plant safety can be assured even with the direct cycle system, and since the potential feasibility of the indirect cycle system could be high if the direct cycle system is realized, the conceptual design of the direct cycle system has been conducted in our present research.

5.2 Single-shaft type and twin-shaft type

The twin-shaft system is composed of two independent shafts; the power turbine directly connected to the generator and the turbines to drive the compressors. Because this selection allows a free selection of the rotational speed of the drive turbine for the compressor, elevated efficiencies of the compressor and its drive turbine can be obtained. However, it requires the installation of a motor to drive the compressor during startup and there is a problem of the difficult control of its overspeed for various assumed accidents. The MHTGR has a helium tank and the compressors can also be started by utilizing the high-pressure helium gas stored in this tank[3]. In the single-shaft system, the rotational speeds of the turbine and the compressor are needed to set to the same speed of the generator, which enlarges outside diameters of the turbine and the compressor, reducing the blade height which leads to difficulties in the design of efficiency improvements. Also, assurance of the rotational shaft stability can be a major problem because the shaft length is extended with the natural frequency of the shafting system being low.

The aforementioned problems have been examined by performing outline designs of two cases of heat outputs 450 MWt and 1200 kWt. In the case of the heat output 450 MWt, the efficiency difference in the compressor between the twin-shaft system and the single-shaft system is 1 to 1.5% and the efficiency difference in the turbine is around 0.5% and the efficiency improvement and the compact design are possible, but the benefit of the twin-shaft system is small in terms of the shaft system vibration.

In the case of the heat output 1200 kWt, the single-shaft system shows a poor efficiency because of a reduced blade height, and because the shaft becomes very small and long, it is determined that maintaining the shaft system stability can be difficult.

6. Design Requirements and Specification

The design requirements have been decided based on our study results of the cycle efficiency calculations, etc., as described earlier and on the study results in U.S.A.[4 et al.]. The results of these design requirements are also applicable to the cases slightly deviating from the given conditions. The specifications of the respective components derived from our conceptual design are compared with another data[4, 5, 6, 7] in Table 2. Both data are nearly identical except for the details of estimate and distribution of losses.

7. Design of the Turbomachine

The fundamental technologies required for the design of the turbomachine have already been proven technologies of aerospace and industrial gas turbines and the turbomachine can be designed using those technologies. However, the gas turbine for the MHTGR requires a design simultaneously provided with the lightweight design for aerospace application and the long-term endurance capability for utility service, and there has been no such design experience, which can be deemed a major developmental issue. The turbomachine should most adequately be designed and its periodical inspection interval should be established by examining and analyzing official permit and authorization laws in various countries, regulations regarding the periodical inspection, etc., proven lives of the components for industrial and aerospace gas turbines, and the factors to limit their lives, etc.

7.1 Turbomachine Type

With regard to the turbomachine type, different types of the compressor and their advantages or disadvantages are shown in addition to the differences between the single-shaft and twin-shaft types as described earlier. Although earlier conceptual designs have dealt with the axial flow turbine and the axial flow compressor, the axial flow compressor has an extended shaft length and a large number of its stages, which increases its costs. Accordingly, our conceptual design has also been done on the case employing the centrifugal compressor where the number of required stages and the manufacturing costs can be curtailed and piping to the intercooler is more easily made available. Although the common practice of the twin-shaft type

Table 2 Specification of MHTGR-GT

System			MHI-GT	GT-MHR	MHTGR-GT
Core	Thermal Power	MWt	450.0	600.0	450.0
	Coolant Pressure	MPa	7.07	7.01	7.03
	Flow Rate	kg/s	244.0	328.0	243.0
	Outlet Temp.	℃	850.0	850.0	850.0
	Inlet Temp.	℃	494.0	497.0	494.0
	Gas Turbine Power	MWe	208.0	301.0	228.0
	Net Cycle Efficiency	%	46.3	47.2	47.7
Effectiveness	Compressor	%	88.0	90.0	90.0
	Turbine	%	92.0	92.0	92.0
	Recuperator	%	95.0	95.0	95.0
	Mechanical Loss*1	%	2.5	2.5	2.5
	Cooling Loss*2	%	1.0	0.3	0.3
	Generator Eff.	%	98.0	97.5	97.5
	Pressure Loss*3	%	3.2	4.4	3.5
Turbine	Inlet Temp.	℃	850.0	850.0	850.0
	Outlet Temp.	℃	515.8	517.8	514.1
	Pressure Ratio	—	2.66	2.64	2.67
Compressor	Outlet Temp.	℃	35.0	33.0	33.0
	Inlet Temp.	℃	86.5	110.8	110.8
	Pressure Ratio	—	2.80	2.80	2.80
Recuperator	Heat Load	MWt	516.0	658.0	484.0
	Hot Inlet Temp.	℃	515.8	517.8	514.1
	Hot Outlet Temp.	℃	108.0	131.1	130.9
	Cold Inlet Temp.	℃	86.5	110.8	110.8
	Cold Outlet Temp.	℃	494.3	497.5	493.9
Precooler	Heat Load	MWt	92.0	167.0	124.0
	Inlet Temp.	℃	108.0	131.1	130.9
	Outlet Temp.	℃	35.0	33.0	33.0
Intercooler	No.	—	2	1	1
	Heat Load	MWt	65.2	65.2	41.4
	Inlet Temp.	℃	86.5	110.8	110.8
	Outlet Temp.	℃	35.0	33.0	33.0

Note *1 Ratio of House Load to Gas Turbine Power for GT-MHR
*2 Ratio of Generator Windage Loss to Gas Turbine Power for GT-MHR
*3 Total Pressure Loss to Turbine Inlet Pressure

is to use a common shaft for the compressor and its drive turbine and use the power turbine to drive the generator, the use of a common shaft for the low pressure stage of the compressor and the power turbine has been proposed to make available the compressor startup by the generator and to preclude its overspeed, and the latter method has been studied in this paper.

7.2 Compressors

The results of our conceptual designs of helium gas turbines employed axial flow and centrifugal compressors are compared with the conventional designs in Table 3 and in Fig.5. In our design of the compressors, the axial flow type has been employed for the single-shaft type and the axial flow and centrifugal types for the twin-shaft type. In all the cases, the peripheral speed has been raised and the number of compressor stages reduced as compared with the conventional designs. The case of the single-shaft type has been divided to the H.P. group and the L.P. group with an intercooler provided between the two groups, consisting of 13 stages of axial flow compressors, respectively, for the H.P. group and the L.P. group. In the case of the twin-shaft type, the number of stages has been reduced to five stages, one-fifth of that for the axial flow type, by employing the centrifugal type. The outside diameter of the impeller for the centrifugal compressor is as large as approximately 1.6m for the L.P. stage and the maximum peripheral speed is as high as approximately 500 m/s for the H.P. stage, and therefore, stainless steel or titanium alloy is used for the impeller material. Also, the twin-shaft type uses the shaft configuration of the shaft for the L.P. stage penetrating through the shaft for the H.P. stage such that the turbomachine can be started by the generator which drives the L.P. compressor at the startup to raise the pressure in the system.

Table 3 Turbomachine Salient Features

System	Nuclear Gas Turbine			Power Generation Unit	
				Heavy Duty Industrial GT	Aeroderivative Gas Turbine
Plant	MHI-MHTGR-GT		GR-MHR	MS9001F (GE)	LM6000 (GE)
Power (MWe)	208		286	226	42
Working Fluid	Helium		Helium	Air	Air
Thermodynamic Cycle	Recuperated and Intercooled		Recuperated and Intercooled	Simple Cycle	Simple Cycle
Turbine Inlet Temp. (℃)	850		850	1,288	1,243
Compressor Pressure Ratio	2.8		2 8	15	30
Mass Flow Rate (kg/sec)	244		320	613	123
Machine Orientation	Vertical		Vertical	Horizontal	Horizontal
Shafting Type	Twin	Single	Single	Single	Twin
Overall Length (m)	7.9	8.9	13	14.5	4.5
Overall Diameter (m)	2.8	2.8	2 8	4 8	2.5
Overall GT Weight (kg)	65,000	70,000	82,000	300,000	5,600
Rotational Speed (rpm)	3,600/6,000	3,600	3,600	3,000	10,225/3,000
Compressor Number of Stages Max. Tip Diameter (mm)	Centrifugal 3HP+2LP 1,762	Axial 13LP+13HP 1,580	 14LP + 19Hp 1,683	 18 2,515	 5LP + 14HP 1,372/737
Turbine Number of Stages Max. Tip Diameter (mm) Tip Speed (m/sec) Blade Cooling	 2HP+5LP 1,905 506/359 Uncooled	 5 2,500 471 Uncooled	 11 1,778 335 Uncooled	 3 3,251 510 Cooled	 2HP + 5LP 889/1,321 476/249 Cooled
Bearing Type	Active Magnetic		Active Magnetic	Oil Lubricated	Oil Lubricated
Number of Bearings (Thrust/Journal)	2/4	1/4	1/4	1/2	1/6

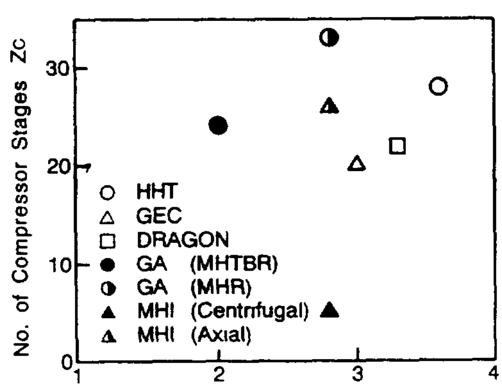


Fig. 5 Comparison of Number of Compressor Stages for Direct Cycle

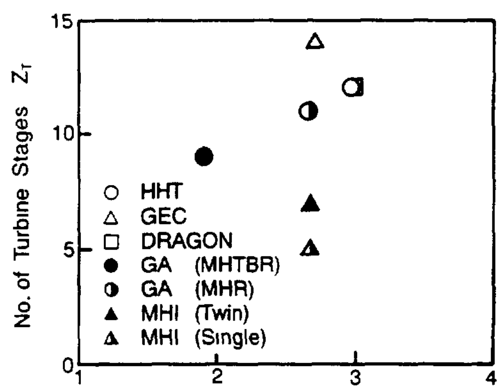


Fig. 6 Comparison of Number of Turbine Stages for Direct Cycle

7.3 Turbines

The results of our conceptual design of the turbine are compared with the conventional designs in Table 3 in Fig.6. The number of turbine stages is five stages for the single-shaft type and seven stages for the twin-shaft type. For the single-shaft type, the increase of the shaft length has been avoided, aimed at a high-load design. The twin-shaft type which shows allowance for the load and the number of stages is found more advantageous over the single-shaft type in terms of performance. The outside diameter of the turbine is 2.5m for the single-shaft type and 1.9m for the twin-shaft type. In either of the types, the gas temperature at the moving blade inlet is 850℃ or less which is below the heat-resistant temperature of the blade material, eliminating the necessity of cooling. For the moving blade, a wide-chord blade to provide a reduced number of blades and improve its vibration-resistant strength and a lightweight hollow blade to alleviate stresses to the disc and the connection between the disc and the blade has been employed. Also, because the blade tip clearance is increased when a back-up bearing is provided accompanied with the employment of the magnetic bearing, the moving blade is provided with shround and tip-fins to minimize leakage from the tip clearance. The disc has a large diameter and the heat-resistant temperature of the material suited to such a large disc is approximately 600℃, which requires the disc to be cooled. The performance drop due to this cooling is estimated to be 0.5 to 1%.

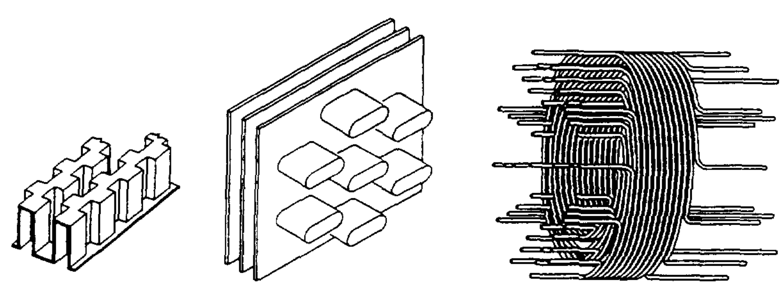
8. Heat Exchangers

Heat exchangers required for the gas turbine are a recuperator, a preheater and an intercooler, and their high efficiency performances and compact designs are demanded. Table 4 lists specifications of these heat exchangers and Fig.7 shows configurations of the heat exchanger elements.

Table 4 Heat Exchanger Salient Feature

Unit	Recuperator			Intercooler / Precooler	
	MHI	GT-MHR	Industrial GT	MHI	GT-MHR
Gas Turbine Power (MWe)	208	286	10 to 60	204	286
Working Fluid (High/Low)	He/He	He/He	Air/Gas	He/Water	He/Water
Unit Thermal Rating (MW)	515	630	15 to 30	65.2/92	131/164
Exchanger Type	Plate-Fin	Plate-Fin	Plate-Fin	Helical Coil	Tubular
Number of Modules	8	6	2 to 4	2/1	—
Flow Rate (kg/sec)	244	320	45 to 240	244	320
Hot Gas Inlet Temp. (K)	516	510	519 to 575	87/108	112/131
Pressure Difference (MPa)	4.5	4.6	0.9	3.6/4.5	4/2.3
Effectiveness	0.95	0.95	0.84 to 0.89	0.83/0.84	0.93/0.95
Overall Pressure Loss (%)	1.6	2.0	3.5 to 4	0.2/0.2	0.47/0.75
Typical Surface Density(m/m)	714	1,906	624	488/295	500
Heat Transfer Coeff. (W/mK)	LP : 1,857 HP : 1,876	LP : 2,555 HP : 3,120	LP : 115 HP : 425	He :1,780/585 Wa: 8600/7230	He : 1,250 Water:11,000
Thermal Density (MW/m)	8.9	17	1	4.0/0.57	3.6
Typical Flux (W/cm)	1.5	2.5	0.1	0.82/0.2	0.75
Material	316 StSt	316 StSt	409 StSt	1/2Cr-1/2Mo	1/2Cr-1/2Mo

*1 : Heat Transfer Area of Hot Gas Side to Total Recuperator Heat Transfer Section Volume
*2 : Typical Specification of First Stage Intercooler
*3 : Low-Fin Tube



(a) Recuperator (Plate-Fin)　(b) Intercooler (Wavy-Fin Flat Tube)　(c) Precooler (Helical Coil)

Fig. 7 Heat Exchanger Elements

8.1 Recuperator

For the recuperator, the plate-finned compact heat exchanger has been selected because a high heat transfer efficiency of 95% is demanded due to a large heat transfer of approximately 520 MW. Depending on the design specification, various fin types are selected for plate-finned heat exchangers and they are broadly used as small heat exchangers. For this system, the offset fin which can provide higher heat transfer efficiency has been employed. The offset fin is a small type of 1.9mm in height, 0.1mm in plate thickness and 1.0mm in fin-to-fin spacing, but this would pose no practical problems because the working fluid is high-purity, inert helium, which will eliminate concerns about oxidation or fouling.

8.2 Intercooler

The intercooler is used to reduce power consumption for the compressors and improve the efficiency of the regenerative gas turbine system. In this section, helium in the pressure rise process of the compressors is cooled by the cooling water running through a wavy-finned flat tube. The system design has been done on the basis of installation of two intercoolers, and Table 5 shows the specification of the first stage of the intercooler.

8.3 Precooler

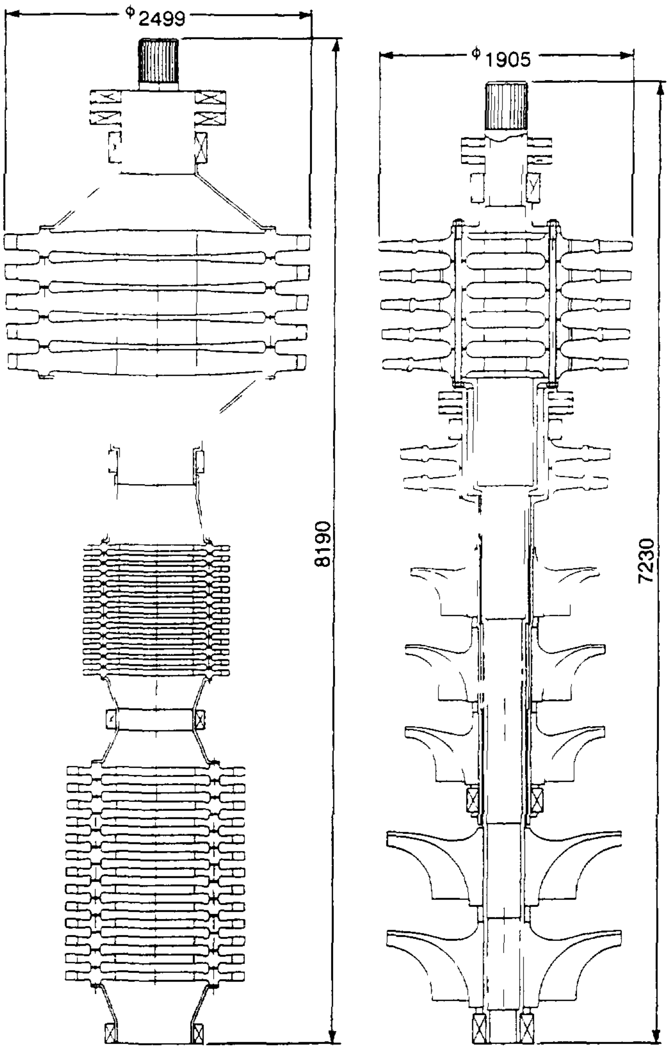
For the precooler, the helical coil type heat exchanger which has proven results for various machine type has been selected and the low-fin tube has been employed to reduce the height. If a plate-finned or fin-and-tube type heat exchanger is employed for the precooler, it can reduce the installation space for the precooler. This subject will be dealt with in the next stage of our research work.

9. Study on Shafting and Bearings

Fig.8 shows the arrangements of shafting and the bearing for the single-shaft type and the applied to the practical unit of 450 MW. There are several examples of the shafting arrangements of the twin-shaft type for the gas turbine, and the fundamental arrangement uses one shaft for the gas generator and the other for the power turbine, which can reduce the overall shafting length and allow the rotational speed of the compressor shaft to be freely selected, providing option for a higher efficiency, but this arrangement requires addition of a motor, etc., to drive the gas generator at startup. As shown in the figure, the twin-shaft type in this design uses the low-pressure compressor shaft inserted through the high-pressure shaft because a common shaft has been designed for the low-pressure compressor and the power turbine for the compressor to be driven by the generator.

9.1 Shafting

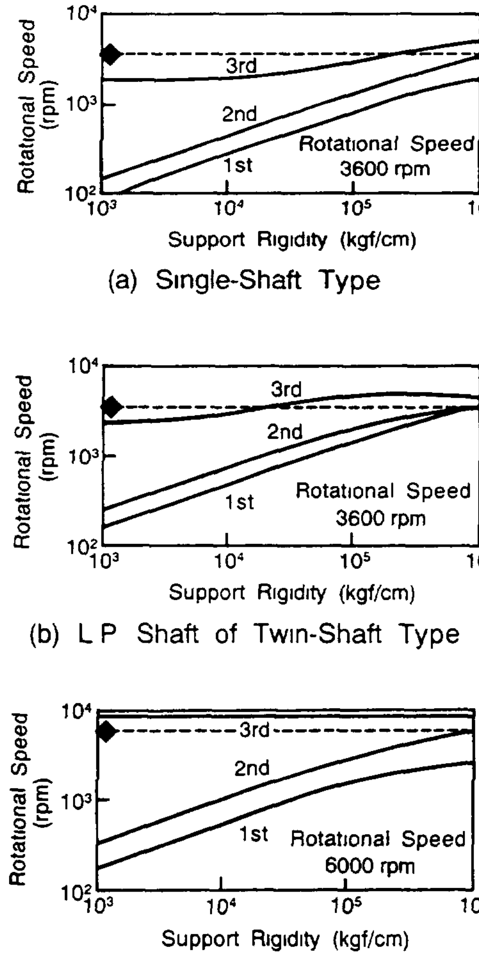
Fig.9 shows the calculated results of the rotational shaft vibrations. The first mode is the parallel mode, the second X mode and the third the bending mode. As shown in Fig.9, the designed rotational speeds of the L.P. shaftings of the single-shaft type and the twin-shaft type are in a region of rotational speeds higher than the third mode and it is difficult to exceed the natural frequency of the third mode when oil lubricated bearings are used, but the natural frequency of the third mode can be exceeded if the magnetic bearing system which can control the shaft vibration is employed. However, it is desirable to maintain the designed rotational speed below the natural frequency of the third mode, and hence, it is needed to devise the elevation of the natural frequency by reducing the shaft lengths of the turbine and the compressor in order to raise rigidity of the shaft bending. For the two-shaft type, the L.P. shaft is placed through the hollow H.P. shaft because the startup using the generator has been made available as already described. This causes the length of the L.P. shaft to be extended, resulting in a reduced natural frequency. If a common shaft were used for the gas generator turbine and the compressor with the power turbine separated from this common shaft and if the high-pressure helium stored in the helium tank were used for the startup, the length of the L.P. shaft would be reduced, causing the natural frequency of the third mode to exceed the designed rotational speed. In this paper, the most serious problem of shafting has been examined.



(a) Single-Shaft, Axial Compressor

(b) Twin-Shaft, Centrifugal Compressor

Fig 8 Helium Turbine Rotor



(a) Single-Shaft Type

(b) L P Shaft of Twin-Shaft Type

(c) H P Shaft of Twin-Shaft Type

Fig 9 Calculated Results of Rotational Shaft Vibration

9.2 Magnetic bearing

The magnetic bearing is provided with the following features; ① no lubricant is needed, ② a small loss, ③ the shaft vibration is controllable and ④ the load capacity does not depend on the rotational speed. For the closed cycle gas turbine, ① the gas bearing using the working fluid as a lubricant and ② the non-lubricant magnetic bearing can be pointed out as candidate bearings to avoid ingress of the lubricant to the working fluid. The gas bearing poses problems of the cooling method and small deformations of the bearing and the shaft because the gas bearing has a small load capacity and a large loss. Meanwhile, the thrust bearing of the magnetic bearing system can support a large load because it has a small loss and provides a large support area. In the journal bearing, control of the shaft vibrations from the low to high modes is available. Therefore, it is best suited for the bearing system for the vertical shafting arrangement of an extended shaft length where the thrust bearing supports an overall weight of the rotor.

The selection chart of the outside diameters of the magnetic thrust bearing is shown in Fig.10. Although the dimensions of the thrust and the journal bearings are over the range of proven experience, they can still be manufactured in terms of technical capability. However, the magnetic bearing is supposedly needed to be provided with ball bearings as the backup

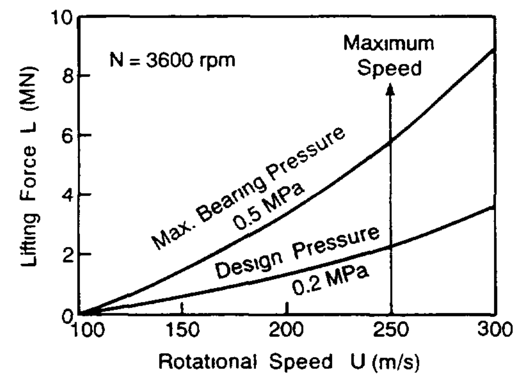


Fig. 10 Lifting Force of Magnetic Thrust Bearing Relative to Peripheral Speed

bearings at the shaft ends in order to avoid contacts at startup and stop and a large clearance must be provided to avoid the contact of this bearing with the shaft during the normal operation. Accordingly, it becomes necessary to provide large clearances for turbine and compressor tips, which reduces their performances. It is desirable to use a backup power supply as an alternative to the backup bearing and design alleviation of damage to the contact area between the shaft and the bearing and their developments are expected for.

10. Design of the Overall System Configuration

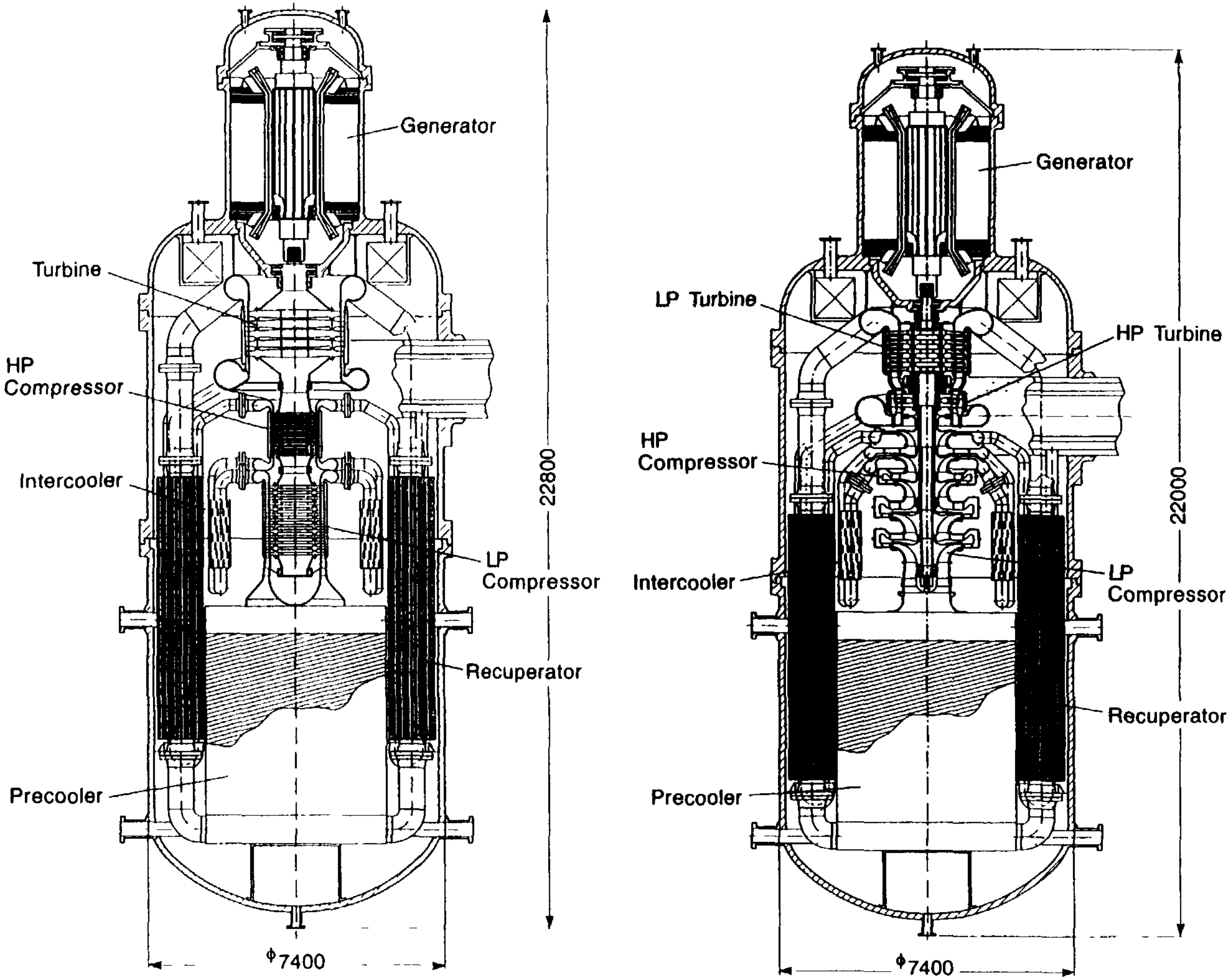
The structural sections of the utility unit and the experimental model unit as designed in this research are shown in Figs.11 and 12. For the utility unit, a configuration of the generator and the gas turbine contained in one module, the recuperator in one module, the intercooler in one module and the preheater in one module, has been employed to carry out the maintenance servicing and the periodical inspection after removing each module from the pressure vessel in order to minimize the required work in the vessel. Flanges will be provided between the respective modules to preclude gas leakage from between the modules. The flange connection method must take into account thermal expansions and deformations, etc., of the assemblies and components, and they will be examined in the stage of the detail design work.

The support structure for the gas turbine will be retained on a support plate between the generator and the turbine and will use the hanger type. Accordingly, the casing structure of the gas turbine is to have a rigidity required to support the shafting and to be of the self-containment structure which could contain fragments of a damaged blade if such a damage should happen.

For the assembly method of the modules, the flange connection or the insert type can be conceived. The former has a problem of the alleviation method of stresses due to deformations caused by thermal elongation, etc., while the latter has a problem of the sealing method. In addition, the arrangements of pipings and the components must be carefully designed, taking into considerations access of maintenance personnel to the vessels, the working space and the workability, etc.

11. Development Program

The main flow of this development program is shown in following table. The preliminary research work in the initial period in this program has already been shown in this paper and the overall system configuration and the outlined configurations of the respective components have conceptually designed, based on the existing technological information and MHI's own design database. In Phase 1 in the next step, the fundamental design of the utility unit will be performed and the component technologies and the design technology required for the development of the utility unit will be fulfilled. In Phase 2, verifications of performances



(a) Single-Shaft Type

(b) Twin-Shaft Type

Fig. 11 Power Conversion Modules

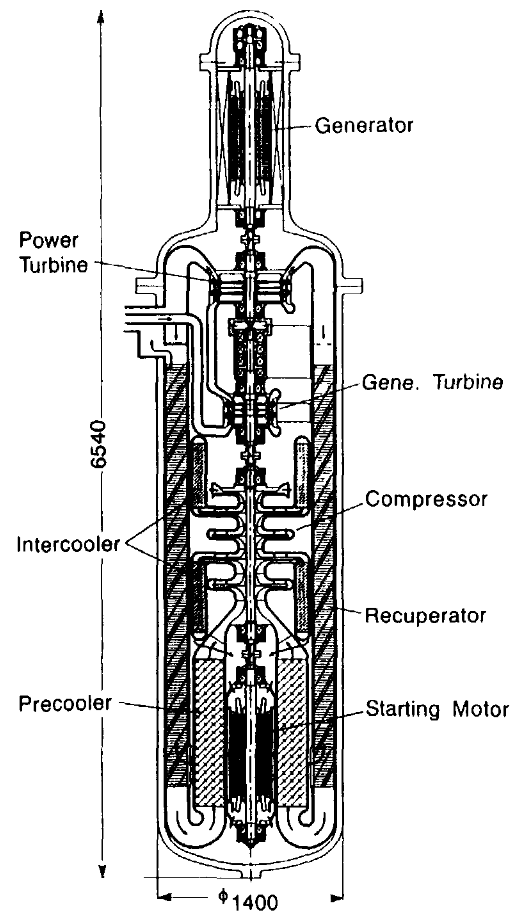


Fig. 12 1200 kWt Test Model Unit

and functions of the required components as well as the design of a demonstration unit will be carried out, ready to respond to the examination of the Safety Counsil, etc. In Phase 3, fabrication and construction of a demonstration plant will take place.

Development Program

Phase 1	Phase 2	Phase 3
· Fundamental Design of Utility Unit · Fulfillment of Component & Design Technologies	· Design of Demonstration Unit · Demonstration of Performances & Functions of Components	· Fabrication & Construction of Demonstration Unit

12. Conclusions

Assuming a full-scale development of MHTGR-GT, the authors have completed the conceptual design of the recuperated cycle intercooled helium gas turbine of the closed cycle on the basis of MHI's own technologies and various other information currently available. As a result, study results nearly identical to the results already obtained in the U.S.A. and other countries have been obtained and a potential for a more compact design has also been confirmed. Furthermore, developmental problems regarding the respective components and the total system have also been clarified and the development program up to the stage of practical use of this gas turbine has been summarized. In future, we intend to realize the Modular High-Temperature Gas-cooled Reactor in cooperation with Japan Atomic Energy Research Institute and other organizations concerned as well as manufacturers in this field both domestic and abroad.

ACKNOWLEDGEMENTS

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INVESTIGATION OF GT-ST COMBINED CYCLE IN HTR-10 REACTOR

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Abstract

In the Chinese HTR-10 project, two phases of the high temperature heat utilization are planned During the second phase, the reactor core outlet temperature is increased up to 950°C and a GT-ST combined cycle is planed to be constructed This paper presents the investigation results of GT-ST combined cycle in INET Two patterns of GT-ST cycles, i e a parallel GT-ST cycle and a series GT-ST cycle, are referred Based on the state-of-the-art technology, the current activities of HTR-10 GT-ST combined cycle focus on a parallel combined cycle in which GT and ST cycle are independently parallel in the secondary side

Keywords HTGR, Reactor, Gas Turbine

1. Introduction

There is currently an increased worldwide interest in the new generation of nuclear power plants which have extremely safe features, small size, technical and economical benefits Module High Temperature Gas Cooled Reactor(MHTGR) becomes new power supplier for the next century based on its excellent safety features, high gas temperature and high thermal efficiency by combining the MHTGR and gas turbine technology

MHTGR is a passively-safe nuclear reactor It is expressed as a large negative temperature coefficient which naturally shuts the reactor down during heat up accident and is inherent safe in the reactivity transient The reactor low power density and huge heat absorbed capacity of the graphite structure assure that the decay heat will be dissipated passively by heat conduction and heat radiation even under the most severe accident condition MHTGR has also capability to produce very high helium outlet temperature up to 950 °C The combination of gas turbine and MHTGR (GT-MHTGR) represents the ultimate in safety and economy. The high temperature heat source produced by MHTGR can be used to provide high thermal efficiency The direct gas turbine (GT) cycle with MHTGR could enhance system thermal efficiency up to 50 % The high temperature helium coming from MHTGR drives the gas turbine cycle directly The application of compact plate-fin recuperator boosts the thermal efficiency

Considering the problems of the possible radioactivity deposition in the turbine blades in the direct GT-MHTGR, and the lower inlet helium temperature (250°C -300°C) in the indirect gas turbine cycle MHTGR, the indirect gas turbine and steam turbine combined cycles (GT-ST-MHTGR) were studied The GT-ST-MHTGR can be used to generate electricity more efficiently The reactor heat is transferred to the secondary loops by means of an Intermediate Heat eXchanger (IHX) and a Steam Generator (SG). In this paper. two patterns of GT-ST cycles, i.e. a parallel GT-ST cycle and a series GT-ST cycle, are referred. In the former pattern, IHX and SG are connected with the reactor primary loop orderly, the gas turbine cycle and the steam turbine cycle are arranged in parallel manner. In the latter pattern, IHX & the gas turbine cycle are arranged in the secondary loop, while the SG & the steam turbine cycle are arranged in the third loop. The system thermal efficiencies of a parallel GT-ST cycle and a series GT-ST cycle are 47% and 45.6 % respectively.

In China , studies are in progress to develop a standard MHTGR plant design. A 10MWth Test Module(HTR-10) for investigation of MHTGR is under construction at site of Institute of Nuclear Energy Technology of Tsinghua University (INET). There are two phases of the high temperature heat utilization in the project. In the first phase, the reactor core outlet temperature is 700°C and a conventional ST turbine cycle is used in the secondary loop. In the second phase, the reactor core outlet temperature is increased up to 950°C and a GT-ST combined cycle is planed to be constructed.

Based on the state-of-the-art technology, the current activities of HTR-10 GT-ST combined cycle focus on an independently parallel Combined Cycle in which GT and ST cycle are independently parallel in the secondary side. This selected configuration will make full use of the current MHTGR design with a relatively low core inlet temperature (i.e.,250~300°C), change smoothly from the ST cycle in the first phase to the GT-ST cycle in the second phase. A conventional helical IHX is used the system.

2. GT-ST combined cycle for a modular high temperature gas cooled reactor

Two kinds of indirect GT-ST cycle are investigated based on Siemens 200 MW pebble-bed modular high temperature gas cooled reactor(HTR-Module). Fig. 1 shows the HTR-Module and its primary loop.

2.1, Series GT-ST combined cycle

Figure 2 gives the series GT-ST-MHTGR system. Two units of the 200 MW HTR-Module reactors are used as power sources. In the second gas turbine cycle, nitrogen is used as working fluid. Table 1 gives the main thermodynamic parameters. The reactor heat is transferred to the secondary loops by means of an IHX and then transferred to the water in the third ST cycle. The gas turbine consists of high pressure turbine and lower pressure turbine. The high pressure turbine and compressor are mounted on a single shaft. The total net electric power is 182 MWe, including gas turbine output of 88 MWe and steam turbine output of 109MWe. The busbar efficiency is 45.6% . In order to optimize system, the effects of the system parameters on the thermal efficiency are investigated. Results show that increasing reactor outlet helium temperature will boost system efficiency. If it is less than 800°C, combined GT-ST cycle will lose its advantage.

2.2, Parallel GT-ST combined cycle

Figure 3 gives the parallel GT-ST-MHTGR system. The difference is that the GT cycle and ST are parallelly arranged in the second side. The high temperature helium is partly used for the GT cycle, and the rest for ST cycle. The thermodynamic data are shown in

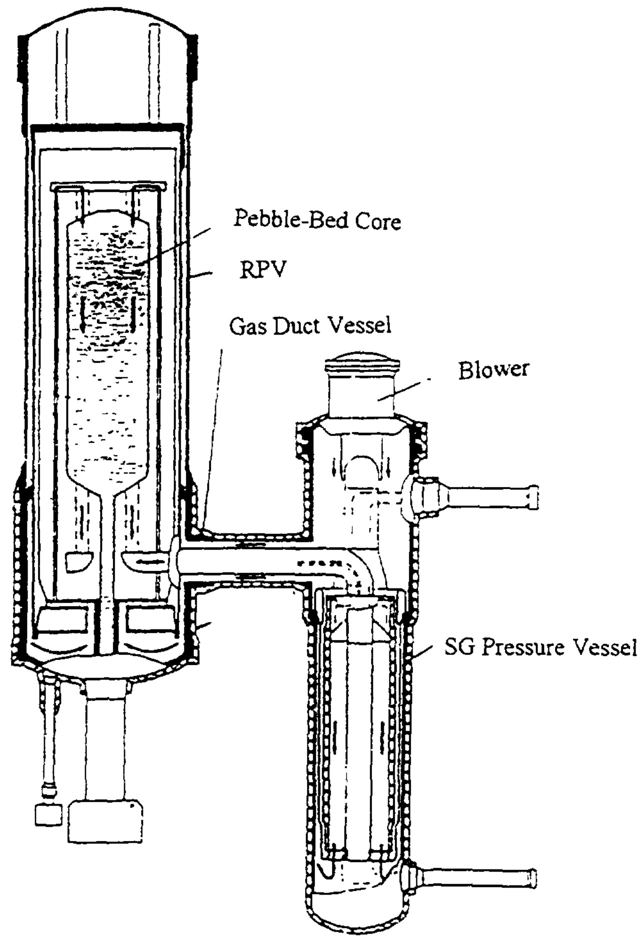
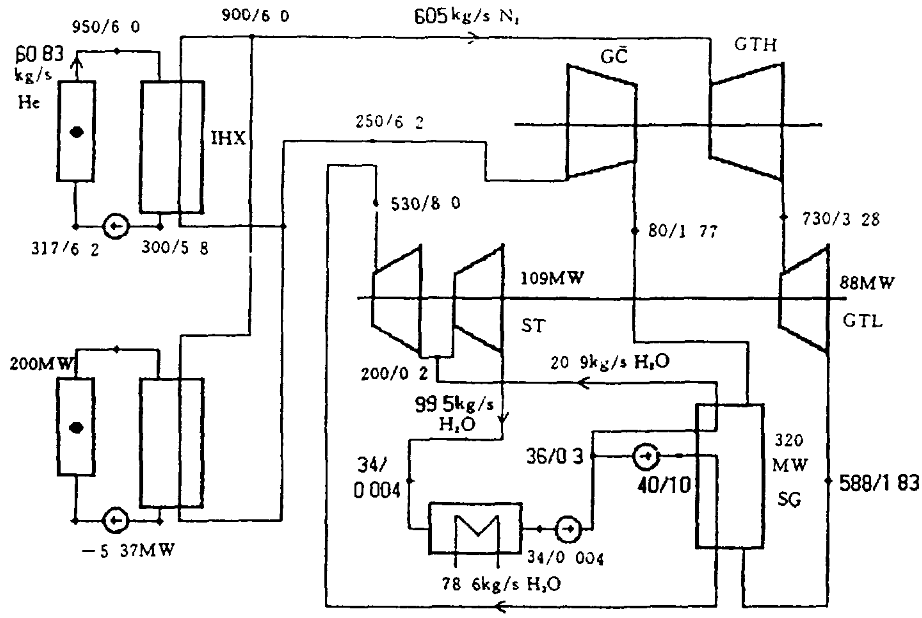


Fig.1 Flow Diagram of Steam Supply System of the 200MW HTR

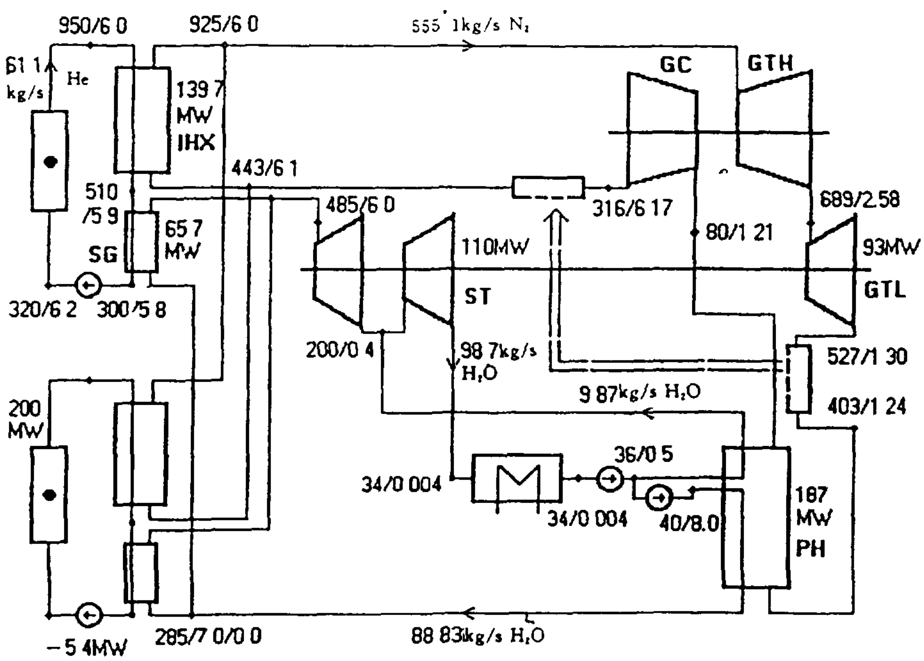
Table 1. Parameters of GT-ST combined cycle

parameter	*series*	*parallel*
Reactor thermal power (MWt)	2×200	2×200
Heat loss (MW)	5.6	5.6
Power output of gas turbine (MWe)	88	93
Power output of steam turbine (MWe)	109	110
Power for plant use (MWe)	10.9	11.0
Power loss of generator (MWe)	3.7	4.0
Net output power (MWe)	182.4	188.0
Net system efficiency (%)	45.6	47.0
Pressure ratio of gas turbine	3.5	5.1
Reactor power density (MW/m^3)	3.0	3.0
Reactor coolant temperature (°C,inlet/outlet)	317/950	320/950
Helium pressure (MPa)	6.0	6.0



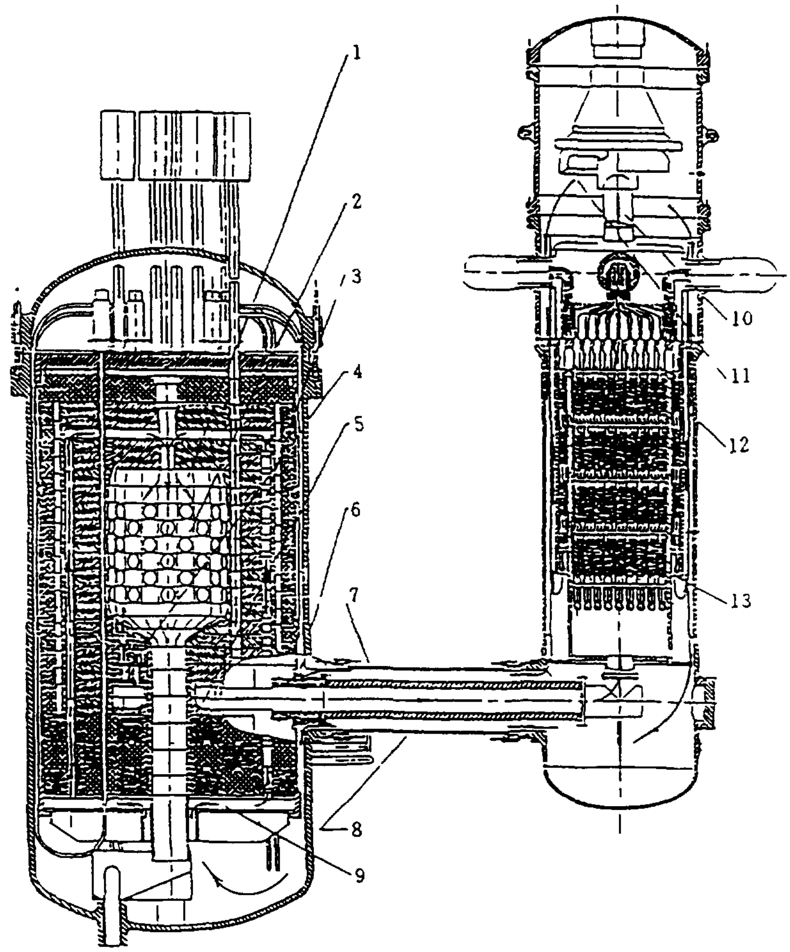
IHX-intermediate heat exchanger GTH-high pressure turbine
GTL-low pressure turbine GC-compressor
ST-steam turbine SG-steam generator

Fig 2 Configuration of Series GT/ST Combined Cycle for 200MW MHTR



IHX-intermediate heat exchanger GTH-high pressure turbine
GTL-low pressure turbine GC-compressor
ST-steam turbine SG-steam generator PH-preheater

Fig 3 Configuration of Parallel GT/ST Combined Cycle for 200MW MHTR



1	core cavity	7	hot gas duct pressure vessel
2	pebble bed	8	internal tube of hot gas duct
3	cold helium hole	9	core support structure
4	hot helium hole	10	inlet of He circulator
5	hot helium plenum	11	outlet of He circulator
6	annular space between reactor pressure vesseland the core sleeve	12	annular space between the pressure vessel and the sleeve
		13	steam generator

Fig 4 The Cross-Section of the HTR-10 Primary Circuit

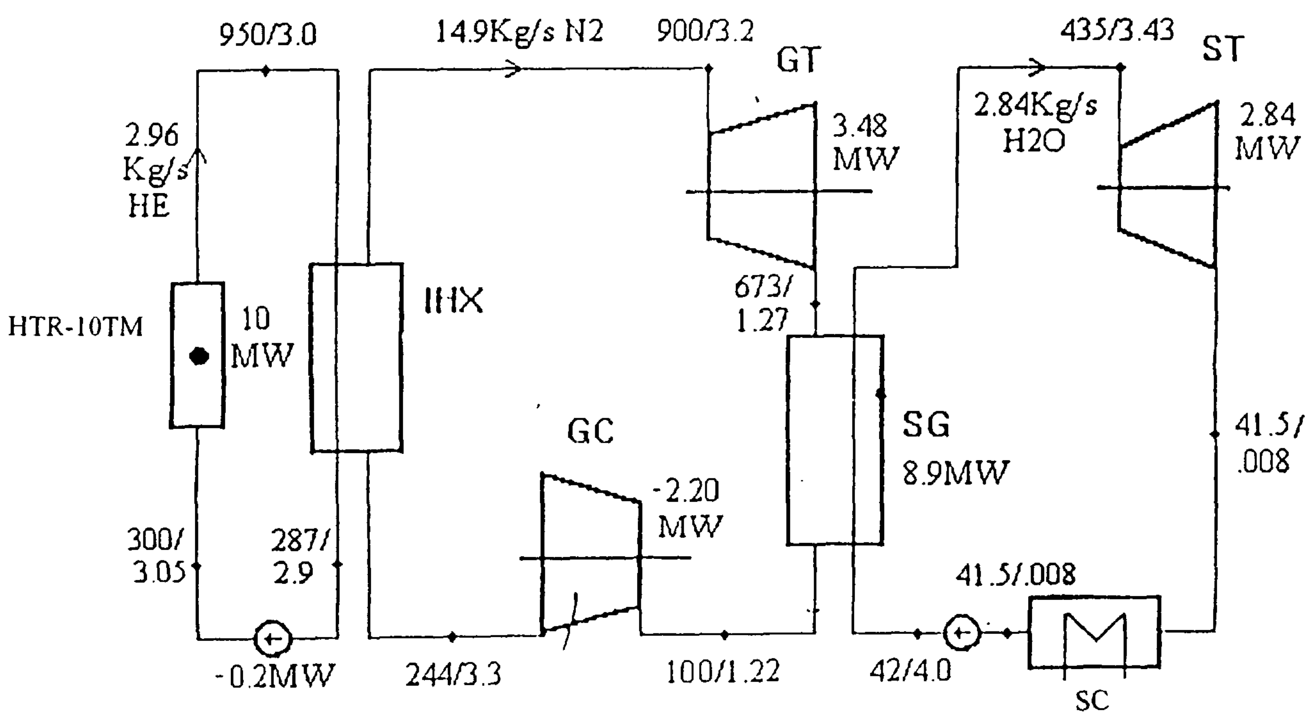
Table 1 The total net electric power is 188 MWe, including the gas turbine output 93 MWe and steam turbine output 110 MWe The busbar efficiency is 47% The higher power efficiency and smaller IHX are the advantages of the parallel GT-ST cycle But on the other side, the water ingress accident will be caused probably by the rupture of the steam generator tubes

3. GT-ST combined cycle of HTR-10

HTR-10 uses spherical fuel elements with ceramic coated particles, graphite as the core structure material and helium as the coolant The fuel elements are charged from the core top and removed from the core bottom via a discharge tube with multi-pass recycling Fig 4 shows the cross section and primary circuit of the HTR-10 The primary system consists of a reactor pressure vessel and IHX-SG vessel are arranged in reactor confinement cavity side by side and connected by a coaxial gas duct vessel. The reactor core, reflector, carbon brick, control rods, thermal shielding as well as absorber ball system are arranged in the reactor pressure vessel. The intermediate heat exchanger and steam generator are included in the IHX-SG vessel. The steam generator consists of 37 small coil pipes isolated with each other and locates in the outer annular region of IHX-SG vessel. The helical tubes of IHX are located in the central region. The helium flows through the reactor core in downward direction then through hot gas duct to IHX in upward direction and SG in downward way, then flows into blower. The cold helium is pumped from outer gas annular duct into the reactor. In the first phase of HTR-10, the reactor operates at a outlet temperature of 700 °C, only the SG cycle is used for power generation. In the second phase, the reactor outlet temperature reaches 900°C. Several kinds of GT-ST cycle are studied, including series GT-ST cycle, parallel GT-ST cycle and independent parallel combined cycle.

3.1, Ideal serial and parallel GT-ST combined cycle of HTR-10

Fig.5 and Fig.6 show the schematics of GT-ST combined cycle. In principle, the ideal serial and parallel GT-ST of HTR10 combined cycle at similar background as GT-ST of MHTGR ,i.e., the same core inlet and outlet temperature and similar system , also can get high thermal efficiency of 41% and 43% corresponding .The main parameters are shown in table 2.

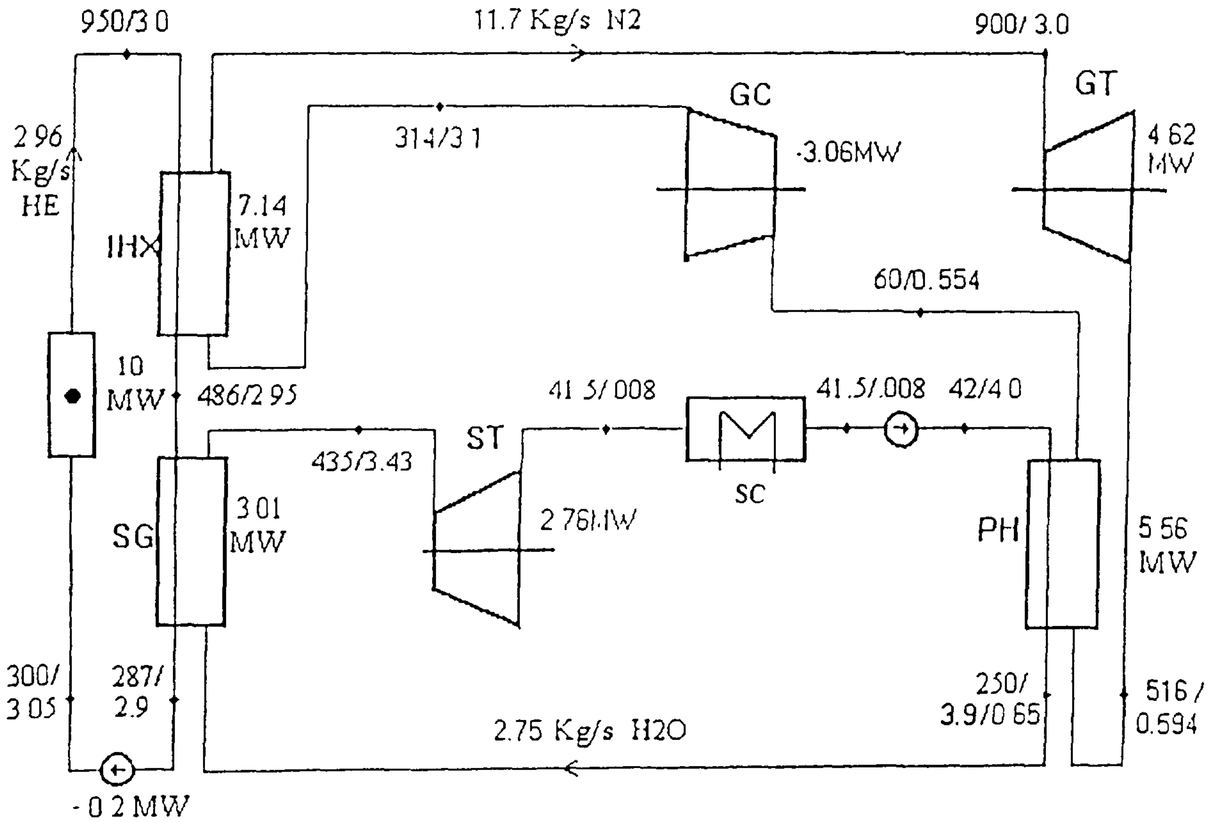


IHX-intermediate heat exchanger GT-gas turbine

GC-gas compressor SG-steam generator

ST-steam turbine SC-steam condenser

Fig.5 Configuration of Series GT/ST Combined Cycle for HTR-10TM



IHX-intermediate heat exchanger GT-gas turbine
GC-gas compressor SG-steam generator
ST-steam turbine SC-steam condenser PH-preheater

Fig.6 Configuration of Parallel GT/ST Combined Cycle for HTR-10TM

Table 2. Parameters of combined GT-ST cycle of HTR-10 system

parameter	*series*	*parallel*
Reactor thermal power (MW)	10	10
Thermal power of IHX (MW)	10	7.14
Thermal power of SG (MW)	8.9	3.0
Output power of gas turbine (MW)	3.48	4.62
Output power of steam turbine (MW)	2.84	2.76
Net output power (MW)	4.1	4.3
Net system efficiency (%)	41	43
Reactor power density (MW/m3)	2.0	2.0
Reactor coolant temperature (°C,inlet/outlet)	300/950	300/950
Helium pressure (MPa)	3.0	3.0

3.2, Independent parallel GT-ST combined cycle of HTR-10

Based on state-of-art technology and consideration of the purposes of both phases in the HTR-10 project, the independent parallel GT-ST combined cycle of HTR-10 is more suitable. It will make reasonable utilization of HTR-10 heat resource with a

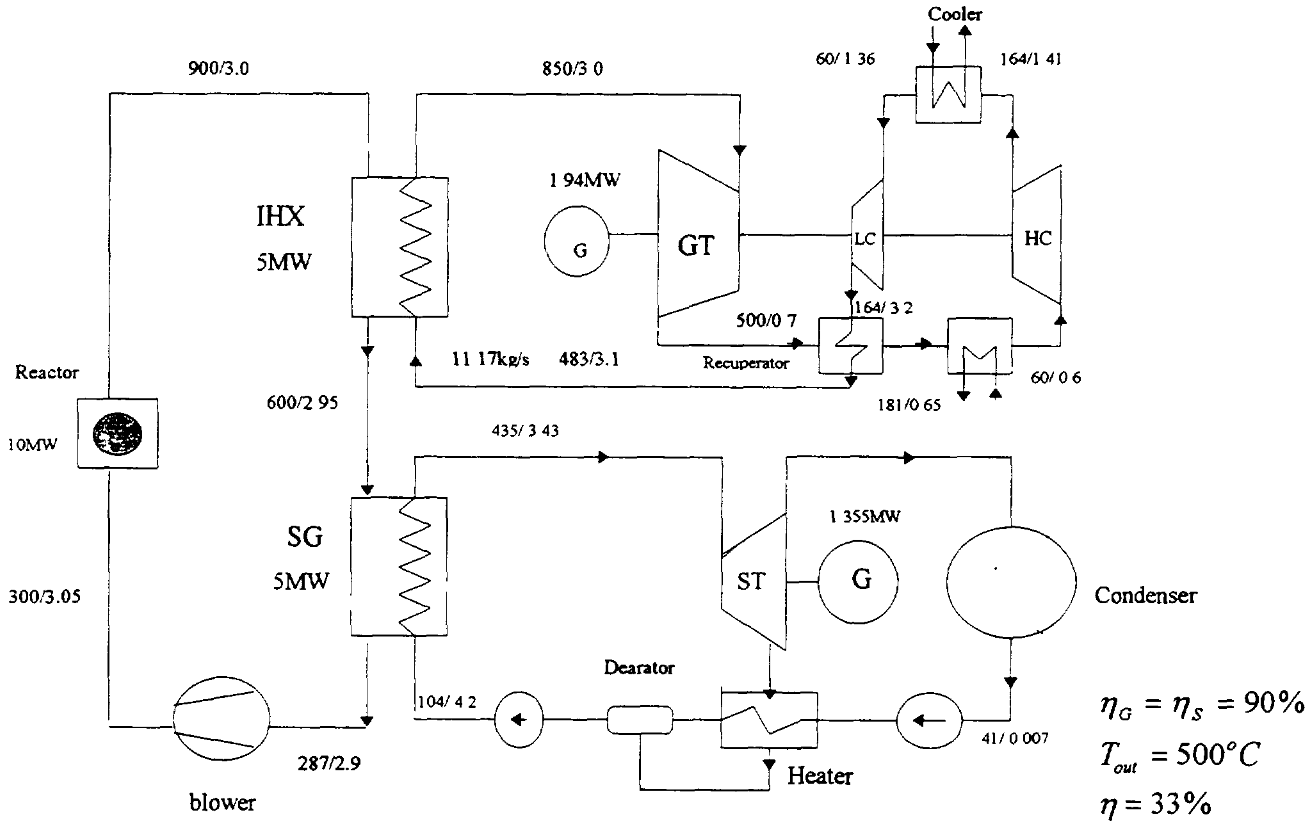


Fig.7 The HTR-10 GT-ST Combined Cycle

Table 3. Parameters of independent parallel combined GT-ST cycle of HTR-10 system

Contents	*1*	*2*	*3*	*4*	*5*	*6*	*7*	*8*
Reactor thermal power (MW)	10	10	10	10	10	10	10	10
Thermal power of IHX (MW)	5	5	5	5	5	5	5	5
Thermal power of SG (MW)	5	5	5	5	5	5	5	5
Working fluid of the GT cycle	He	He	He	He	N2	N2	N2	N2
Inner efficiency of the gas turbine and compressor (%)	90	86	90	86	90	86	90	86
Net power output of gas turbine (MW)	2.01	1.73	1.92	1.61	2.08	1.78	1.94	1.63
Power output of steam turbine (MW)	1.36	1.36	1.36	1.36	1.36	1.36	1.36	1.36
Net power output (MW)	3.36	3.08	3.27	2.96	3.42	3.14	3.30	2.98
Net system efficiency (%)	33.6	30.8	32.7	29.6	34.2	31.4	33.0	29.8
Reactor coolant temperature (°C,inlet/ outlet)	300/ 900	300/ 900	300/ 900	300/ 900	300/ 900	300/ 900	300/ 900	300/ 900
GT Nitrogen temperature (°C,inlet/outlet)	850/ 550	850/ 550	850/ 500	850/ 500	850/ 550	850/ 550	850/ 500	850/ 500
Pressure ratio of GT	2.36	2.46	2.80	2.94	3.33	3.53	4.29	4.57
Pressure ratio of HC & LC	1.68	1.71	1.75	1.89	2.09	2.09	2.35	2.43
Helium pressure (MPa)	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0

relatively low core inlet temperature (i.e.,250-300°C). The system power efficiency reaches acceptable value of 33%. Fig.7 gives the schematic of independent parallel GT-ST combined cycle. The main parameters are listed in table 3. Table 3 shows that if the inner efficiency of GT & GC change from 90% to 86%, the efficiency loss is about 3%.

The current work associated with the second phase development of HTR-10 includes the design of IHX-SG vessel, the key accidents analysis of the reactor operating in up to 900°C~950°C outlet temperature. The license for the second phase must be applied in the future.

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HTGR GAS TURBINE-MODULE DEMONSTRATION TEST - A PROPOSAL

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Abstract

The concept of HTGR (High Temperature Gas-Cooled Reactor) plant with Closed Cycle Gas turbine is taking an increasing interest because of its unique characteristics such as highly excellent safety and high thermal efficiency which reaches about 50 percents. The necessity of the demonstration test of the full scale power conversion module to be used in this plant has been pointed out. This paper presents a new proposal of demonstration test facility concept.

1. Introduction

As for the module type high temperature gas-cooled reactor (HTGR), the range of application as the thermal energy source is very wide because the primary coolant temperature at the reactor outlet is much higher compared with other types of reactors. Therefore, utilization plans in various fields have been examined.

Among these, Direct Cycle Gas Turbine Generation Plant recently attracts attention as the next stage power generation plant due to its excellent thermal efficiency.

Now, power generation plant named GT-MHR; which consists of 600MWth HTGR connected with Power Conversion Module (closed cycle regenerative gas turbine system contained within a single vessel) is under development by the US. and Russian cooperation.

Necessity of The Integrated Test Facility which verifies performance of Power Conversion Module has been pointed out from the view point that before the Module is incorporated to the reactor system, its integrity and soundness of structural design , and of completely assembled full scale hardware should be demonstrated under non-radioactive full temperature and full speed conditions.

2. The gas turbine system to be tested

To establish the concept of the Test facility, the gas turbine system to be aimed at as the target plant was examined in the following way.

Two types of closed cycle systems illustrated in Fig.1 were considered, and thermal efficiency was surveyed for various turbine inlet temperatures and pressure ratios.

Other parameters which affect thermal efficiency were kept constant to the value tabulated in Tab.I.

As a result, parameters shown in Tab.I were selected. This system yields thermal efficiency of 50% under turbine inlet temperature of 850℃.

In the following study, thermal output of the reactor will be assumed to be 200MWth. This value was selected considering that choosing somewhat smaller unit output and expect mass-production effect of the gas turbine system would be the more realistic and faster way to realize gas turbine HTGR.

3. Concept study of the Test Facility

3.1 Examination of the pressure level

Closed cycle gas turbine system has a feature that by controlling the pressure level of the loop, the same thermodynamic condition as that at the design rated point can be retained even under partial load condition. If the temperatures at each point of the cycle are kept to the same values as those at design condition, pressure ratio and thermal efficiency are maintained ,while shaft-end output varies in proportion to loop the pressure.

$$\text{Power} \propto P/P^* \qquad (1)$$

here,

Power : Turbine shaft-end power output

P : Fluid pressure

P* : Fluid pressure at real machine design point

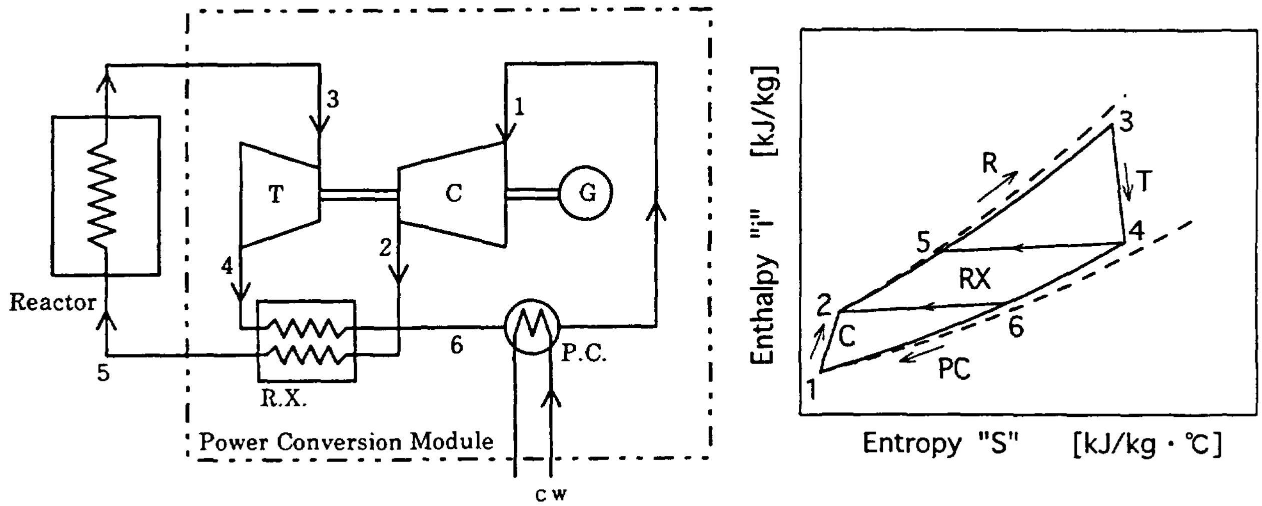
From this relation, thermodynamic performance of the system can be verified by operating it under lower pressure level than the design pressure. When such a way was chosen, the capacity of the heater used instead of reactor and the heat sink can be made much smaller in proportion to P/P*.

$$Q(H) \text{ or } Q(S) \propto P/P^* \qquad (2)$$

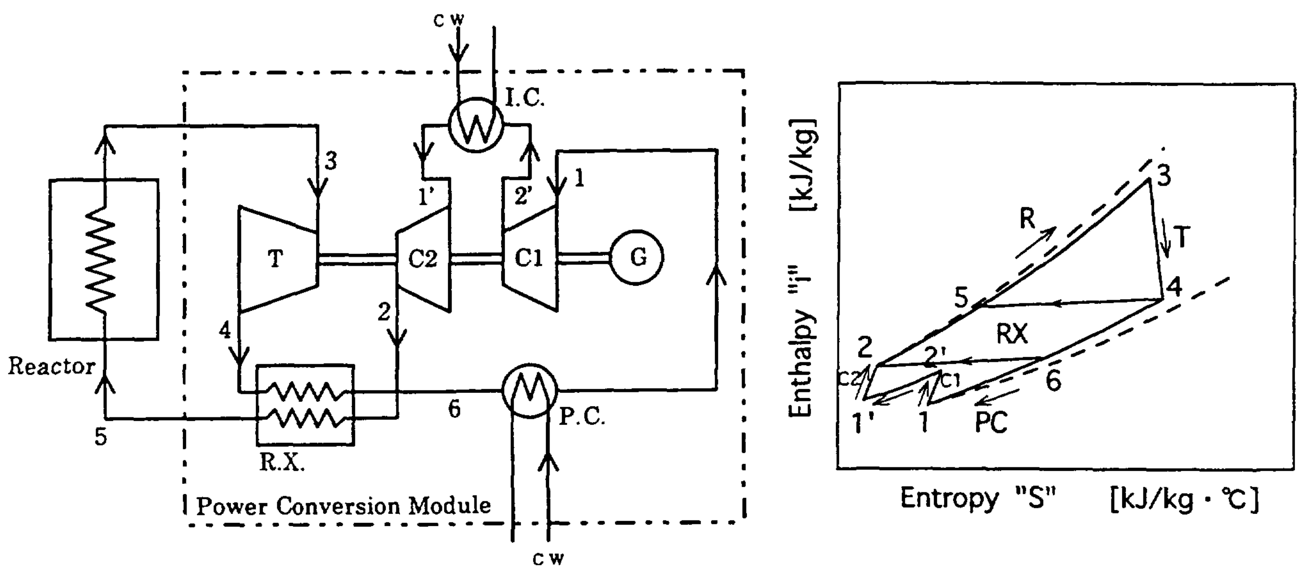
here,

Q(H) : Thermal power of heater (nuclear reactor)

Q(S) : Capability of heat sink



(1) Non Intercooled Regenerative Cycle



(2) 1-Stage Intercooled Regenerative Cycle

Fig.1 Two Types of Closed Cycle Systems Studied in the Plant Selection

TABLE I. MAJOR PARAMETER OF TARGET PLANT

Reactor (Heater) Power	MWt	200
Compressor inlet pressure	MPa	2.86
Compressor outlet pressure	MPa	8.0
Cycle Pressure ratio	—	2.80
Compressor inlet temp.	℃	30
Turbine inlet temp.	℃	850
Turbine adiabatic efficiency	—	0.91
Compressor adiabatic efficiency	—	0.91
Recuperator efficiency	—	0.95
Number of Intercooling	—	1
Turbine shaft-end power output	MW	102
Thermal Efficiency at shaft-end	%	50.8

Although above expressions are theoretically correct, there are actually some essential or inevitable losses such as mechanical loss, auxiliary loss, and etc. ,which can not be neglected in extremely small partial load. Therefore it is necessary to examine to what point P/P* can be reduced.

The change of the thermodynamic characteristics due to relative pressure level, (X=P/P*) can be shown in Fig.2.

Here,
X=P/P* (relative pressure level)
Y=Power/Power* (relative turbine shaft-end output)
η / η *=y/x (relative thermal efficiency)
Power ; Turbine shaft-end output
(* : design point value)

Fig.2 suggests if inevitable losses of the system can be assumed 3% or less, X=0.2 or more seems sufficient to predict full power efficiency of the system.

3.2 Flow Diagram and Energy Balance of the Test Facility

When the relative pressure level X=0.2 is adopted, required heat input to the Test Facility can be reduced to 40MW ;20% of the target plant thermal output. In addition, when considering that the gas turbine system itself has a thermal efficiency of 50%, a half of the required heat input can be recovered by the turbo generator connected to the gas turbine. Fig.3 and 4 show the flow diagram and

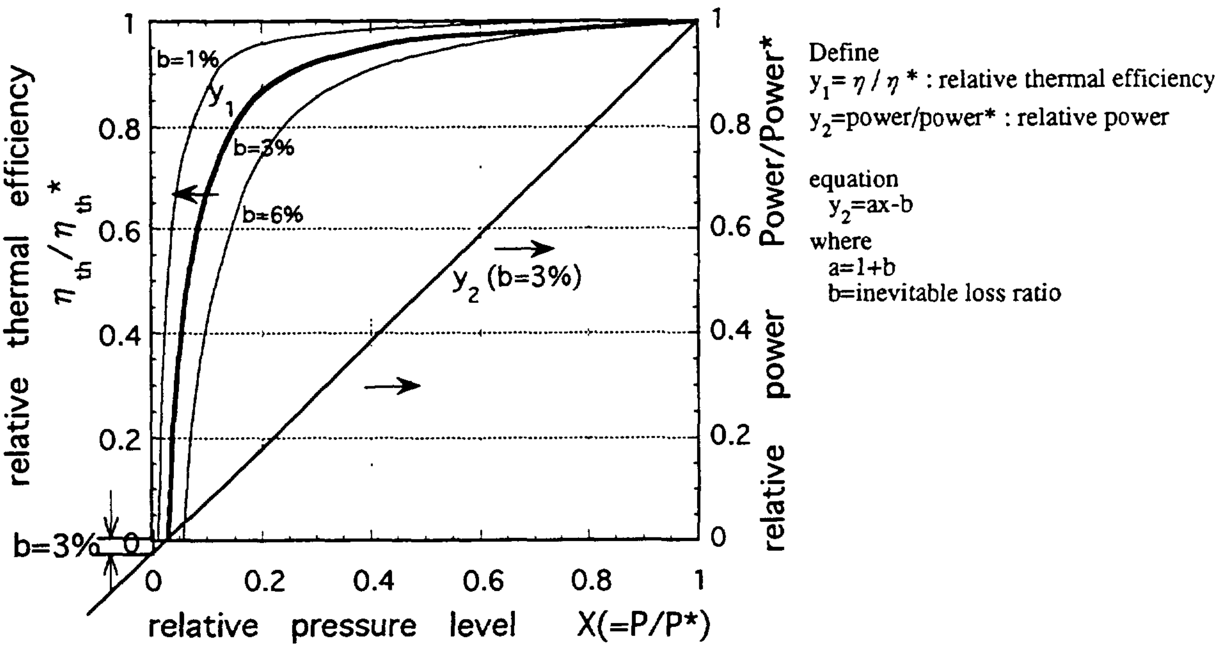


Fig.2 Effect of "inevitable loss" in Real Plant Partial load Characteristics of Closed G.T.

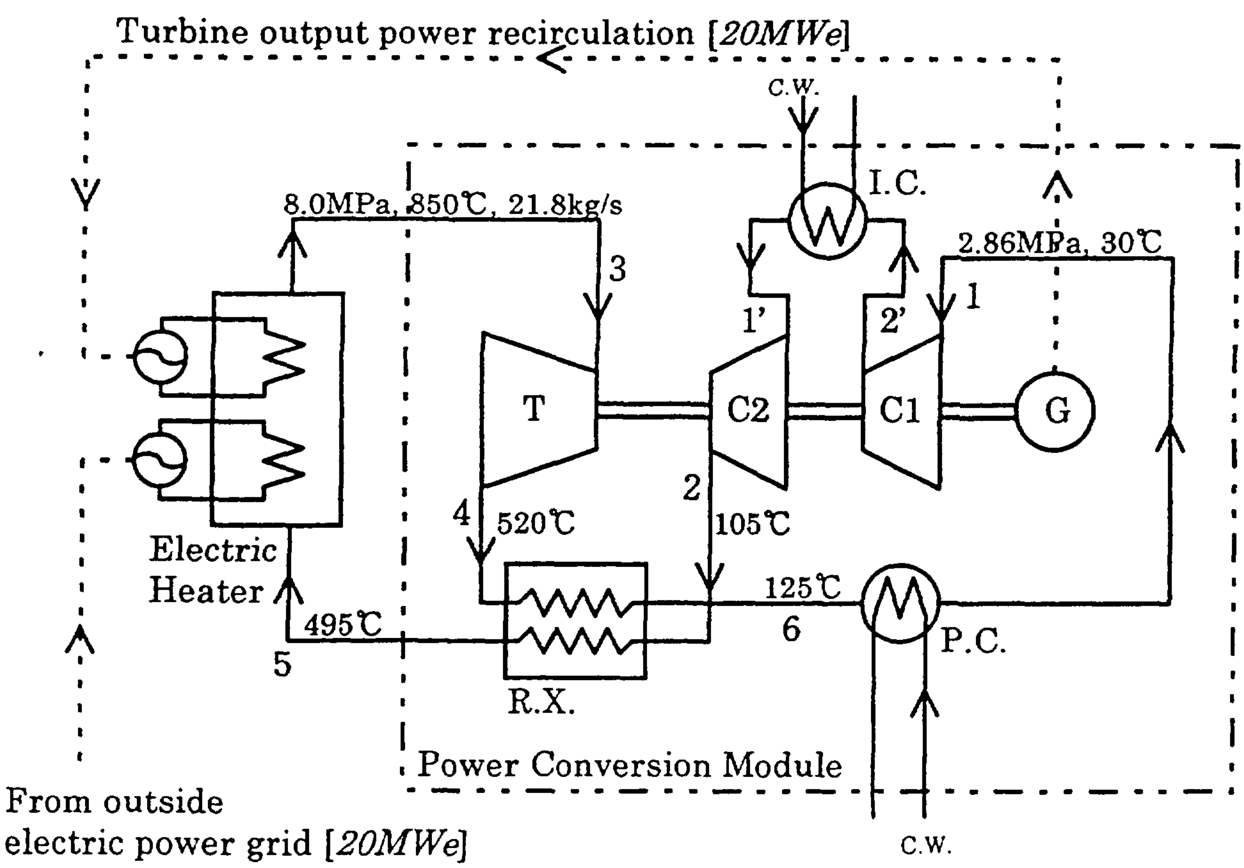
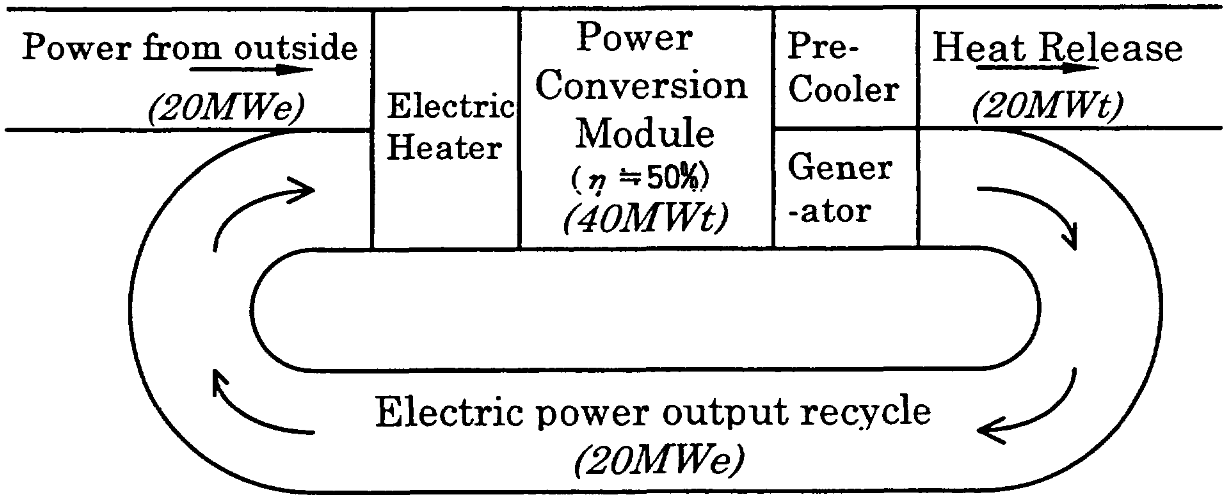


Fig.3 Flow diagram of Integrated Test Facility



Major Specification of Test Facility	
Heater Power Output	40MWt
Compressor inlet pressure	0.57MPa
Compressor outlet pressure	1.6MPa
Cycle Pressure ratio	2.80
Compressor inlet temperature	30.0℃
Turbine inlet temperature	850℃
Turbine adiabatic efficiency	0.91
Compressor adiabatic efficiency	0.91
Recuperator efficiency	0.95
Number of Intercooling	1
Turbine shaft-end power output	20.3MW
Thermal Efficiency at shaft-end	50.8%

Fig.4 Energy balance of Integrated Test Facility

energy balance of this method. 20MWe electric power from outside and 20MWe from the turbo generator are supplied to the test loop electric heater. Of these, 20MW is recovered by the turbo generator of the power conversion module and the rest are released to the environment.

Electric heater was preferred from the view point that it is the easiest way to recover turbine output, and when compared with fossil fired high temperature heater it seems to be the proven technology in our experience. It is also expected that by using electric heating, various test conditions including that for load variation test can be controlled more easily than in case of fossil fired heating.

3.3 Test Items

By using this facility, in addition to the performance verification under partial load, but full temperature and full speed condition, the following tests can be performed in a clean helium environment prior to connect the nuclear reactor.

① Responses of the system under various transient conditions

Response of the system under startup, shutdown operation, ramp or step change of load ,loss of electrical grid load ,loss of cooling water supply etc. will be tested.

② Flow distribution measurement

Flow distribution in various flow paths such as those among recuperator banks, at inlet and outlet of the turbo machines etc. will be measured.

③ Interface problem resolution

If the interfaces between each components are proper, if they are not damaged, or if the unexpected leak paths and flows are formed by differential thermal expansion and other operating loads, etc. will be verified.

④ Maintenability verification

Methods and remote handling equipments (if necessary) for removal or installation of turbo-machines can be tested.

⑤ Operator training

4. Conclusion

It is said that no fatal problems are foreseen in developing each components of the Power Conversion Module such as turbo-machinary, magnetic bearings, high performance recuperator, and etc. However, for the complex system like this, it is absolutely necessary to understand the characteristics of the entire system by assembling and operating them prior to deployment as a nuclear plant system.

The Integrated Test Facility proposed in this paper will provide realistic and economical way to meet such a requirement.

LICENSING, FUEL AND FISSION PRODUCT BEHAVIOUR

Session 3

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A STUDY OF SILVER BEHAVIOR IN GAS-TURBINE HIGH TEMPERATURE GAS-COOLED REACTOR

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Abstract

In High Temperature Gas-cooled Reactors (HTGRs), some amounts of fission products (FPs) are released from fuel elements and are transported in the primary circuit with primary coolant during normal operation. Condensable FPs plateout on the inner surface of pipings and components in the primary circuit and the gamma-ray emitted from the plateout FPs becomes main source during maintenance works. In the design of a Gas-turbine High Temperature Gas-cooled Reactor (GT-HTGR), behavior of FPs, especially of silver, is considered important from the view point of maintenance works of the gas-turbine. However, the behavior of silver is not well known comparing with that of noble gases, iodine and cesium. Then a study of silver behavior in the GT-HTGR was carried out based on experiences of High Temperature engineering Test Reactor (HTTR) design. The purposes of this study were (1) to determine how important the silver in the GT-HTGR, (2) to find out countermeasures to prevent silver release from fuel elements and (3) to determine the items of future research and development which will be needed. In this study, evaluations of (1) inventory, (2) fractional release from fuel elements, (3) plateout distribution in the primary circuit and (4) radiation dose at the gas-turbine were carried out. Based on this study, it was predicted that the gamma-ray from plateout silver contributes about a half of total radiation dose during maintenance work of the gas turbine In future, it is expected that more detail data of silver release from fuel, plateout behavior, etc. will be accumulated through the HTTR operation.

1. INTRODUCTION

In high temperature gas-cooled reactors (HTGRs), some amounts of fission products (FPs) are released mainly from fuel with coatings and are transported to the primary cooling system with the primary coolant during normal operation. In that case, condensable FPs plateout on the inner surface of pipings and components in the primary cooling system[(1),(2)]. On the other hand, since the HTGRs use helium gas, which is not activated itself, as primary coolant, almost no amount of corrosion products is generated. Then, the gamma-ray emitted from the FPs becomes main source in shielding design of the primary cooling system of HTGRs. In order to calculate the plateout distribution in the primary cooling system of HTGRs, a computer code, PLAIN, has been developed[2], and the applicability of the code to the design has been examined[3],[4].

Because of high diffusion coefficients in coating layers and graphite[5] and emitting relatively high energy gamma-ray, ^{110m}Ag would be important in the shielding design and planning of maintenance work of gas-turbine HTGRs (GT-HTGR)[6],[7]. Sufficient plateout distribution has not been observed in plateout experiments at JAERI because ^{110m}Ag is an activation product of ^{109}Ag which has very small yield by uranium fission (two orders smaller than that of ^{131}I or ^{137}Cs). A preliminary study of silver behavior in the GT-HTGR was carried out based on experiences of High Temperature engineering Test Reactor (HTTR) design. The purposes of this study were

(1) to determine how important the silver in the GT-HTGR,

(2) to find out countermeasures to prevent silver release from fuel elements and

(3) to determine the items of future research and development which will be needed.

This paper describes evaluation results of (1) inventory, (2) fractional release from fuel elements, (3) plateout distribution in the primary circuit and (4) radiation dose at the gas-turbine.

2. INVENTORY CALCULATION

The characteristic features of important FP category in HTGR systems can be summarized below.

(1) Noble gases

These can be retained by intact coating layers, then these are released from failed particles. Noble gases exist in the gas phase and can be removed by primary coolant purification system. Though some of them have condensable daughter FPs, this effect is not large at failure fraction of HTGR fuel.

(2) Halogens

These behavior is approximately same as the noble gases with respect to release from fuel. However, because of a modest chemical affinity for metals, halogens tend to chemisorb on solid surfaces. In this study, the behavior of ^{131}I was investigated.

(3) Alkali metals

These have a higher chemical affinity for graphite than the halogens. The sorption is sufficiently strong such that graphite is an effective barrier to release from the core under normal operating conditions. The most significant feature of cesium chemistry relative to plateout is its high affinity for oxides. These form on diffusion of cesium into the oxide protective layers on steel and tend to fix the plated material in place. Since ^{137}Cs has 30 years of half life, its plateout amount is accumulated and has been an important nuclide from the view point of shielding design of the HTGRs.

(4) Rare Metals

Because of high diffusion coefficients in coating layers and graphite[3] and emitting relatively high energy gamma-ray, ^{110m}Ag would be important in the shielding design of HTGRs. Though ^{110m}Ag is an activation product from ^{109}Ag which has very small yield by uranium fission (two orders smaller than that of ^{131}I or ^{137}Cs), it has larger yield by plutonium fission. It means that in the low-enriched uranium cycle up to high burnup, the inventory of ^{110m}Ag becomes large. .

Table 1 shows fission yields and half-lives of ^{131}I, ^{137}Cs and ^{110m}Ag (including fission yields and absorption cross section of ^{109}Ag). Inventories of ^{131}I and ^{137}Cs are calculated by the following equation.

Table 1 Comparison of fission yield and half-life of ^{131}I, ^{137}Cs and ^{110m}Ag.

Nuclide	Fission yield	Half-life	Cross section
^{131}I	3.23 %	8.1 days	
^{137}Cs	6.35 %	30 years	
^{109}Ag	0.03 % (^{235}U) 1.66 % (^{239}Pu)	∞	93.5 barns
^{110m}Ag	0	252 days	

$$\frac{dN}{dt}=3.2\times10^{10}PY-\lambda N(t). \tag{2.1}$$

Where, N :number of nuclide (atoms),

P :reactor power (W),

Y :fission yield (-),

λ :decay constant (s-1).

Inventory of ^{110m}Ag is calculated by the following equations.

$$\frac{dN_{109}}{dt}=3.2\times10^{10}P(aY_{Pu}+(1-a)Y_U)-\sigma N_{109}(t). \tag{2.2}$$

$$\frac{dN_{110m}}{dt} = \sigma \Phi N_{109}(t) - \lambda N_{110m}(t). \qquad (2.3)$$

Where, N_{109} :number of ^{109}Ag (atoms),
N_{110m} :number of ^{110m}Ag (atoms),
a :fraction of plutonium fission,
Y_{Pu} :yield of ^{109}Ag by plutonium fission,
Y_U :yield of ^{109}Ag by uranium fission,
σ :capture cross section of ^{109}Ag,
Φ :thermal neutron flux ($m^{-2}s^{-1}$).

Figure 1 shows calculated result of ^{110m}Ag inventory in GT-HTGR core. In the calculation, plutonium fission fraction and thermal flux are treated as parameters and the result shows that inventory of ^{110m}Ag depends on plutonium fission fraction and thermal neutron flux. **Table 2** indicates predicted core design parameters of GT-HTGR together with those of HTTR. With higher burnup, higher power density and longer irradiation time, plutonium fission fraction, thermal flux and irradiation time in GT-HTGR core could be 60 %, 4×10^{14} $cm^{-2}s^{-1}$, 6 years, respectively. Based on **Fig. 1** and **Table 2**, ^{110m}Ag inventory in GT-HTGR core, which thermal power is 450 MW, will be about 70 times larger than that of HTTR.

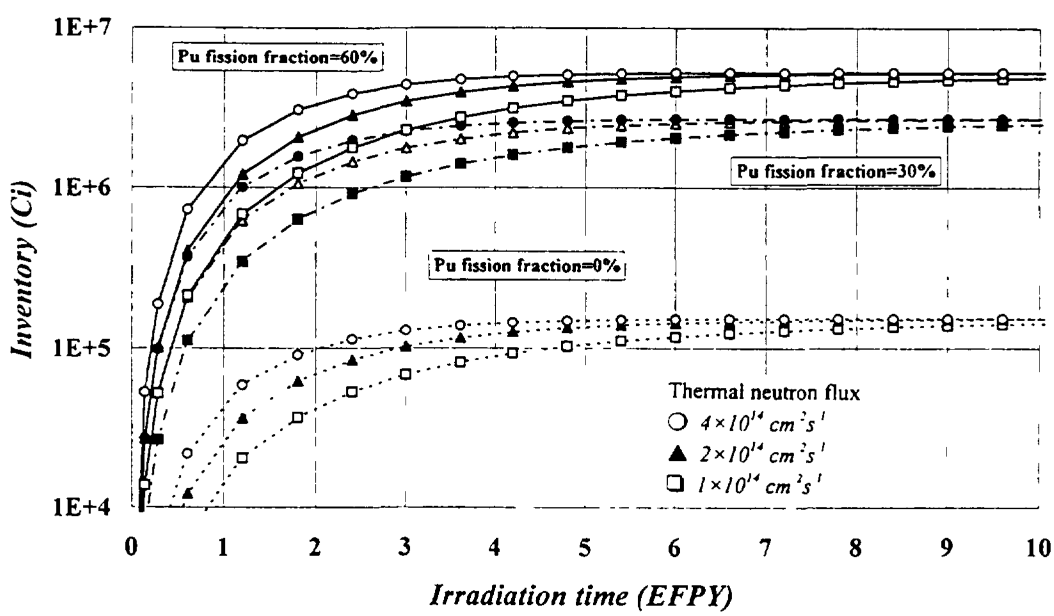

Fig. 1 Calculated inventory of ^{110m}Ag.

Table 2 Core design parameters for ^{110m}Ag inventory calculation.

Parameter	HTTR	GT-HTGR
Pu-fission fraction	30	60
Thermal flux ($\times 10^{14} cm^{-2}s^{-1}$)	0.6	4
Irradiation time (EFPY)	3	6

3. RELEASE FROM FUEL ELEMENT

The coating layers of fuel particles provide barriers to the release of iodine. Since short-lived FPs disintegrate during diffusion process in coating layers, it can be considered that iodine is mainly released from fuel particles with defects in their coating layers (i.e. failed particle). A calculation model has been developed to evaluate the ratio of release to birth, (R/B), of iodine from the fuel compact containing failed particles(5),(8).

The release process of the cesium and silver from the fuel rod can be described by two consecutive steps: the first release occurs from fuel particles and then it occurs from the fuel rod. The former release is controlled by two physical mechanisms - diffusion and recoil - and the latter release, diffusion in the interior of the fuel compact and sorption/evaporation on the surface. In the analysis of fractional release of metallic FP, three types of particles are considered: intact, degraded and failed particles. Since the SiC layer acts as the primary barrier to the release of metallic FPs, the particle which has only a failed SiC layer should be also modeled in addition to the intact and failed coated particles. Then the degraded particle corresponds to the particle with failed SiC layer. Moreover, they can be released from the intact coated fuel particle which is irradiated at higher than about 1300 °C during long period. It is known that fractional release of ^{110m}Ag is higher than that of ^{137}Cs.

Fractional releases of ^{131}I, ^{137}Cs and ^{110m}Ag are shown in **Fig. 2** as a function of failure fraction. In the figure, X-axis shows through coating fraction and in the calculation, failure fraction of SiC layer is assumed to be 10 times of that of through coating failure fraction. Since fractional release of iodine is proportional to failure fraction, it can be reduced with minimizing the through coating failure fraction. The fractional release of cesium can be reduced as low as 10^{-4}~10^{-5} by minimizing failure fraction, however, diffusive release from intact particle should be reduced to attain lower fractional release than 10^{-5}. For silver, since diffusive release fraction from intact particle is higher than that of cesium, the fractional release does not depend on the failure fraction. This is a reason why silver could be taken into account in the GT-HTGR design.

The countermeasures to decrease diffusive release from intact particle are, for example,

(1) reduce fuel temperature (the maximum temperature ≤ 1100 °C), and/or

(2) adoption of diffusive resistant material as coating layer (ZrC layer).

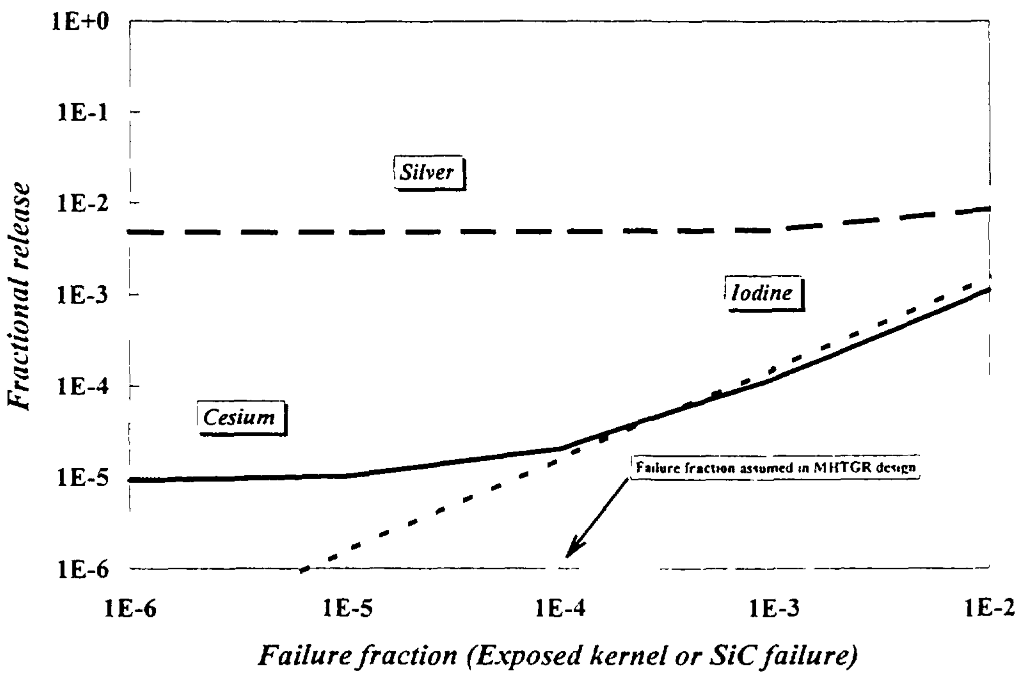

Fig. 2 Fractional release of fission products.

The fuel temperature reduction depends on core design. In order to attain higher thermal efficiency, reactor outlet gas temperature cannot be reduced and it is difficult to reduce the fuel temperature. The diffusion coefficient of ^{137}Cs in ZrC layer is about two orders lower than that in SiC layer. For silver, smaller diffusion coefficient in ZrC layer than in SiC was obtained(9), however, sufficient data have not been accumulated.

4. PLATEOUT DISTRIBUTION

A computer code, PLAIN, has been developed in JAERI to calculate FP plateout distribution in the primary cooling system of HTGRs(2). PLAIN is based on Iniotakis model(10) but has a modified model in the FP penetration process into the base metal(2).

The verification works were carried out by comparing the calculated FP plateout distribution with experimental data obtained in Oarai Gas Loop No. 1 (OGL-1)(1), which is installed in the Japan Materials Testing Reactor (JMTR) in JAERI and simulates the primary cooling system condition of HTGR. In the OGL-1, the plateout distribution has been measured from the outside of the primary pipes after every operational cycle using Ge-detector. The range of helium gas temperature is from 1000 °C at test fuel exit to room temperature. Flow condition is turbulent. Flow diagram of primary cooling system of OGL-1 is shown in **Fig. 3**(1).

Examples of verification results for ^{131}I and ^{137}Cs are shown in **Fig. 4**. In the figures, black circles and lines show measured and calculated plateout densities, respectively. The main

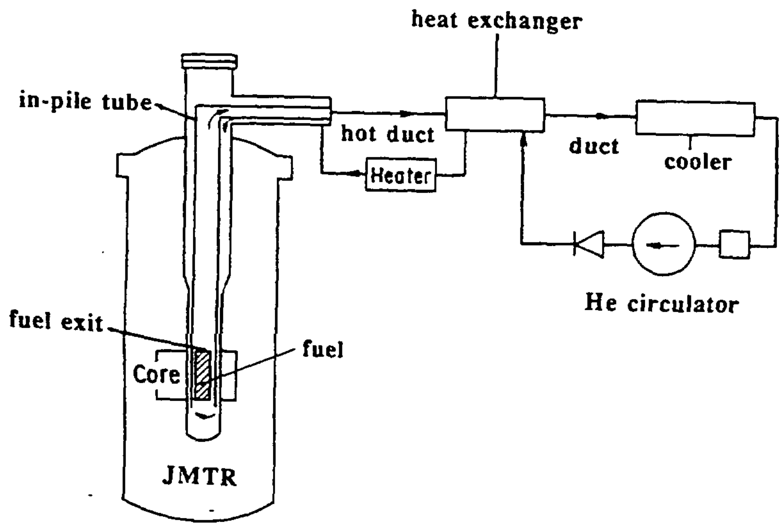

Fig.3 Flow diagram of OGL-1.

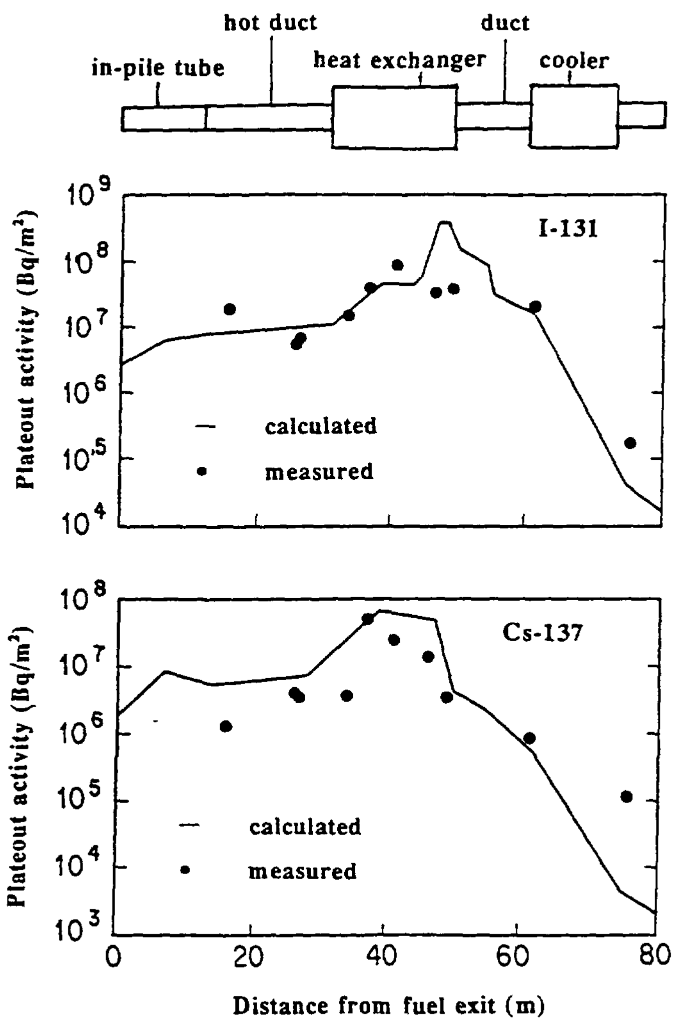

Fig. 4 Measured and calculated plateout distribution in OGL-1.

Table 3 Main plateout condition of OGL-1.

Parameter	OGL-1	HTTR
Materials	Stainless steel Hastelloy-X	Stainless steel (PWC heat transfer tube) Hastelloy-XR (Inner pipe, IHX heat transfer tube) Cr-Mo steel (Annulus of heat exchanger, RPV)
Coolant temperature	950°C~RT*	950~395 °C
Wall temperature	950°C~RT*	950~160 °C
Helium gas pressure	3 MPa	4 MPa
Helium gas velocity	10~60 m/s	20~40 m/s

* Room Temperature

plateout condition in OGL-1 is shown in **Table 3** with that of HTTR. From the results, though about an order of difference is locally observed between measured and calculated values, the calculated plateout profiles show good consistency with the measured ones as a whole in spite of the complicated temperature distribution and flow diagram as shown in **Fig. 3**. Then, it is concluded that the analytical model and the physical constants are applicable to predict the plateout distributions in the primary cooling system of HTGRs.

Since the HTTR takes out high temperature helium gas from the core to the heat exchangers, it can accumulate good operational data to predict the plateout distributions of other HTGRs such as the GT-HTGR. The thermal output of the HTTR is 30 MW, and it is cooled by helium gas of 395 °C at the reactor inlet which flows downward through the core. The reactor outlet coolant temperature is 850 °C at the rated operation and 950 °C at high temperature test operation. The number of main cooling loop is one, and the heat is removed by an intermediate helium/helium heat exchanger (IHX) and a pressurized water cooler (PWC) loaded in parallel. The pressure of primary coolant is 4 MPa. Temperatures and flow rates of the components in the primary cooling system are shown in **Table 4**. In the table, values written in the parenthesis indicate those at 950 °C operation.

Plateout distributions in the parallel loaded operation at rated power operation, i.e. 850 °C of outlet coolant temperature, are shown in **Fig. 5**. It is predicted that both ^{131}I and ^{137}Cs plateout mainly on the heat transfer tubes of PWC where temperature is relatively low in the primary cooling system of the HTTR. On the other hand, there are very small amounts of plateout FPs on the inner pipe of co-axial double pipe where temperature is very high and therefore, desorption rate is large by its high lattice excitation frequency. From the calculation results, it is predicted that

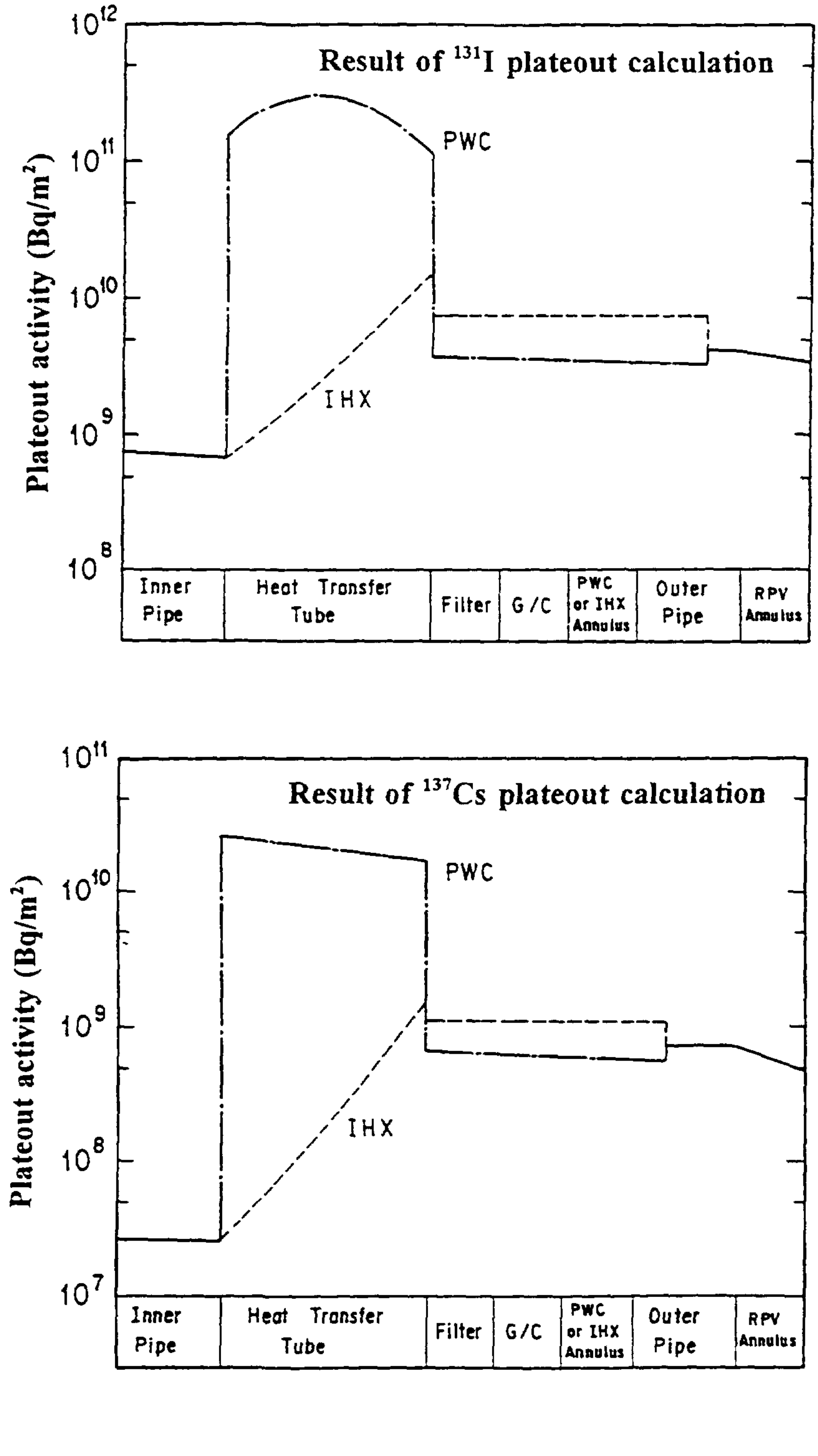

Fig. 5 Predicted plateout distribution in HTTR.

Table 4 Main calculation parameter of primary cooling system of HTTR (Parallel loaded operation).

Region	Coolant temperature (°C)	Wall temperature (°C)	Flow rate (t/h)
Inner pipe	850 (950)	850 (950)	44.6 (36.5)
PWC heat transfer tube	850~385 (950~385)	160~165	29.7 (24.3)
PWC G/C	385~395	385~395	9.9* (8.1*)
IHX heat transfer tube	850~385 (950~390)	805~320	14.9 (12.2)
IHX G/C	385~390	385~395	14.9 (12.2)
RPV annulus	395	395	45.3 (36.5)

( ) shows values in 950 C operation.
* sum of 3 lines

cesium and iodine will plateout mainly on the wall of component whose temperature is relatively low (about 200-400 °C), in the concrete, the pre-cooler of GT-HTGR.

The plateout distribution of silver was not observed in the OGL-1 experiment because the maximum fuel burnup in the experiment was about 5% FIMA and there was no enough silver inventor to measure its plateout distribution. **Figure 6** shows plateout distribution of silver which was measured in the VAMPYR-II experiment[11]. The measured data show that silver tends to plateout on the relatively high temperature wall of 500~600 °C. Though no verification has been carried out, the calculated result of silver plateout distribution in HTTR by the PLAIN code is shown in **Fig. 7**. This result shows that silver plateout on high temperature wall of 850 °C. These results suggest that silver will plateout on the gas-turbine in the GT-HTGR system.

5. RADIATION DOSE

Radiation dose in the gas-turbine was evaluated based on these studies. The calculation was carried out by QAD code[12]. The evaluation conditions are as follows.

(1) Inventory

The thermal power is 450 MW. For ^{110m}Ag inventory calculation, plutonium fission fraction and thermal flux are assumed to be 60% and 4×10^{14} $cm^{-2}s^{-1}$, respectively.

(2) Fractional release from fuel

The through coating and SiC coating failure fractions are assumed to be 4×10^{-5} and 1.6×10^{-4}, respectively, which are one order lower than expected values in the HTTR. The maximum fuel temperature is assumed to be 1300 °C and SiC layer thickness is 35 μm.

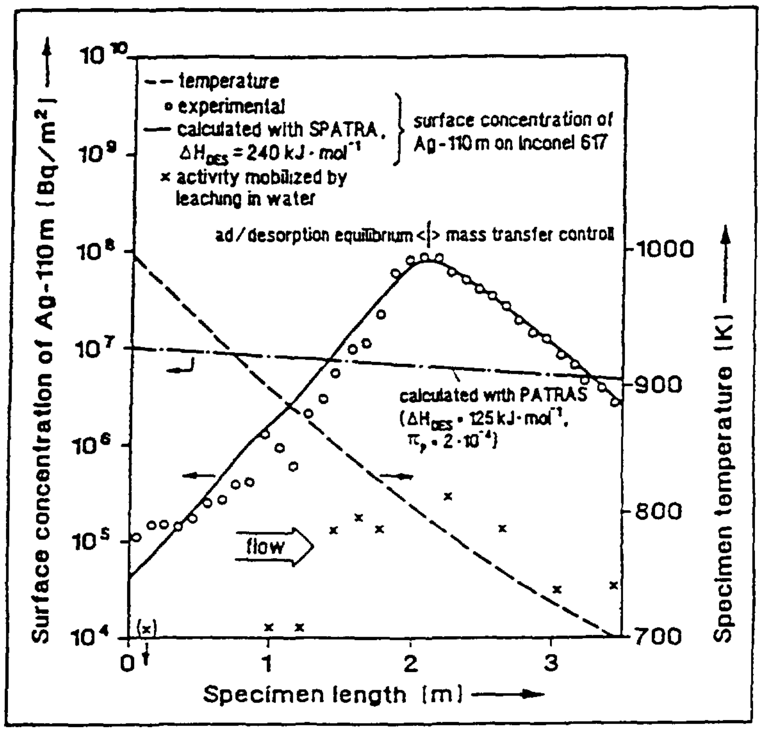

Fig. 6 Plateout distribution in VAMPYR-II[11].

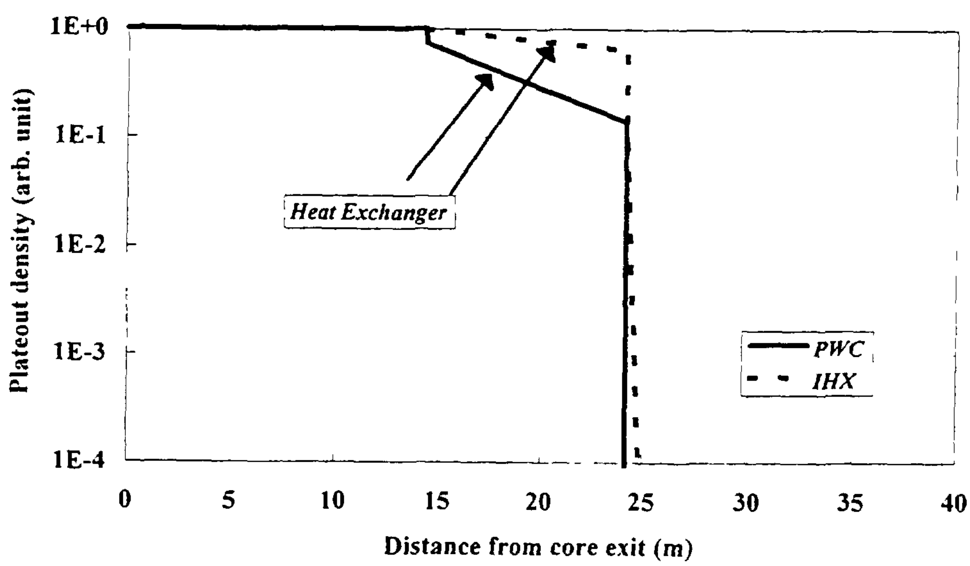

Fig. 7 Calculated plateout distribution of ^{110}Ag in HTTR.

Table 5 Result of dose rate.

Nuclide	Dose rate (μSv/h)
^{137}Cs	12
^{131}I	5
^{110m}Ag	125
Others	100
Total	242

(3) Plateout distribution

The plateout fraction in the gas-turbine is the same as that in the inner pipe of the HTTR where temperature is about 850 °C, namely 0.03 % for ^{131}I and 0.01 % for ^{137}Cs (from **Fig. 5**). For ^{110m}Ag, it is assumed that all silver plateout on the gas-turbine components.

(4) Radiation dose

The radiation dose from gas-turbine depends on its detail design configuration, arrangement, shieldings, etc. In this calculation, it is assumed that (a) FPs plateout on the inner surface of infinite tube, (b) an evaluation point is 1 m from the tube and (c) shielding material is 3 cm of iron.

The dose rates from ^{137}Cs, ^{131}I, ^{110m}Ag and other FPs are shown in **Table 5**. The result shows that the total dose near the gas-turbine is about 250 μSv/h and about the half of that is contribution from ^{110m}Ag.

6. CONCLUSIONS

The calculated result of ^{110m}Ag inventory in GT-HTGR core showed that with higher burnup, higher power density and longer irradiation time, plutonium fission fraction, thermal flux and irradiation time in GT-HTGR core could be about 70 times larger than that of HTTR.

The fractional release of cesium from fuel can be reduced as low as 10^{-4}~10^{-5} by minimizing failure fraction, however, diffusive release from intact particle should be reduced to attain lower fractional release than 10^{-5}. For silver, since diffusive release fraction from intact particle is higher than that of cesium, the fractional release does not depend on the failure fraction. The countermeasures to decrease diffusive release from intact particle are reduction of fuel temperature (the maximum temperature ≤ 1100 °C) and/or adoption of diffusive resistant material as coating layer (ZrC layer).

It is predicted that both ^{131}I and ^{137}Cs plateout mainly on the heat transfer tubes where temperature is relatively low in the primary cooling system of HTGR. The measured data show that silver tends to plateout on the relatively high temperature wall of 500~600 °C. The calculated result of silver plateout distribution in HTTR by the PLAIN code shows that silver plateout on high temperature wall of 850 °C. These results suggest that silver will plateout on the gas-turbine in the GT-HTGR system

The dose rates from ^{137}Cs, ^{131}I, ^{110m}Ag and other FPs shows that the total dose near the gas-turbine is about 250 μSv/h and about the half is contribution from ^{110m}Ag.

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COMEDIE BD1 EXPERIMENT: FISSION PRODUCT BEHAVIOUR DURING DEPRESSURIZATION TRANSIENTS

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Abstract

An experimental program in the CEA COMEDIE loop has been carried out to obtain integral test data to validate the methods and transport models used to predict fission product release from the core and plate-out in the primary coolant circuit of the Modular High Temperature Gas Cooled Reactor (MHTGR) during normal operation and liftoff, and during rapid depressurization transients.

The loop consists of an in-pile section with the fuel element, deposition section (heat exchanger), filters for collecting condensible Fission Products (FP) during depressurization tests and an out-of-pile section devoted to chemical composition control of the gas and on-line analysis of gaseous FP.

After steady state irradiation, the loop was subjected to a series of in-situ blowdowns at shear ratios (ratio of the wall shear stress during blowdown to that during steady state operation) ranging from 0.7 to 5.6.

The results regarding the FP profiles on the plate-out section, before and after blowdowns are given. It appears that :

- the plate-out profiles depend on the FP chemistry
- the depressurization phases have led to significant desorption of I 131, but on the contrary, they have almost no effect for the other FP such as Ag 110m, Cs 134, Cs 137 and Te 132.

1. Introduction

The COMEDIE BD1 experiment to support the MHTGR (Modular High Temperature Gas cooled Reactor) design has been performed in the COMEDIE loop located in the SILOE reactor at CEN Grenoble during the period of Sept. 92 through Nov. 92 for the irradiation phase and December 92 - June 93 for the post irradiation examination.

This experimental program has been carried out in the framework of a US/CEA contract sponsored by US DOE and managed by OAK RIDGE NATIONAL LABORATORY. Its goal was to give measured data to GENERAL ATOMICS to validate computer codes developed for the design of MHTGR reactors.

The primary objective of the BD-1 test is to obtain representative data on the release, transport, plateout, and liftoff of condensible fission products in an in-pile test loop under nominally "clean" conditions. These data will then be used to validate that the design methods used to predict fission product transport in the MHTGR have the required predictive accuracies. Typically, these transport codes contain multiple component models which are derived from differential single effects tests performed in the laboratory or in-pile experiments. The purpose of these in-pile loop tests is not to provide fundamental data from which transport models may be derived but rather to provide integral test data to assess the validity of these integral computer codes.

The primary test objectives can be divided into three parts :

1. To obtain data on fission product release from a fuel element with a known fuel failure fraction, in particular to confirm the effects of high pressure and high flow on release.

2. To obtain data on plateout in an in-pile loop under nominally "clean" conditions expected during normal operation.

3. To obtain data on fission product liftoff over a range of shear ratios

2. Description of the COMEDIE loop :

The COMEDIE loop (fig. 1) is composed of an in-pile and an out-pile sections.

2.1. In-pile section :

This section consists of :

- the fuel element (fig. 2) is the source of FP which were entrained by the coolant gas after release from the particles and diffusion through the graphite.
  Some "designed to fail" (DTF) particles (LEU UCO kernels with a 23 µm pyrocarbon seal coat) which were expected to fail along the first cycle, seeded in fuel compacts with TRISO-coated particles.

- The H-451 graphite reflector block simulates the lower (core exit) reflector of a MHTGR and it can determine the deposition of condensible FP on graphite structures.

- The plate-out section (fig. 3), or heat exchanger, consists of three parallel tube bundles. One of the three bundles was isolated prior to the blowdown test and it is considered as a reference.

- Four cartridge filters (fig. 4-5) which collected FP during the depressurization tests.
  A different cartridge filter is used for each depressurization test.

- 2 probes located on the upstream side and on the downstream side of the heat exchanger make possible sampling of coolant gas.

- An electrical helium heater to adjust the gas temperature at the gas inlet in the fuel element.

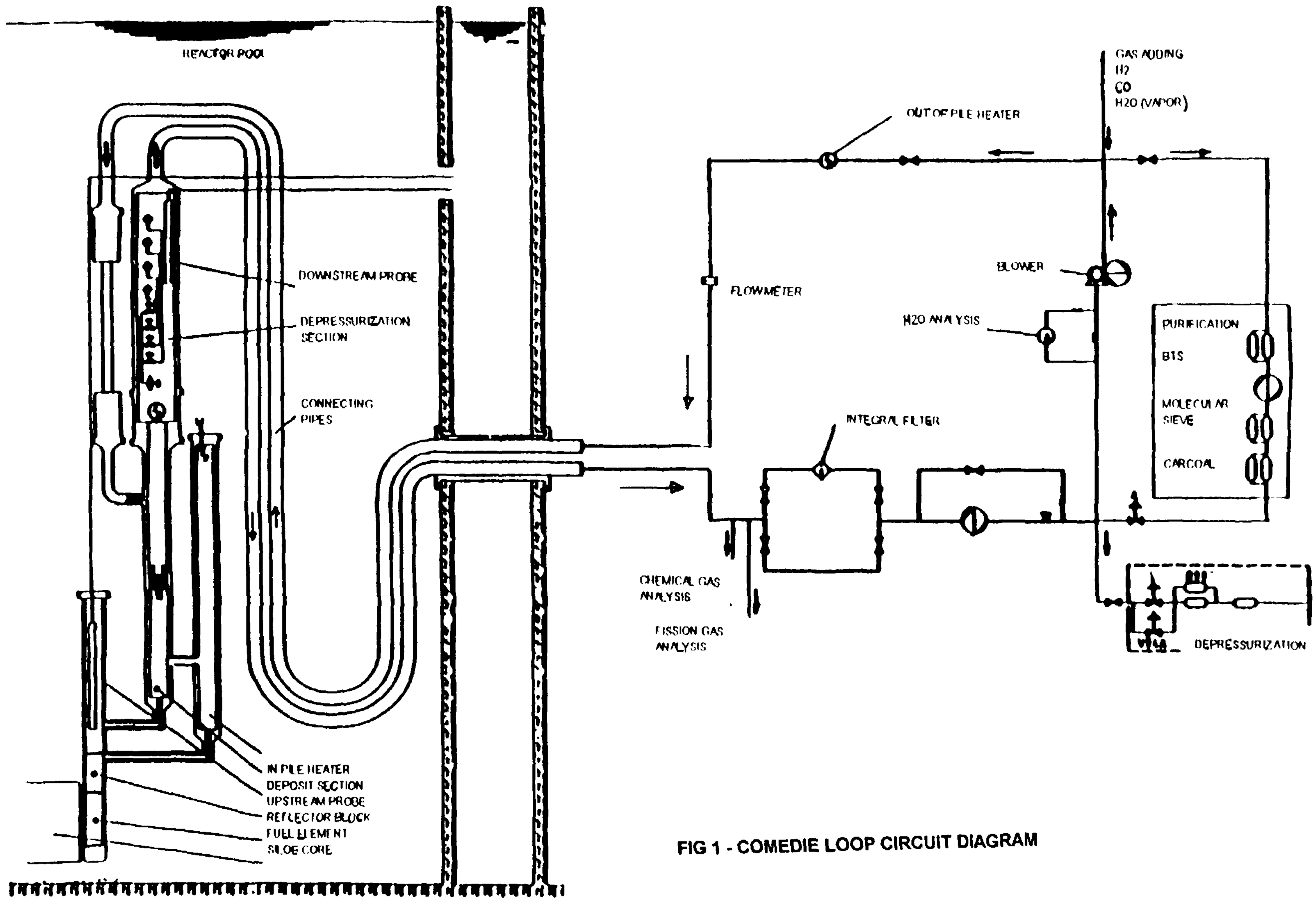

FIG 1 - COMEDIE LOOP CIRCUIT DIAGRAM

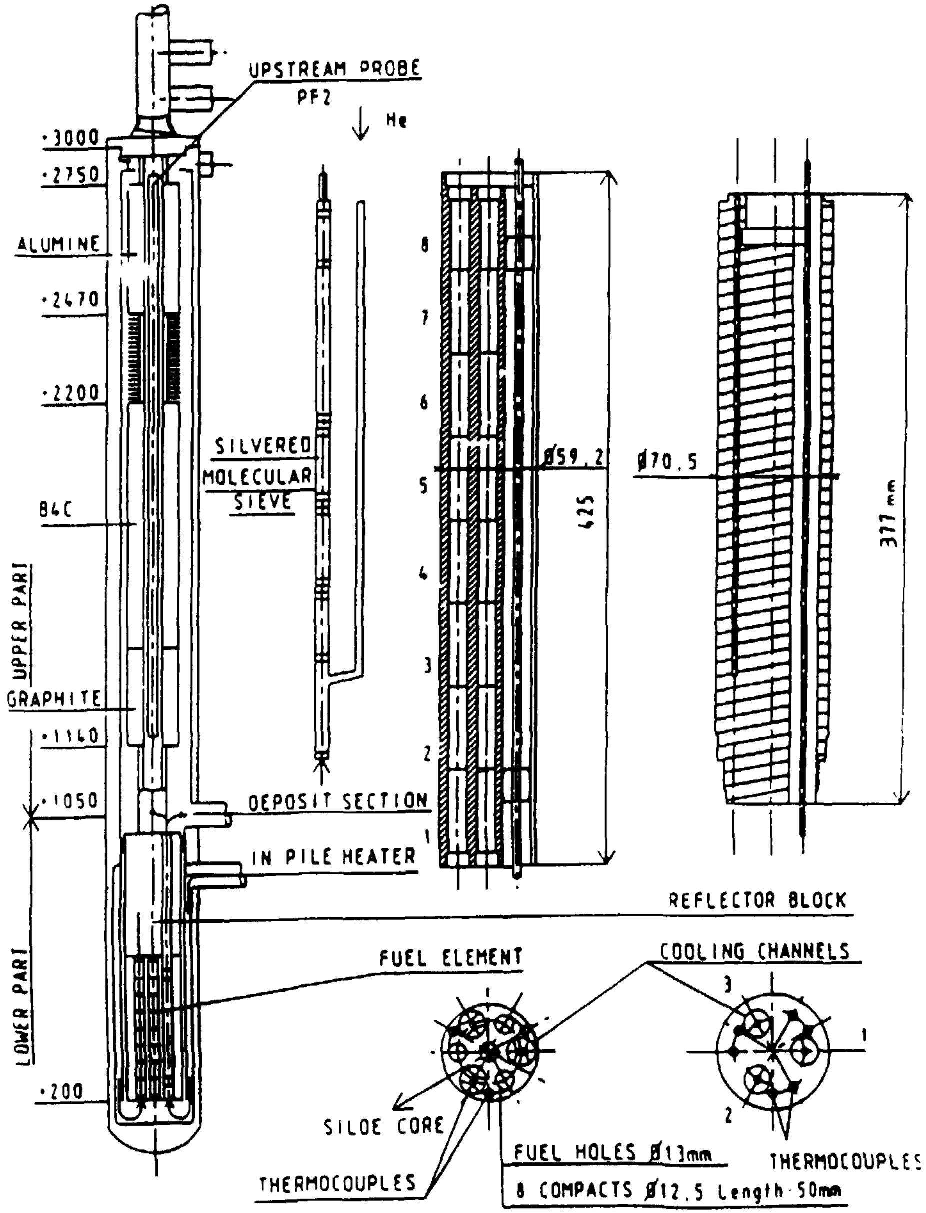

FIG 2 - COMEDIE BD1 FUEL AND REFLECTOR ELEMENTS

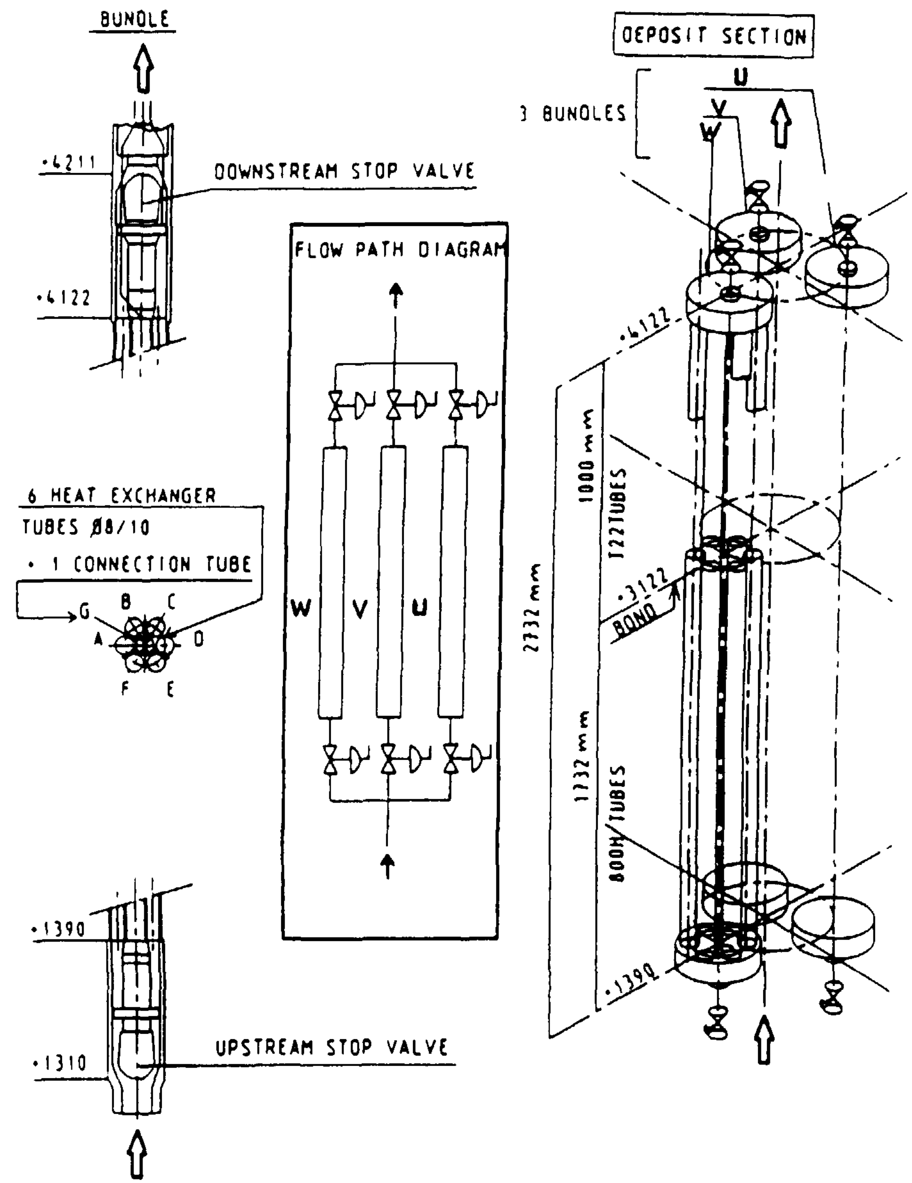

FIG 3 - COMEDIE BD1 HEAT EXCHANGER

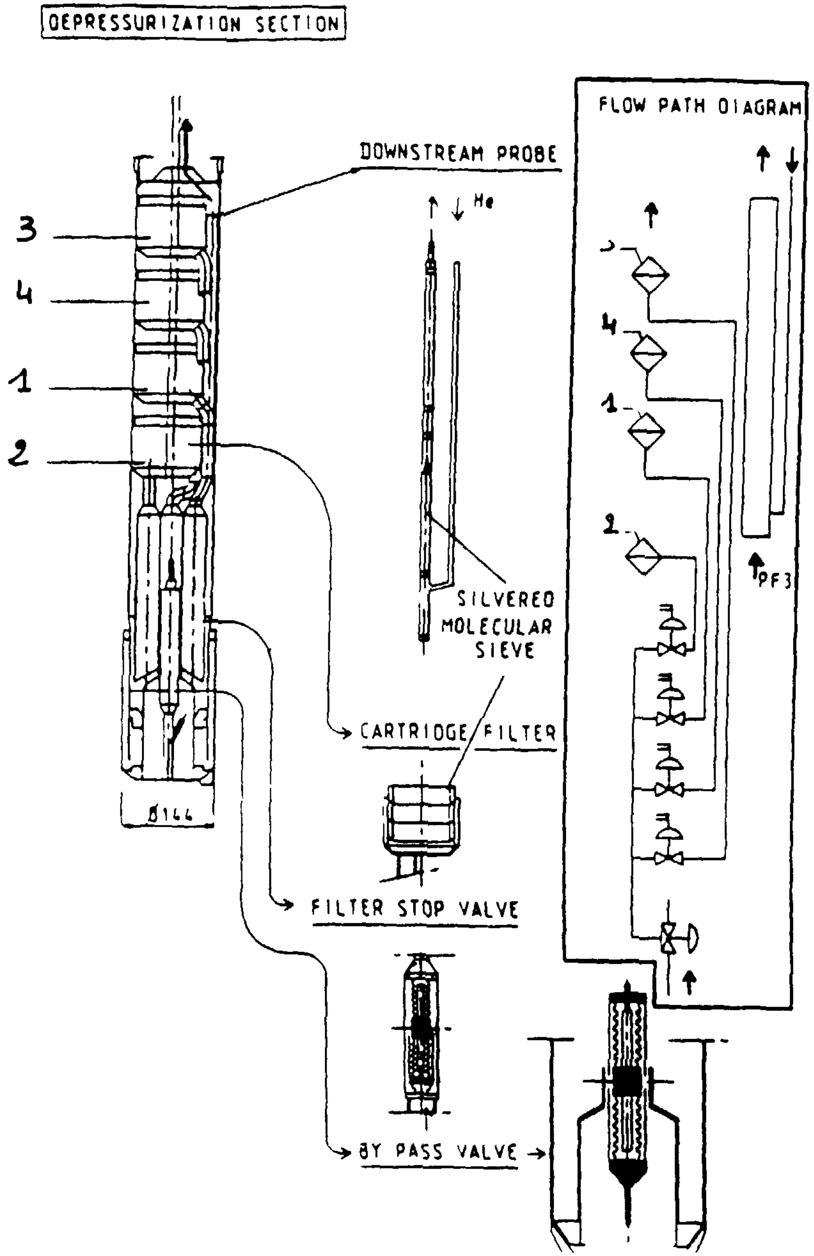

FIG 4 - COMEDIE BD1 BLOWDOWN FILTERS

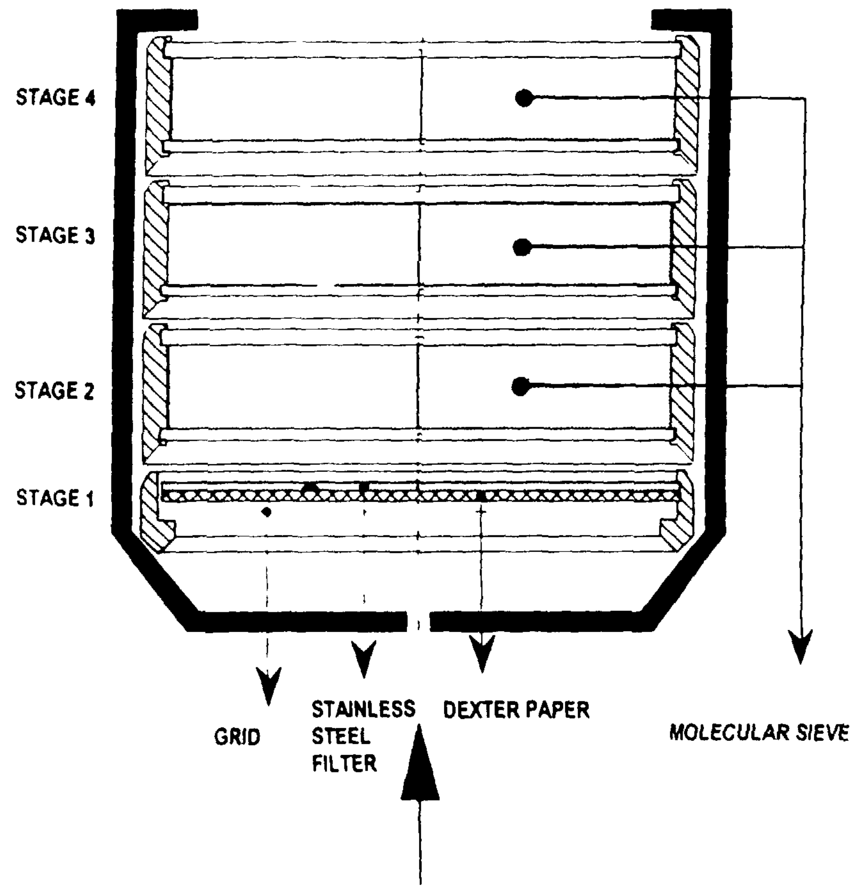

FIG 5 - DEPRESSURIZATION SECTION FILTER

2.2. Out-of pile section

The role of this section is to set up the flow rate and the chemical composition of the coolant gas. It consists of :

- the full flow filter which facilitates the operation under clean conditions by trapping solid particles ("dust") and condensible FP.
- the blower, or gas circulator, which is capable of achieving the mass flow rate up to 70 $g.s^{-1}$ helium under 60 bar pressure
- facilities for gas analysis, gas purification and injection of desired chemical impurities.

3. Irradiation and depressurization phases

3.1. Steady state irradiation :

This phase is to establish and measure steady state operating conditions prior to initiating the transient or blowdown phase of the experiment.

It was conducted for 3 cycles (63 days) in the SILOE reactor under clean conditions. The operating conditions were maintained effectively constant throughout the irradiation except to the coolant impurities which varied somewhat.

These parameters are :

coolant gas pressure	:	60 bars
coolant gas temperature	:	
fuel element	:	$\begin{cases}\text{inlet } 650°C \\ \text{outlet } 720°C\end{cases}$
heat exchanger	:	$\begin{cases}\text{inlet } 720°C \\ \text{outlet } 295°C\end{cases}$

Chemical composition of the coolant gas (typical) :

H_2O	2.3$\pm$ 1 ppm
H_2	12 $\pm$ 3 ppm
Co	6 $\pm$ 1 ppm

During the steady state irradiation, the release of noble gases (Xe and Kr) has been measured continuously on line.

The noble gas release was first very low :

- 2 % of prediction for the 1st cycle
- a rapid increase at the beginning of the 2nd cycle
- the predicted values were reached at the beginning of the 3rd cycle.

This release profile shows us that the DTF particles have started to fail significantly during the 2nd cycle and at the beginning of the 3rd cycle. This is a slower rate than in previous tests with DTF particles (in HFR PETTEN and HFIR at ORNL).

The release of I 131 during the steady state irradiation and deposited in the whole loop is about 650 mCi.

3.2. Depressurization phases (fig. 6) :

These phases called blowdown tests were devoted to the lift off of the FP deposited principally on heat exchanger walls, to transport and to trap them on specifically designed filters.

One of the three bundles of the exchanger was isolated prior to the blowdowns and was considered as a reference (i.e., the plateout distribution at the end of steady-state irradiation phase).

4 blowdowns of increasing levels have been carried out in a very short time after the end of the irradiation phase in order to keep a maximum of short half life FP.

For turbulent flow inside a circular duct (i.e. the heat exchanger tubes), strength of blowdowns is given by the shear ratio (SR) :

$$SR = \left(\frac{P_B}{P_N}\right)^{0.75} \left(\frac{V_B}{V_N}\right)^{1.75} \left(\frac{TG_B}{TG_N}\right)^{-0.58}$$

with P : pressure of the coolant gas
V : flow velocity at the outlet of heat exchanger
TG : gas temperature at the outlet of heat exchanger
B : value under blowdown condition
N : value under steady state condition

The shear ratio is defined to be the ratio of the transient wall shear stress to the wall shear stress during steady-state irradiation. It is used to determine the amount of radioactive plate out that is lifted off from the circuit surfaces and entrained in the coolant circuit.

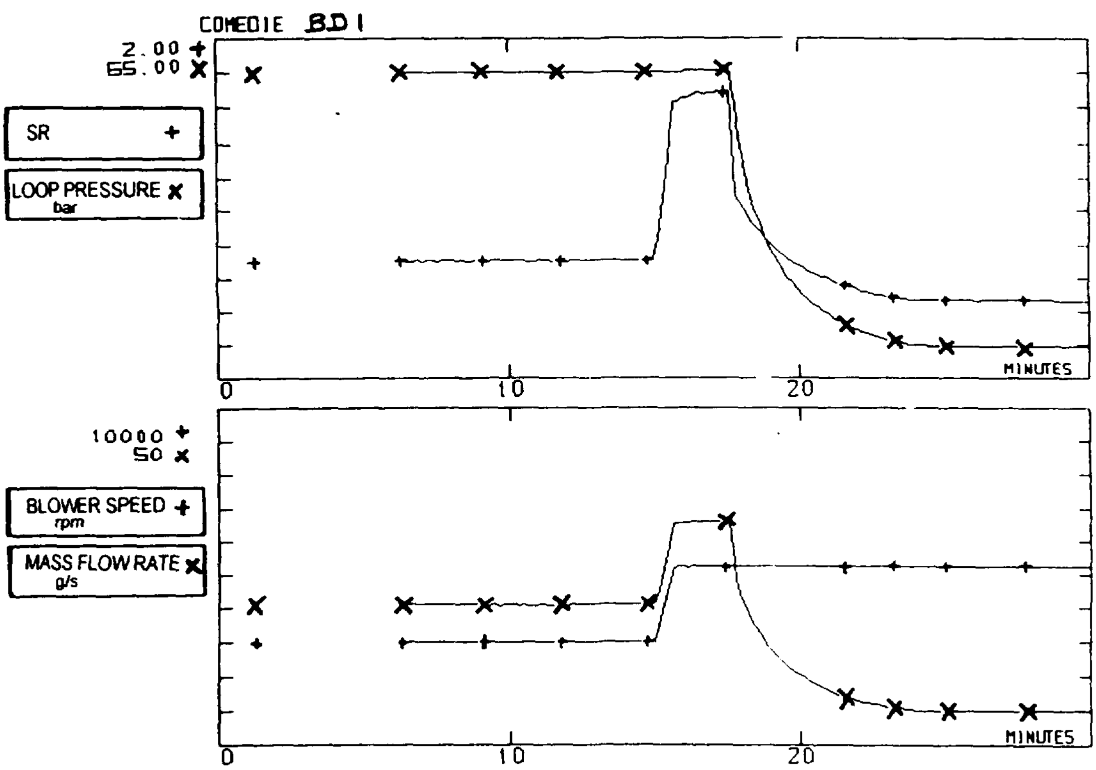

FIG 6 - BLOWDOWN SEQUENCE SR = 1.7

4 blowdowns corresponding to :

SR = 0.72 1.7 2.8 5.6

have been performed, mainly by varying the gas flow velocity by raising the blower speed and maintaining it at a constant level for 2 minutes. A rapid depressurization from 60 bar to 6 bar has taken place just after this short holding period.

4. Plateout and reentrainment of FP

Post irradiation examination has been performed by γ and β spectrometries to measure the amount and the distribution of radionuclides within the various components of the loop.

4.1. Heat exchanger :

The figures show the FP profiles on bundle W (reference, isolated prior to blowdowns) and on bundles U and V (subjected to 4 blowdowns).

I 131 (fig. 7) is preferentially deposited in the coldest zone on the bundles but > 90% of this nuclide passed through the heat exchanger.

I 131 activity of U + V is : 4795 µCi to compare to 5322 µCi for W. If we take into consideration global I 131 activity (650 mCi) within the loop we can say that :

- only 2 % of I 131 have been deposited on the heat exchanger
- during blowdowns, 55 % of iodine deposited on U and V bundles has been moved,as a result of the chemical desorption. However, this is < 1 % of the total I 131 plateout in the loop.

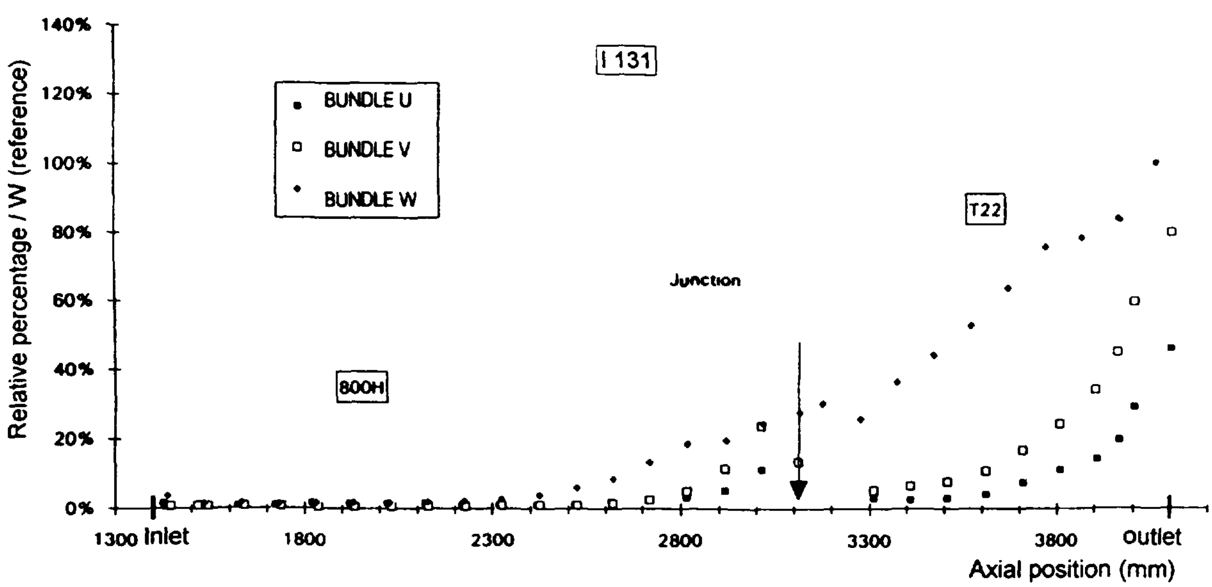

FIG 7. I 131 DISTRIBUTION PROFILES FOR BUNDLES U, V, W

Ag 110m (fig.8)

The main part of Ag 110m is concentrated in the inlet of the bundles (the hottest zone). The heat exchanger has been very efficient in trapping that FP. The amount of Ag 110m is equivalent in the three bundles (nearly 380 µCi on each of them). Blowdowns had no influence, neither on the deposited amount nor on the profile of Ag 110m located inside the bundles.

Cs 134 and 137 (fig. 9 - 10) :

Those two nuclides deposit the same way in the inlet zone of the tubes, profiles and amounts being nearly the same in the three bundles. The same as Ag 110m, total activities in each bundle are very similar, 441 to 478 µCi Cs 137 and 59 µCi Cs 134.

Blowdowns have not led to a noticeable modification ot those nuclides deposited in *U and V bundles.*

Te 132 (fig. 11)

Deposition profiles of that nuclide are the same for the three bundles. It is located in the hot zone of the bundle for the main part. The general behaviour is the same as the one of Ag 110m, including the effect of blowdowns.

Total activities in each bundle are also very similar : 8,6 mCi Te 132 for bundle W, 8 mCi Te 132 for U and V.

4.2. Depressurization section

I 131 (fig. 12)

756 µCi, that is 13 % of I 131 having migrated out of U and V bundles, have been trapped on the four filters, 60 % of which (451 µCi) being found of the filter n° 2 (SR = 0.72). However, this represents < 1 % of the total I 131 plateout in the loop.
The activity (451 µCi) can't be representative of the first blowdown (SR = 0.72) but, more probably, is linked to Iodine desorption phenomena, during the rather long time (5.8 h) when the loop was brought back to operating temperatures with the electrical heaters, prior to the first blowdown.

Ag 110 m (fig. 13)

The trapped activities are very low (2,38 µCi on the 4 filters) that is in agreement with what have been seen when examining the bundles. We find 80 % of the total Ag 110m trapped on the filter corresponding to the highest shear ratio (SR = 5,6).

Cs 134 and 137 (fig. 14 - 15) :

The collected activities are very low :
Cs 137 : 1.36 µCi
Cs 134 : 0.156 µCi
70 % of measured Cs are linked to the SR = 5.6 blowdown.

Te 132

About 1.2 µCi has been trapped on the four filters.

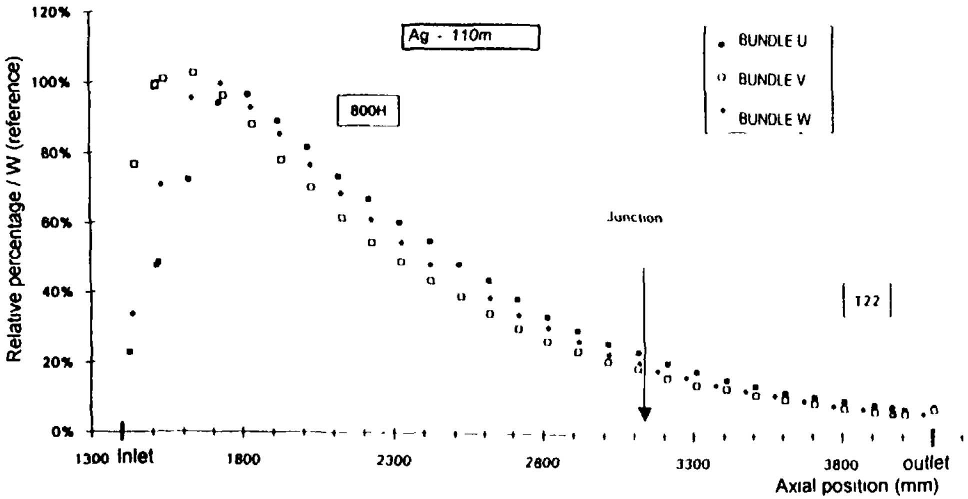

FIG 8. Ag 110m DISTRIBUTION PROFILES FOR BUNDLES U, V, W

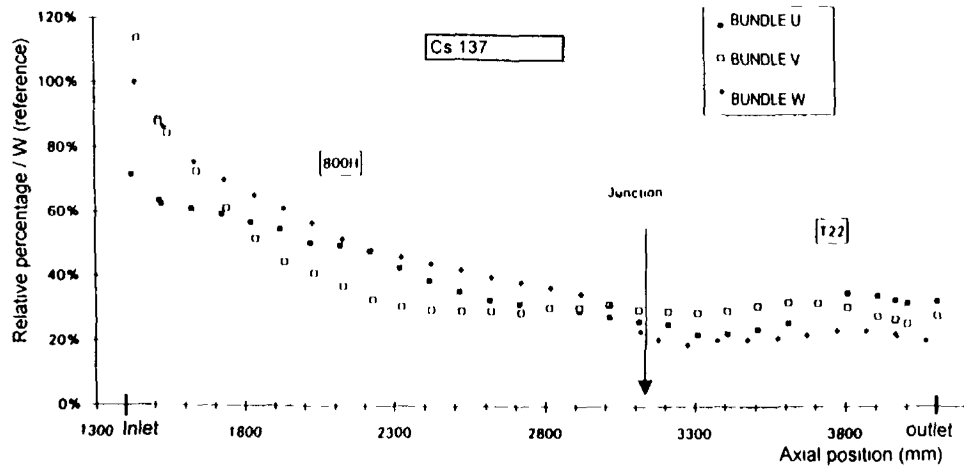

FIG 9. Cs 137 DISTRIBUTION PROFILES FOR BUNDLES U, V, W

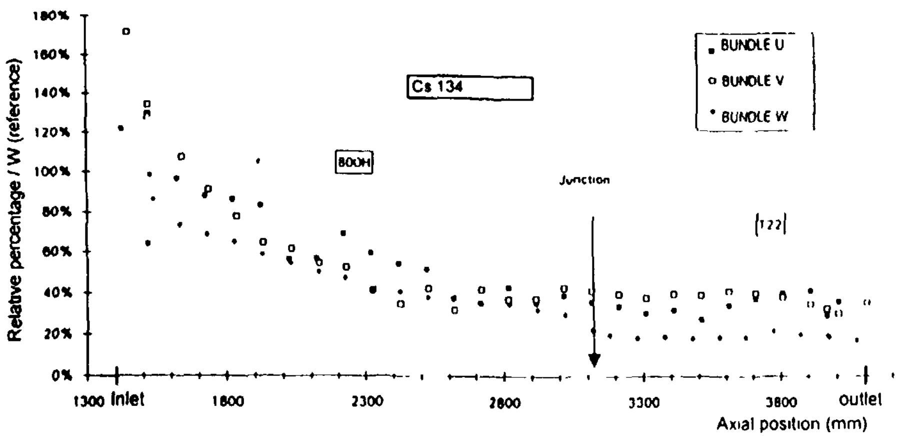

FIG 10. Cs 134 DISTRIBUTION PROFILES FOR BUNDLES U, V, W

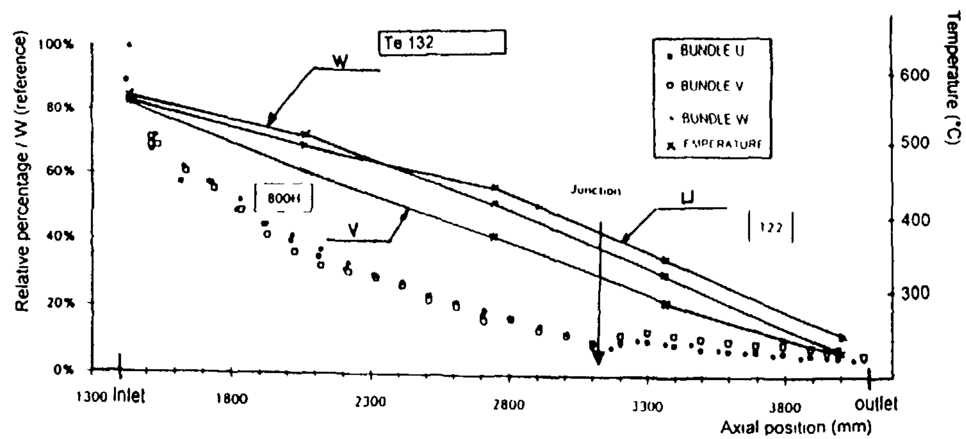

FIG 11. Te 132 DISTRIBUTION AND TEMPERATURE PROFILES FOR BUNDLES U, V, W

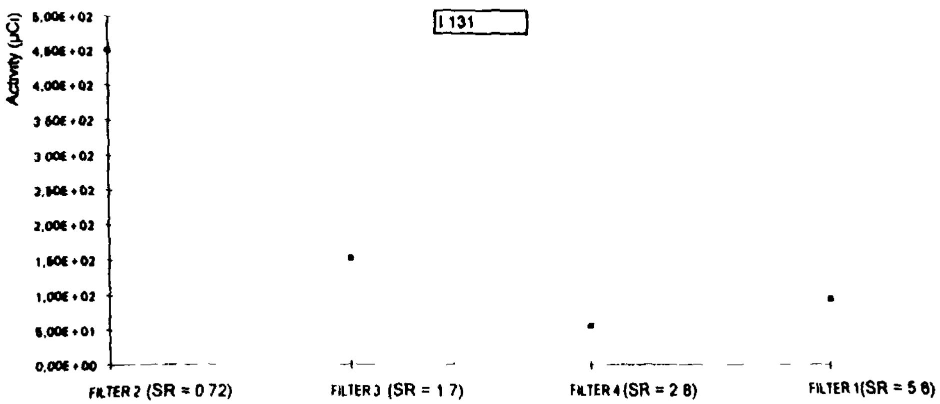

FIG 12. DEPRESSURIZATION SECTION : TRAPPED I 131 ACTIVITY IN FILTERS 1 to 4

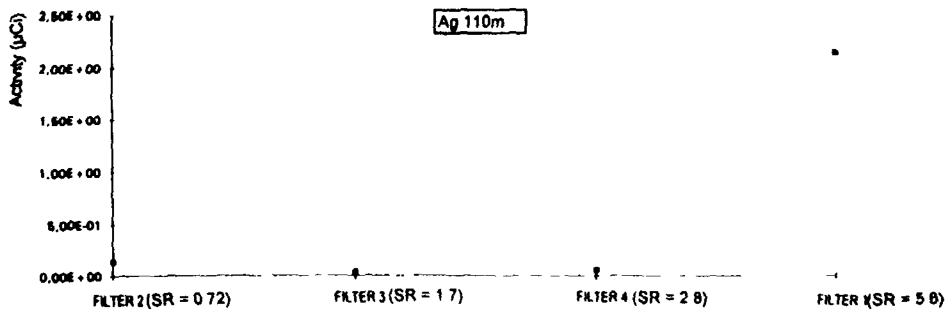

FIG 13. DEPRESSURIZATION SECTION : TRAPPED Ag 110m ACTIVITY IN FILTERS 1 to 4

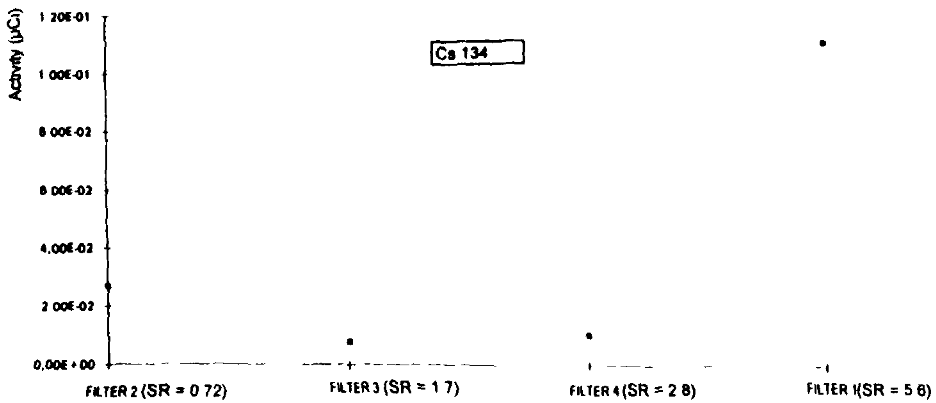

FIG 14. DEPRESSURIZATION SECTION : TRAPPED Cs 134 ACTIVITY IN FILTERS 1 to 4

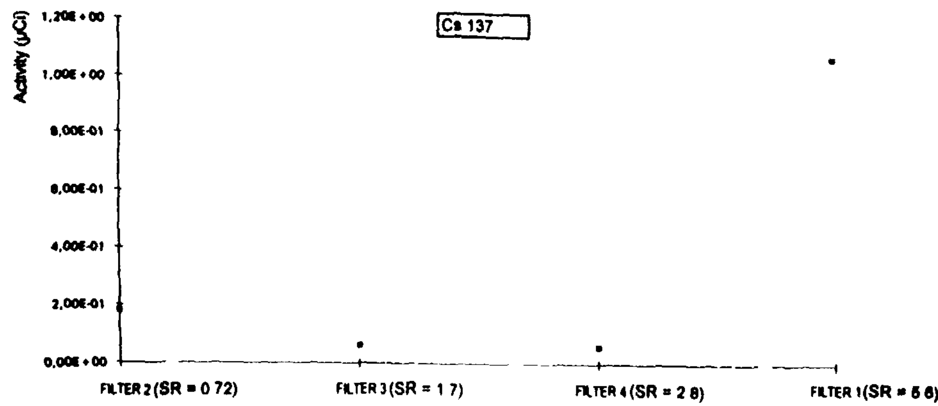

FIG 15. DEPRESSURIZATION SECTION : TRAPPED Cs 137 ACTIVITY IN FILTERS 1 to 4

5. Conclusions

The results show that :

- deposition profiles are dependant on the chemical nature of fission products
- an important redistribution of I 131 (55 % of the initial heat exchanger bundles U and V inventories but < 1 % of the total I 131 plateout in the loop) occured during the thermal conditioning of the loop before the first blowdown.
- Ag 110m, Cs 134, Cs 137 and Te 132 are well fixed within the deposition section and blowdowns did not move significant quantities of them.
- by extrapolation, the fractional liftoff during a rapid depressurization of an MHTGR should be < 1 % for all radionuclides, including I isotopes, since the peak shear ratio is < 1.1. and the blowdown is complete in a few minutes.

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[1] MEDWID W. , Specification for COMEDIE Test BD1 - DOE HTGR - 87095, Rev. H, November 92

General Atomics, San Diego, Ca.

LICENSING EXPERIENCE OF THE HTR-10 TEST REACTOR

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Abstract

A 10MW high temperature gas-cooled test reactor (HTR-10) is now being projected by the Institute of Nuclear Energy Technology within China's National High Technology Programme. The Construction Permit of HTR-10 was issued by the Chinese nuclear licensing authority around the end of 1994 after a period of about one year of safety review of the reactor design.

HTR-10 is the first high temperature gas-cooled reactor (HTGR) to be constructed in China. The purpose of this test reactor project is to test and demonstrate the technology and safety features of the advanced modular high temperature reactor design. The reactor uses spherical fuel elements with coated fuel particles. The reactor unit and the steam generator unit are arranged in a "side-by-side" way. Maximum fuel temperature under the accident condition of a complete loss of coolant is limited to values much lower than the safety limit set for the fuel element. Since the philosophy of the technical and safety design of HTR-10 comes from the high temperature modular reactor design, the reactor is also called the Test Module.

HTR-10 represents among others also a licensing challenge. On the one side, it is the first helium reactor in China, and there are less licensing experiences both for the regulator and for the designer. On the other side, the reactor design incorporates many advanced design features in the direction of passive or inherent safety, and it is presently a world-wide issue how to treat properly the passive or inherent safety design features in the licensing safety review.

In this presentation, the licensing criteria of HTR-10 are discussed. The organization and activities of the safety review for the construction permit licensing are described. Some of the main safety issues in the licensing procedure are addressed. Among these are, for example, fuel element behaviour, source term, safety classification of systems and components, containment design. The licensing experiences of HTR-10 are of great reference value for the modular reactor concept.

1 Introduction and Background

Presently, a 10MW high temperature helium cooled test reactor (termed HTR-10) is being projected by the Institute of Nuclear Energy Technology (INET) of Tsinghua University. The reactor will be erected on the site of INET which is about 40km to the north of Beijing city. The HTR-10 test reactor is a major project in China's High Technology Programme.

The HTR-10 test reactor uses spherical fuel elements which are made completely of ceramic materials. Uranium dioxide as nuclear fuel is in the form of coated particles which are dispersed in the graphite matrix of the fuel elements. Graphite serves as neutron moderator and reflector as well as the main structural material of the reactor core, so that the reactor has a practically full-ceramic core which has a large heat capacity and is high-temperature-resistant As coolant serves the inert gas helium which causes practically no corrosion problems and plays no part in the reactor neutron balance

For the HTR-10 test reactor, decay heat removal does not rely at all on any active cooling systems At a complete loss of coolant accident, the maximum fuel temperature remains under the limit value with a big margin Reactor shut down systems are placed only in the side reflector No control rods would have to be inserted into the pebble bed so that damages to the fuel elements are avoided The reactor unit and the steam generator unit are arranged in a so-called "side-by-side" way, so that the reactor and the steam generator can be easily isolated from each other under accident conditions in order to protect the ceramic core of the reactor and the metallic structure of the steam generator These design features of the HTR-10 test reactor represent the advanced modular design of high temperature gas cooled reactors (HTGR)

In terms of reactor types, HTR-10 is the first of its kind to be built in China Besides, the reactor design incorporates many advanced features in the direction of passive safety From these points of view, the HTR-10 reactor represents a big licensing challenge both to the regulator and to the applicant On the one side, there exist not enough standards, codes and guides in China directly applicable to gas cooled reactors And on the other side, it is now a world-wide problem for regulators how to properly treat passive features in the advanced reactor designs In the next sections, the settlement of the main safety issues in the licensing procedure will be addressed

For the licensing of the construction permit (CP) of the HTR-10 test reactor, the licensing authority is the National Nuclear Safety Administration (NNSA) which is technically backed up by Suzhou Nuclear Safety Center and Beijing Nuclear Safety Center The applicant is the Institute of Nuclear Energy Technology of Tsinghua University

2 Regulatory Criteria

As stated above, up to now there exist in China not enough nuclear standards, codes and guides specifically compiled for high temperature gas cooled reactors which are directly applicable to the HTR-10 reactor Before the CP licensing started, two documents had been compiled under the organization of NNSA Design Criteria for the 10MW High Temperature Gas-cooled Test Reactor[1] and Standard Content and Format of the Safety Analysis Report of the 10MW High Temperature Gas-cooled Test Reactor[2], which were supposed to serve as the licensing basis of the test reactor In the actual licensing procedure, stronger reference is made to the second document than to the first one

From the nuclear point of view, the HTR-10 reactor is much smaller than nuclear power plant reactors since its thermal rating is only 10MW The main purpose of the reactor project is to test and prove its main technical and safety features rather than to provide commercial power But HTR-10 has not been regarded as a research reactor because the overall purpose of the reactor is to test power generation technology Based on the above considerations, following principal guidelines have been followed in the licensing procedure concerning what standards and/or codes are taken as licensing basis or references

- As top level regulatory criteria, the nuclear safety related standards, codes, regulatory guides issued by NNSA (HAF series), which covers site selection of nuclear power plants (NPP), design of NPP, quality assurance of NPP, etc., should serve as fundamental basis for licensing criteria.
- References are made to international and foreign standards, codes and guides, e.g. of USA (RG series, ASME, General Design Criteria of High Temperature Gas Cooled Reactors), of Germany (KTA Regeln) and of France (RCC series).
- HTR-10 deserves special treatment in specific cases considering its lower power rating, its nature as a test reactor and especially its designed inherent safety features.

3 Licensing Activity and Procedure

3.1 Pre-application activities

Since the HTR-10 test reactor is the first high temperature helium cooled reactor to be built in China, there are less experiences both from the regulator side in licensing and from the applicant side in design. Therefore, there had been many interactions between the licensing experts and the design engineers in the form of e.g. seminars given by the design engineers to introduce the HTR-10 design and general features of high temperature gas cooled reactors to the licensing experts. These communications helped the involved personnel to exchange ideas and opinions on an early stage.

As mentioned above, NNSA had organized to establish technical documents [1,2] before the licensing procedure started. The second document, namely the Standard Content and Format of the Safety Analysis Report of the 10MW High Temperature Gas-cooled Test Reactor, which defines the content framework of the Preliminary Safety Analysis Report of the 10MW High Temperature Gas-cooled Test Reactor (PSAR)[3], has guided the compilation of the document.

Following the principle of "earlier involvement", some licensing staff had been involved in some pre-application work such as in the site selection activities to assure a more smooth licensing procedure.

Before the licensing procedure started, the applicant had got the HTR-10 project approval from the State Education Commission and the approval of the Feasibility Study Report of HTR-10 from the State Science and Technology Commission. The "Environmental Impact Report of the 10MW High Temperature Gas-cooled Test Reactor" had been approved by the State Environmental Protection Administration.

3.2 Licensing activities and procedure

The application for the Construction Permit of HTR-10 was submitted to NNSA in December 1993 with the attached documents, of which the PSAR[3] and the Quality Assurance Programme of the 10MW High Temperature Gas-cooled Test Reactor (for Design and Construction Periods) (QAP)[4] are the main technical documents to be reviewed by the licensing personnel. From then on the licensing procedure started formally and lasted for one year. The applicant, namely the Institute of Nuclear Energy Technology, got the Construction Permit in December 1994.

The licensing procedure goes with the following main activities undertaken:

- The reviewers (licensing personnel) raise technical questions in written form. Altogether, more than 700 questions were raised on PSAR and QAP in several batches.
- The raised questions are answered by the HTR-10 design engineers in written form.
- Meetings are held between reviewers and design engineers to discuss and address technical questions and issues. During these meetings, clearances or solutions to some questions or problems are found, or agreements upon some technical issues are reached. Unresolved issues are documented in the form of so-called work-sheets. These work-sheets are worked on further by the designers for further interactions.
- Topical meetings are held between reviewers and design engineers on some special issues in site selection, digital protection system design, quality assurance.
- The Nuclear Safety Expert Committee is consulted about some special issues and about the overall evaluation of the licensing personnel of the HTR-10 test reactor on the CP application stage.
- The licensing authority finally reaches a favorable Safety Evaluation Report which leads to the issuing of the construction permit of the HTR-10 test reactor

4 Main Licensing Safety Issues

4.1 Fuel elements

The designed passive safety features of the HTR-10 test reactor are fundamentally based on the excellent fission product retaining capability of the fuel elements. For all the reasonably postulated accidents, both within the design basis and beyond that, radioactive nuclides are retained in the fuel elements well enough so that unallowable radioactive release into the environment will not take place. Therefore, it has been a core issue during the licensing to make sure whether the fuel elements used for the HTR-10 reactor will really behave as good as they are designed for. It is planned that sample fuel elements are to be irradiated to designated conditions covering burn-up, fast neutron dose and irradiation temperature before the fuel elements are operated in the real reactor to those conditions. An oxidation test of the fuel elements under severe accident conditions is also planned to be made.

4.2 Source term

A mechanistic methodology is adopted for determining the radioactive source term. Severe core damages are not arbitrarily postulated, as it is done for large water cooled power reactors, where large amount of radioactivity would have to be postulated to be released out of fuel elements due to severe core damages, and which would lead to the requirement of a pressure-containing and leak-tight containment design.

The release of radioactivity is calculated specifically for those individual demanding accidents which lead to the most release of radioactive nuclides from the fuel elements. The release of radio-nuclides from the fuel kernel through the surrounding coatings and the matrix graphite to the helium coolant is calculated on a diffusion-sorption basis, where the selection of calculation parameters is based mostly on the German experimental results and literature. Where necessary, conservative factors are put into the analysis.

4.3 Accident analysis

As usual, design basis accidents (DBA) are classified in several categories These are

- increase of heat removal capacity in primary circuit
- decrease of heat removal capacity in primary circuit
- decrease of primary flow rate
- abnormality of reactivity and power distribution
- rupture of primary pressure boundary tubes
- anticipated transients without scram (ATWS)

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

Great attention and effort has been given to the treatment of severe accidents A number of highly-hypothetical postulated accidents are selected to be analyzed These hypothetical events 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 circulator shut-down
- simultaneous rupture of all steam generator tubes
- rupture of the cross duct vessel

In selecting these severe accidents, reference is made to the licensing experience of MHTGR in USA[5] and of HTR-Modul in Germany The analyses of severe accidents show that under these highly-hypothetical circumstances, severe damages to the fuel elements would not be expected which would lead to impermitted release of radioactivity into the environment

4.4 Safety classification

Because the HTR-10 test reactor is designed on the inherent safety philosophy, safety classifications of systems and components departure from the way it is done for water cooled power reactors For example, primary pressure boundary is defined to the first isolation valve Steam generator tubes are classified as Class II component Diesel generators are not required to be as highly qualified as those used for large water cooled power reactors, since no systems or components with large power demand would require an immediate start of the diesel engines at a plant black-out accident

4.5 Containment design

Based on the characteristics of inherent safety of the HTR-10 test reactor, no pressure-containing and leak-tight containment is designed The concrete compartments, which houses the reactor and the steam generator as well as other parts of the primary pressure boundary and which is preferably called as confinement, together with the accident ventilation system, serve as the last barrier to the radioactivity release into the environment During normal operation, the confinement is ventilated to be kept sub-atmospheric When the integrity of the primary pressure boundary is lost, the primary helium coolant is allowed to be released into the environment without filtering because of its low radioactivity content Afterwards the confinement is ventilated again, gases in it will be filtered before they reach the environment

5 Summary

The licensing activities of the construction permit of the HTR-10 test reactor are overall well organized in a rather tight time-framework The evaluation of the licensing body on the safety favorites the reactor safety design and has led to the issuing of the construction permit of the HTR-10 test reactor

Experiences in licensing HTR-10 are of great reference value for the modular concept of high temperature gas cooled reactors. The main safety issues would be roughly the same with the modular concept and the methodology used in licensing the HTR-10 should be to great extent applicable when licensing a modular power reactor

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GAS-TURBINE POWER CONVERSION SYSTEM DEVELOPMENT

Session 4

DEVELOPMENT OF COMPACT HEAT EXCHANGER WITH DIFFUSION WELDING

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Abstract

A plate fin type compact heat exchanger (PFCHX) normally uses brazing for connecting plates and fins. However, the reliability of brazing is insufficient when PFCHXs are used for a long duration as primary or secondary components in nuclear plants. Particularly, PFCHXs used as a recuperator of gas-turbine plant with the High Temperature Gas-cooled Reactor (HTGR) or Intermediate Heat Exchanger (IHX) in future generation HTGRs need high reliability in a high temperature region.

We have been developing the PFCHX with diffusion welding between plates and fins. The tensile and creep strength in the diffusion welds are superior to those in the brazing especially in high temperature condition. The developing PFCHX consisting of Ni based Hastelloy XR plates is expected to be used over 900 °C.

Prior to the development of a full scale PFCHX, the small PFCHXs with the diffusion welding were designed, manufactured and installed in a test loop to investigate the welding strength and reliability. The early tests showed that reliability of the diffusion welding is very high, and the PFCHX with the diffusion has a possibility for the IHX or recuperator. Thermal performance tests were also carried out to obtain effective thermal conductance. This paper describes the design and test results of the small compact heat exchanger with the diffusion welding.

1. Introduction

The High Temperature engineering Test Reactor (HTTR) [1] being constructed in Japan Atomic Energy Research Institute (JAERI) is a graphite-moderated and helium-cooled reactor with an outlet gas temperature of 950°C. The HTTR will be used to establish and upgrade the technological basis for advanced HTGRs and to conduct various irradiation tests for innovative high temperature basic researches. It can be also used for demonstration of various nuclear process heat applications that are strongly required from various standpoints such as CO_2 emission reduction, efficient use of energy and so on.

The Intermediate Heat Exchanger (IHX) [2] which can be used over 900°C condition is an essential component for process heat applications. The Helically coiled IHX (HCIHX) of 10MW has been developed and installed in a primary cooling system (PCS) in the HTTR. Various R&D tests and design works were carried out to develop materials and component structures in the HCIHX. The HCIHX will be tested in high temperature condition during the HTTR operations in order to investigate thermal-hydraulic characteristics and structural integrity. The technology basis for the HCIHX will be established in these tests in the HTTR.

The problems of the HCIHXs are their size and cost. In the HCIHX used in a 400-600 MW thermal HTGR plant, its manifold to support heat transfer tubes cannot be easily manufactured because a welding structure is not proper to keep creep strain and creep fatigue damage lower than an allowable level. Even if a big forging manifold is manufactured, the plant loses economical competitiveness. On the other hand, a compact heat exchanger (CHX) has 20-30 times larger heat transfer area per volume than the HCIHX. The CHX is expected to be the best heat exchanger in the future generation HTGRs with process heat application. It can be also used as a recuperator in a Gas-Turbine with the Modular High Temperature Reactor (MHTGR).

In our study, a new CHX can be used over 900°C condition is designed and manufactured. The CHX consists of only plates welded with diffusion. The tensile and creep strengths in the diffusion welding are superior to those in the brazing especially in high temperature condition.

Major features of the HCIHX in the HTTR and present CHX are described in Chapter 2 and 3, respectively. The design characteristics of the new CHX with diffusion welding and its test and results are described in Chapter 4.

2. Intermediate Heat Exchanger for the HTTR

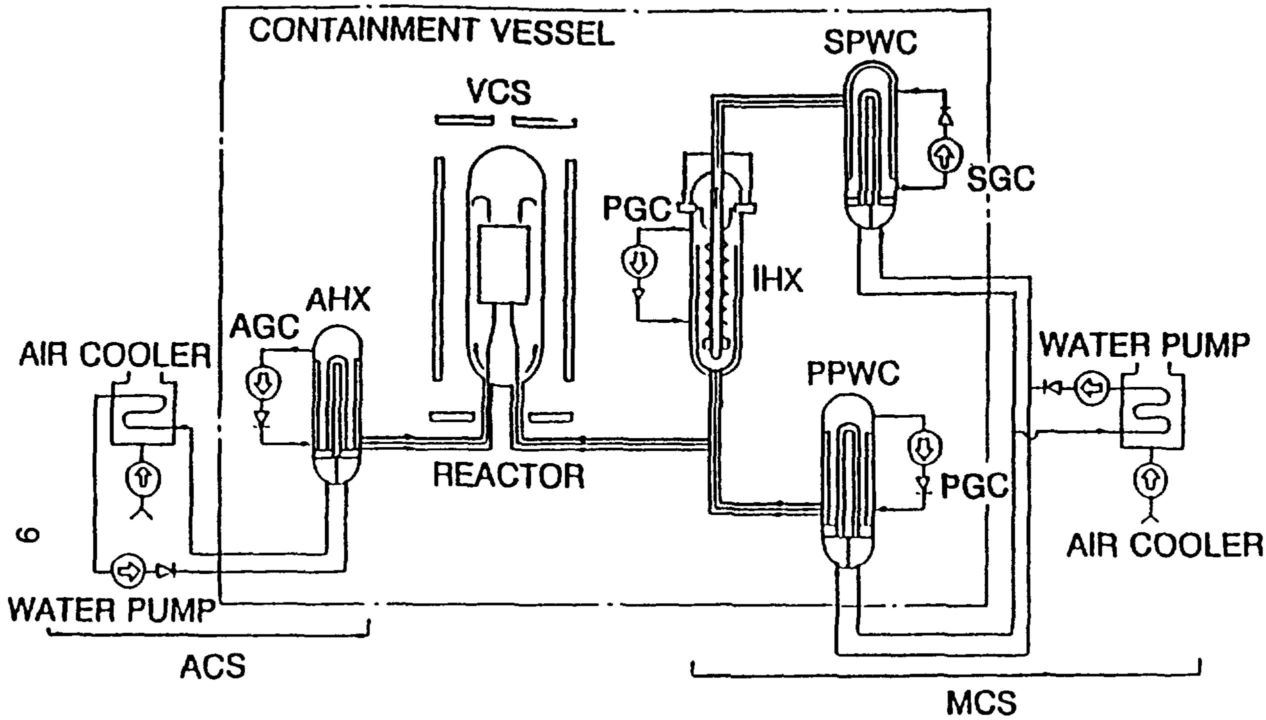
Figure 1 shows a reactor cooling system of the HTTR. The reactor cooling system of the HTTR consists the PCS, a secondary helium cooling system, pressurized water cooling system and vessel cooling system. The PCS consists of a main cooling system, an auxiliary cooling system. The main cooling system has two heat exchangers, the 10MW HCIHX and a 30MW pressurized water cooler.

The HCIHX for the HTTR is a vertical helically-coiled counter flow type heat exchanger in which the primary helium gas flows on the shell side and the secondary in the tube side as shown in Fig.2. The major specification is shown in Table 1. The primary helium gas of the maximum 950°C enters the HCIHX through the inner tube in the primary concentric hot gas duct. It is deflected under a hot header and discharged around the heat transfer tubes to transfer primary heat to the secondary cooling system. It flows to the primary circulator via an upper outlet nozzle and returns between the inner and outer shell in order to cool the outer shell.

On the other hand, the secondary helium gas flows downwards inside the heat transfer tubes and is heated up to 905°C. It flows upwards inside the central hot gas duct. The inner insulation is installed inside the inner shell to maintain the temperature of the inner shell under the allowable one. The thermal insulator outside and inside the central hot gas duct prevents the heat transfer between the primary and secondary coolant except the heat transfer area on the heat transfer tubes so that high heat transfer efficiency can be obtained, and the temperature of the central hot gas duct is maintained under the allowable one. The pressure of the secondary helium gas is controlled higher than that of primary helium gas for prevention of fission product release even if the heat transfer tube should be broken.

A tube support assemblies hold the heat tubes. Both central hot gas duct and heat tube support assemblies are hanged with a vessel top so that thermal expansion cannot be constrained.

Material of the heat transfer tubes and the hot header is Hastelloy XR, and inner and outer shells is made of 2 1/4Cr-1Mo. The Ni-base Cr-Mo-Fe superalloy Hastelloy XR was so developed as to have a superior corrosion resistance under the exposure to



MCS : Main cooling system
IHX : Intermediate heat exchanger
PPWC : Primary pressurized water cooler
PGC : Primary gas circulator
SPWC : Secondary pressurized water cooler
SGC : Secondary gas circulator
ACS : Auxiliary cooling system
AHX : Auxiliary heat exchanger
AGC : Auxiliary gas circulator
VCS : Vessel cooling system
CS : Concrete shield

Figure 1 Reactor Cooling System of the HTTR

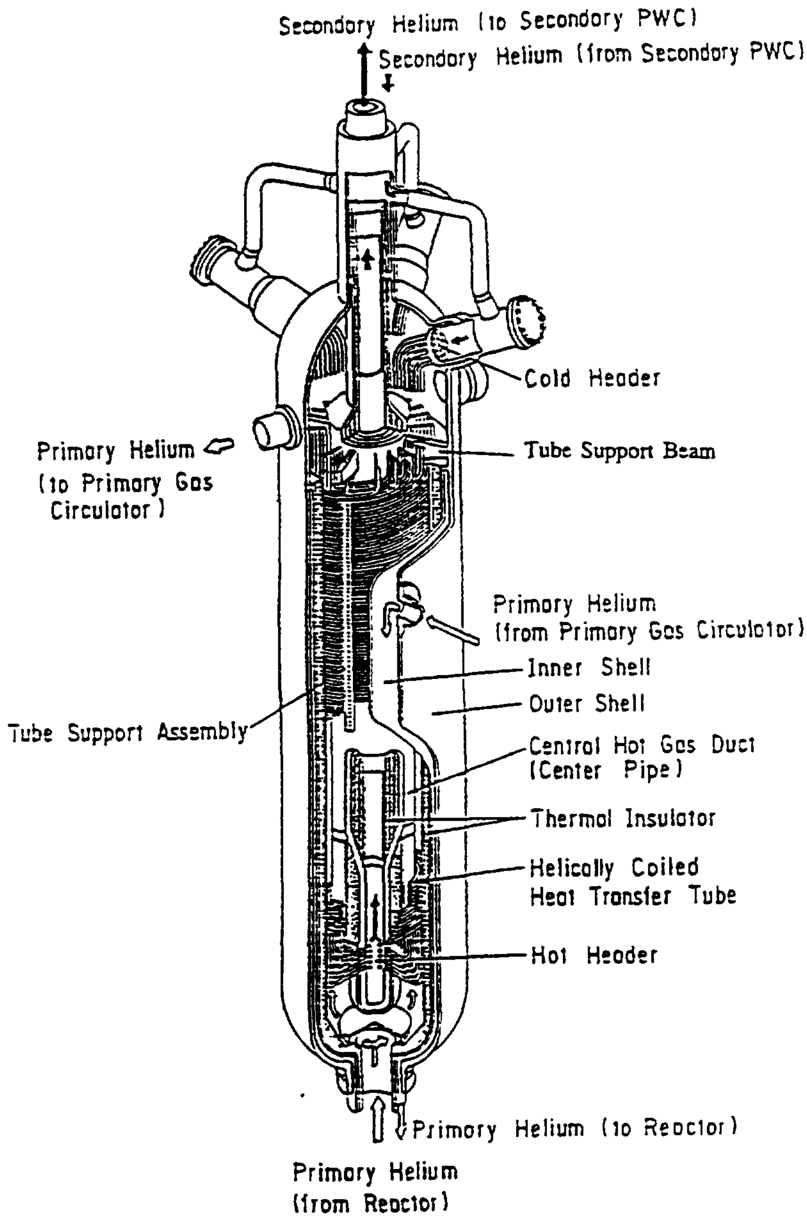


Figure 2 Bird's eye view of the IHX in HTTR

the HTTR primary coolant. The 2 1/4Cr-1Mo is also superior in anti-corrosion and strength in high temperature condition.

3. Compact Heat exchanger

The CHXs widely used in conventional plants such as fuel cells power generation plant, co-generation plant and gas-turbine plant have several advantages described as follows;

(1) A large heat transfer area per heat exchanger volume can be offered.
(2) Various kind of fin types and flow patterns can be selected.
(3) Compact size and high thermal performance can be obtained.

Table1 Major specification of IHX

Type	Helically-coiled counter flow	
Design pressure Outer shell Heat transfer tube etc.	4.7 MPa 0.29 MPa	
Design Temperature Outer shell Heat transfer tube	430°C 955°C	
Operating condition	Rated operation	High temp. operation
Flow rate of primary He	15 t/h	12 t/h
Inlet temp. of primary He	850°C	950°C
Outlet temp. of primary He	390°C	390°C
Flow rate of secondary He	14 t/h	12 t/h
Inlet temp. of secondary He	300°C	300°C
Outlet temp. of secondary He gas	775°C	905°C
Heat capacity	10 MW	
Dimension Diameter of heat transfer tube Thickness of heat transfer tube Outer diameter of outer shell Total height	31.8 mmOD 3.5 mm 2.0 m 10.0 m	
Material Outer and inner shell Heat transfer tube	2 1/4 Cr - 1 Mo Hastelloy XR	

Figure 3 shows a typical CHX used in conventional plants [3]. It consists of a plates and fins core, header and nozzles. The plates, fins, and blazing metals, are bonded by brazing in a vacuum furnace. Brazing technology for the CHX was well developed, and strength and anti-corrosion characteristics of the brazing material were confirmed by several tests. Their composing materials are stainless, aluminum and so on.

These CHXs have never used as primary heat exchangers in unclear power plants. However, they have a possibility to be used less than 600°C.

4. Development of Compact Heat Exchanger with diffusion welding

4.1 Development items and schedule

A final goal of this study is to develop the CHX that can be used as a primary intermediate heat exchanger in a future generation HTGR plant or recuperator in the MHTGR with the direct gas-turbine (GT-MHTGR).

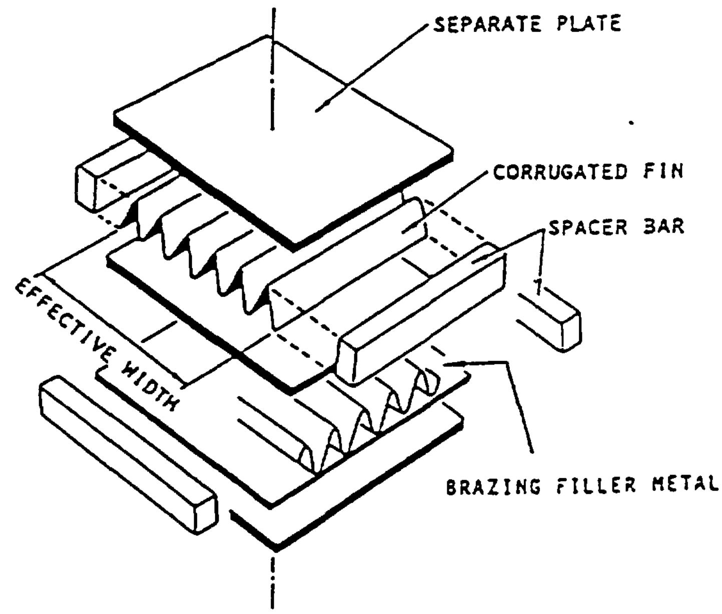


Figure 3 Typical CHX used in conventional plants

Requirements to the IHX are to keep primary coolant during normal operations and off-normal conditions and transfer heat efficiently. The boundary between primary and secondary coolant must be assured when a significant system failure occurs in a secondary system.

The recuperator in the GT-MHTGR is not used in higher temperature condition than the IHX. However, it is not easy to meet their requirements because the recuperator is a key component to improve efficiency for the GT-MHTGR[4]. Requirements to the recuperator in the gas-turbine system are as follows,

(1) Thermal effectiveness is over 90%.
(2) Pressure drop is less than 2% of total pressure drop.
(3) Pressure difference between low and high temperature helium gas is the maximum 5MPa.
(4) Life time is 40 years.
(5) Structural integrity is assured in off-normal transients, start-up and shut-down.
(6) In Service Inspection is necessary when it is used in a direct cycle plant.

Effectiveness, pressure drop and differential pressure between two sides for the recuperator used in an existing gas-turbine plant are approximately 88%, 6% of GT plant total pressure drop and 1.4MPa, respectively. In addition, components with the brazing welding have never been used in primary components. The CHX with diffusion welding is one of the best candidates to solve above problems.

In the first stage, the feasibility study of the CHX with the diffusion welding will be carried out by designing and manufacturing a small CHX. In addition, small specimens of the diffusion welding will be manufactured to measure their strength and observe their welding surfaces. In the second stage, after comparison between the CHX with the conventional brazing and new diffusion welding is conducted, the best CHXs for the GT-MHTGR and IHX are selected. In the final stage, the full scale model test will be conducted if necessary.

4.2 Design characteristics of the CHX with diffusion welding

In order to establish technological base for the CHX with the diffusion welding, the small CHXs with the diffusion welding were designed, manufactured and installed in an air loop to investigate the welding reliability and confirm the structural integrity. Table 2 shows major design characteristics of the small CHXs with diffusion welding. They only consist of Hastelloy XR plates with flow paths. The flow paths are machined by a numerical control machining system. Those plates are welded with the diffusion. They do not use the separate fins shown Fig. 3. These are unique characteristics of the new CHX and it is easy to be manufactured comparing with the CHX with brazing diffusion. Moreover, considering that it is used over 900°C condition, Hastelloy XR is selected for the plates material. Hastelloy XR is the same material as that used in the HCIHX in the HTTR and superior in creep and tensile strength in a high temperature region. Figure 4 shows the cross sectional view of the CHX.

4.3 Diffusion welding method and results

First, the best condition for the diffusion welding is determined by the tests using two small thin plates. Those plates are set in a chamber and pressurized each other by a hydrostatic generator outside the chamber. Figure 5 shows a schematic diagram of the chamber and its affiliated system.

In our study, a contact pressure, ambient temperature and holding time, which significantly affect the diffusion welding, are selected as a parameter. Besides these parameters, the ambient pressure is determined as low as possible so that the impurity composites are not included between two plates. The ambient pressure is set to 6×10^{-3} Pa by vacuum pumps in this test.

After the diffusion process, those plates are removed from the chamber and their surfaces are observed by an optical electron microscope and a scanning electron microscope(SEM).

Figure 6 shows the enlarged cross section between two plates by the optical electron microscope. Voids between two plates decrease as the contact pressure increases. The voids are not found when the contact pressure is more than 49MPa.

Effects of ambient temperature to the welding are significant. As the ambient temperature changes from 1100°C to 1150°C, voids and inclusions between two plates disappear. However, when the ambient temperature is more than 1150°C, the

Table 2 Major specification of the new CHX

Type	Plate heat exchangers with diffusion welding
Size	170mm× 90mm× 49mm
Material	Hastelloy XR
Design pressure	8.0MPa
Differential pressure between low and high temperature fluid	5.0MPa
Number of stages	10
Inlet temperature	Maximum 950°C

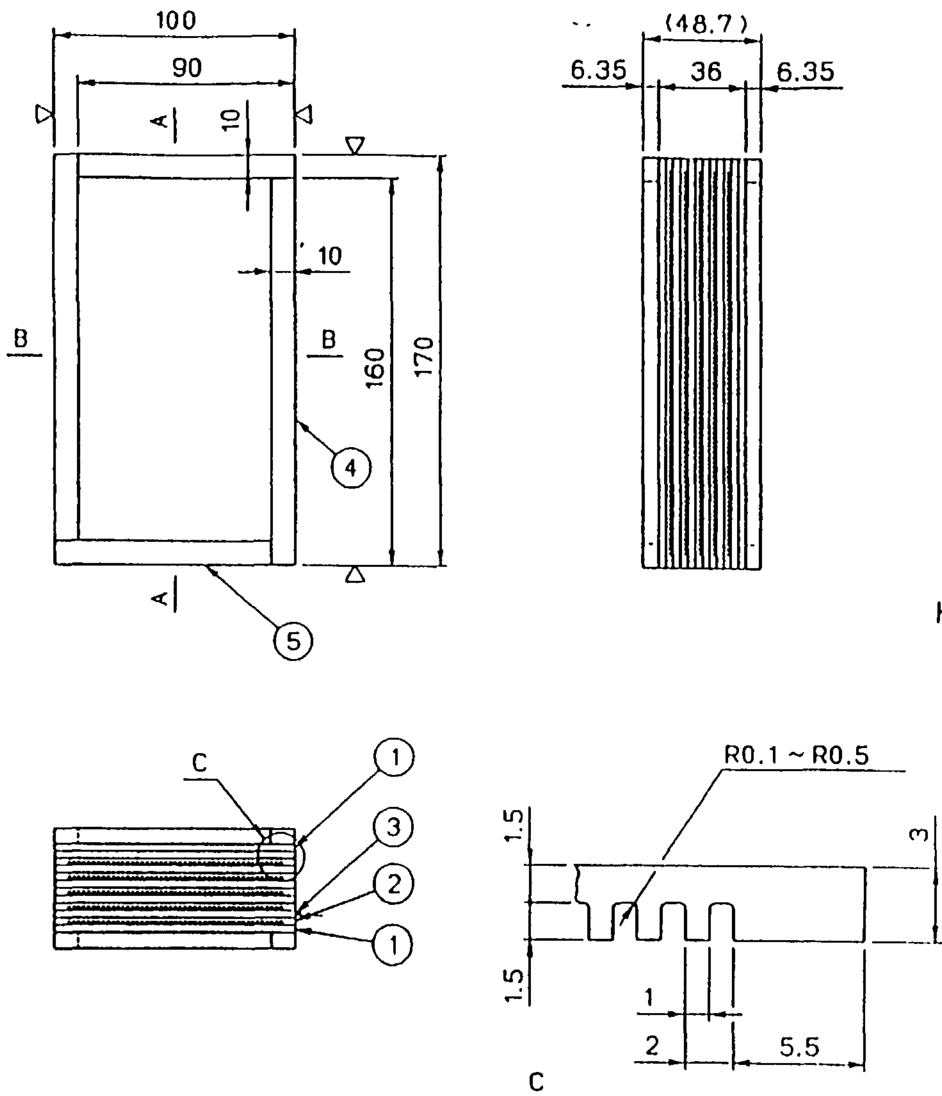


Figure 4 Cross sectional view of the new CHX

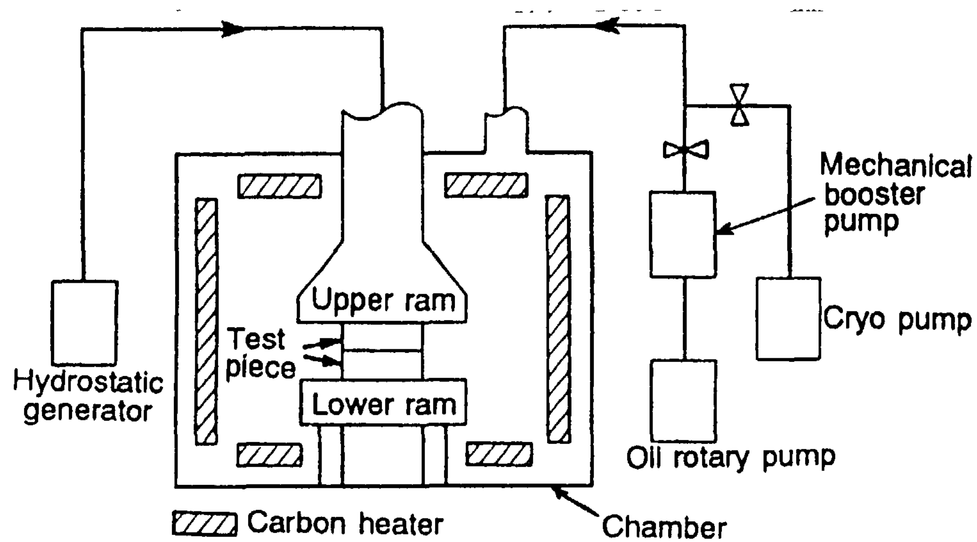


Figure 5 Schematic diagram of the chamber and its affiliated system

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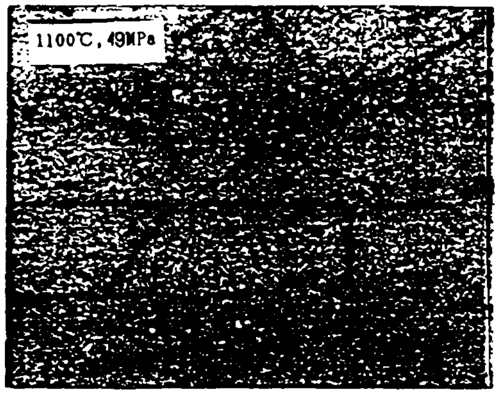

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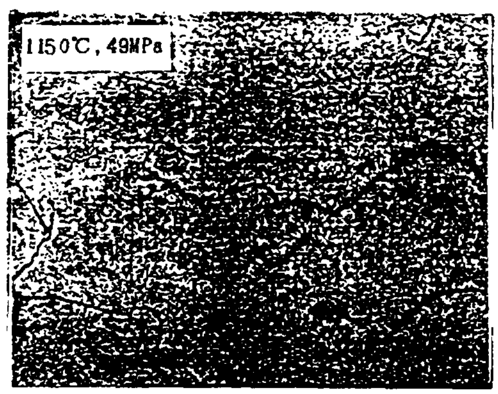

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Figure 6 Cross section of two thin plates by the optical electron microscope

deformation occurs at the edge of the plates. As for the holding time, if the holding time is over 30 minutes, the deformation at the edge is not negligible. The deformation deteriorates not only the thermal performance but also structural integrity for long duration of operations. This results prove that the ambient temperature of 1150°C, contact pressure of 49MPa and holding time of 30 minutes are selected as the best condition for the diffusion welding.

In the next tests, test specimens of the diffusion welding will be manufactured and tensile and creep strength will be obtained. The Electron Probe Micro Analyzer analysis will be also conducted to investigate inclusions in the diffusion welding.

4.4 Thermal performance of the CHX

The CHX is installed in an air loop to obtain its thermal conductance. Figure 7 shows a flow sheet of the air test loop. A compressor transfers low temperature air into the CHX. After that, it is heated up to the maximum 120°C in an electric heater and flows back to the CHX. The heat is transferred from high to low temperature air in the CHX. An exhausted air from the CHX is released to the atmosphere. Four K-type thermocouples are installed at each inlet and outlet nozzle of the CHX, and the flow rate of air is measured by a vortex flowmeter. As a result, the thermal conductance is calculated by an inlet and outlet temperatures, flow rate and specific heat of air.

Table 3 shows the results of the thermal performance test. The thermal conductance in the CHX obtained by this experiment is approximately 10% higher than that of the analytical result. The pressure drop of the experiment is 7.3% higher than that of the analysis. Internal air flow in the CHX is more turbulent than expected because the length of the flow paths in the CHX is so short that air flow cannot be stable.

Optimization of heat transfer characteristics has not been carried out because the focal point of this study is to develop the diffusion welding. The thermal conductance of this CHX does not meet the requirement from the GT-MHTGR. However, the thermal conductance is sufficiently improved by modification of the plate shape or installation of turbulent promoters in the flow paths. For example, a modified CHX where a twisted tape[5] is installed in the flow paths is assumed and its thermal conductance is evaluated. Figure 8 shows relationship between the effectiveness and the CHX height. The modified CHX with diffusion welding can achieve effectiveness of 95% and its size meet the requirement (1.5m/width × 1.5m/length × 5.5m/height/unit × 6 units). The CHX with diffusion welding has the possibility to be used for the recuperator in the GT-MHTGR.

The CHX with the diffusion welding would also become feasible for the IHX because the strength of the diffusion welding in high temperature condition is apparently higher than that of brazing, and the thermal performance and cost are superior to the HCIHX.

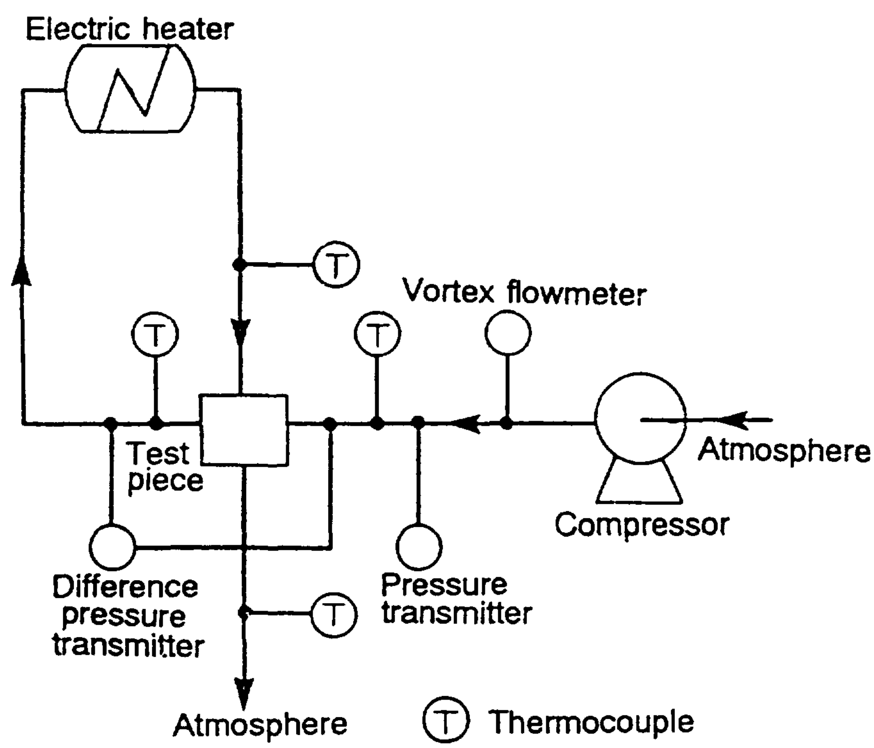

Figure 7 Flow sheet of an air test loop

Table 3 Comparison between experimental and analytical thermal conductance and pressure drop of the new CHX

Amount of Heat transfer:	533W
Inlet and outlet temp. of lower fluid:	23.9°C, 66.0°C
Inlet and outlet temp. of higher fluid:	116.0°C,73.7°C

Items	Experiment	Analysis	Error
Thermal conductance	66.8 W/m^2K	74.2 W/m^2K	9.8%
Pressure drop	5.9 kPa	5.5 kPa	7.3%

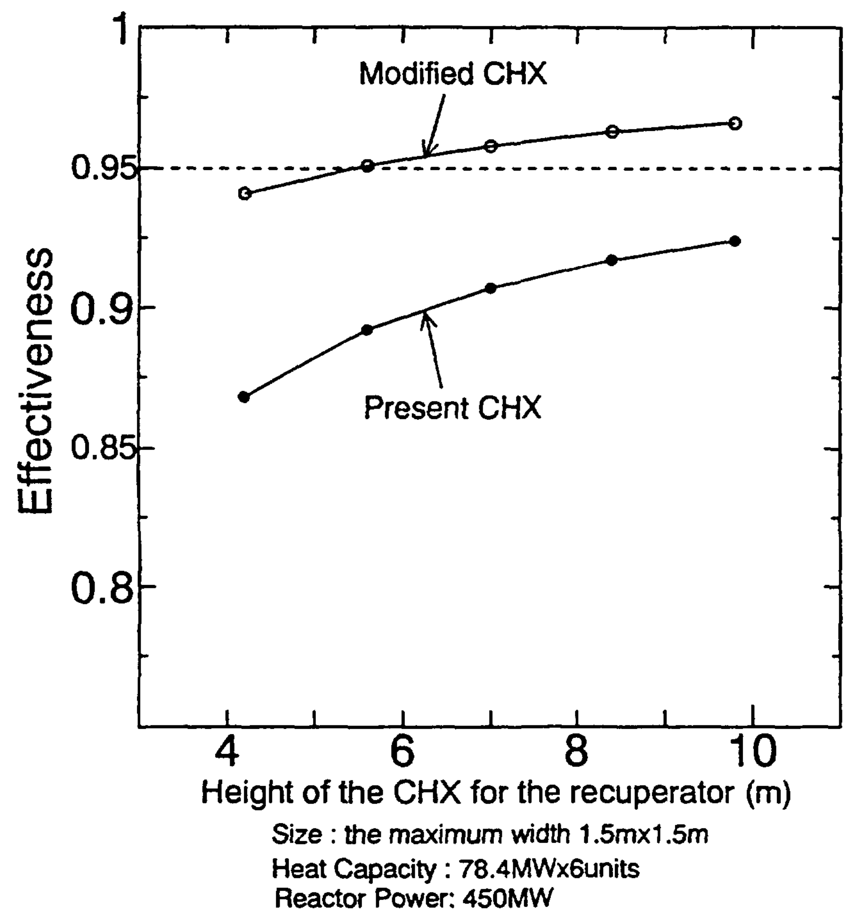

Figure 8 Relationship between the effectiveness and height of the CHX

5. Concluding Remarks

It is confirmed that the CHX with diffusion welding is one of the best candidates for the IHX in the future generation HTGR and the recuperator in the GT-MHTGR.

The early tests prove that the ambient temperature of 1150°C, contact pressure of 49MPa and holding time of 30 minutes are selected as the best condition for the diffusion welding. The welding surfaces between plates are sound in this case. The tensile strength and creep strength of the welding will be tested this year.

As for the thermal performance, the thermal conductance would meet the requirement from the GT-MHTGR by modification of the plate shape or installation of turbulent promoters in the flow paths.

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XA9642789

SUMMARY REPORT ON TECHNICAL EXPERIENCES FROM HIGH-TEMPERATURE HELIUM TURBOMACHINERY TESTING IN GERMANY

XA9642789

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Abstract

In Germany a comprehensive research and development program was initiated in 1968 for a Brayton (closed) cycle power conversion system. The program was for ultimate use with a high temperature, helium cooled nuclear reactor heat source (the HHT project) for electricity generation using helium as the working fluid. The program continued until 1982 in international cooperation with the United States and Switzerland. This document describes the designs and reports the results of testing activities that addressed the development of turbines, compressors, hot gas ducts, materials, heat exchangers and other equipment items for use with a helium working fluid at high temperatures.

The program involved two experimental facilities. The first was an experimental cogeneration power plant (district heating and electricity generation) constructed and operated by the municiple utility, Energieversorgung Oberhausen (EVO), at Oberhausen, Germany. It consisted of a fossil fired heater, helium turbines, compressors and related equipment. The second facility was the High Temperature Helium Test Plant (HHV) for developing helium turbomachinery and components at the Research Center Jülich (KFA). The heat source for the HHV derived from an electric motor-driven helium compressor. These facilities are shown in the photos on the following pages.

In both facilities negative and positive experiences were gained. At initial commissioning operation difficulties were encountered with the EVO facility, including failure to meet fully the design power output of 50 MW. The reasons for these difficulties were identified and as far as economically feasible the difficulties were corrected. At the HHV some start-up problems also occurred, but were soon corrected. The research and development programs at both facilities can be judged successful and fully supportive of the feasibility of the use of high temperature helium as a Brayton cycle working fluid for direct power conversion from a helium cooled nuclear reactor.

Unfortunately, the HHT project was terminated in Germany and both test facilities have been shut down. Except for information on life testing the facilities accomplished their missions. If helium turbomachinery technology is reconsidered for power conversion from a helium cooled nuclear reactor, no unresolvable problems have been identified in these turbomachinery test facilities.

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1. Introduction

This report gives a survey of the design and operational experiences gained from high temperature, helium turbomachinery testing in Germany, which formed a portion of the high temperature turbine (HHT) development program. In addition appendices have been provided on other experimental programs relevant to the HHT project and on a short history of the HHT project (Appendix I).

This report describes the initial difficulties, the improvements made and the results achieved.

Some of the experimental results achieved are not publicly available, because they are classified as "proprietary information" within the gas-turbine-technology and high-temperature-technology industrial communities of Germany. Therefore in some cases only a short survey could be given.

2 Design of Test Facilities and Description of Turbomachinery

2 1 Helium Turbine Cogeneration Power Plant (EVO)*

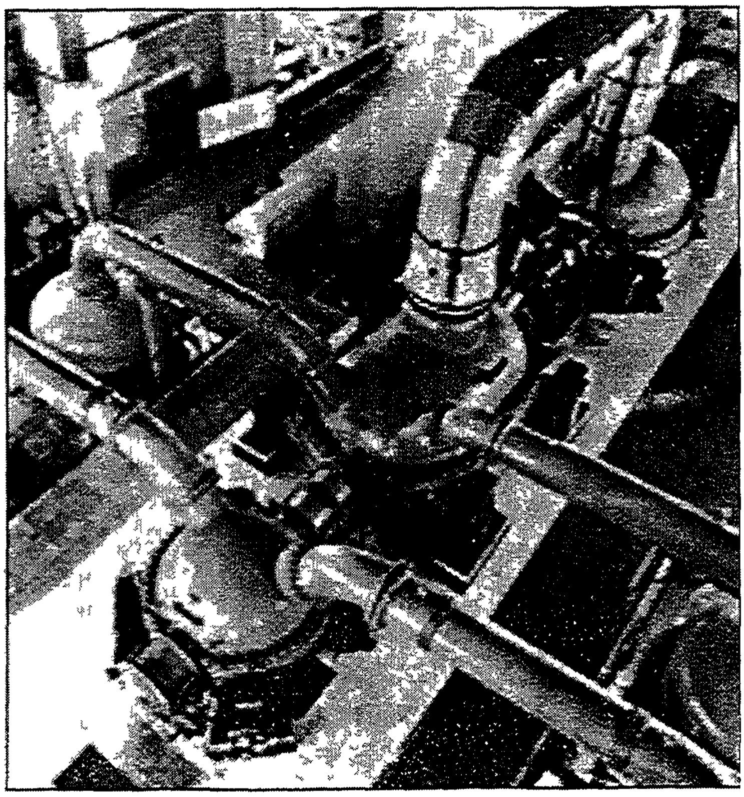
2 1.1 Development Goals

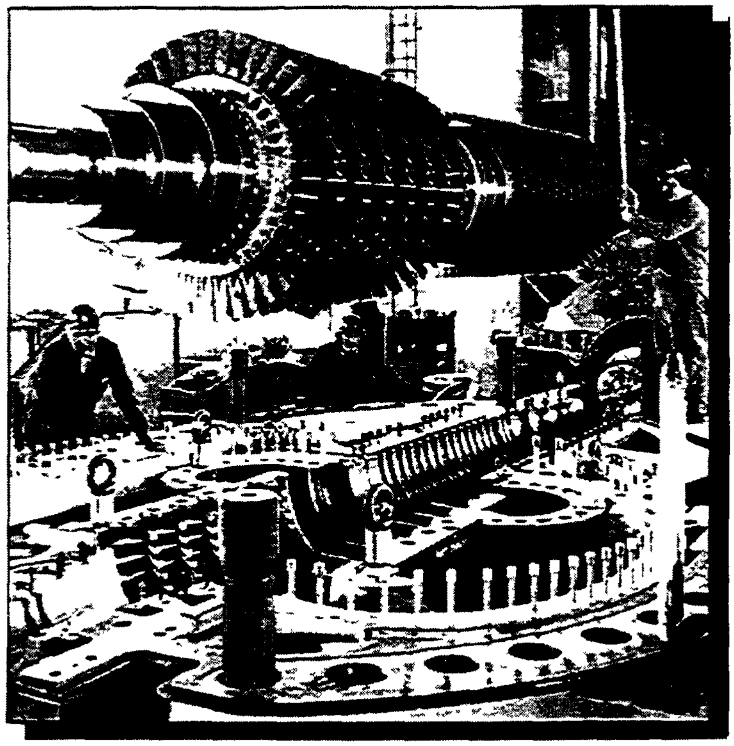
(Ref. [1], [2], [3])

Beginning in 1960 the municipal energy utility of Oberhausen (EVO) operated on its grid a closed-cycle, hot-air turbine plant for the combined generation of electricity and district heat. Due to increasing demand, an extension of the power plant capacity was decided upon in 1971.

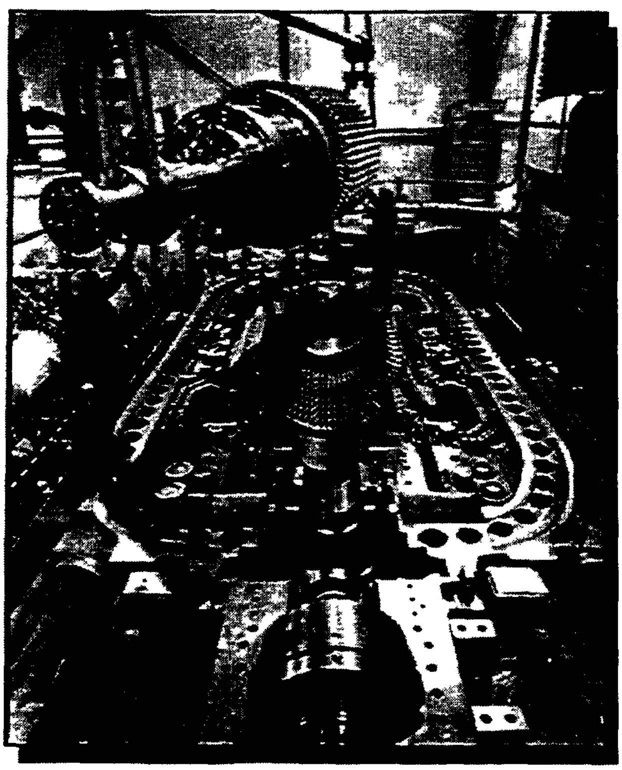
* EVO: Energieversorgung Oberhausen AG

OBERHAUSEN II
HELIUM TURBINE PLANT





HIGH PRESSURE ROTOR
FROM OBERHAUSEN II
HELIUM GAS TURBINE



ROTOR FROM HHV
TEST FACILITY

The EVO decided to apply a closed-cycle helium turbine cogeneration power plant in consideration of the requirements for a commercially operated power plant for the cogeneration of electricity and district heat. This decision recognized the favorable thermodynamic properties of helium, which promised an attractive application of closed-cycle gas turbines, as well as the goals for the international development of the High Temperature Reactor (HTR) using a direct cycle High Temperature Helium Turbine (HHT) power conversion system, as defined in the 4th Atomic Program of the Federal Republic of Germany. With this plan a large-scale practical contribution to the German nuclear program in a commercially operated power plant could be made. In order to optimize the experiences to be gained, it was decided to provide a "nuclear design" for the helium-systems and the turbomachinery and to use as high helium-temperatures and helium-pressures to the greatest extent possible, taking into account the state of the art of materials and components at the decision date.

2.1.2 Test Operation Goals

(Ref. [3])

The following experiences were to be gained with the test plant:

- Nuclear design of the helium systems and of the turbomachinery
- Mechanical, thermal, aero-dynamic and dynamic behaviour of the helium-turbine and of the low-pressure (LP) and high-pressure (HP) compressors, including rotor shafts, blades, shaft seals and cooling systems
- Acoustic emissions and propagation within the plant and into the environment
- Behaviour of hot gas ducts, bellows, valves and insulation under thermal loads, pressure gradients, acoustic emissions and helium impurities
- Experience with different operating conditions (start-up, steady state power and shut-down)
- Power regulation
- Remote operation (simulating nuclear reactor applications)
- Behaviour of all other helium-components including heat exchangers and purification system (except radioactive impurities)
- Longterm operational behaviour of all helium components and helium systems as well as of the turbomachinery under the conditions of a commercially operated, cogeneration plant

2.1.3 **Design and System Description**

(Ref. [4], [5], [6], [7], [8], [9], [10], [11], [12], [13], [14], [15],[16])

a) <u>Overall Design</u>

The design of the EVO test plant provided for an electrical power of 50 MW and a heating power (district heat) of 53.5 MW. The inlet temperature into the turbine was limited to 750 °C, taking into account the life endurance of the materials for the first turbine stage blades and of the helium heater.

A two-shaft design was selected for the turbomachinery. The high- pressure turbine drives the compressors on the first shaft and the low-pressure turbine drives the generator by a separate shaft. Both rotors are interconnected by a gear.

Air was applied as the working media for all previously operated closed-cycle gas turbines at EVO, because the physical properties of air had been well known. Compared to air, helium has distinct advantages for a closed-cycle gas turbine. First of all it is a chemically inert gas. Moreover, it has very advantageous physical properties as can be seen from Table 2.1./1.

Table 2.1./1: Physical Properties of Air and Helium

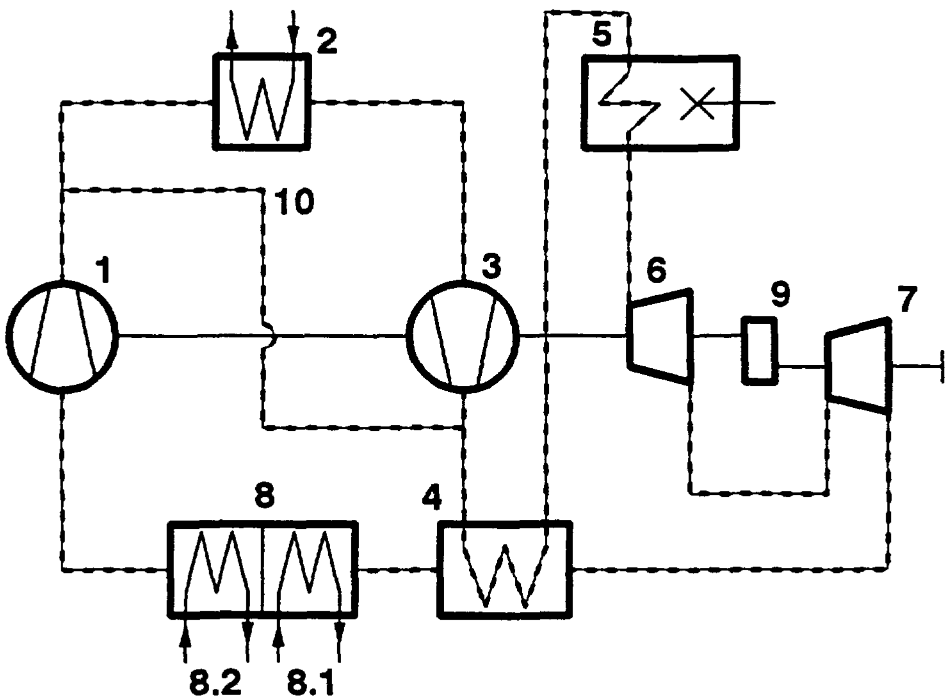
		Air	Helium
Sound velocity at	20 °C 600 °C	343 m/s 584 m/s	1007 m/s 1738 m/s
Specific heat cp at	1 bar 20 °C 30 bar 600 °C	1010 J/kg K 1120 J/kg K	5274 J/kg K 5274 J/kg K
Isentropic coefficient	1 bar 20 °C 30 bar 600 °C	1.40 1.36	1.665 1.665
Heat conductivity	1 bar 20 °C 30 bar 600 °C	0.0265 W/m K 0.0628 W/m K	0.1466 W/m K 0.3287 W/m K
Dynamic viscosity	1 bar 20 °C 30 bar 600 °C	1.82 10^{-5} kg/ms 3.87 10^{-5} kg/ms	1.98 10^{-5} kg/ms 4.18 10^{-5} kg/ms

The specific heat of helium (a measure for the heat capacity and heat transport) is five times larger than that of air, thus requiring smaller heat transfer areas. The sound velocity is three times as large, resulting in design advantages for the turbomachinery. In particular, the permissible circumferential velocity of the heliumcompressor is no longer limited by sonic speed consideration but by the centrifugal stresses of the blades. A characteristics of helium's large specific heat is that the enthalpy difference between preselected temperatures is accordingly large, resulting in a larger number of stages for a helium than for an air turbine.

b) Flow Scheme

The basic flow scheme of the EVO helium turbine plant is shown in Fig. 1.

The helium working media is initially compressed in the low pressure compressor (1), recooled in the intercooler (2) and additionally compressed to the plant design pressure of 30 bar in the high pressure compressor (3). Preheating occurs in the recuperater (4) and final heating to the plant design temperature of 750 °C occurs in the gas-fired heater (5). The compressed and heated helium then expands in the high-pressure (HP) and low-pressure (LP) turbines, (6) and (9) respectively. The compression ratio between the low-pressure and high-pressure compressors was selected in such a way as to optimize the heating power for district heating, which is transferred within section (8.1) of the precooler, while section (8.2) further cools the helium working fluid prior to its entering into the low-pressure stage of the compressor (3).



Legend:

1	Low pressure compressor	7	Low pressure turbine
2	Intercooler	8	Precooler
3	High pressure compressor	8.1	District heat removal section
4	Recouperator	8.2	Precooler section
5	Heater	9	Gear
6	High pressure turbine	10	Regulation bypass

Fig. 1: Flow-scheme

The thermodynamic cycle point conditions for the plant are shown in Table 2.1/2.

Table.2.1/2: Thermodynamic Cycle Point Conditions Data

No.	Component	Temperature ° C	Pressure bar
1	Low Pressure Compressor	25	10,5
2	Intermediate Cooler	83	15,5
3	High Pressure Compressor	25	15,4
4	Precooler High Pressure Side	125	28,7
5	Heater	417	28,2
6	High Pressure Turbine	750	27,0
7	Low Pressure Turbine	580	16,5
8.1	Heater Inlet	460	10,8
8.2	Cooler Inlet	169	10,6

The inlet pressure of 27 bar into the HP turbine was selected as a compromise. The largest pressure for closed cycle gas turbines with air was 40 bar at that time. Conditions for a nuclear heated plant provided with a helium-turbine were foreseen as about 60 bar, a helium-temperature of 850 °C and a plant capacity of 300 MWe. The low pressure turbine was selected to deliver power at a frequency of 50 Hz or 3000 rpm and is connected by a gear arrangement to the high pressure turbine shaft that rotates at 5500 rpm. It has dimensions and stress loadings for the rotor shaft for the blades and for the turbine housing which are very similar to those of a 300 MW helium-turbine plant considered at the time as the reference size. In addition to optimization requirements a reason for the limitation of the primary pressure was the permissible stresses for the material of the tubes of the fossil-fired heater.

2.1.4 **Turbomachinery Description**

(Ref. [4], [5], [6], [7], [12], [13], [14], [15])

The rotor shafts of the LP compressor (see Fig. 2) and the common rotor shaft of the HP compressor and HP turbine (see Fig. 3) were coupled by a gear assembly for the nominal revolution of the HP shaft is 5500 rpm. The LP compressor was located in its separate housing and the HP compressor and the HP turbine were located within a common housing. The split-up of these three machines into two housings was made to avoid natural frequencies of the rotor shafts at run-up, operational speed and rundown.

In order to ensure an easy accessibility to the bearings between the LP compressor and the HP compressor without requiring the removal of the main housings the two housings were interconnected by a pressure-tight tunnel.

As shown in Fig. 2, a ball-shaped housing, horizontally divided and welded from 8 ball-shaped segments, was provided for the LP compressor. Thus led to minimizing both the housing wall thickness and the helium-leak rate. The LP compressor had 10 stages. The blading was designed according to the potential vortex theory. Along the complete blade length a reaction ratio of 100 % was provided.

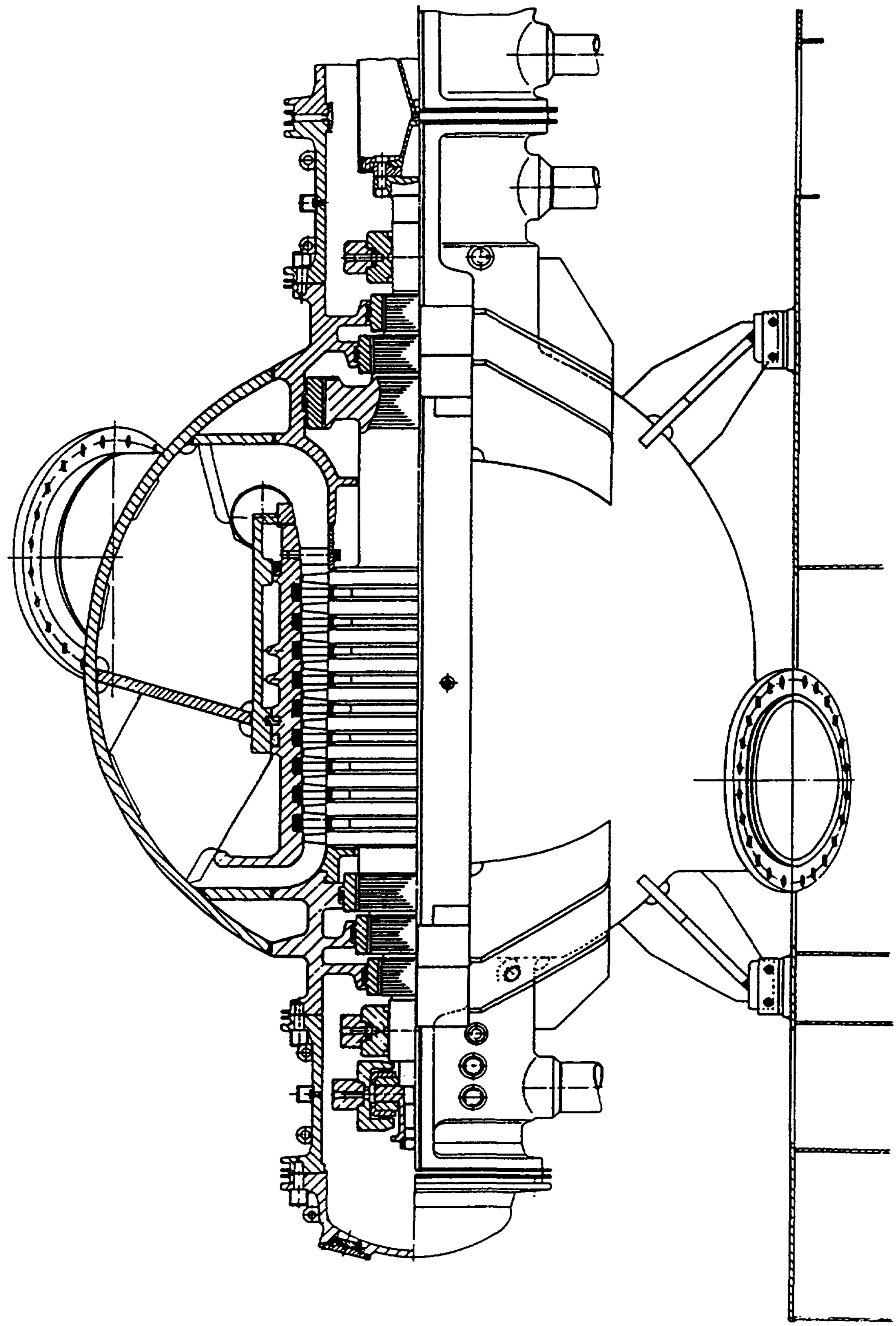


Fig. 2: LP-Compressor

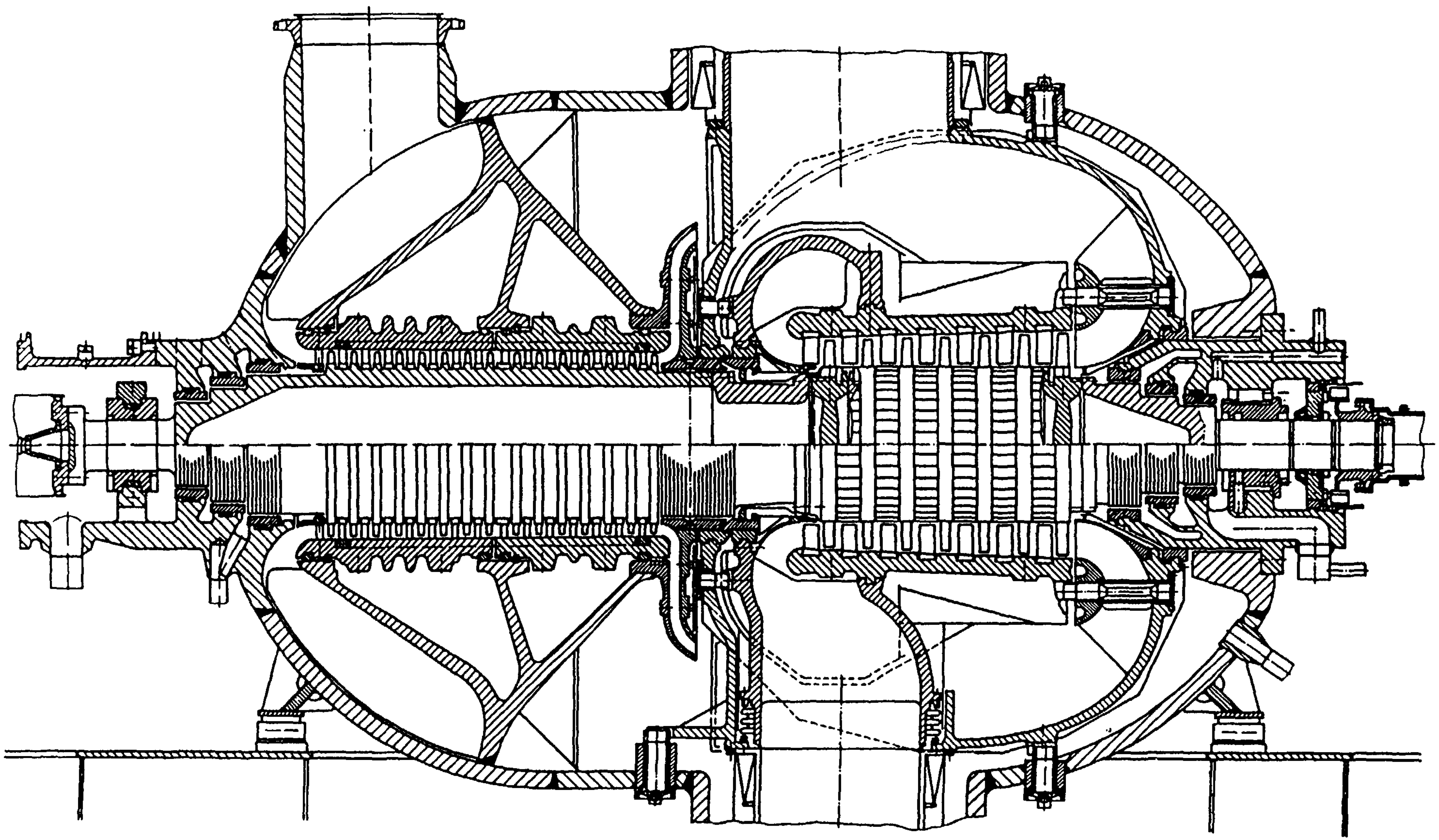


Fig. 3: HP-Compressor and HP-Turbine

As shown in Fig. 3, a common housing was provided for the HP compressor and the HP turbine, similar to the provisions for the proven hot air turbine plants. In view of the natural frequencies, the rotor shaft of the HP compressor was provided with a bore, whereas the rotor of the HP turbine was formed by single disks connected via a Hirth-type coupling. Both rotor sections were firmly bolted to each other by a tube-shaped bolt. The design of the HP turbine housing was made with regard to the design of large-scale turbines for nuclear heated plants. Therefore no inner insulation, be it foils or fibres, was provided. Rather a separate inner turbine housing was provided. which was cooled by the passage of outlet helium from the HP compressor.

The common housing of this unit is made of 8 ball-shaped segments at the ends and a cylindrical center section.

The HP compressor has 15 stages, which like in the case of the LP compressor are designed for a reaction ratio of 100 %. The HP turbine has 7 stage blading with a reaction ratio of 50 % for the medium blade-length of the final stage. The turbine rotor is provided with cooling for the rotor disks and for the blade feet. Since at nominal power the HP turbine power is just sufficient to drive the LP compressor as well as the HP compressor, the gear interconnected between HP section and LP turbine transfers only a small amount of power.

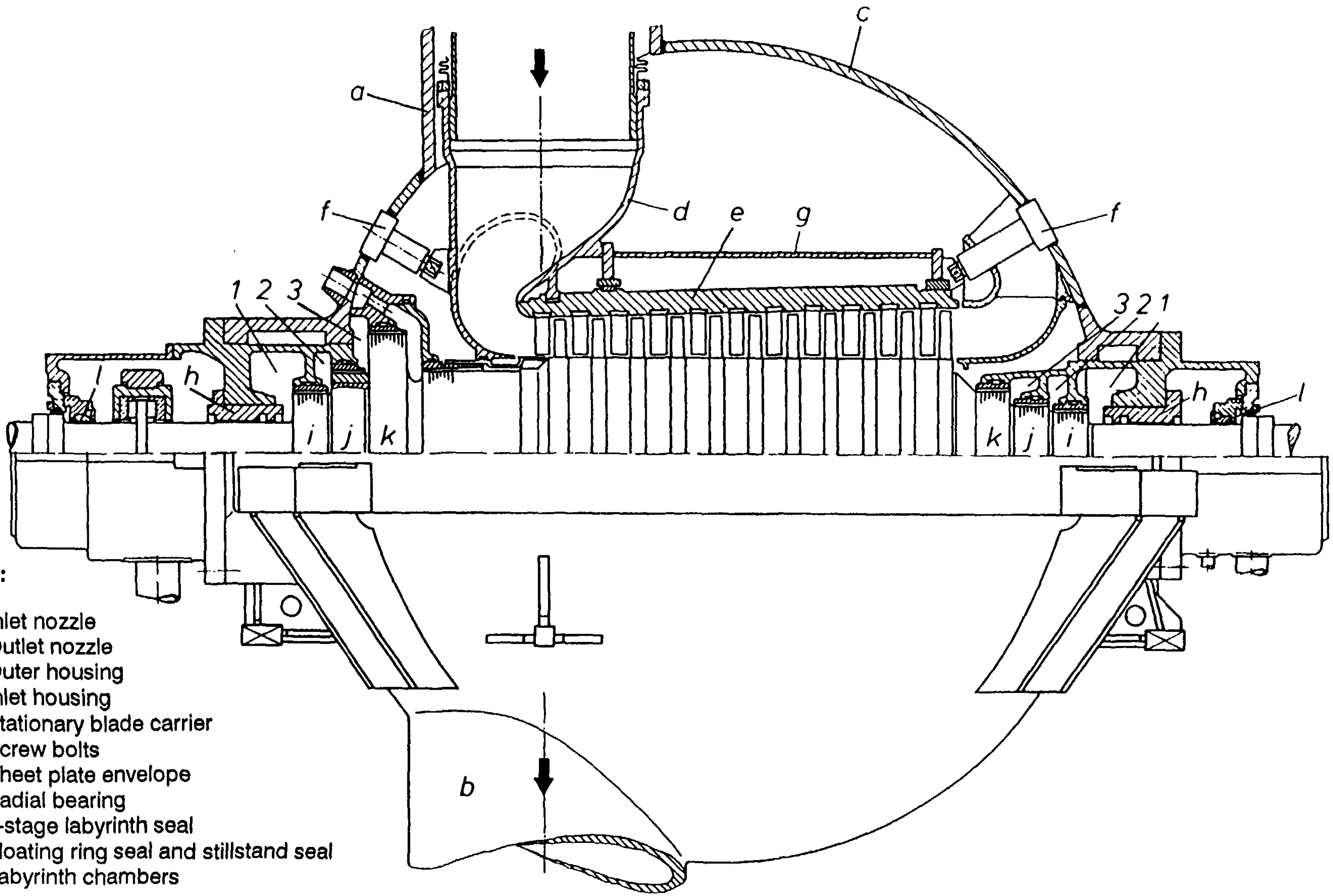
Fig. 4 shows the 11-stage LP turbine provided with a ballshaped housing also made of 8 ball-shaped segments. In accordance with the prevailing grid frequency of 50 Hz the LP turbine operated at 3000 rpm. The inlet nozzle (a) was located at the top of the housing and the outlet nozzle (b) was at the bottom, thus minimizing the forces on the connecting pipes. Like in case of the HP turbine, no inner insulation was provided. However an outer insulation on the outer housing made of mineral wool was provided. The inner housing as well as the inlet nozzle and inlet housings, located inside the outer housing, were cooled by the outflowing helium. The inlet section of the housing (d) as well as the stationary blade carrier (e) were accurately adjusted by means of several screw bolts (f) located on the outer housing. In order to ensure a better thermal insulation the stationary blade carrier (e) was enclosed by a thin sheet-plate envelope (g).

For the sealing between bearings and the helium-circuit, a seal gas system consisting of 3 labyrinth seals (i, j, k) was chosen. Into the chamber 2, provided between the labyrith seals (i) and (j) sealing helium was introduced. (In the case of a nuclear plant, clean helium would there be introduced). In the labyrinth seal (i) the helium-flow was directed toward the bearing (chamber 1). The sealing helium was mixed there with the oil drained from the bearing and was subsequently regained in an oil-helium-seperation system.

Helium from the primary side was directed into chamber 3 and mixed there with the sealing helium flowing from chamber 2 through the labyrinth seal (i). From there it flowed through the labyrinth seal (k) into the primary helium-system.

By means of an adequate regulation of the pressure levels within the chambers provided between the labyrinths seals it was ensured (for future application to a nuclear heated system) that under no conditions of operation would primary helium carry radioactive impurities into the oil circuit. In order to seal the shaft penetration at the housing, a floating ring seal and a static seal were provided. This sealing principle was used for all 3 turbomachines.

The common rotor shaft of HP turbine and HP compressor was supported on two bearings provided with multiple plain surfaces. The compressor section of the rotor shaft was made of one forged piece using low-alloyed steel. This semi-finished forging was then machined to a



Legend:

a) Inlet nozzle
b) Outlet nozzle
c) Outer housing
d) Inlet housing
e) Stationary blade carrier
f) Screw bolts
g) Sheet plate envelope
h) Radial bearing
i), j), k) 3-stage labyrinth seal
l) Floating ring seal and stillstand seal
1, 2, 3 Labyrinth chambers

Fig. 4: LP-Turbine

drum-shaped rotor. The turbine rotor shaft was made of high-alloyed austenitic steel disks connected by means of a Hirth-type coupling and screw bolts to form the turbine rotor. HP compressor rotor and HP rotor were both connected by screw bolts to form the complete common rotor shaft. In spite of the provided blade foot cooling, austenitic steel was chosen for the turbine rotor in order to ensure an appropriate behaviour in case of all credible operationing conditions.

The blades for the HP compressor were made of forged semi-finished parts, which subsequently were precisely machined. The blades for the HP-turbines were made by precision forging.

Due to the high sonic velocity of helium, it was possible to provide higher circumferential speeds without closely approaching the sonic range. Therefore, it was possible to design the compressor at the same circumferential speed for a 100 % reaction ratio, resulting in a higher enthalpy increase step than in case of a compressor stage with a 50 % reaction ratio.

The bearing housings of all three turbomachines could be removed separately without opening the horizontal flanges of the main housings. The flanges were provided with welded lip seals.

For the cooling of the housings of the three turbomachines, a multiple housing system with a flow of low-temperature helium at the outer section was provided in order to avoid inner insulation. In order to cool the HP turbine rotor and blade feet, a cooling flow of helium was extracted from the HP compressor outlet and guided through the bore of the rotor shaft to the ring-shaped chambers of the turbine rotor disks and then finally, after the last turbine stage, ducted into the main helium-stream. The ring-shaped chambers were formed by the outer surfaces of the rotor disks and the blade shoulder.

2.1.5 Power Regulation

(Ref. [4], [5])

The power regulation of the closed-cycle, helium turbine plant was designed using the same principles used for the closed-cycle air turbine plants. In the normal case the power was regulated by means of a pressure level regulation. When lowering the load, helium was extracted from the main stream after the HP compressor through a regulation valve into storage tanks. In case of a load increase, helium was returned from the storage tanks through an inlet valve arranged ahead of the LP compressor into the main system.

For plant operation prior to grid-synchronization the regulation of the machine revolutions occurs by means of a regulated bypass line provided between outlet of the HP compressor and outlet of the LP compressor. Thus the high-pressure section of the heat exchanger and the helium-heater as well as the turbines would be bypassed. Since the required compressor power remained nearly constant, only about a third of the circulated helium must be bypassed in order to achieve zero net power output.

2.2 High Temperature Helium Test Plant (HHV)

2.2.1 HHV Goals

(Ref. [17], [18], [19])

The HHV plant was to be an integral part of the development project for a high-temperture reactor with a direct-cycle, helium turbine of large capacity (HHT). This project was carried out in an international cooperation between the Federal Republic of Germany, Switzerland and the United States.

Although proven gas turbine technology was used to the largest possible extent, the development work for HHT has shown that experimental tests of the new and vital HHT components were required. Since no test facility of sufficiently large size was avaible, it was decided to set up a new test facility at the Research Center Jülich (KFA). This should meet the HHT requirements with regard to the necessary test conditions (sufficiently large helium flow, high helium temperature).

2.2.2 Test Operation Goals

(Ref. [16], [17], [18])

The goal HHV plant was to test HHT components of sufficiently large size to permit the extrapolation for HHT use.

The following test loop conditions were specified:

- Helium mass flow:approximately 200 kg/s
- Helium temperature: 850 °C with the possibility to reach 1000 °C for short time periods
- Helium operating pressure:50 bar
- Adequate helium atmosphere with regard to non-radioactive impurities
- Sufficiently large test section for employing sufficiently large components to enable an extrapolation to the HHT.

Test results were to be achieved in the HHV for the following main HHT components and systems:

- Helium turbine with gas seals, oil seals, bearings, cooling systems, insulation
- Helium compressor with gas seals, oil seals, bearings
- Helium/helium heat exchangers/recuperators, coolers
- Hot gas ducts with valves, butter fly valves, bellows, bends
- Helium purification
- Coatings in helium atmosphere
- Material behaviour in helium atmosphere at high temperatures
- Instrumentation
- Acoustic emissions and propagations
- Dust and particles in the helium-circuit

2.2.3 Design and System Description

(Ref. [17], [18])

a) Basic Design

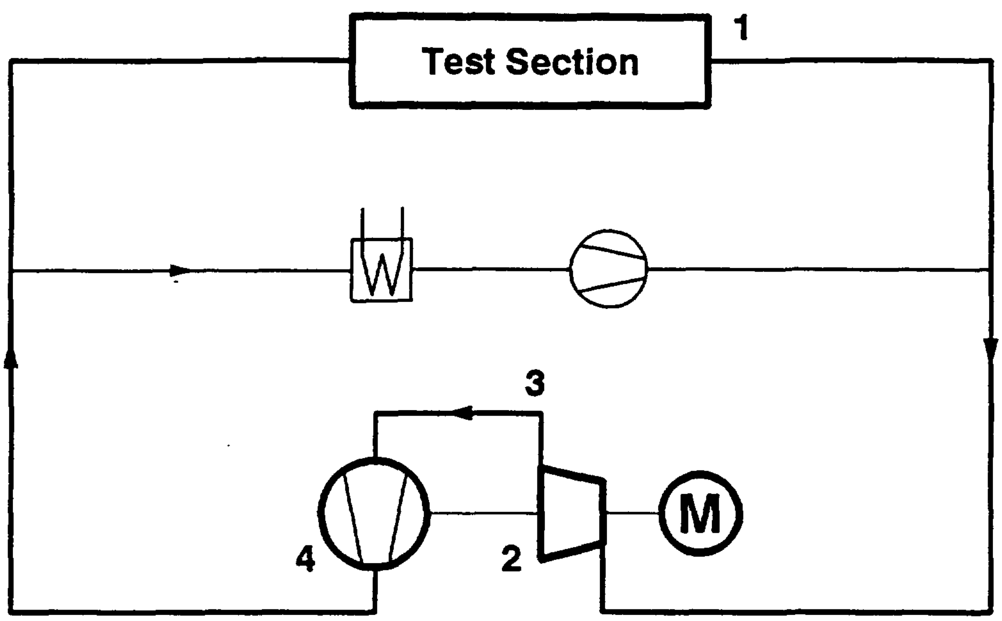
The first design considerations for the HHV plant began with a system, where a fossil-fired heater replaced the nuclear heat source. However, because of the desired peak temperatures of 850 °C (1000 °C for shorter periods) and the design pressure of 50 bar there were feasibility concerns about the lifetime capability of such a fossil-fired heater. Thus a test circuit was chosen comparable to a closed-cycle gas turbine plant as shown in Fig. 5.

In this arrangement the compression heat from the compressor is used to heat up the helium to the desired temperature. The required compressor power was 90 MW with 45 MW generated and regained by the expansion in the gas turbine and 45 MW introduced by the electric drive motor. The selected combination of the turbine and compressor on one shaft resulted in dimensions being comparable with a helium turbine of 300 MW capacity (the reference plant size at the time).

b) Design Description

The flow scheme of the HHV test loop is shown in Fig. 6.

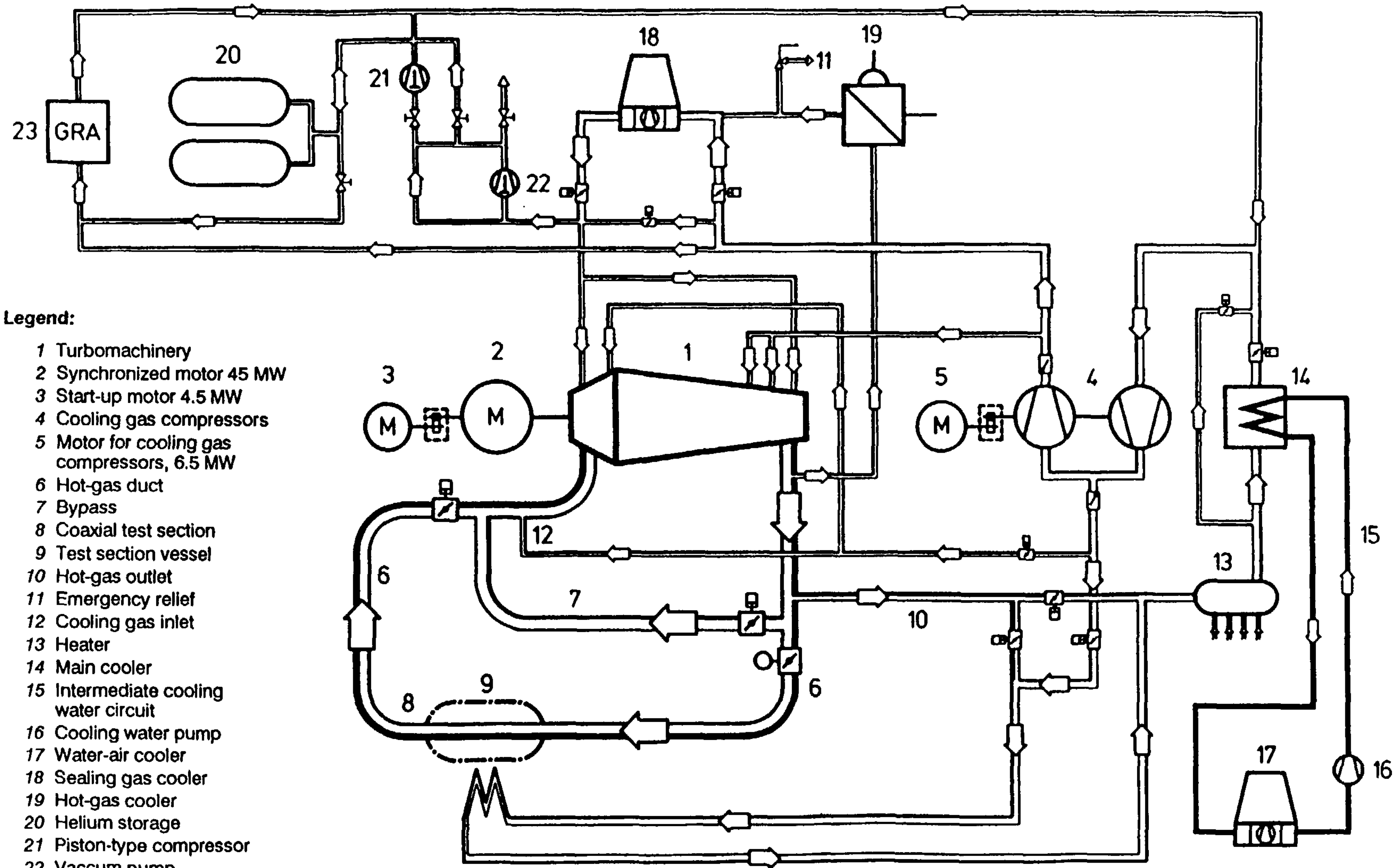
Helium was circulated in the hot gas system by means of the electrically-driven turbomachinery. A synchronous-motor at 3000 rpm was used as the electrical drive.



Legend:

1 Test section
2 Helium-turbine
3 Duct to compressor inlet
4 Compressor

Fig. 5: HVV test circuit



Legend:

1 Turbomachinery
2 Synchronized motor 45 MW
3 Start-up motor 4.5 MW
4 Cooling gas compressors
5 Motor for cooling gas compressors, 6.5 MW
6 Hot-gas duct
7 Bypass
8 Coaxial test section
9 Test section vessel
10 Hot-gas outlet
11 Emergency relief
12 Cooling gas inlet
13 Heater
14 Main cooler
15 Intermediate cooling water circuit
16 Cooling water pump
17 Water-air cooler
18 Sealing gas cooler
19 Hot-gas cooler
20 Helium storage
21 Piston-type compressor
22 Vaccum pump
23 Helium purification system

Fig. 6: Overall flow scheme of the HHV

The highest helium-temperatures were achieved at the HP compressor outlet. Hot helium could be conducted completely or partially through the test section or directly bypassed back to the turbine for expansion by means of hot gas ducts provided with regulation valves.

A helium/water cooler was provided in order to ensure the desired temperature equilibrium between added and removed heat as well as to supply cooling gas and sealing gas.

The principal design parameters for the helium circuit are given in the following Table 2.2/1.

Table 2.2/1: Principal Design Parameters of Helium Circuit at Nominal Power

	Temperature °C	Pressure bar	Flow Rate kg/s
Compressor outlet	850	51.20	212.0
Test section inlet	850	51.20	201.0
Turbine inlet	826	49.70	209.0
Main cooler inlet	390	49.50	53.5
Cooling gas compressor inlet	236	49.00	56.8
Cooling gas for coaxial hot gas duct	300	51.20	22.9
Sealing gas outlet	50	52.75	2.3

The total helium content in the circuit was approximately 8000 Nm^3 and the helium throughput for purification was approximately 1000 Nm^3/h.

2.2.4 Description of Turbomachinery

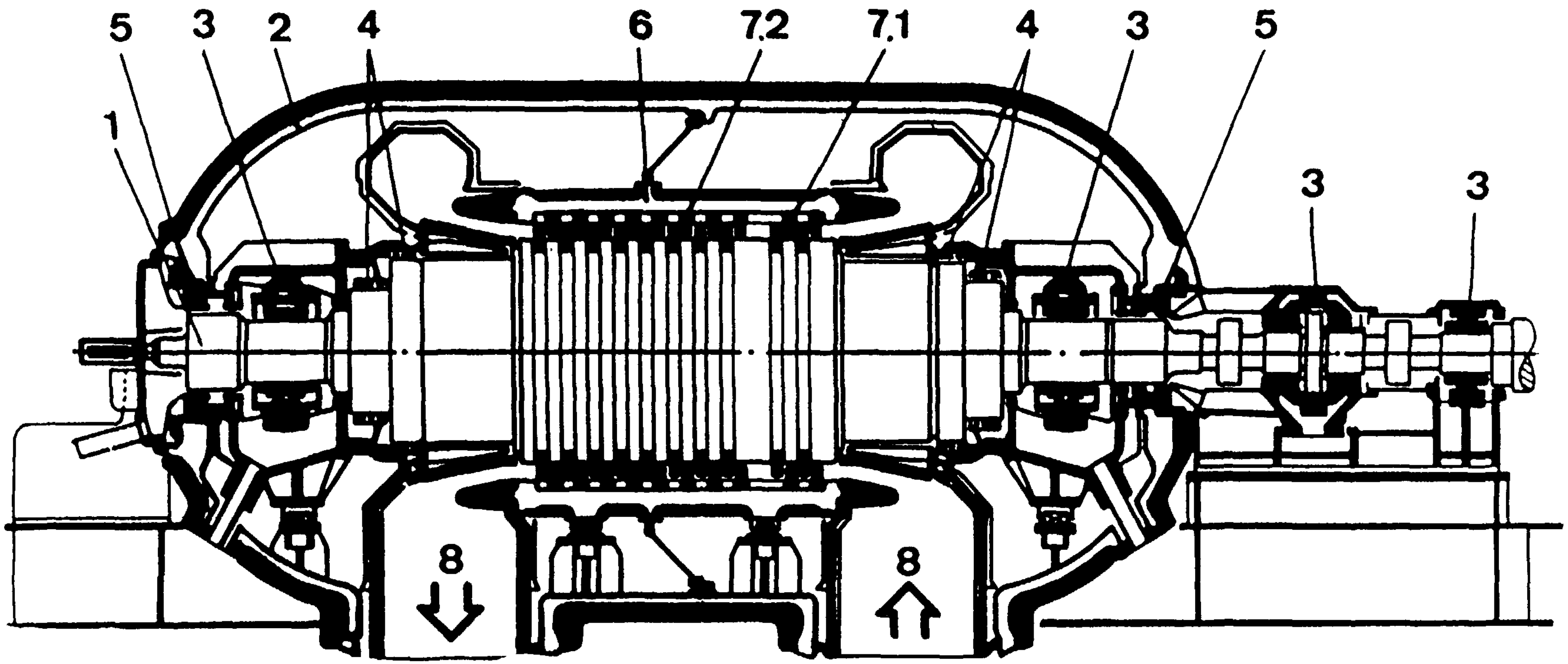
(Ref. [17], [18])

As shown in Fig. 7, the turbine and the compressor were arranged on a single shaft. The turbine has two stages, the compressor eight.

Helium flowed through the inlet nozzle into the turbine section. Behind the compressor outlet helium flowed through the outlet diffusor into the outlet nozzle and from there into the hot gas duct. Inside the turbomachinery only the inlet and outlet nozzles, the diffusors and the blading channel were exposed to high temperatures. All other sections of the machine (blade feet, rotor and housing) were cooled by means of the cooling gas system or the sealing gas system (see Fig. 8).

The rotor provided for the turbomachine had a weight of 60 Mp, a bearing distance of 5.7 m and a total length of about 8 m. The blading channel had an inner diameter of 1.6 m and an outer diameter of 1.8 m at an overall length of about 2.3 m.

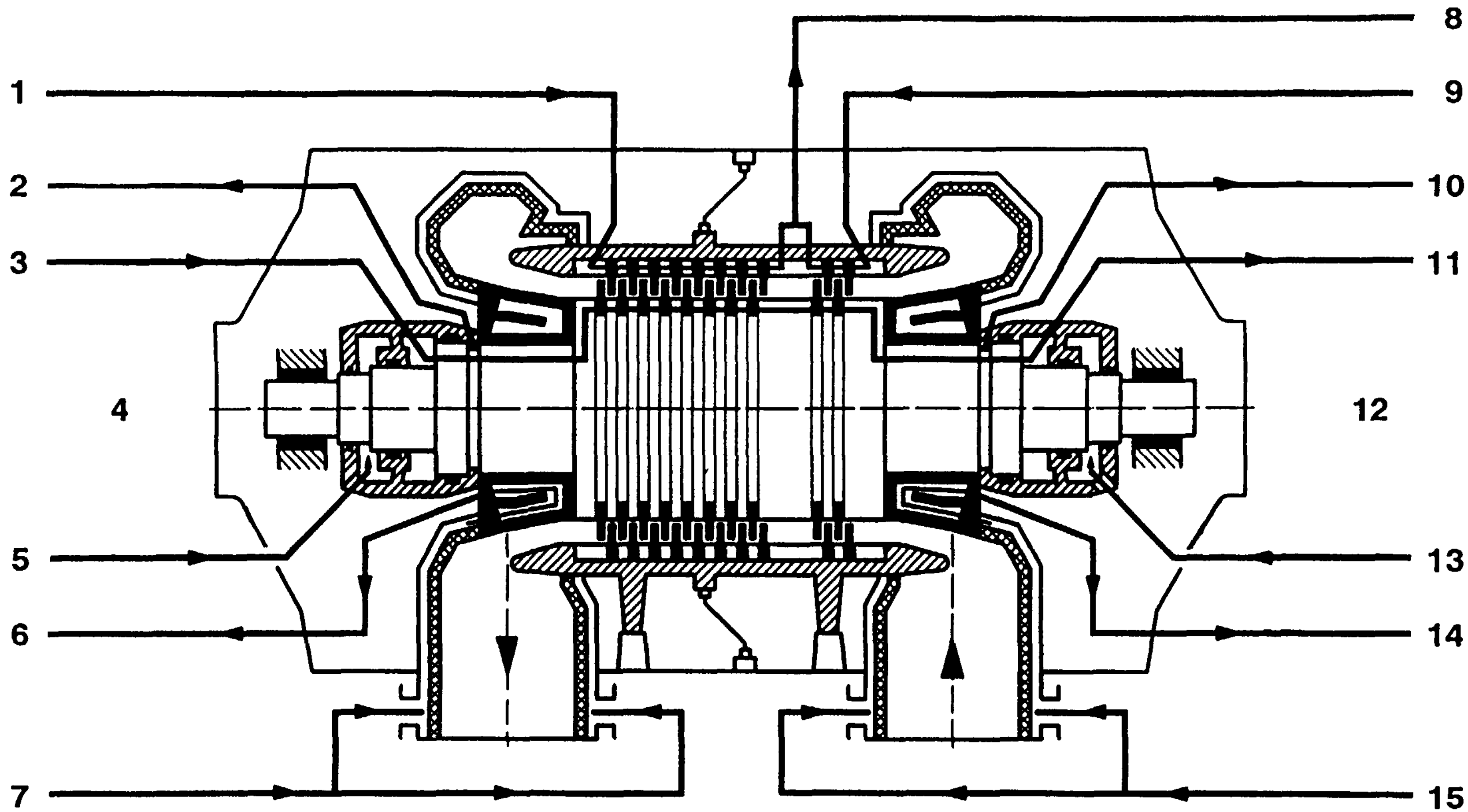
The rotor was constructed by welding several forgings of heat-resistant ferritic material and was provided with longitudinal grooves for attaching the blading. The turbine was equipped with two blade rows, each consisting of 84 rotor blades and 90 stationary blades, while the



Legend:

1 Rotor	*6* Stationary blade carreir
2 Outer housing	*7* Blading
3 Bearing	*7.1* Turbine section
4 Labyrinth seals	*7.2* Compressor section
5 Shaft seal	*8* Connecting nozzles to and from hot-gas ducts

Fig. 7: Longitudinal section of turbomachine



Legend:

1 Inlet at stator compressor section
2 Sealing gas outlet
3 Inlet into rotor

4 Low pressure side
5 Sealing gas inlet
6 Gas outlet nozzle

7 Gas inlet nozzle
8 Outlet at stator
9 Inlet at turbine section

10 Sealing gas outlet
11 Outlet from rotor
12 Drive-motor side

13 Sealing gas inlet
14 Gas outlet nozzle
15 Gas inlet nozzle

Fig. 8: Cooling and sealing gas flow in the turbomachine

compressor had 8 blade rows each with 56 rotor blades and 72 stationary blades. All blade feet were provided with a helium cooling. The blades were manufactured from Nimocast 713 LC using a vacuum precision casting procedure.

The rotor was supported on two segmental shoe bearings with forced oil lubrication. The one bearing located outside the turbomachine between drive motor and turbomachine at the interim shaft was designed as the axial fixed point of the rotor shaft (thrust bearing). The fixed point for the turbomachine housing was provided at the compressor outlet stud.

The turbomachine housing was made of cast steel and was horizontally divided at a flange joint. The flange surfaces had been growned and a sealing paste (Poly-Butyl-Cuprysil) was used to ensure the tightness. However, at the first leak tests it became evident that this sealing was not sufficient. Thus three additional grooves for accommodation 0.3 mm diameter gold rings and a purge groove were provided (see. Fig. 9).

The schematic designs of the shaft seals for operation and for shutdown, and for the bearing lubrication oil supply are shown in Fig. 10. The shaft seals are connected to the front flanges of the turbine housing.

The chosen floating ring seal design corresponds to the proven technology used for hydrogen-cooled generators. In addition, this design has been pretested at Brown Boverie Cie (BBC) - now Asea Brown Boverie (ABB) - in a special seal testing stand.

2.2.5 Power and Temperature Regulation

(Ref. [17], [18])

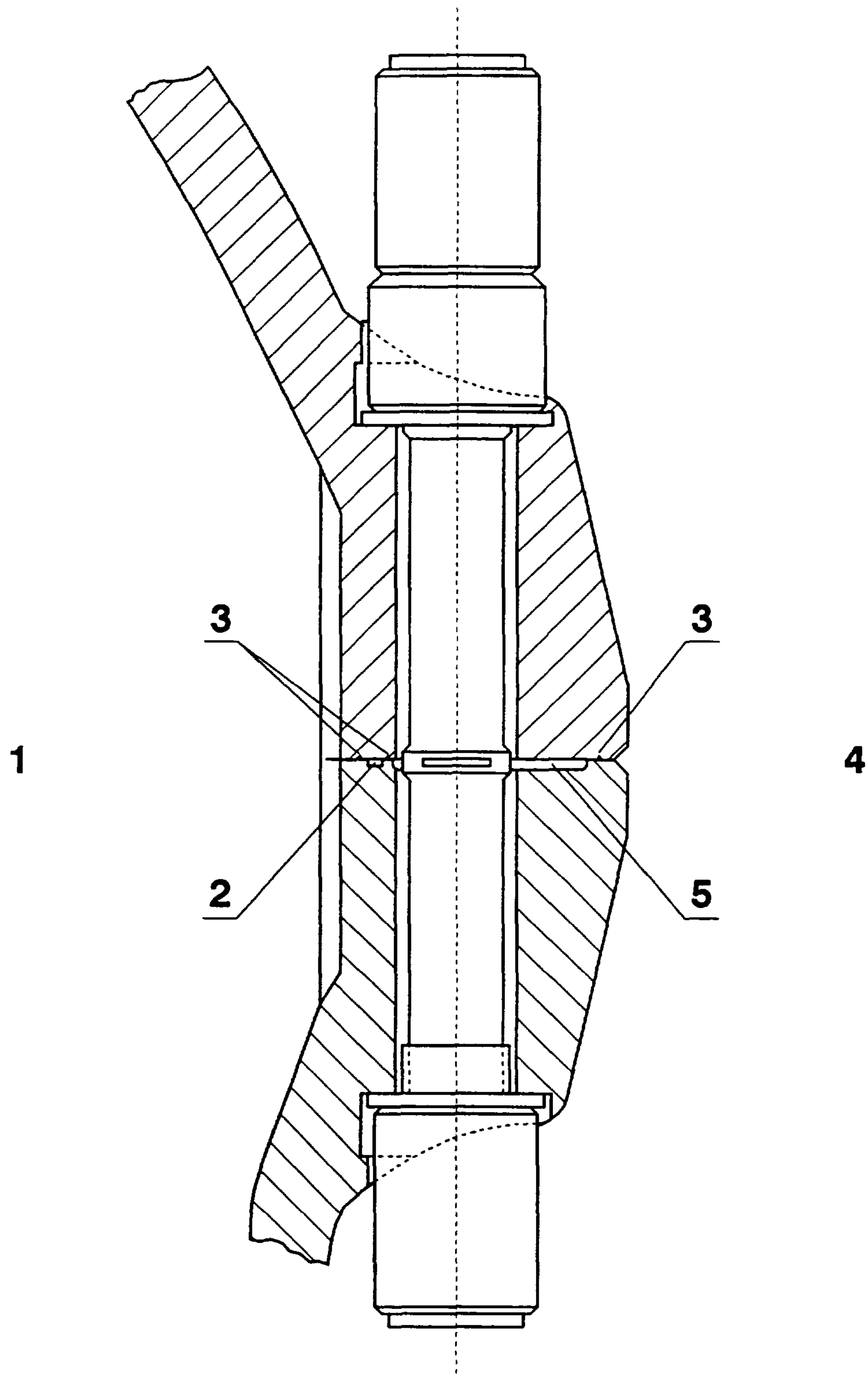
The power and temperature regulation of the test plant occured at constant revolutions of the turbomachinery. The temperature as well as the pressure were regulated by adjusting the helium flowrate, respectively the cooling water massflow at the cooler in the helium bypass line (see Fig. 5).

The nominal operating temperature of the helium main circuit could be regulated between 850 °C and 1000 °C. The nominal pressure of 50 bar remained constant.

3. Technical Experiences gained from the Test Plants

General Remarks

The technical experiences gained from EVO and HHV test facilities are summarized from numerous reports. It was to be expected that the judgements of different companies or persons on the same subject would not always be consistent. The largest portion of those discrepancies might be explained by a different background of experiences or different technologies known and applied by the different companies and persons. Nevertheless it was attempted to select the most competent judgements on the specific items. But in case of the most important items (e. g. the turbomachinery) different judgements or statements or details are retained with respect to the same subject in order to ensure that no important aspects were omitted.



Legend:

1 System pressure

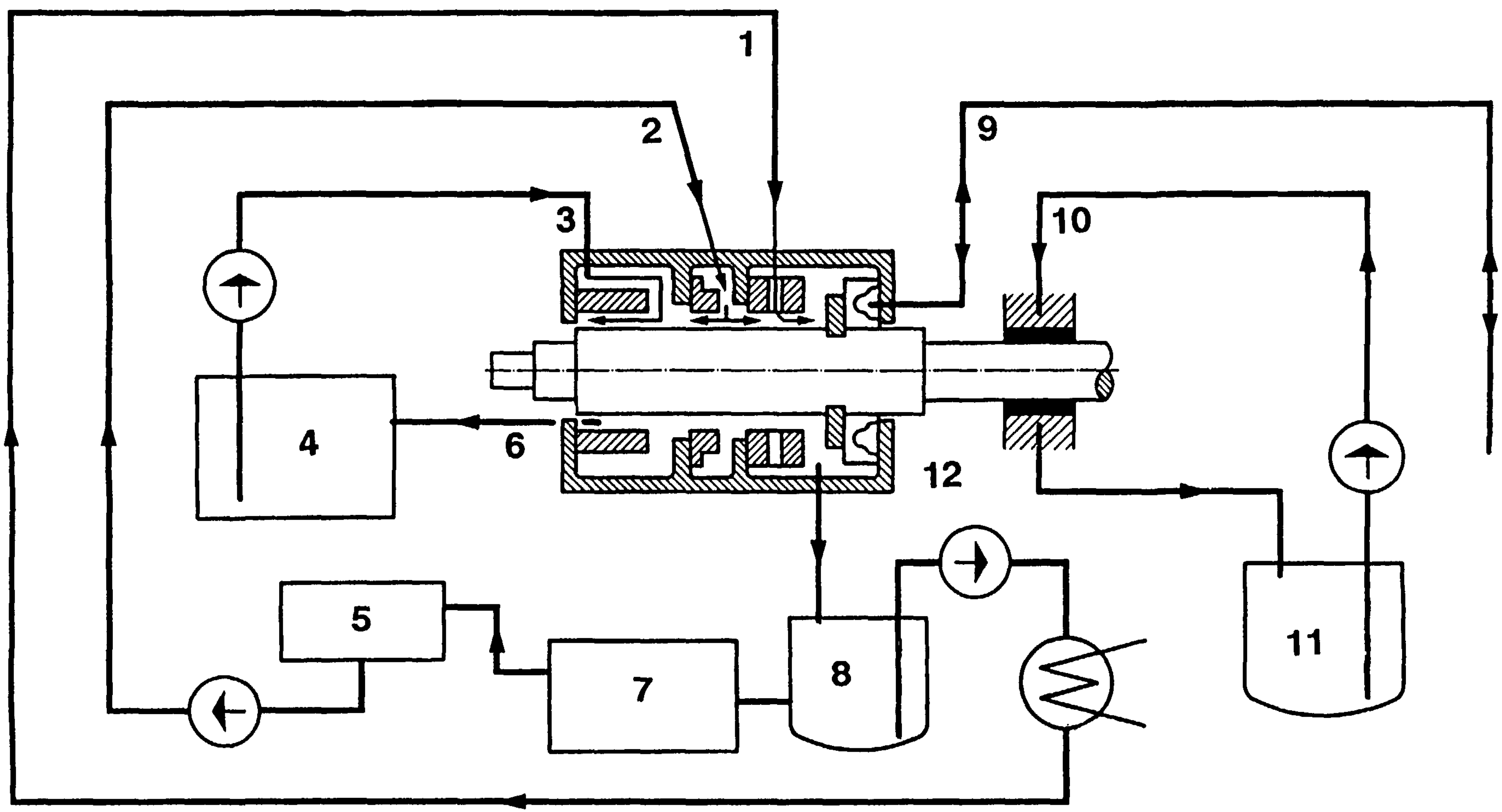
2 Purge groove

3 Sealing wire

4 Outer atmosphere

5 Bolt chamber

Fig. 9: HHV flange joint seal



Legend:

1 Helium sealing oil
2 Vacuum sealing oil
3 Air sealing oil
4 Air oil storage tank
5 Vacuum oil storage tank
6 Air
7 Helium separation tank
8 Helium sealing oil storage tank
9 Standstill seal
10 Helium/bearing oil
11 Helium/bearing oil storage tank
12 Helium

Fig. 10: Schematic designs of the shaft seals for operation and for stillstand and of the lubrication oil supply for the bearings

3.1 Turbomachinery

3.1.1 EVO - Turbomachinery

3.1.1.A Summary on the operational experiences as published by EVO

(Ref. [19], [20], [21], [22], [23], [24], [25], [26], [27], [28])

General Operation Experiences

The turbogenerator was initially synchronized to the grid on November 8, 1975. Up to the end of 1988 (final shut-down) the helium-turbine plant had achieved approximately 24,000 hours of operation, not including periods when it was used for test purposes in isolated operation. A total of 11,500 hours of operation had been at the design temperature of 750 °C.

During the operation many components and systems showed good performance. However, for some components unexpected problems arose.

Vibration Problems

Since it was known and expected that the long and slender rotor of the HP-set would be susceptible to vibrations, the vibration behaviour was carefully precalculated in the planning phase. The dynamic behaviour of the rotor was distinguished as follows:

- forced oscillations (due to unbalances and/or thermal distortions)
- resonance frequencies and
- self excitated oscillations (sealing gap excitations, hydromechanical forces in the lubrication oil film, elastic hysteresis, shrink fit friction)

The first resonance frequency of the HP-set was originally calculated to be about 2050 rpm, assuming stiff bearings. The first resonance frequency of the LP-set was calculated to be about 1800 rpm.

These calculations indicated that a large safety margin for the stability limit was to be expected. Nevertheless, at the first startup and as soon as the pressure and temperature had been increased a sudden rise of the shaft oscillation amplitudes at the rotor of the HP-set was detected. These oscillation amplitudes were so large that the design value of speed and power could not be attained. Moreover severe damages occured at the sliding surfaces of the bearings including partial tearing the bearing surfaces.

At about 1450 rpm excessively large oscillations due to self-excitated vibrations were observed. The measured first resonance frequency was about 1950 rpm (as against 2050 rpm in the precalculation). The conclusion is, that approximately 100 to 200 rpm must be deducted from the theoretical calculations for stiff bearings to meet the realistic conditions.

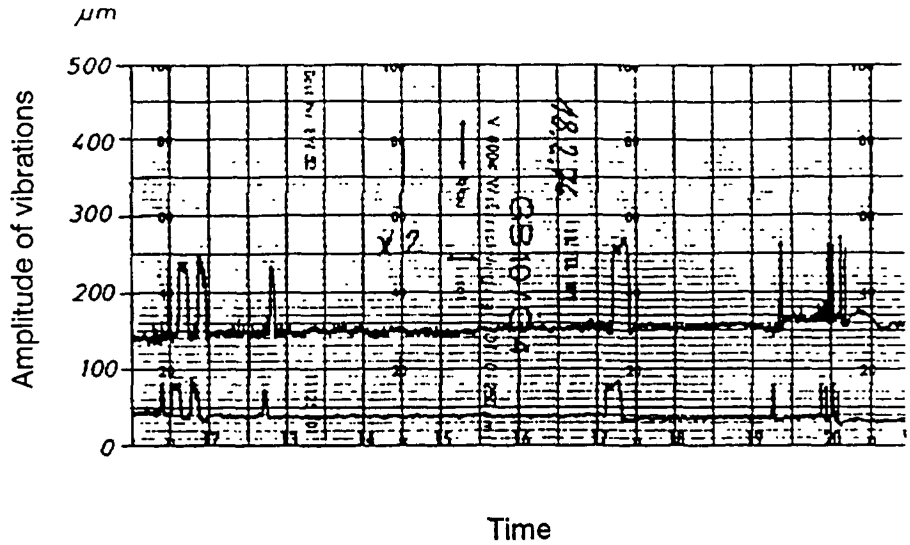


Fig. 11: Measurements of the vibrations actually accrueing at the bearings of the HP-turbine (measuring points staggered by 90°) before performing modifications

In order to overcome the large oscillation amplitudes which mainly were due to large imbalances and thermal distortions of the rotor, new balancing was carried out and the gear was modified to permit a lower running speed of the HP-set. As further reasons, a gap excitation was suspected together with a "unfavorable design" of the bearings. To clarify these latter items special measurements for different types of bearings were performed.

Fig. 11 shows a record of the overall amplitude of the rotor oscillations measured at two measuring points staggered by 90°. These measured oscillations were typical for the vibrations observed with the EVO-turbomachinery and indicated self-excitated vibrations. Taking into account the known effects that influence a self-excited vibration, various countermeasures were taken including providing transverse strips in the labyrinth seals.

Instead of orginally provided cylindrical bearings various types of bearings, multiple plain surfaces bearings with/without grooves, different wedge-type of bearings with different width/diameter ratios and various tilting-segment-type bearings, were also tested in the course of further operation.

It was also decided to modify the rotor support that had a large influence on the vibration behaviour. The rotor design was changed in order to shorten the bearing distance and to achieve a stiffer drum shaft within the compressor section. Fig. 12 shows a longitudinal section of the rotor of the HP-set before and after the modification. This change led to a substantial increase of the critical rotor frequency and the running behaviour became very quiet and satisfactory. The theoretically calculated resonance frequency (assuming stiff bearings) was then approximately 2600 rpm.

The result of these efforts was that there was a distinct rise of the stability limit with regard to higher operating pressures and higher speeds (speeds increased in steps from 4532 rpm to 5136 rpm, to 5300 rpm and finally to 5500 rpm). The power of the turbo-set did not reach the nominal power, however.

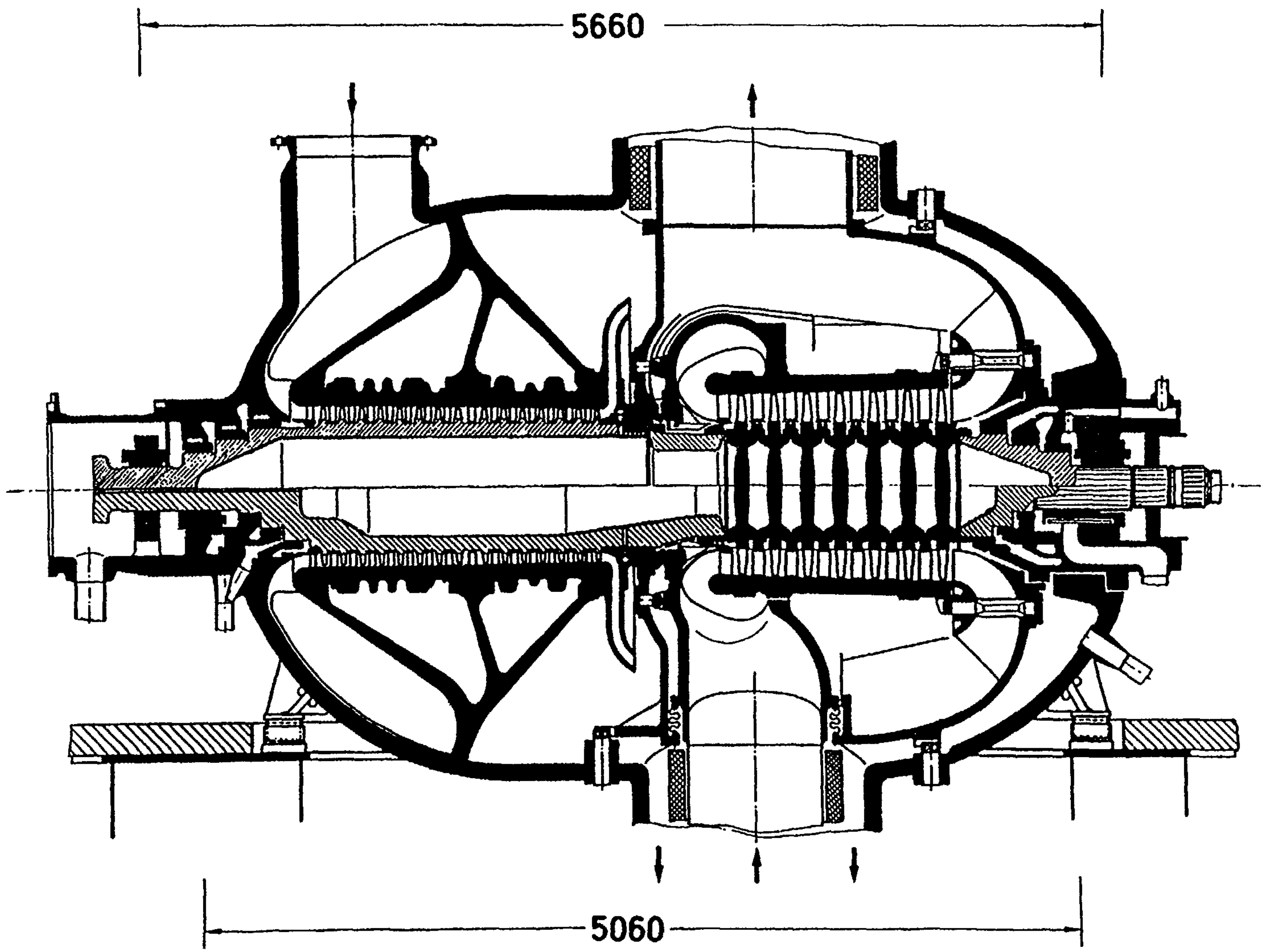


Fig.12: Modifications of the rotor shaft of the HP-set

The unexpected vibration problems and severe bearing damages were a large part of the cause of the unplanned plant shutdowns. However, the countermeasures taken were so effective that the problem of the rotor oscillations is considered solved. After having carried out the above modifications new measurements were taken for the total turboset and especially for the HP-set.

Fig. 13 shows the measurements of the vibrations actually accrueing at the bearings of the HP turbine (measuring points staggered by 90°) after the modifications.

This figure shows distinctly that the turboset was running quietly and that the oscillation amplitudes still observed, in the order of 10 μm, were sufficiently low for turbosets of this size. Only at one measuring point was an amplitude of 110 μm measured. Overall, the vibration pattern was smooth and the machine ran quietly.

Fig. 14 shows the runup spectrum measured by a sensor located at the turbine side of the HP shaft rotor (measuring point VO 4). This is further proof of the success of the improvements. The indicated retention times in Fig. 14 have been shortened substantially. This figure too shows, that the oscillation amplitudes at all frequencies have become reasonably low and acceptable.

Analysis of Vibration Problems

The self-excitated and gap-excitated oscillation and bearing problems of the turbine revealed a surprising discrepancy between theory, state of the art at the time when the helium-turbine was designed, and the actual results. It must be pointed out, however, that fundamental investigations regarding gap excitation and bearing characteristics had only been initiated at the time when the helium-turbine was under construction. Moreover, during the design the excitation forces resulting from the blading and accrueing in the labyrinth seals could only be roughly estimated. The excitation forces accrueing in the labyrinth seals were also known only approximately.

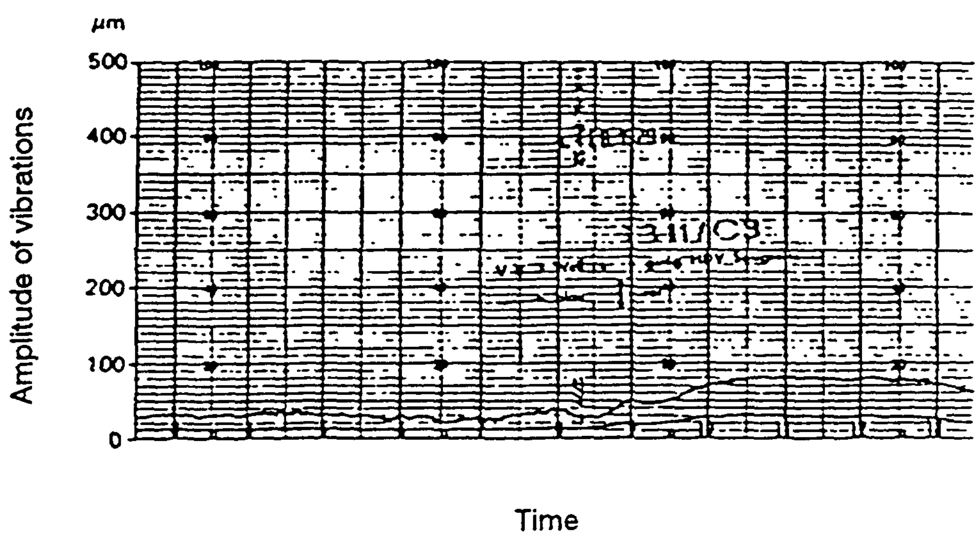
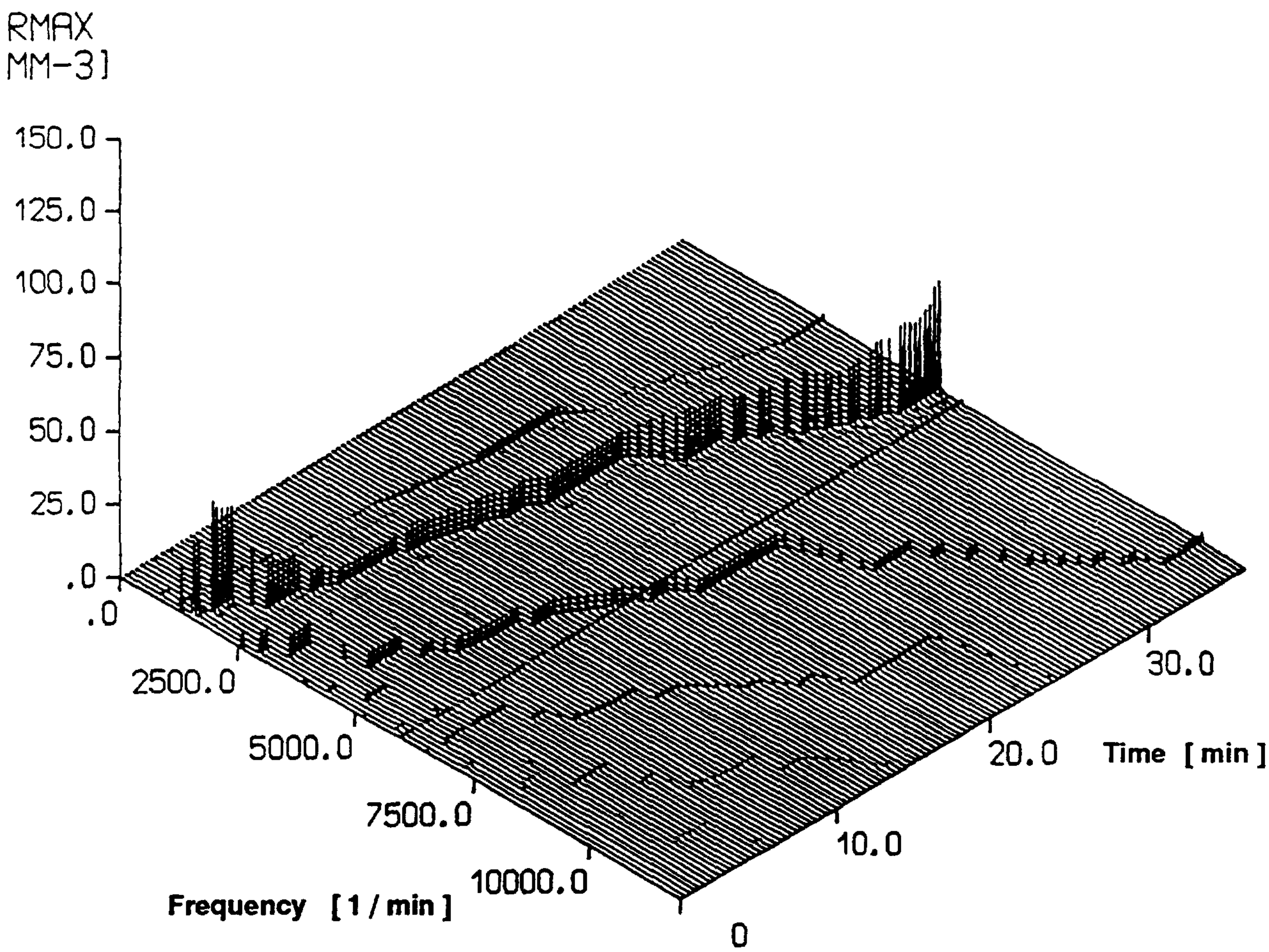


Fig. 13: Measurements of the vibrations actually accrueing at the bearings of the HP-turbine (measuring points staggered by 90°) after the modifications



Measuring point: VO 4

Fig. 14: Runup spectrum of the EVO-turbomachinery (October 1980)

The bearing characteristics knowledge was only improved continuously from the beginning of the 1970's onward. In 1987 EVO initiated a comprehensive summarizing analysis and judgement of the bearing problems, carried out by competent experts. The characteristics of many types of bearings and bearing geometries applied in the course of the operation of the turbine, were recalculated in terms of a permissible gap excitation coefficient, q_{er}. The summarizing results are given in Fig. 15.

The so-called gap excitation coefficient was introduced to characterize the ratio of gap excitation force and rotor bending amplitude. The gap excitation coefficient to be achieved by the bearings, as calculated, is given on the ordinate. It can be seen that the original bearings even had a negative or extremely low q_{er} being the cause for the unfavorable vibration behaviour. From bearing No. 5 onwards q_{er} is distinctly larger than zero. The measures subsequently taken led to a further improvement of the q_{er}. Thus it can be seen that the countermeasures taken were correct.

A theoretical calculation of the prevailing gap excitation coefficient on the basis of the excitation forces resulted in a value of about 18 KN/mm as indicated by the horizontal line in Fig. 15. This shows that the bearing modifications using the original rotor were not qualitatively sufficient.

A distinct improvement of the overall oscillation behaviour becomes evident only after having decreased the bearing distance on the rotor.

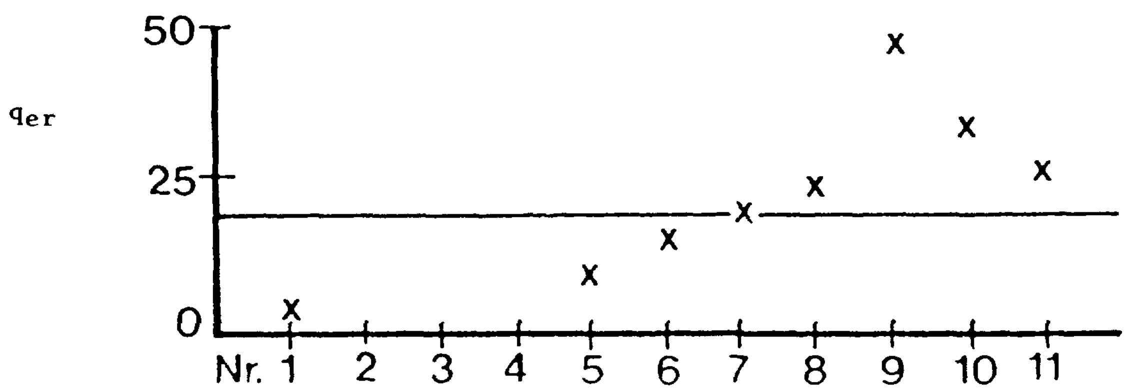
For the design of new helium turbines, it is recommended that to ensure a quiet running and stable behaviour the first critical frequency should be as high as possible and that turbulence straightening sheets arranged in the shaft seals should be provided. In addition, non-sensitive bearings such as tilting segment bearings, should be used whereever possible.

Non-Achievement of the Nominal Power

Within the first three years a maximum electrical power of 28 MW could not be exceeded because of the seal-gap-excited oscillations, the bearing problems, the shaft imbalances and the thermal distortions. After having modified the rotor of the high pressure stage, nominal values of pressure, temperature and could be reached, but nominal power could not be reached. In order to understand the reasons for this power deficit, measurements were taken at various steady-state operating levels. Of particular interest was that level of operation where the actual operating values largely corresponded with the nominal design parameters.

No.	1	2	3	4	5	6	7	8	9	10	11	Unit
B/D	0,72	0,72	0,72	0,72	0,5	0,5	0,5	0,5	0,5	0,3	0,5	–
SV	–	–	2,75	3,5	4,0	4,0	4,5	4,5	4,5	–	–	–
ψ	1,4	1,4	1,4	1,4	1,4	1,6	1,6	1,7	1,6	1,6	1,6	‰
q_{er}	0,68	–	–	–	10,12	15,40	18,67	21,82	47,95	31,63	23,57	kN/mm

B/D Ratio bearing width / shaft diameter
SV Ratio bearing gap / shaft diameter
Ψ Minimum relative bearing gap



1 - 2 Bearings consisting of multiple plain surfaces with/without groove
3 - 9 Wedge-type bearings
10-11 5 tilting segment bearings

Fig. 15: Permissible gap excitation coefficient q_{er} for the different applied bearing types resp. bearing geometries

At these conditions the nominal speed, the helium-inventory, temperatures at the inlets into the compressors and the HP turbine, and the inlet pressure into the compressor corresponded with the design. The plant should have reached its nominal design power, but the actual electrical power of the plant amounted to 30.7 MW (nominal: 50 MW) and the heating power reached 66.5 MW (nominal: 53.5 MW).

An accurate analysis has shown, that the deficiency of power production was due to deficiencies inside the turbo-set. The blading efficiencies did not reach the nominal design values and the helium-mass flow rates for the cooling and the sealing gas (which are deducted after the compression) had been far larger then predicted. Since this lost helium-flow does not produce power in the turbine, an accordingly large power loss resulted. Additionally, large losses occured due to insufficient flow guidance from the inlet diffusor into the first blade row.

Finally, EVO had decided to accept the helium turbine from the manufacturer in spite of non-conformance with the supply contract. This decision was made because helium-specific experiences could nevertheless be gained with the actual plant conditions and because the plant could be operated (in spite of the lower electrical power) as a cogeneration plant satisfying the required district heat demand.

Design Modifications to Reduce the Power Deficit:

As the observed power deficit had been explained to a very large extent, several measures were identified to enable its reduction. One measure would be to optimize the flow conditions for the inflow and outflow areas of the compressor and turbine sections in order to minimize the pressure drop losses and to achieve an optimum inflow into the blading. Moreover, a reduction of the blade gap losses would be required. That could be achieved by a reduction in rotor vibration and better selection of materials. The materials for both the rotor and the stationary blade carrier should be selected to optimize thermal expansions to achieve minimum gaps at operating conditions. One approach would be the replacement of the non-cooled austenitic stationary blade carrier in favor of a ferritic one, with cooling provided at necessary locations. Preferably no cooling at 750 °C should be provided at all, taking into account possible improvements in avaible blade and rotor materials. A third approach that seems to be possible would be a further optimization of the stationary blade profiles and rotor blade profiles.

The modifications required to achieve the nominal power for the current design would have required such large expenses, that a new design would have been economically preferable. Therefore a new concept for the turboset had been prepared, as shown in Fig. 16. The essential difference in this new design is that the turbine is no longer split up into high pressure and low pressure sections. Moreover both the high pressure and low pressure sections of the compressor have been optimized. A larger number of blade rows, compared to the former design, because a reaction ratio of 50 % instead of 100 % was selected for the blading design. All turbomachines run at a speed of 90 Hz, so that a gear must be provided between turboset and generator. If this concept is applied the nominal power could be reached at the previously provided circuit parameters.

Rotor Seals

The sealing systems of the rotors at the penetrations through the housings are a good example for the numerous systems which have proven an excellent functionality since the commissioning of the EVO facility.

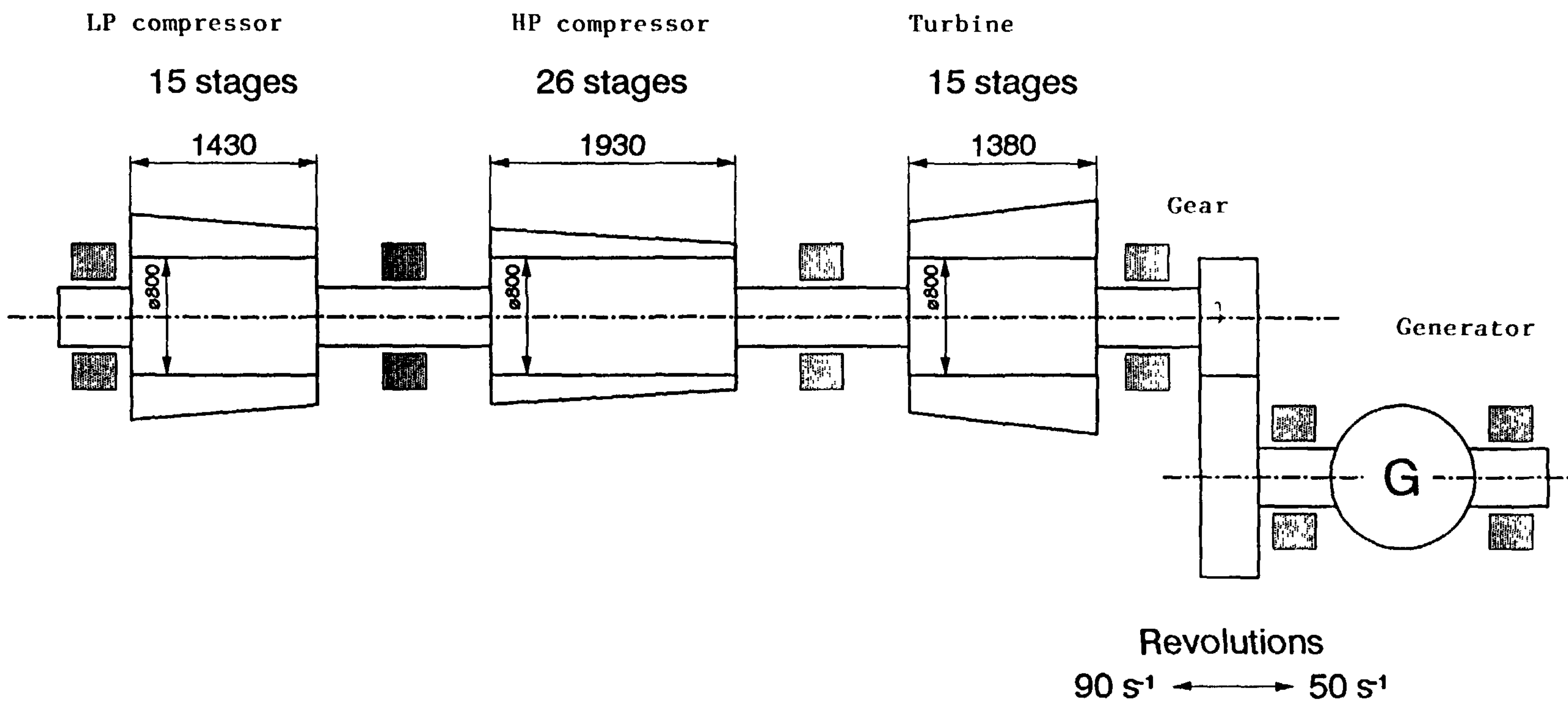


Fig. 16: New concept of the 50 MW helium turbine

Fig. 17 shows the schematic design which seals the main circuit against the bearing oil circuit, and the bearing oil circuit against the atmosphere.

Sealing against the atmosphere occurs by means of a sealing oil type gasket. One half of the introduced sealing oil flows together with the discharged bearing oil from chamber (6) into a closed low pressure oil tank. The other half of the sealing oil gets into contact with the ambient atmosphere and returns through pipe (8) directly into the sealing oil tank.

For the orderly functioning of the system it is most important that the pressure differentials indicated are maintained. Since the power of a closed gas turbine cycle is regulated by means of a pressure change in the main circuit, it is necessary to regulate the pressures in the auxiliary circuits accordingly in order ensure the appropriate sealing function. The provided system has proven its orderly function without any operational problems.

The last stage of the HP turbine can be seen on the left side of the figure. Between that stage and the bearing three labyrinth seals, separated by the inner chamber (2) and the outer chamber (3), are provided. Pure helium at high pressure is introduced with a defined overpressure into chamber (2). From there it flows to the left into the main circuit as well as to the right, in the direction of the bearing. From the outer chamber (3) one portion of the sealing helium is directed to the corresponding system at the LP turbine. The other portion of helium flows to the right through a third labyrinth seal and there, together with a part of the bearing oil, through pipe (4) to a helium-oil-separation stage, where both media are returned to the corresponding circuits.

Summary

At the design and construction stage of the EVO plant new and still insufficiently known technologies had to be applied. Nevertheless this prototype plant demonstrated, after having settled initial difficulties, that such a helium turbine plant can run in a continous and reliable operation.

3.1.1.B Shortened Summary of the Necessary Improvements, as proposed by Siemens/Kraftwerksunion (KWU)

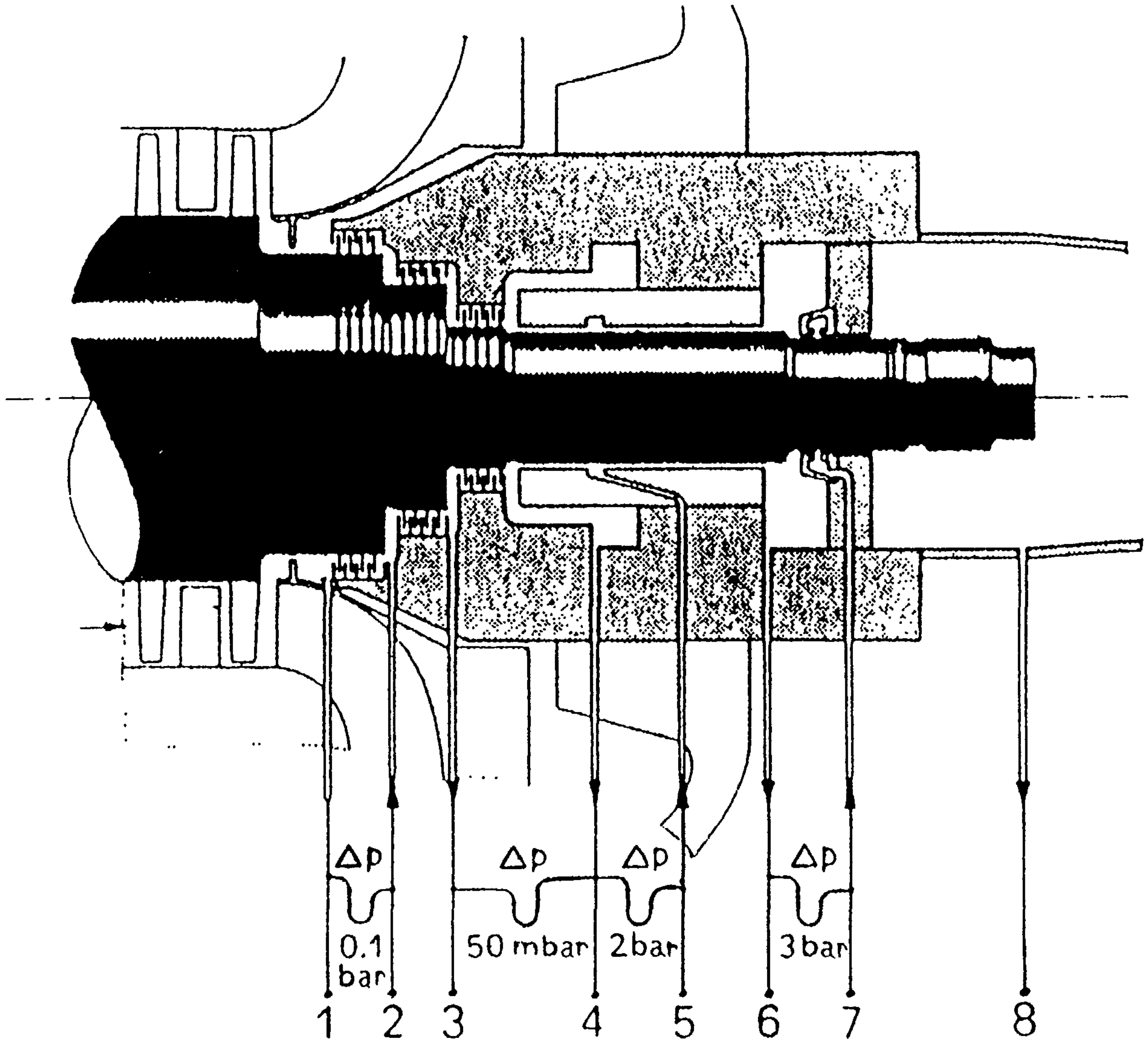
(Ref. [21], [29])

The operational experiences with the turboset as well as the results of the power measurements of the EVO-turbomachinery are summarized below. Based on these results improvements and modifications have been proposed by Siemens/KWU (see Fig. 16).

Operational Experiences

Self-excited shaft bending vibrations, induced by seal-gap excitations at the HP-turbine rotor and amplified due to an unfavorable bearing design, resulted in a rubbing within the labyrinth seals and at the radial blade and caused blade defects. The radial gaps had to be enlarged, leading to a large increase of the sealing gas and cooling gas losses, that were not considered in the original thermodynamical design.

Subsequently, the bearing axial distance at the HP rotor was reduced by 600 mm in order to improve the running behaviour of the rotor. In addition, the slide plain bearing design was modified in order further to stabilize the rotor.



Legend:

1 Sealing helium and HP-circuit-helium

2 Sealing helium inlet

3 Towards LP - sealing helium circuit

4 Sealing helium and bearing oil outlet

5 Bearing oil inlet

6 Bearing oil and sealing oil outlet

7 Sealing oil inlet

8 Sealing oil outlet

Fig. 17: Sealing helium and sealing oil systems at the HP-turbine

2. Power Measurements

After having carried out the above described modifications, there still were large deficits of the turboset power generation compared to the original design. Whereas the nominal power was 50 MW, the actual maximum power was 30.5 MW. The nominal design heating power of 53.5 MW was exceeded however and amounted to 66.5 MW. Power measurements and recalulations were carried out for three different operating conditions. For the nominal operating conditions, it was confirmed that the applied KWU computer code came to the same result for the nominal power of 50 MW, when using the same assumptions for the thermohydraulic and mechanical design of the turbomachinery and circuit that had formerly been used by Gute Hoffnungshütte (GHH). When inserting the latest measured actual parameters of the turboset and of the circuit, the measured electrical power of 30.5 MW at nominal conditions was recalculated to be 30.7 MW, which is nearly identical.

The following power losses have been measured and recalculated for the 100 % nominal circuit conditions:

Turbomachinery:

• Flow losses in the inlet diffusor and in the blading of the LP compressor	1.3 MW
• Flow losses in the inlet diffusor and in the blading of the HP compressor	4.0 MW
• Blade gap losses, flow losses in the HP-turbine	3.9 MW
• Profile losses due to remachined blades after having detected damaged blades in the LP turbine	2.4 MW
• Increased sealing and cooling gas flow rate in all four machines	5.3 MW
Total Turbomachinery Loss	16.9 MW
Additional loss due to higher pressure drop in the circuit	2.6 MW

The total power loss in the circuit was 19.5 MW .

This means that in order to overcome the power deficit, it was necessary to modify the turbomachinery and its components as well as the main circuit in such a way that the theoretically assumed design parameters would be fulfilled by the mechanical design of the turbomachinery and of the circuit.

Flow and Blading Losses in the Turbomachines

In order to overcome the power losses due to flow and blading losses, the following countermeasures could have been taken:

- Modification of the inflow and outflow sections of the compressors and turbines (experimental tests would be required).

- Reduction of the blade gap losses by designing essentially vibration-free rotors. The materials for the rotor and the stationary blade carrier should be related to each other in such a way that minimum gaps will be achieved at operating conditions (e. g., replacement of the non-cooled, austenitic stationary blade carrier by a ferritic one). Cooling must be provided as needed when the ferritic material requires lower operation temperatures.

- Optimization of the applied profiles for the stationary blades and the rotor blades by using a reaction ratio of 50 % at both compressors instead of 100 %.

Sealing and Cooling Gas Losses

Due to the excessive vibrations of the rotors, parts of the labyrinth seals had to be removed by machining, thus increasing their clearances. This resulted in the additional sealing and cooling gas losses amounting to 5.3 MW.

Summary

Proposed improvements:

- By designing nearly vibration-free rotors, the gaps required in the labyrinth seals could be reduced. Rotor improvements could have been achieved by reduction of the rotor weights, provision of a single shaft turboset (see Fig. 16), reduction of the bearing distance, and the use of bearings with a degree of high damping.

- The required cooling gas flow rates should be minimized. The currently available technology for the cooling of rotors and blades as well as the availability of high temperature resistant materials permit reduction of the cooling gas flow rates.

3.1.1.C Generation of Acoustic Emmissions by the Turbomachinery and Propagation into Circuit (EVO)

(Ref. [30], [31], [32], [33])

The EVO plant is located close to the center of the city Oberhausen. Therefore it was necessary to ensure a nearly noise-free plant operation. The operator measured a noise of 70 - 80 dB directly in the area around the turboset and outside of the machine hall of < 50 dB.

Like for all turbomachines the largest portion of the acoustic emissions of the EVO turbomachineries is propagated, even against the helium flow direction, into connected piping, where it can excitate vibrations and thus develop mechanical loads on portions of the piping, thereby possibly leading to fatigue fractures. In order to investigate these problems numerous measurements were carried out. In this report only one typical example, namely the coaxial gas duct between recuperator and HP turbine, is described. Figure 18 shows the schematic arrangement of this piping including the flow baffles and the measuring points.

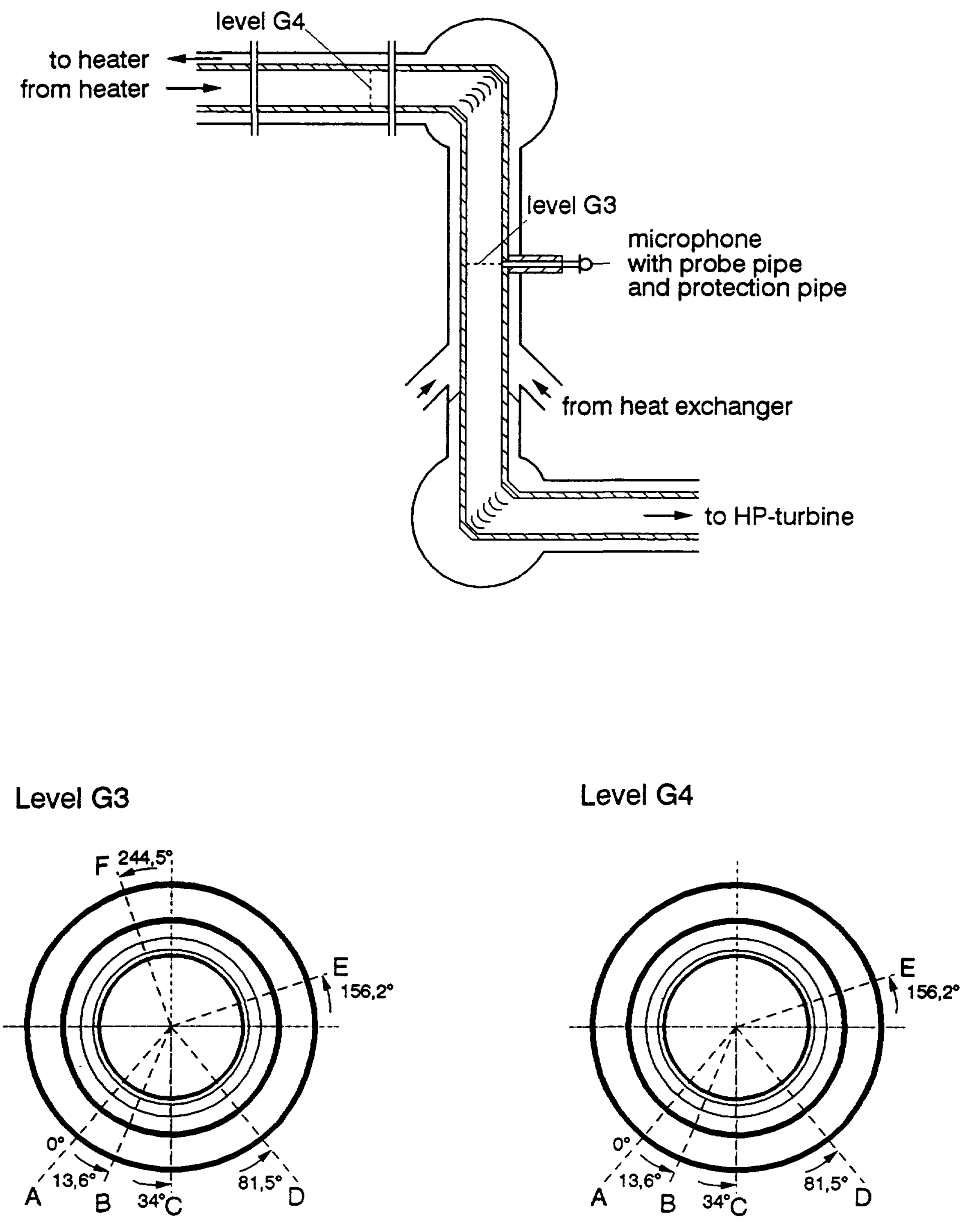


Fig. 18: Koaxial gas duct between heater and HP-turbine

Fig. 19 shows the noise spectrum level at measuring G3 at nominal power parameters at a total sound power level of 140 dB emmitted from the turbomachinery.

The measurements have shown that the overall level of the acoustic emissions rises with increasing electric power. At nominal plant parameters the maximum sound level within the piping section investigated was 148 dB. Fig. 20 shows the total sound power level, averaged over the level circumference of measuring point G3 depends on the power level.

The measured frequency values of the averaged rotary sound power at the maximum achieved electrical power ($\approx$ 30 MW) was 3770 Hz. Fig. 21 shows the rotary sound power averaged over the circumference as dependant on the electric power level (measuring point G3). These measurements have a broad range of scatter but generally increasing with the generator power. Small changes in operating conditions result in large changes of the sound propagation, so that at the same measurement position large fluctuations of the rotary sound power at the same modal composition of the sound field have been measured.

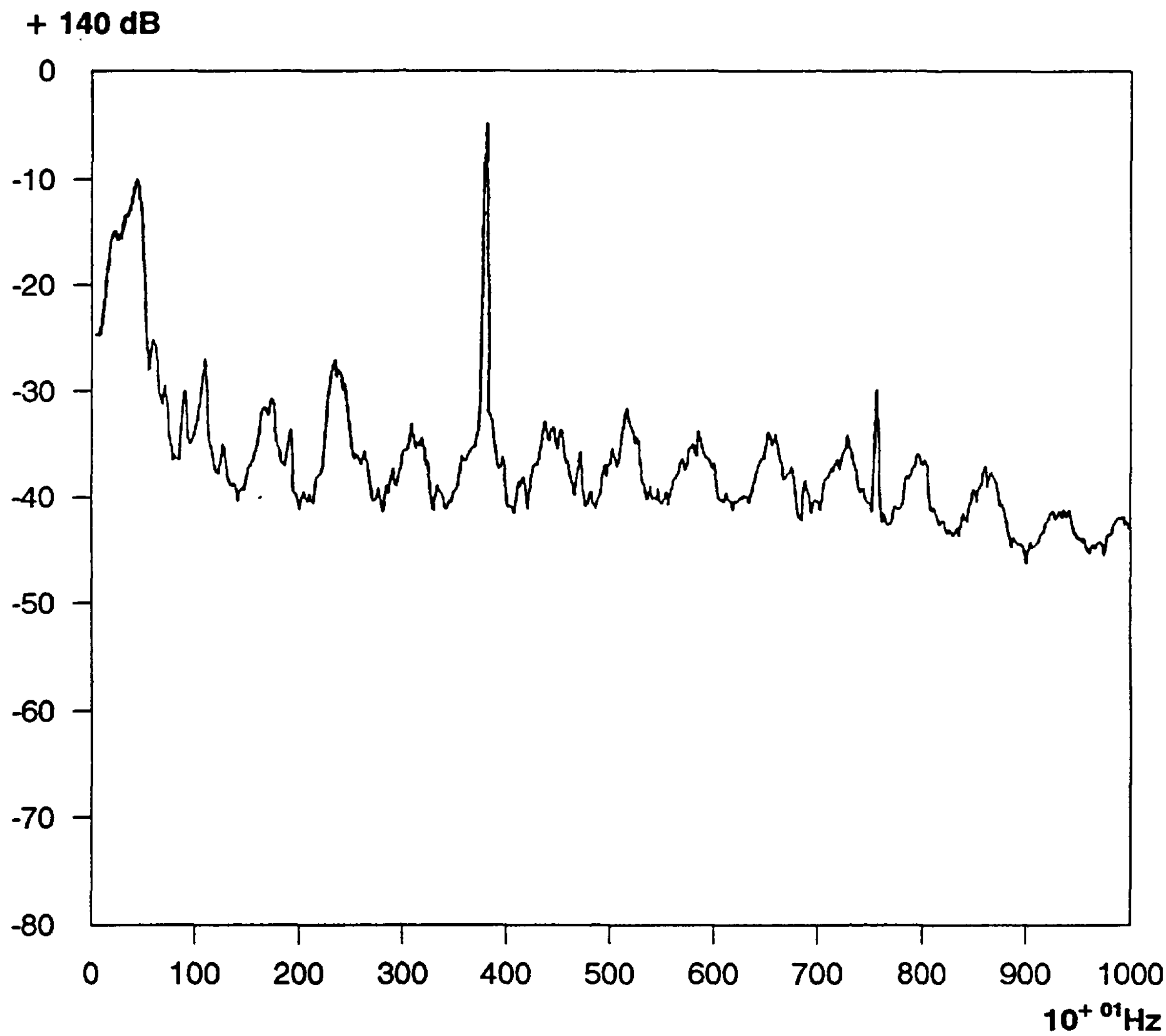


Fig. 19: Noise level spectrum at measuring point G3 at nominal power parameters

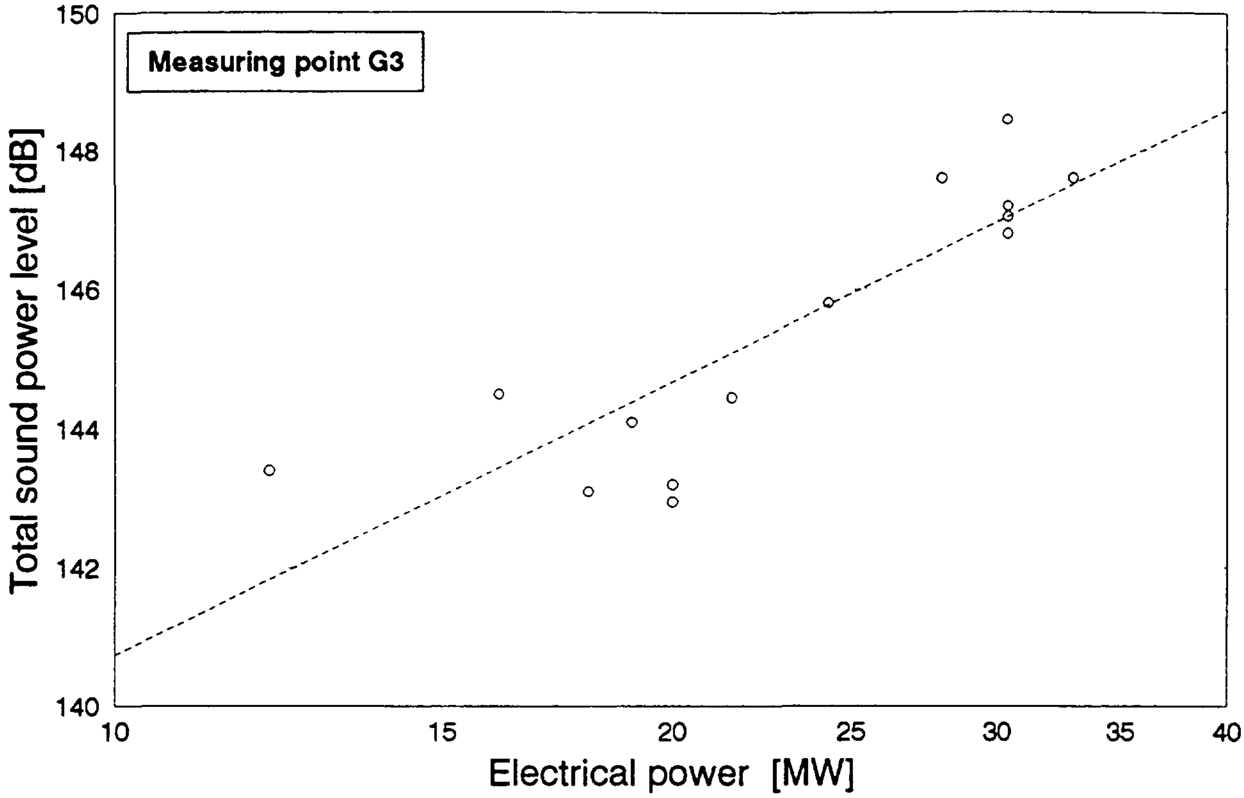


Fig. 20: Total sound power level averaged over the circumference in dependance on the power level

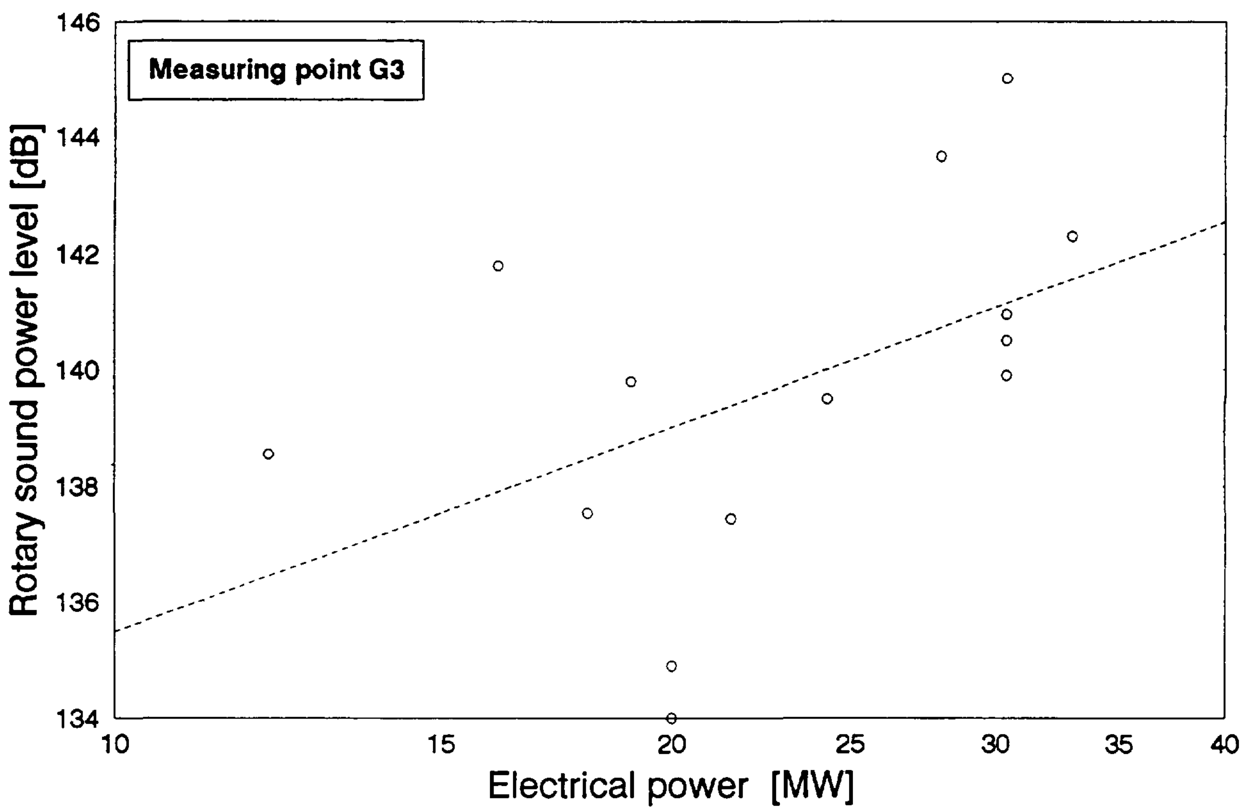


Fig. 21: Rotary sound power level (averaged over the circumference) in dependance on electric power level

Knowledge of the modal composition of the sound field enables the calculation of the spatial distribution of the sound pressure, which is the cause for the piping vibrations. It has been observed, that large differences occurred in the circumferential pressure level distribution, with observed peaks at the bottom. The measured values at measuring point G3 showed a good coincidence between all measurements and the precalculated values for the sound pressures.The effect of acoustic emmissions and their propagation into the piping have been investigated regarding their influence on the mechanical loads induced by excitated vibrations. These investigations have shown that a fatigue failure of the liner guiding the flow between the fossil-fired heater and HP-turbine is not credible, because the largest stresses have been determined to be approximately 25 N/mm^2.

A fundamental fact that has been found is that oscillation amplitudes are larger in the low-frequency range than those measured at the rotary tone frequency of the HP turbine.

3.1.2 **HHV - Turbomachinery**

(Ref. [23], [24], [34], [35], [36])

The HHV facility was operated by a consortium of German utilities, participating in the HHT-project.

Main Problems at the Commissioning

The main difficulties occurring at the commissioning in 1979 were:

Oil Ingresses

Oil ingress into the main helium circuit occurred twice from the turboset seals. The first ingress, which amounted to between 600 to 1200 kg of oil, was due to a serious operator error during the commissioning phase. The HHV-plant was shut down over the weekend, but one auxiliary oil pump was unfortunately not switched off, transporting this oil quantity. This accident made evident that an interlock and a detector were needed. The second oil ingress event occurred due to a mechanical defect of a sealing element. However, in that case the ingressed oil quantity was negligibly small and was immediately indicated by the detector. At the first incident the ingressed oil was partly coked and formed thick deposits on the cold and hot surfaces of the turbomachinery and of the circuit, especially in the hot gas ducts and in the test section. The effected surfaces had to be cleaned mechanically, by baking-out (partial evaporation or coking of oil), or by chemically cracking with the addition of hydrogen or other additives. At the second incident, as the quantity of ingressed oil quantity was very small, it was removed by cracking at 600 °C (with the use of additives).

Excessive Helium Leakrate

After having modified the main turbine flange joint (see chapter 2.2.4) the pressure and leak tests of the HHV at ambient temperature showed a good leak tightness for the flange joints of the turboset and of the main and auxiliary helium circuits. But at operating conditions (850 °C) large helium leaks were detected, and comprehensive work and countermeasures had to be taken. A first measure was to weld the lip seals provided at the flange joints of the main circuits. Later a large leak was detected at the front flanges of the turboset, caused by a non-uniform temperature distribution during operation resulting in thermal stresses forming local gaps of about

1/10 mm. To overcome this problem, the cooling gas distribution and flow rates within the turboset housing were modified and improved and the temperature distribution was improved sufficiently so that the local gaps were prevented. In the course of the commissioning runs another large leak occurred at the main flange joint of the turboset (when the lip seals had not been welded). The outer sealing ring was partly ruptured due to operator's overpressurization of the purge groove. Modifications to the interlock system were then made.

Trial Run: Demonstration of Safe Quick Shutdown

After having overcome the leaks of oil and helium, and the associated cooling problems, the HHV was stepwise brought to full operating conditions at 850 °C and 51 bar. During a 60 h trial run the functioning of the instrumentation, control and safety systems and the general operating function of the HHV were demonstrated. A principle task was to demonstrate the emergency shutdown and the reliability of countermeasures in case of incidents. For this demonstration the turboset and the cooling gas compressor had been switched off at full operating conditions. The turboset must be slowed down electrically within 90 s to a full stop in order to prevent an unpermissible heatup of the circuit by the rotation energy of the rotor shaft. Subsequently, the HHV was returned to the full operating condition. This safety demonstration for the turbomachinery is shown in Fig. 22.

Overall Operational Performance

After having overcome its initial problems, the HHV was sucessfully operated for about 1100 hours,of which the turbomachinery operated for about 325 hours at 850 °C.

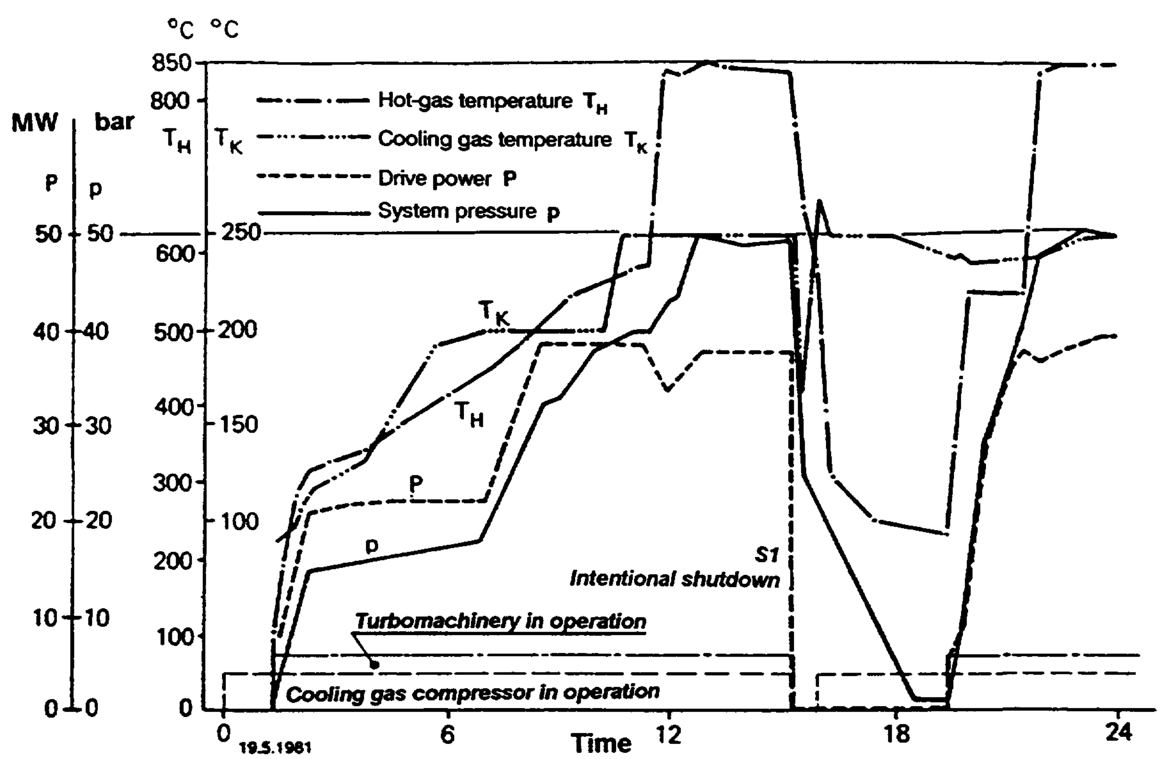


Fig. 22: Startup of the HHV from cold condition and after a quick shutdown (hot condition)

During the design of the HHV turbomachinery comprehensive precalculations were made regarding the operational behaviour at various partial loads and at nominal load. These precalculations were be counterchecked during the experimental operation of the HHV plant for the nominal plant parameters. Comparisons between the precalculated and the measured parameters are given as follows:

Table 3.1.2/1: Plant Parameters at Nominal Load

	Precalculated	Measured
Turbine inlet	49.4 bar 820 °C	49.1 bar 818 °C
Between turbine and compressor	45.1 bar 785 °C	44.8 bar 780 °C
Behind compressor	51.6 bar 850 °C	51.9 bar 849 °C
Required motor power	45 MW	39.2 MW
He mass flow	212 Kg/s	211 Kg/2
Pressure drop	2.2 bar	1.9 bar

The agreement between precalculations and measurements was excellent. When the following uncertainties in the precalculations and the restrictions at the measurements during the operation are considered:

- mass flow in the turbomachine cannot be measured directly
- mass flows in stuffing boxes and labyrinth seals can only be estimated
- cooling gas flows in the rotor and the stator can only be estimated

it can be concluded that the thermodynamic design data were achieved and even exceeded. For example the compressor and the turbine had a better efficiency than assumed at the design. This was derived from the measurement of the electrical drive power and from the direct or indirect measured mass flows, pressures and temperatures at the turbine and compressor inlet and outlet

Dynamical Behaviour

Extensive measurements were made during the commissioning phase as well as during the 60 h trial run with regard to the shaft rotor oscillations of the turboset with the measured results compared with the precalculated values. Thus the first resonance frequency was calculated to be approximately 1700 rpm, whereas the measurement showed, that it is in the range between 1700 and 2000 rpm. The second resonance frequency was precalculated to be far above the nominal speed, which was confirmed by operation.

The rotor shaft was running quietly at all operating conditions. The running behaviour was better than preplanned with the shaft oscillation amplitudes at nominal speed in the order of 40 to 60 μm. This is usual for large steam turbines. A detailed comparison was carried out for three startup and shutoff runs as well as for the 60 h trial run. The measurements and comparisons concerned rotation-frequency, double-rotation-frequency and self-excitated vibrations. The coincidence of measurements and calculations was good. The measurements confirmed that the rotor shaft was satisfactorily balanced and yielded valuable information regarding the rotor shaft behaviour at the startup and at the resonance frequencies. However, it should be pointed out that the good agreement between measurements and precalculations most probably occurred, because the rotor shaft had only slight imbalances and only slight thermal distortions. Other concerns of the precalculations were the bearing properties. More details about the dynamical properties and behaviour are shown in the following figures.

Fig. 23 shows the mass distributions of the HHV rotor shaft and indicates the measurement points. Fig. 24 shows the dynamical behaviour of the main rotor shaft at the runup, measured at point A. The figure shows clearly, that the rotor shaft was excellently balanced. Fig. 25 shows the dynamical behaviour of the main rotor shaft at rundown, also measured at point A. The figure indicates, that there had been some slight thermal distortions of the rotor shaft. Fig. 24 and 25 show moreover that the first resonance frequency is at 1700 - 2000 rpm.

Required Modification of Instrumentation Monitoring the Temperature of the Rotor

The continous temperature measurement provided at the rotor, which consisted of a slip-ring-transmitter, did not prove to be reliable and was replaced by a telemetric measurement.

Cooling of Turbomachinery

Rotor, stator, blade feet as well as the inlet and outlet nozzels of the turbomachine were cooled by means of cold helium conducted through cooling channels. Satisfactory rotor cooling was especially important for the safe operation of the plant. The newly designed rotor cooling proved to be very effective, assuring that the disk temperatures could easily be kept below 400 °C. The measured coolant gas flow rates were 10 to 20 % larger than the planned flow rates, so that the actual temperatures of the rotor disks and blade feet were distinctly lower than forecasted. Only at the turbine inlet (where the cooling helium temperatures are the highest), were the measured values 20 - 30 °C higher than estimated by the design, but distinctly below permissible values. Fig. 26 shows the pressure and temperature patterns in the blading channel and in the cooling channels as well as the rotor disk and blade feet temperatures. It may be noted that in addition:

- The cooling for the stator and for the nozzles was functioning well. The temperature at the inlet nozzle flanges was lower by 30 to 35 °C than precalculated.

- The only modification of the cooling system that was required was for the front flanges of the turboset (equilization of the cooling, in order to prevent flange splitting).

- The cooling system functioned so well, that an operating temperature of 1000 °C for the turbine seems to be possible.

Shaft Seals; Helium-Tightness

Shaft seals generally performed well as follows:

- The 3-circuit floating ring seal used for the rotor shaft seal, which had been pretested in a special test stand, demonstrated adequate behaviour under helium operating

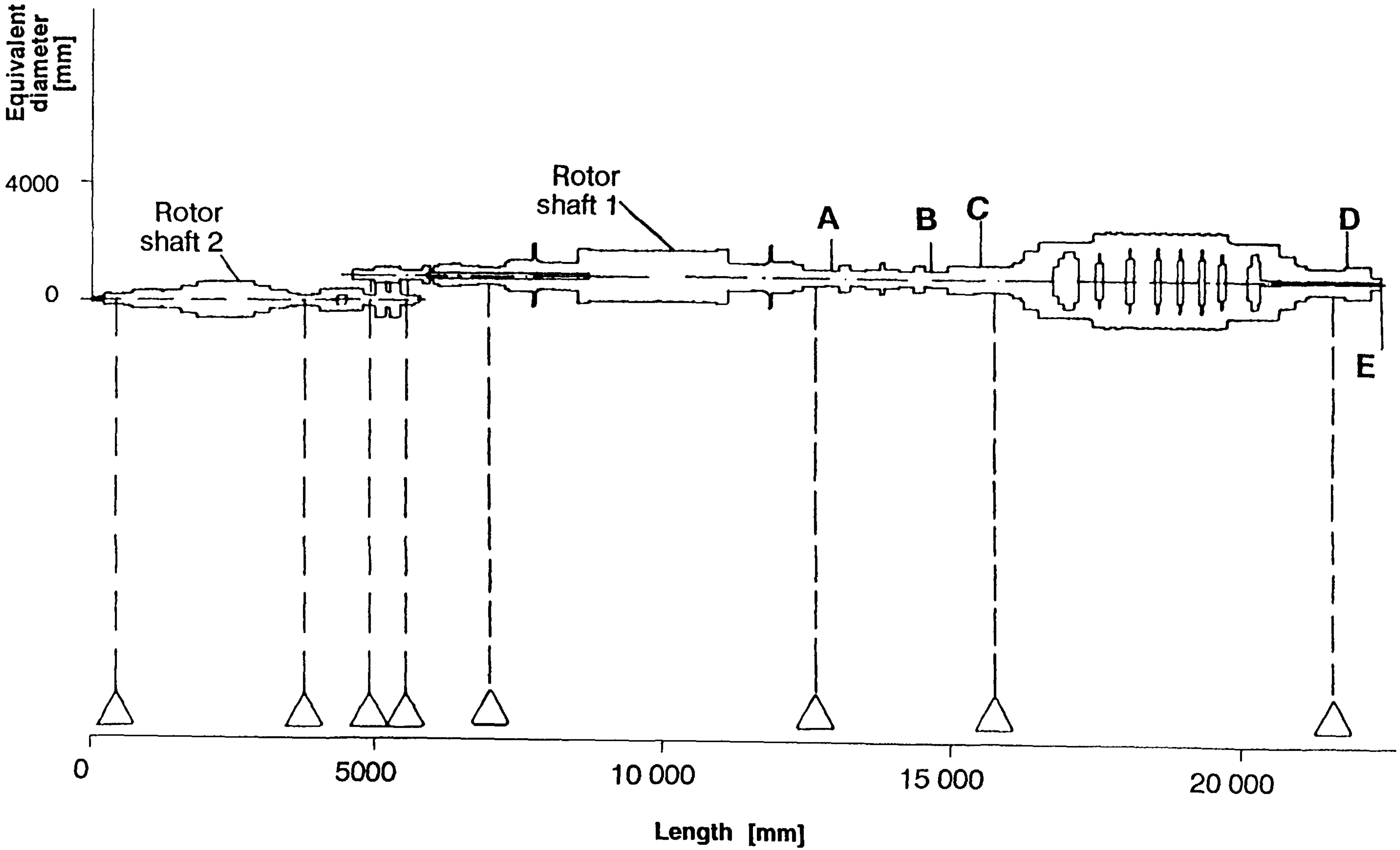
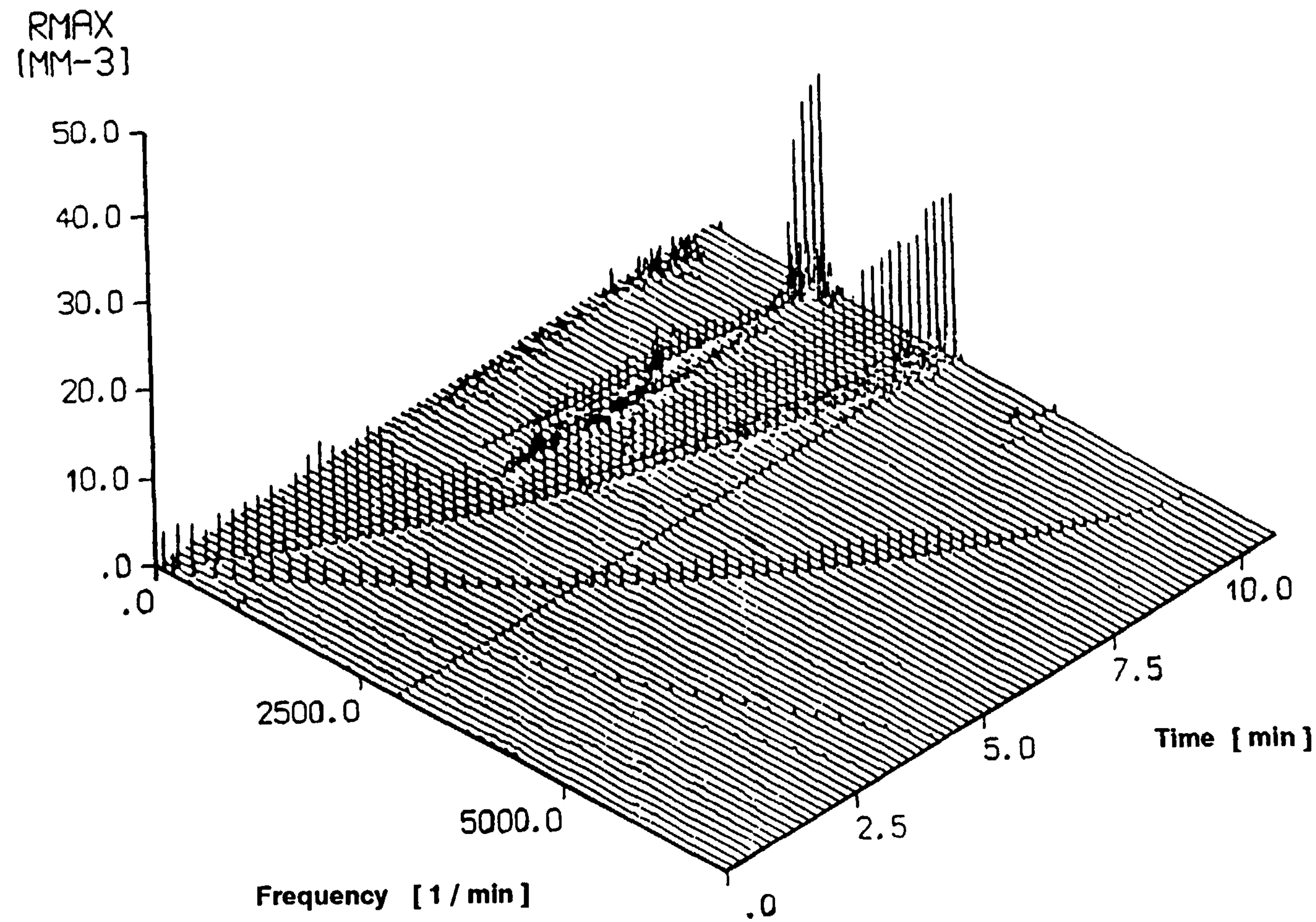
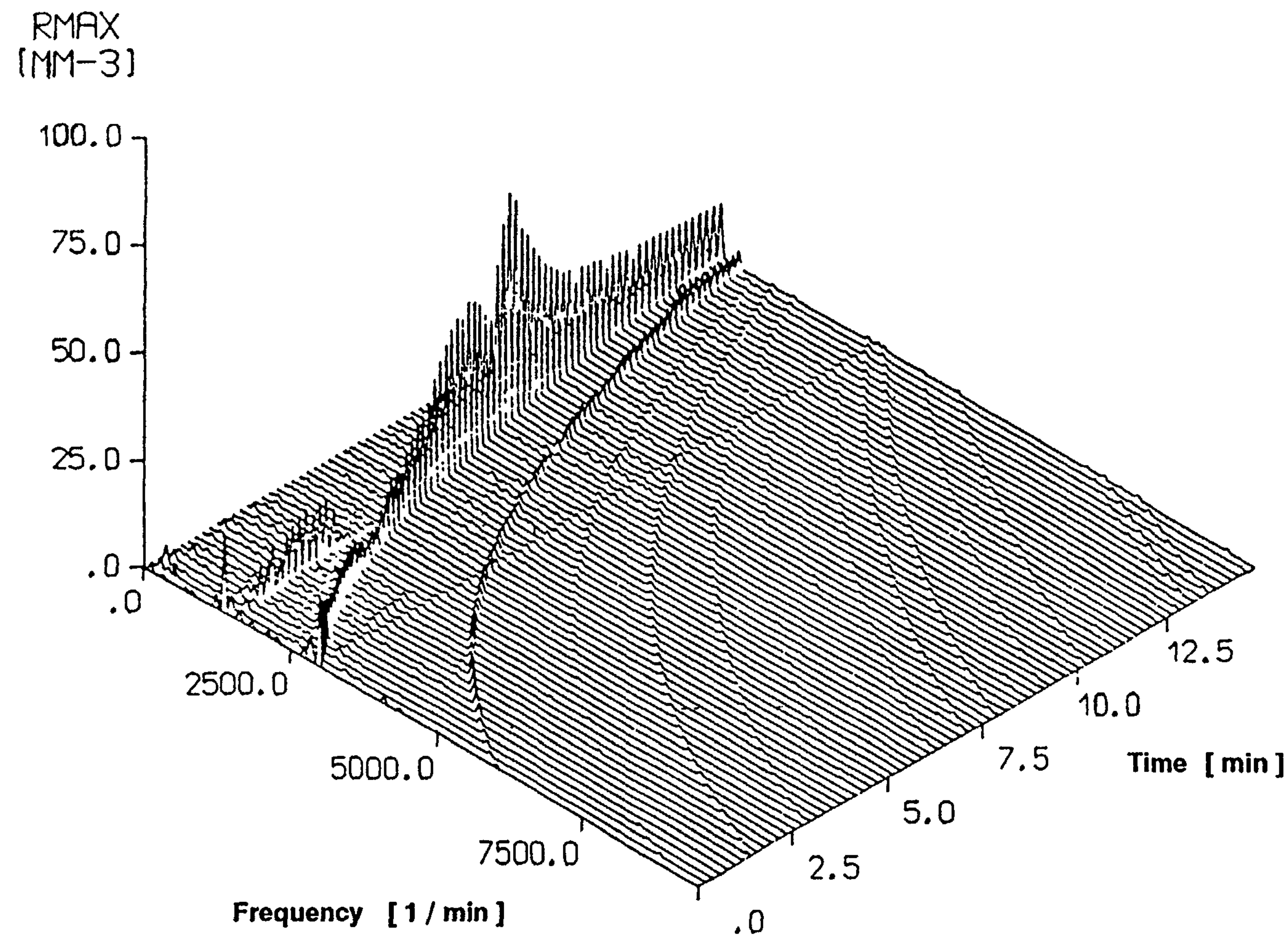


Fig. 23: Mass distribution of the HHV rotor shaft



Measurement point A

Fig. 24: Dynamical behaviour of the main HHV rotor shaft at the runup



Measurement point A

Fig. 25: Dynamical behaviour of the main HHV rotor shaft at the rundown

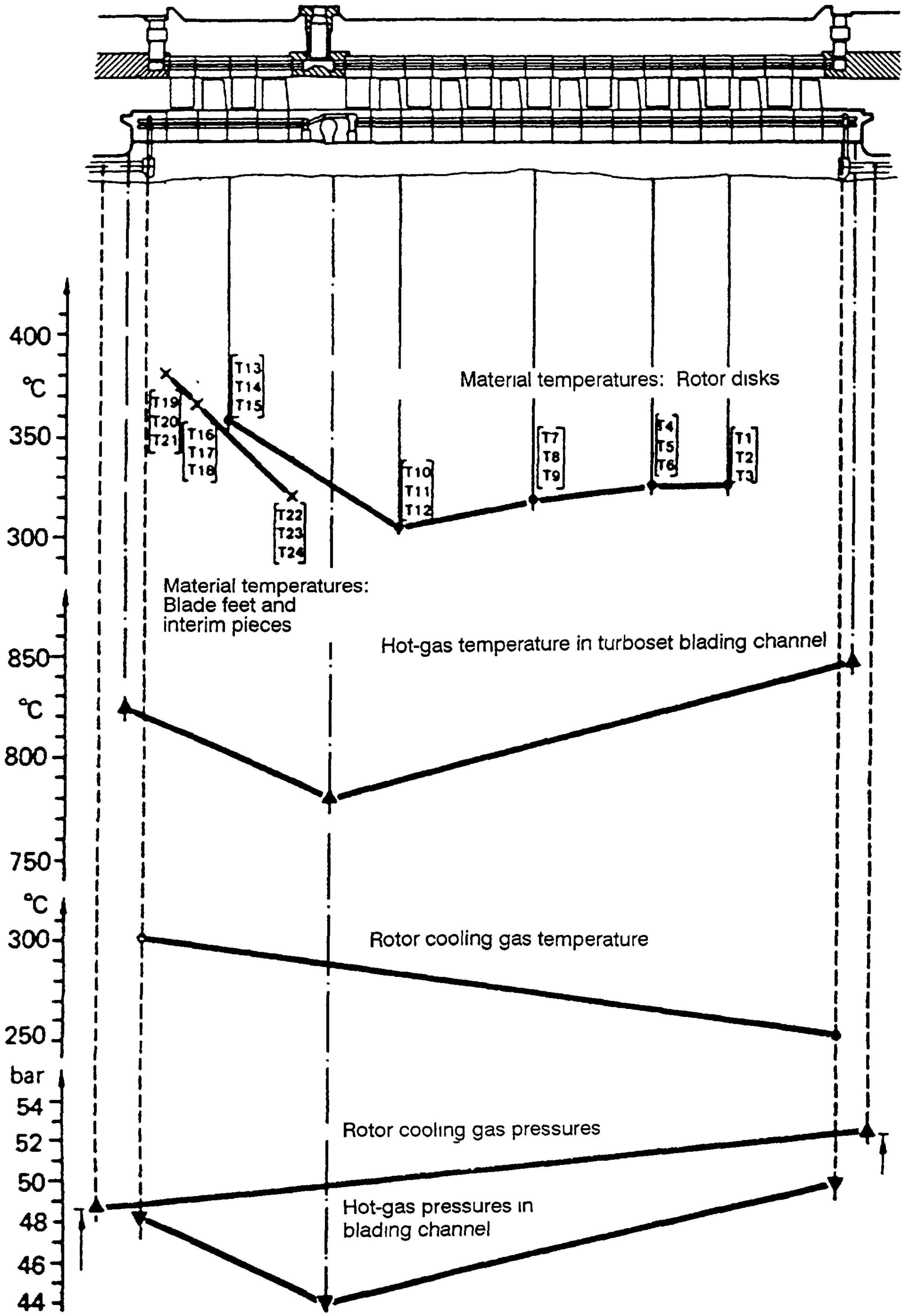


Fig. 26: Temperature- and pressure-patterns in blading channel, cooling channels, and rotor disks

conditions of 850 °C and 51 bar. Minor modifications had been required for the regulation of the sealing oil system in order to eliminate the disturbing effect of gas bubble releases. In total, the HHV operation showed that the selected shaft seal fulfilled expectations. The observed rotor shaft oscillations did not have any effect on the sealing behaviour.

- The static seal at the rotor shaft also functioned without any problems.

- After performing the above modifications, the sealing and cooling gas system including the cooling helium compressor, functioned well. The purification system for the circulating sealing gas did not function satisfactorily, however. The sealing gas flowing from the labyrinth seals to the bearing pedestal chamber becomes loaded with oil aerosols ($\approx$ 50 grams per hour oil at steadystate operation) and must be cleaned by means of a separater and filter system in order to reach the required cleanliness of the gas before it is returned into the circuit. This oil separation system, consisting of a cyclone separator and a wire-mesh and down stream fibre filters, needed further improvement. An oil ingress into the inner helium circuit was no longer observed after performing the above modifications.

- After performing the above described modifications the seal of the horizontal turboset flange as well as the front seal of the shaft seal carrier proved their reliability. The measured leak tightness of these flange joints was better than 10^{-3} Torr l per second. The weld lip seals of the hot gas duct were tight. Thus the helium losses against the ambient environment amounted to 10 - 20 Nm^3/d at 51 bar (out of total helium inventory of about 8000 Nm^3).

Generation of Acoustic Emissions by the Turbomachinery and Propagation into the Circuit and Environment

The noise level specified outside the HHV concrete building was less than 50 dB. The actual noise level the HHV-turbomachinery had inside the building (in close vicinity of the turboset) was so low (estimated approximately 50 - 60 dB) that no further measurements were necessary.

During the trial run measurements of the sound power spectrum were taken at four different measuring points (two at the connection nozzles of the turboset and two at the coaxial test section) of the helium-circuit. This was done to determine the spectrum and the intensity of the noise generated and propagated by the turbomachinery and to investigate its influence on resulting vibrations of the wall and insulation elements of the overall circuit (see Fig. 27).

The frequency spectrum of the analysed microphone signals consists of a stochastic, broad spectrum of noise contributions and of the narrow spectrum of rotation frequencies of the turbomachinery. The compressor section contributes a fundamental frequency of 2800 Hz and the turbine section a fundamental frequency of 4200 Hz. The averaged total sound power within the helium-circuit and its piping increases with the power of the drive motor up to a maximum level of 160 dB. The total sound power as dependant on the drive power at measuring point 13 is shown in Fig 28.

The modal composition of the acoustic emissions released into the hot gas duct was determined also. Fig. 29 shows the modal compositions measured at different pressures and temperatures during the trial run.

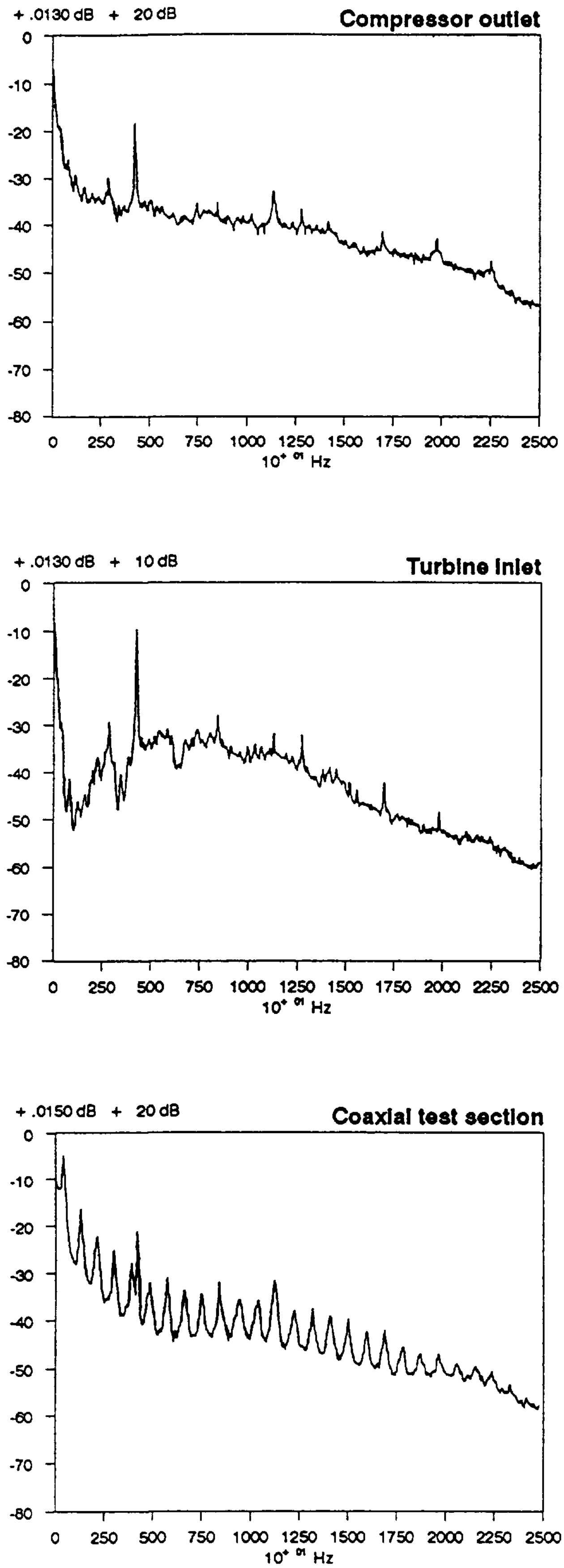


Fig. 27: Sound power spectrum at various locations of the HHV-plant

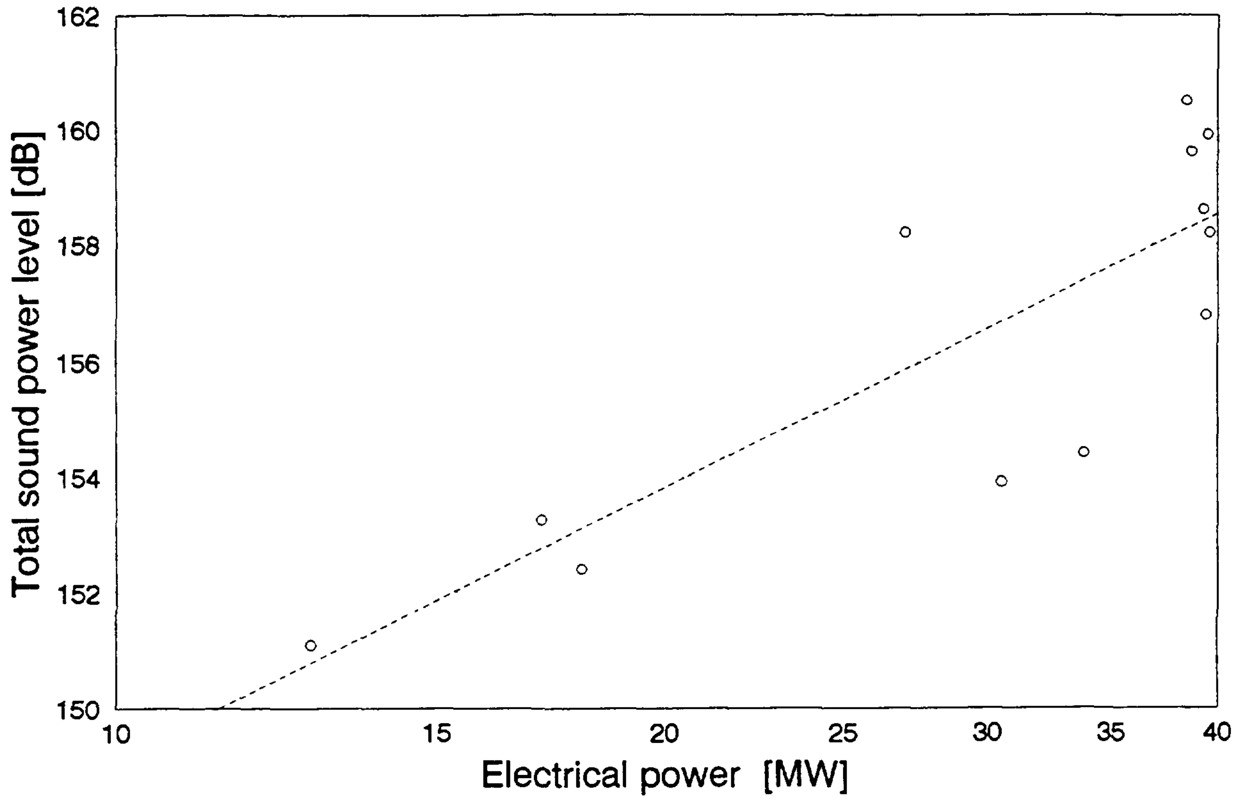


Fig. 28: Total sound power level in dependance on the drive power

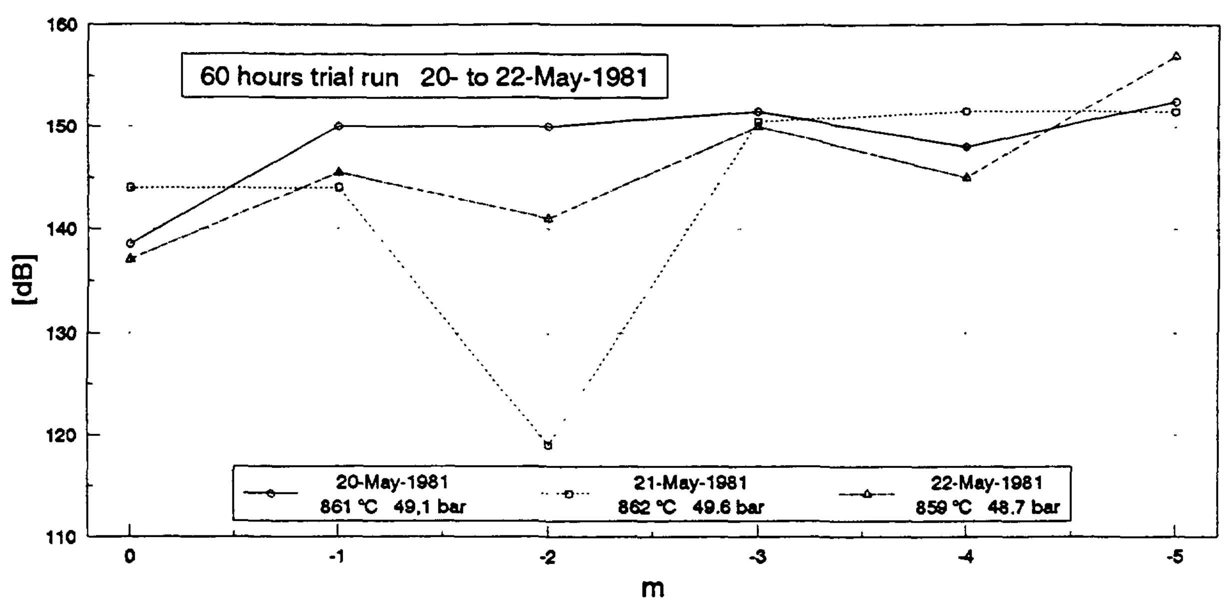


Fig. 29: Modal composition of the acoustic emmissions into the hot-gas duct

The knowledge of the modal composition in sound fields including the magnitude and phase angle enables the calculation of the total spatial distribution of the sound pressures. Using the developed calculation procedure, the dynamic response of the circuit insulation resulting from the excitation by the sound field can be calculated.

3.2 Systems/Components/Materials

3.2.1 EVO

(Ref. [16], [19], [20], [36])

Helium purification

In order to gain experience, the helium purification-system of the fossil-fired EVO plant was designed in accordance with the requirements for a nuclear heated plant, except without provision for the removal of radioactive impurities. The design throughput was 100 kg/h and the required cleanliness was < 1 ppm for any substance. The flow scheme of the system is given in Fig. 30.

The inflowing helium containing impurities was conducted through a dust filter (F) into the gel-filled molecular sieve T (adsorption of H_2O and CO_2). Subsequently, it flowed through a recuperator, where it was cooled down to about 90 ° K, into the liquid N_{2liqu}-cooled heat exchanger where it was cooled down further to 80 ° K and where N_2 and O_2 were partially removed. From there it flowed to a separator (separation of gaseous and liquid N_2 or O_2) into the low temperature absorbers (GA), which were filled with several types of gels for the removal of residual N_2 and O_2 as well as of other gaseous impurities.

During the first operating phase the required and expected cleanliness of the helium could not be reached even with a continously operating purification system. The cause was found in the sealing oil circuit, since at various locations the sealing oil contacted and dissolved air. This combination of sealing oil and dissolved air found its way into the main helium circuit. The problem was solved by providing an additional degasifier for the sealing oil.

Table 3.2.1/1 shows the impurities measured in the helium main circuit of the EVO plant. The first line shows the values without degasification and the second line those with degasification of the sealing oil. The third line shows the content of impurities in the main circuit if the helium purification system is not operated. Lines 4 and 5 permit a comparison with AVR and Dragon (except for radioactive impurities). Line 6 gives the maximum permissible values as specified for HHT.

The overall performance of the helium purification system was judged satisfactory. Since no oxidation bed for the removal of H_2 was provided, no statement on the removal of that type of impurity can be made.

Forces from Gas Ducts and Heat Exchangers on the Turbomachinery

At the initial startup of the EVO plant in May 1975 it was suspected that large unplanned and undefined forces from the ducting and from the heat exchangers were acting on the turbomachinery. These large forces had an adverse effect on the running behaviour of the turbomachinery and moreover these large forces acted adversely on the horizontal flange joints of the turbomachinery, affecting the helium leak tightness due to local splitting of the flange joint.

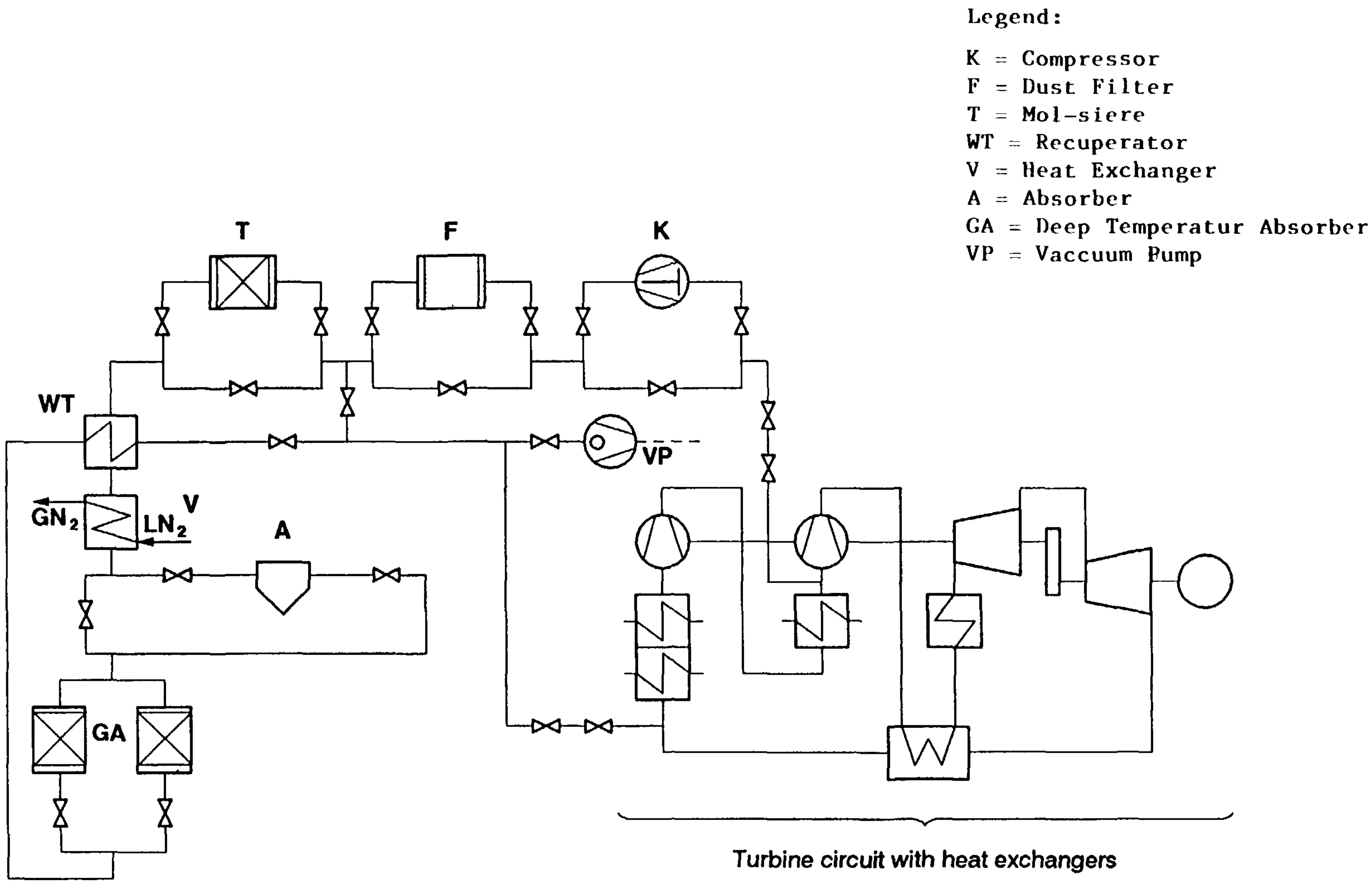


Fig. 30: Helium purification system for EVO

Table 3.2.1/1: Impurities in Helium Circuits

	H_2O	CO_2	H_2	CO	CH_4	N_2	O_2
	cm^3/m^3						
EVO without degasification of sealing oil	< 0.1	20	2	2	< 0.5	240	20
EVO with degasification of sealing oil	15	1.4		5.5	< 0.5	8	< 0.1
EVO after a 2-week operation of the main circuit without He-purification operation	15	22.5		22	13	177	1
AVR Juelich *	0.1	0.5	30..50	125	0.4	20	
Dragon *	0.05	0.02	0.5	0.4	0.05	0.6	
Specified impurity limits of HHT *	5	1	50	50	5	5	0

* Radioactive impurities are not included

Operational measurements showed that the turbomachinery foundation moved at startup first in one direction and when reaching the operation temperature in the opposite direction.

Subsequent inspections, which included disconnection and reconnection of the ducting to the LP turbine and of the main heat exchanger to the ducting showed large deviations of the initial alignment of the rotor axis of the LP turbine, which initially was accurately aligned. A realignment was required. It became evident that the design provided for compensating the thermal expansions and torsions of the ducting, heat exchangers and turbomachinery was not satisfactory i.e. the original provisions for slide bearings, support springs and bellows were not sufficient.

Under the restriction that the very compact plant arrangement could only be changed slightly, successful additional provisions were taken as follows:

- Provision of an additional bellow in the duct between the heat exchanger and the LP-turbine
- Additional insulation of the foundation steel structure
- Modifciation of the forces of some support springs

After having performed these improvements no further severe problems or misalignments were observed.

Hot Gas Ducts

A coaxial design was selected for the hot gas ducts of the EVO plant. The hot helium (750 °C) flows in an inner liner tube, with a fibre-type insulation provided on the exterior, enclosed by a inner pressure bearing tube. The colder helium flows in a surround annulus, with the outer tube designed for full system pressure.

This design was also applied to the bellows and flange joints. All flange joints are provided with weld lip seals.

The design chosen for the hot gas insulation proved reliable.

Corrosion, Erosion on Turbine and Compressor Blades

No signs of any corrosion or erosion of the turbine and compressor blades or other helium-components could be observed at the inspections, modifications and repairs of the EVO plant and EVO turbomachinery.

3.2.2 HHV

(Ref. [23], [24], [34], [35], [36], [37])

Helium purification

Description of Design

Fig. 31 shows the flow scheme of the continously operating helium purification system of the HHV-plant.

At normal operation a partial flow of 1000 Nm^3/h was extracted from the test plant (total inventory: about. 8000 Nm^3/h) and processed in the helium purification system. The extraction occurs at the HP side of the helium compressor through a dust filter that follows the oxidation bed. This oxidation bed contains two beds, one consisting of copper and copper-oxide on carrier material. In this bed H_2, CO and O_2 contained as impurities are converted to H_2O and CO_2. Before being conducted into the cryogenic section of the purification plant, the helium is cooled and then conducted through an activated carbon filter provided for absorbing carried-over oil and a dust filter provided for retaining activated carbon dust or other dust.

Inside the cryogenic section H_2O and CO_2 are removed in the counter-flow cooler/freezer. Inside the low-temperature absorber N_2, Ar and CH_4 are absorbed by activated carbon. The purified helium is then returned through a dust filter into the main circuit.

The effectiveness of the helium purification system is continously monitored by a gas chromotograph and a humidity measuring device. The specified cleanliness of this system was: $H_2 < 5$ vpm, O_2, Co, Co_2, Ar and CH_4 each < 1 vmp, H_2O and N_2 each < 0.5 vpm and oil < 0.1 vpm.

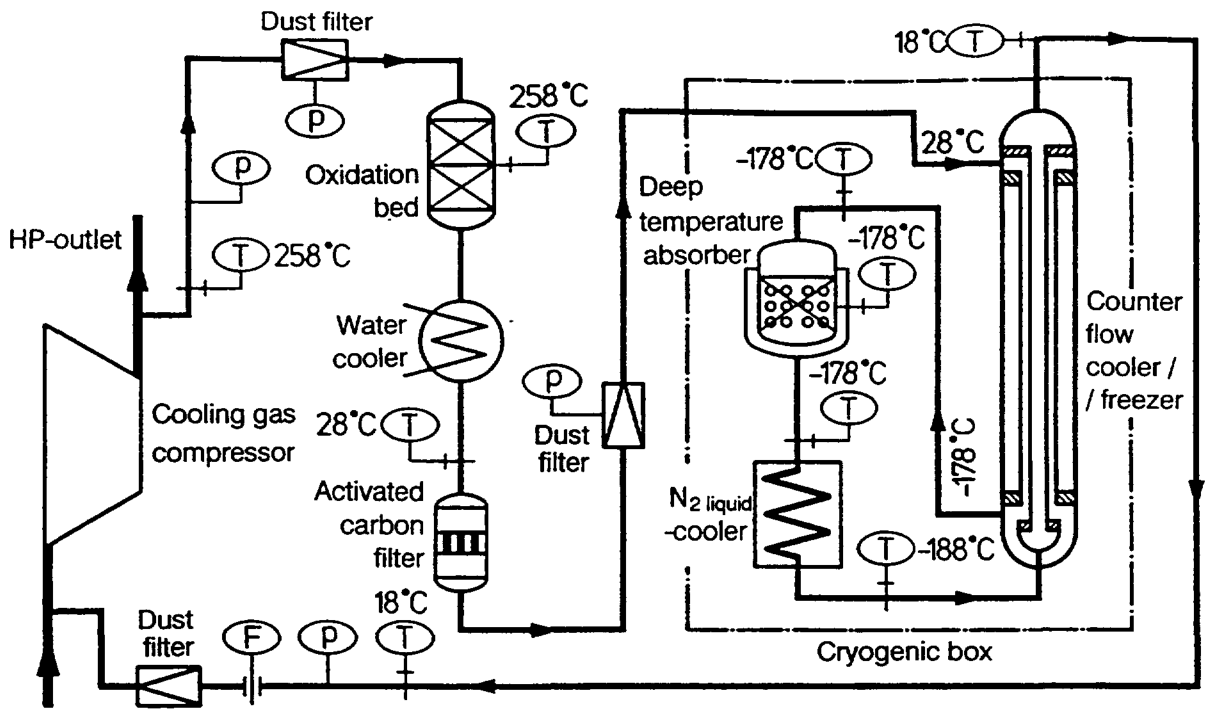


Fig. 31: Helium purification system

Operational Experiences

Operation showed that it was necessary to install within the main helium circuit an additional device for measuring the accrueing special gaseous impurities, that is hydrocarbons and oil vapor. For this purpose a "flame-ionisation-detector" was provided and proved successful.

As a whole, the helium purification system proved to be a reliable and well-functioning system. Although the content of impurities in the helium to be purified was higher than specified and expected, the purification system lowered each of the gaseous impurities to values < 0.1 ppmv. This latter value is below the lower detection limit of the highly sensitive gas chromotograph used, which means that the actual function was better than specified.

Hot Gas Ducts with Insulation, Regulation Valves, Bellows

Hot gas Ducts

For development purposes three different types of hot gas duct sections were used and evaluated:

- One having an inner liner, providing flow guidance and a outside pressure-tight wall with Kaowool-insulation stuffed in between.

- One with coaxial flow guidance. An inner tube carries hot helium and an outer annulus carries cold helium. The inner tube is provided on its outer surface with a Kaowool-mat-insulation and the inner wall of the pressure bearing tube with a Kaowool-mat-insulation also.

- One test section of the hot gas duct with inner metal foil insulation instead of the Kaowool insulation and an outer pressure-tight wall.

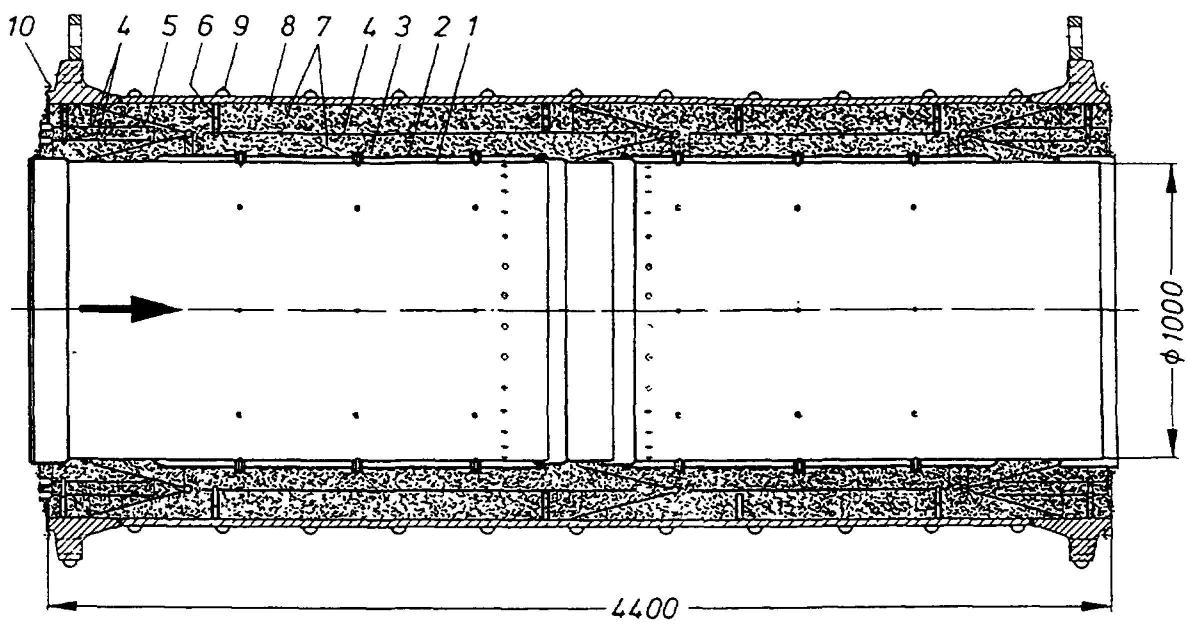
Fig. 32 shows the design of the hot gas duct with inner liner (made of Inconel 625), the surrounding volume stuffed Kaowool-insulation and the outer pressure tube made of ferritic steel, which is water cooled by means of welded-on semi-tubes.

Fig. 33 shows the coaxial design of the hot gas duct shown with the example of a bend. It consists of a inner flow guidance tube, the surrounding Kaowool mat insulation, the cold gas tube (where the cold helium flows in counterflow to the hot helium), the outer Kaowool-insulation and the water cooled pressure bearing tube.

The test sections of the three hot gas ducting designs were instrumented in order to get the necessary information.

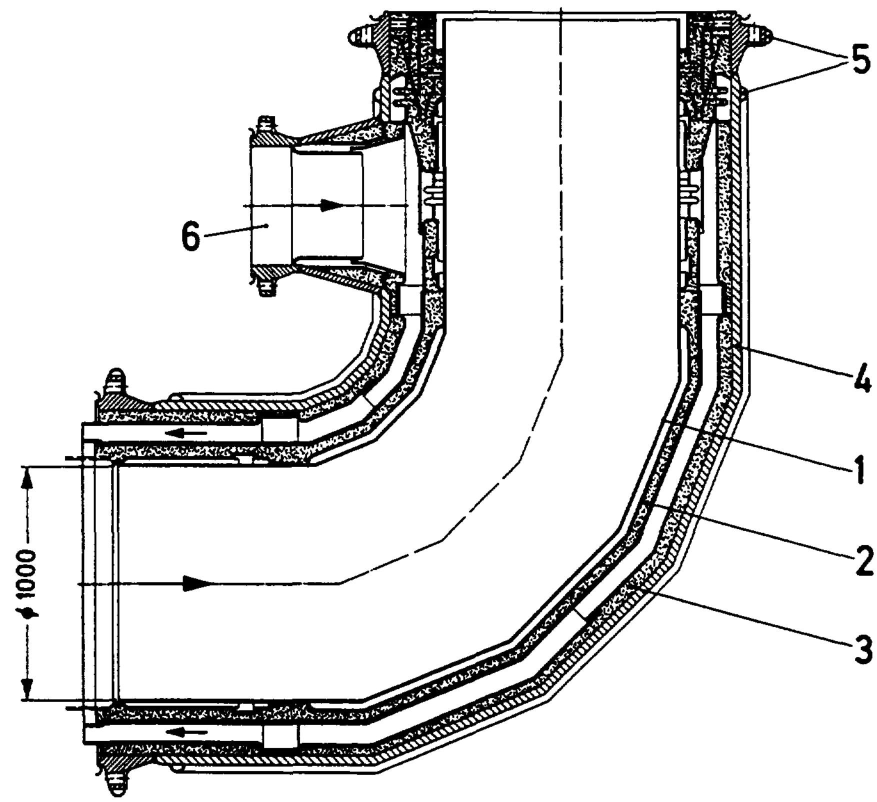
Flow Regulation

Two valves to regulate the hot gas flow through the test section were provided. They have a nominal width of 1000 mm and are arranged in the hot gas duct behind the bypass branch and before the bypass reentrance. There are also two smaller valves in the bypass line with a nominal width of 830 mm and also in the hot gas extraction line with a nominal width of



1 Inner liner
2 Perforated sheet plate
3 Mesh wire
4 Interim sheet plate
5 Support element
6 Fixture bolt
7 Stuffed insulation
8 Outer wall
9 Water cooling
10 Weld lip seal

Fig. 32: Longitudinal section through a hot gas duct with inner insulation



Legend:

1 Inner liner
2 Inner insulation
3 Outer insulation
4 Outer pressure tube
5 Water cooling
6 Cold-gas inlet

Fig. 33: Longitudinal section through a hot gas duct bent with coaxial flow

400 mm. The valves are not required to be completely leak-tight. The typical design of such a regulation valve is shown in Fig. 34.

Bellows

Two bellows are provided in the main hot gas duct in order to compensate for thermal expansions in the horizontal direction. The bypass line is also provided with three bellows for compensating the horizontal thermal expansions.

Operational Experiences with the Hot Gas Duct

The 60 h trial run was defined and used for the assessment and the comparison of the insulation properties of the different types of hot gas ducts. During the measurements the helium circuit conditions had been maintained nearly constant. The results are summarized as follows:

Different Test Sections

- Tests with the hot gas duct with the inner liner, Kaowool insulation and outer pressure tube showed that only small gas flows were observed within the undisturbed axial length of the insulation. However, at the location of transition to the flange joint, where the Kaowool stuffing density is probably less homogenous, larger temperature differences inside the insulation as well as on the outer wall were measured. The average Nusselt-number coincided well with former measurements. However, the distribution of the Nusselt number over the circumference was less favorable than expected. This deficiency can be explained by the stuffing density, which was 15 % lower than normal, in order to implant the instrumentation devices.

- The coaxial hot gas test section with cold gas in the outer annulus and provided with Kaowool-mat insulation showed a very satisfactory performance.

- For the foil insulated duct larger temperature differences were expected between the bottom and the top sections. The measurements showed that close to the flange connections some temperature peaks occurred on the cooled outer wall, which were slightly above the maximum permissible peaks of 80 °C. The measured circumferential distribution of the Nusselt number led to the conclusion that natural convection was the probable reason for the temperature peaking, especially in the vicinity of the flange joints. A satisfactory modification of the as-built design was not considered to be feasible.

Insulation

The comparison of both insulation types, stuffed Kaowool insulation versus metal foils insulation, led to the unanimous conclusion that the Kaowool insulation type is definitely preferable. However, further confirming tests at other pressures and temperatures are needed as well as tests for long-term behaviour.

Cooling

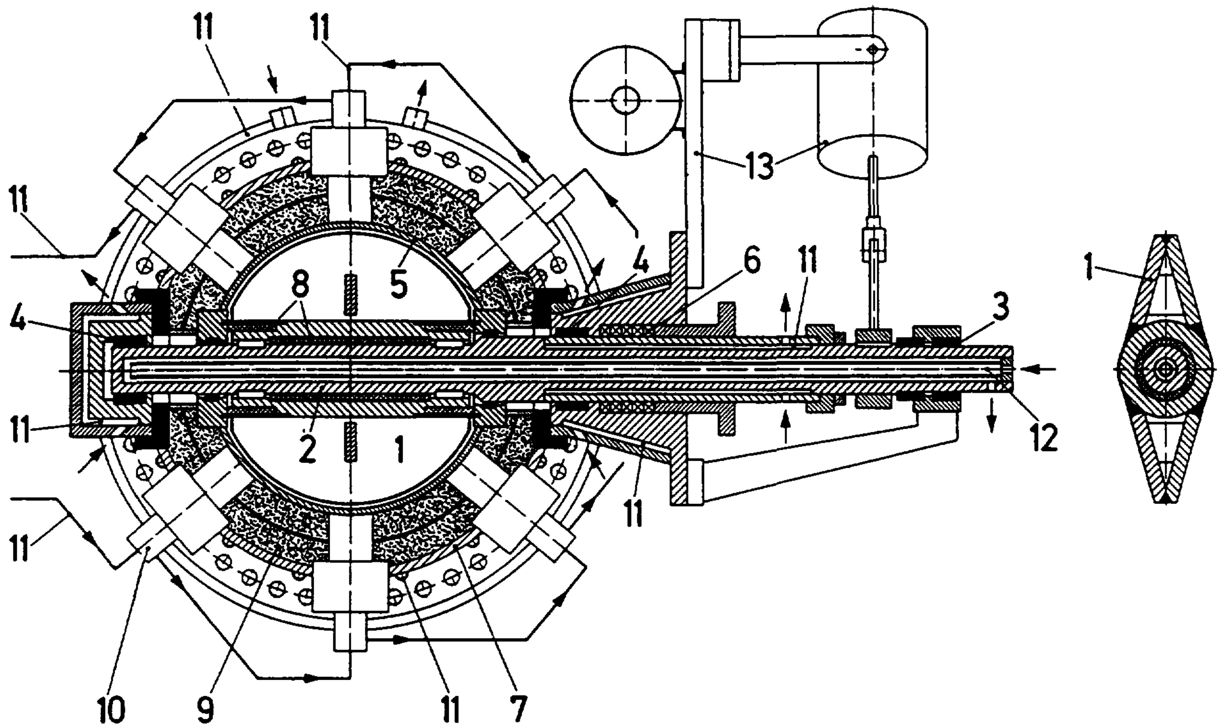
The outside water cooling, with the cooling water flowing through semi-tubes or cover shells, functioned very satisfactorily. Nowhere did the surface temperature of the hot gas duct exceed 60 °C, except for two uncritical spots with metal foil insulation, where local temperatures of about 100 °C were measured.

Regulation Valves

The regulation valves provided for regulation demonstrated reliable functioning.

Thermal Expansions and Flange Joints

The thermal expansion behaviour of the hot gas ducting corresponded with the precalculated behaviour. The horizontal expansions were taken up by the bellows and the sliding supports located inside. Vertical expansions were accomodated by support springs. The welded lip seal flange joints maintained their helium tightness even after multiple pressure and temperature cycling.



Legend:

1	Flap disk	8	Insulation sleeves
2	Flap shaft	9	Stuffed insulation
3	Axial bearing	10	Support bolt
4	Support bearing	11	Water cooling
5	Inner liner	12	Helium cooling
6	Stuffing box	13	Regulation drive
7	Outer housing		

Fig. 34: Cross section through a hot gas regulation flap

Impact of Sound Load on Hot Gas Duct and Insulation

In the tests with the metal foil insulation, numerous strain gages were placed to determine the influence of the sound field on the fatigue strength of the hot gas duct. The main goal of these investigations was the validation of a calculational method to precalculate randomly occuring vibrations from known local distribution and frequencies of the sound pressures. The calculations showed and the measurements confirmed that the flow guidance tube would not exceed its fatigue strength limit during the designed lifetime of 40 years. It is interesting to note that the measured structure responses were of the same order as the precalculated responses. (Also, see chapter 3.1.2 on this subject)

Experimental Experiences with other main Components

Drive Motor, Startup Motor

The synchronous drive motor (45 MW) at 3000 rpm showed a trouble-free operational behaviour. The same applieds to the asynchronous three-phase motor (4.5 MW) for the start-up.

The gear provided between main shaft and startup motor, which is required only for this test plant, released some noise.

Cooling Gas Compressor

Two redundant radial-type compressors were provided. They were arranged in a barrel-type housing and each had a throughput of 56.8 kg/s at an nominal inlet temperature of 236 °C with an inlet pressure of 49 bar and an outlet pressure of 53.5 bar at 258 °C.

At the commissioning large problems had been observed because of the ingress of oil via the labyrinth seals into the gas circuit. As described earlier, after taking adequate countermeasures, no further difficulties with oil ingress or helium leaks were experienced.

Coolers, Oil Pumps and Auxillary Systems

All other vital components in the auxiliary systems operated very satisfactorily.

Cooling Water Flow Monitoring; other instrumentation/interlocks

The test operation of the HHV showed that monitors were necessary in order to ensure that the cooling water flows in each of the approximately 50 cooling water circuits. Moreover operationing experience showed the need for a hydrocarbon detector and for adequate interlocking in the sealing oil lubrication systems.

Behaviour of Protective Coatings

Pretests indicated that coatings were probably required for structural components which have sliding motions during the operation when exposed to high temperatures in pure helium. This was thought to be necessary to prevent excessive friction and seizing, as well to provide for disconnectable joints. As a result of the pretests, coatings used were of chronium-carbide/zirconium-oxide and provided on the surfaces either by a plasmaspray procedure or a detonation coating procedure. Coatings made of boron nitride powder were used for less critical positions, e.g., at the sliding positions of the inner hot gas duct liner.

All these coatings showed a very satisfactory behaviour. However, no judgment on the long-term behaviour can be made on the basis of the short HHV-operational history.

Corrosion and Erosion of Turbine and Compressor Blades

At the inspections, modifications or at the dismantling of the HHV no signs of any corrosion in the helium-circuit or any sign of erosion at the turbine or compressor blades were observed. The sieve provided in the hot gas duct for catching residual particles from the installation or particles accrueing during the operation did not prove to be necessary.

However, the reservation must be made that the HHV operation was too short to make any reliable statement on corrosion as well as on erosion.

Status of Magnetic Bearing Development

At the beginning of the nineteen eighties, the engineering development of magnetic bearings (radial and axial bearings and catcher bearings) for the helium blower and the helium turbine had ben initiated by ABB. A test stand had been set up in Jülich. Test bearings had already been provided.

At the termination of the HHT project the development was reoriented and concentrated on the magnetic bearings for helium blowers. Due to the lack of funds, ABB terminated the development at the beginning ot the nineteen nineties.

However, in the meantime the test stand has been reinstalled and recommissioned at the "Hochschule für Technik, Wirtschaft und Sozialwesen" in Zittau/Görlitz. There the interrupted development for magnetic bearings shall be continued.

3.3 Other Experiences

A number of technical items to be addressed as requested by IAEA for the EVO and HHV plants cannot be treated sufficiently and completely by considering exclusively the experimental experiences from these two plants. That is true for the following reasons:

- the peak temperature of the EVO plant (750 °C) was too low
- the operation time of the HHV-plant was too short

Therefore a number of the items to be addressed like:

- effect of impurities on materials
- performance and wear of coatings
- helium purification
- erosion of blades by graphite particles
- helium leakage
- hot gas ducts including valves and regulation valves
- general operation experiences with high-temperature systems

can be judged better by additionally considering the operational and experimental experiences of the AVR and KVK and EVA II test facilities.

The respective technical experiences additionally gained from the AVR and the KVK and EVA II test facilities beeing relevant for the helium turbine technology are summarized in Annex II.

4. Summarizing Conclusions on Technical Experiences

(Ref. [19], [21], [23], [24], [39])

4.1 Turbomachinery

Both the designs and operational experiences for helium turbines and compressors and their auxiliary equipment for EVO and HHV facilities are described in this report on the basis of publicly available information.

While the orginally planned scope of the experimental work of EVO and HHV could not be executed due to funding limitations and early funding termination, the reasons for the critical problems experienced were found and corrected where economically possible. The principal countermeasures:

- EVO: the shortening of the bearing support distance for the HP turbine/HP compressor, the application of other bearing-types, the provision of an additional bellow in the hot gas ducting to prevent another thermal misalignment of the rotor axis and a splitting of the horizontal flange.

- HHV: successfull modification to prevent (1) the reoccurrence of oil ingress into the main turbomachinery and into the main helium-circuit, (2) to prevent the excessive helium leck rate, and (3) the provision of detectors for hydrocarbons in the main helium circuit.

Positive experiences were achieved with both facilities after overcoming these initial deficiencies:

- Excellent performance of the gas and oil seals
- Low helium leak rate
- Excellent performance of the hot gas ducting, of the turbomachinery cooling, of the helium purification system (except the oil aerosole separation), and of the instrumentation and regulation.

The dynamical performance of the HHV turbomachinery was patently excellent, but the EVO turbomachinery showed at first insufficient dynamical behaviour. This dynamical behaviour was improved sufficiently however, whereas the power deficit of the turbomachinery could not be overcome without significant rebuilding or exchange of the turbomachinery. The reasons for all the experienced problems were well identified and completely redesigned turbomachinery was proposed to replace the first turbomachinery.

The German gas turbine experts at EVO, ABB, Siemens/ KWU and KFA judge the experimental experiences achieved and the accompanying analyses very positive. No unresolveable problems were identified. It is believed, that the results and experiences achieved provide a firm basis if a new initiative is taken for helium turbine power conversion.

Annex I

History of the HHT Project in Germany

(Ref. [23])

Subsequently the history of the HHT-development is described:

- Initiation of the HHT project in 1968 within the 3rd German Atomic Energy Development Program

- Start of the feasibility investigation in 1969 for a 300 to 600 MW-demonstration plant

- End of the feasibility phase: 1972

- Inclusion of the HHT project in the 4th German Atomic Energy Development Program in 1972

- Initiation of phase I of the HHT project by BBC/HRB/KFA in an international cooperation with General Atomic Company, USA and BBC/EIR; Switzerland in 1972

  Reference design: Block-type fuel elements; 3 x 360 MW He-turbine, integrated arrangement of the three turbosets in the prestressed concrete pressure vessel.

- Decision to construct HHV: 1972

- Decision to modify the HHT reference design by providing one single He-turboset with 1240 MW and block-type fuel elements in 1975

- Decision to use the pebble bed core, like for the PNP project, for HHT in 1978; reference power for a commercial power plant: 1240 MW with one helium turboset

- Decision for a demonstration HHT power plant with 676 MW and pebble bed core in 1978; further goal: Assessment of the HHT safety concept by independant experts (which was concluded in November 1981).

- Decision to cancel the HHT project in 1981 and to concentrate instead on a shorter term feasible HTR with the steam cycle: Thermal power 3000 MJ/s with block-type fuel elements and later with pebble bed core.

It should be emphasized that the HHT-project was always actively supported by German and Swiss utilities.

Annex II

Relevant Technical Experiences additionally gained from the AVR and other High Temperature Helium Test Facilities in Germany.

(Lit. [40] to [67])

<u>General</u>

The helium-cooled, high temperature test reactor of the Arbeitsgemeinschaft Versuchsreaktor (AVR) (thermal power: 45 MJ/s, electrical power: 15 MW) was operated for approximately 127,000 hours and approximately 45,000 hours at a helium temperature of 950 °C. Besides its function as an integral test reactor for spherical fuel elements, valuable experiences were gained for the typical helium systems and components.

The component test loop (KVK) was provided for testing the main helium components of a nuclear process heat reactor concept (PNP) at 950 °C (max. 1050 °C for short periods). The main test components were: helium to helium heat exchangers, hot gas ducts, hot gas valves, bellows, steam generators, a helium purification system, helium blowers and their acoustic emissions, helium leakage, control, and other miscellaneous systems and components. The overall operation time of the KVK amounts to more than 20,000 hours at 950 °C or even higher temperatures. Its main operating parameters were:

Thermal power:	10 MJ/s (max. 12.8 MJ/s)
Temperature:	950 °C (max. 1000 °C to 1050 °C for short time periods)
Pressure:	40 bar (max. 46 bar)
Helium mass flow:	max. 4 kg/s on primary helium-side and max. 20 kg/s on secondary helium-side
Velocities:	< 60 m/s
Max. temperature transients for test purposes:	± 200 K/min
Max. pressure transients for test purposes:	± 5 bar/s

A test facility at the Research Center Jülich known as the EVA-II was provided as test stand for testing different types of steam reformer tube bundles and different catalysers for the steam reforming of methane. The EVA-II test stand operated for about 15,000 hours at 950 °C (and for a smaller number of hours at 1000 °C). Its main operating parameters were:

Thermal power:	10 MJ/s
Temperature:	950 °C (max. 1000 °C)
Pressure:	40 bar
Helium mass flow:	4 kg/sec

The main technical experiences relevant to the high temperature helium technology gained in the AVR and at the two related test facilities are summarized below.

1. **Helium Leak Rate (AVR, KVK, EVA-II)**

The following data were obtained at normal operation:

AVR:	10 to 15 Nm^3/d (at an inventory of 2000 Nm^3)
KVK:	< 6 Nm^3/d (at an inventory of 3600 Nm^3)
EVA-II:	2 - 3 Nm^3/d (at an inventory of 1000 Nm^3)

2. **Hot Gas Ducts at 950 °C (KVK)**

Within the framework of the PNP project, the basic development of the hot gas ducts including related construction elements, e. g., flange joints, bellows, bends, and different types of insulation, was performed in special smaller test stands. The integral testing at 950 °C to 1000°C was mainly performed in the KVK. The following test components for hot gas ducts were tested at steady-state and cycling transient conditions:

	Test Time
• Primary helium system hot gas duct	10100 h
• Primar hot gas duct compensator	5300 h
• Hot gas connection at the core outlet header	1700 h
• Secondary helium system, hot gas duct	10600 h
• Bend for secondary helium system, hot gas duct	10100 h
• Axial-type valve NW 750 in secondary helium system	3900 h
• Ball-type valve NW 700 in secondary helium system	5300 h
• Raco-globe valve NW 150	5300 h
• Axial-type valve NW 200	14000 h
• Hot gas throttle valve NW 200	8700 h

Main design features and experimental results:

The reference primary hot gas duct was designed as coaxial gas duct with the hot helium in the inner tube and the cold helium in the outer ring gap. As flow guidance a liner consisting of a graphite tube and alternatively of a carbon-fibre-reinforced graphite tube were used. The insulation around the outside of the liner consisted of a wrapped fibre-type insulation (Al_2O_3 and SiO_2 fibres) and graphite foils arranged to prevent convective flows within the insulation. The pressure was retained by an outer, uncooled (no forced cooling) pressure tube.

The reference, secondary hot gas duct (900 °C) was designed as a duct with an inner flow guidance liner, an fibre-type insulation wrapped on the liner (Al_2O_3 and SiO_2) and an outer uncooled pressure tube.

Bends and bellows required for the primary and secondary hot gas ducts were also provided with similar insulation.

For the primary as well as for the secondary hot gas ducting protective coatings were used. They were provided as a sprayed-on base layer (Ni Cr Al Y), followed by a second sprayed-on $Z_r O_2$-layer (stabilized with $Y_2 O_3$). These coatings were at positions where sliding motions due to thermal expansions were expected. In the course of the test operations (mounting and demounting of test sections), there were some spots were coatings had been damaged mechanically (locally chipped-off layer spots). But this damage did not lead to any difficulties (no seizing or hooking or similar). Otherwise the postexamination showed no damage or wear at the sliding supports.

The most complicated components to be designed had been the isolation valves and the regulation valves in the secondary helium duct. Two different types of isolation valves, namely axial-type and ball-type valves, were tested. The insulation was designed according to the same principle as for the straight secondary hot gas duct.

After experiening some initial difficulties, deficiencies were repaired (e. g., hot spots, defective supports, insufficient prevention of convective flows and defective temperature sensors at the penetrations through the insulation). The performance of the primary and secondary hot gas ducts, including bends and bellows, was excellent even after the cycling transient tests (e. g., 1500 load cycles for the bellows). Except the large hot gas valves, the design of the primary and secondary hot gas ducts with flange joints, bellows and bends were considered as proven and reliable for a long-term operation at 950 °C (max. 1000 °C). A design life of 40 years might be expected. To assure such a life time further endurance tests for the components are required.

It appears possible to use a less expensive version of ducts, with an inner metallic liner (e. g. made of Inconel 617) and no coaxial flow guidance for helium temperatures up to 950 °C or even 1000 °C. Moreover it became evident that a coating at the sliding transitions or sliding supports (thermal expansions) might not necessarily be needed. A final confirmation test is still required however.

The first tests of the large hot gas valves showed very unsatisfactory results concerning insulation, function and seat tightness. Although a large number of modifications leading to essential improvements were made, a further development of the constructive design and subsequent tests would be required before an application is possible.

3. Helium Purification and achieved Helium Atmosphere (AVR; KVK, EVA-II)

a) AVR

The AVR is provided with a helium purification system for removing radioactive and non-radioactive impurities. It consists of three sections:

- A pre-purification section with a dust filter for particles > 0,3 μm (throughput 1000 Nm^3/h) followed by cooler (throughput 50 Nm^3/h)
- A section for removal of radioactive impurities (throughput 50 Nm^3/h) and
- A section for removal of other non-radioactive impurities (throughput 50 Nm^3/h)

The purification system is designed in such a way that the total impurity content (for all substances) at the purification outlet is < 10 ppm (for H_2 < 1 ppm). The dust filters have a separation efficiency of at least 99,95 % for dust particles > 0,3 μm.

Practical experience has shown, that these specified requirements have been always fulfilled. No noteworthy difficulties have been experienced with the purification plant. However it was surprising to discover that only about 1 % of the known dust particles being distributed in all possible locations of the AVR helium circuit could be trapped in the pre-purification dust filter.

b) KVK

The KVK is provided with a redundant helium purification system. In additon to its classical layout, it is additionally equipped with an injection/doping system for injecting such substances as CO, CH_4 and H_2O into the helium circuit in order to adjust a defined helium atmosphere. This is very important for helium systems operating above 850 °C because the metallic materials applied in high temperature helium systems are very sensitive to the chemical attack by helium impurities (i.e., inner oxidation or carburization, decarburization or formation of stable oxide layers on the surfaces). The material tests have indicated that only a very defined narrow band of a permissible helium atmosphere composition can be tolerated and the permissible narrow band becomes smaller with increasing temperatures. In order to test the helium components in the KVK under realistic conditions, it was necessary to operate with a controlled helium atmosphere, i.e., with controlled impurities in the helium.

After having settled some problems at the commissioning, the KVK purification plant generally operated very satisfactorily in spite of the difficult operating conditions of frequent openings of the system and the resulting ingress of air, humidity, dust and impurities. Except for H_2 (< 5 vpm) all other impurities (CH_4, Co, O_2, N_2, H_2O) could be purified to values < 1 vppm each.

c) EVA-II

The helium-purification plant for EVA-II has layout similar to that for KVK, except that no injection/doping device has been provided. The operational experiences correspond with those of the KVK-plant.

4. He-Blowers (AVR, KVK)

a) AVR

The AVR is provided with two speed-controlled radial blowers (400 to 4400 rpm), each with a throughput of 13 kg/s. The blowers have operated at about 275 °C for roughly 130,000 hours without any noteworthy difficulties. Only in the case of a large water ingress in 1979, where water reached the blower axis, did the bearings of one blower became defective. The blower was withdrawn, impected, repaired and reinstalled.

Any signs of corrosion or erosion on the blower blades could not be observed.

b) KVK

The blowers provided in the KVK loops were required for achieving the helium circulation.

Table 3.3.4/1: **Main Operational Data of KVK Blowers**

Blower	1	2	
Mass flow	3	3	kg/s
Suction pressure	39	39	bar
Suction temperature	230	230	°C
Pressure increase	3.30	4.25	bar
Rated speed	8660	15500	min^{-1}
Control range	20-100	20-100	%
Design temperature	350	350	°C
Design pressure	46	46	bar

The blowers are designed as compact uncooled radial compressors of horizontal barrel-type construction They are installed on a steel foundation with an integrated oil system. Drive speed-controlled electric motors with intermediate gearing are provided.

Seals provided at the shaft penetrations of the compressor housing are labyrinth seals, floating ring gaskets, and a static seal.

The shaft with the attached impellers is supported by lubricated bearings arranged outside the compressor housing. The bearing housings are accessible from the outside without dismantling the main housing.

After having settled some initial problems (damage at the floating ring gasket and unsufficient function of the labyrinth seals) the blowers functioned very satisfactorily for more than 20,000 h.

As expected, at the rather low helium temperatures prevailing in the blowers, no corrosion attack or deposits could be observed at the blower blades. Moreover no erosion attack could be visually observed, although the test conditions had been very rough and dust and other impurities had ingressed at numerous openings of the helium systems.

The acoustic emmission of the two compressors into the environment had been very high at the commissioning. Improvements such as a sound insulation had to be made.

c) EVA-II

Similar experiences as described for KVK have been made with the EVA-II blowers.

5. Materials and Effect of Helium-Impurities (AVR, PNP, HHT)

The comprehensive German development and testing of metallic high temperature-resistant materials for the application in helium systems and helium turbines at design temperatures up to 950 °C (max. 1050 °C) have shown the need for a careful control of the helium atmosphere at a narrow limited band for the permissible impurities.

At very high helium temperatures the following problems might arise:

- Inner oxidation of the material leading to a shorter lifetime
- Decarburization or carburization of the material leading to a shorter lifetime
- Destruction or formation of a stable oxide layer on the surface

Therefore the helium atmosphere with the impurities must be adjusted in such a way that stable protecting oxide layers form on the surfaces (without H_2-production) in order to avoid an inner oxidation as well as a carburization or a decarburization, or other corrosion attacks.

The adjustment of the helium atmosphere occurs by purification and/or injection of defined substances, especially defined ratios of CO/H_2O and CH_4/H_2O.

It could be demonstrated in the material test stands as well as in the large test facilities (KVK, EVA-II; EVO, HHV) that the permissible helium atmosphere can be orderly controlled and adjusted by means of the helium purification systems and at very high helium temperatures by the addtional provision of injection systems.

In case of the AVR blower blades, no corrosion attack and no erosion were detected by the visual inspection of one blower in 1979 (after the water ingress and the defective bearing event) considering the background of known and large graphite dust quantities, estimated to be approx. 60 kg, in the AVR-helium-circuit. It was observed that large quantities of graphite dust had been stirred up at every startup and at load changes or other transients. A closer examination showed, that the dust consisted of graphite with only a few metallic particles or other substances. The largest portion of the dust was in the size range of 0.5 μm. However, some graphite particles were found with a size of 1500 μm x 300 μm x 100 μm and some metallic particles with a size of 400 μm x 200 μm x 50 μm. The visual inspection nevertheless had not shown any erosion on the blower blades.

6. General Operation Experiences with AVR and large High-Temperature Test Facilities with Helium Circuit

It is well known that the AVR reactor had an excellent performance record during its 20-year operation (approximately 127,000 hours) including 45,000 hours operation at 950 °C.

The KVK-plant was sucessfully operated for more than 20,000 h with peak temperatures up to 1050 °C and the EVA-II for more than 15000 h at 950 °C. Both test plants demonstrated that large sized, high temperature helium circuits can be operated safely and with a high availability (e. g. in the last three years of the test operation the KVK showed an availability > 95 %!).

Annex III

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WORKSHOP

Session 5

XA9642790

PEBBLE BED MODULAR REACTOR - SOUTH AFRICA

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Abstract

In 1995 the South African Electricity Utility, ESKOM, was convinced of the economical advantages of high temperature gas-cooled reactors as viable supply side option. Subsequently planning of a techno/economic study for the year 1996 was initiated.

Continuation to the construction phase of a prototype plant will depend entirely on the outcome of this study.

A reactor plant of pebble bed design coupled with a direct helium cycle is perceived. The electrical output is limited to about 100 MW for reasons of safety, economics and flexibility. Design of the reactor will be based on internationally proven, available technology. An extended research and development program is not anticipated.

New licensing rules and regulations will be required. Safety classification of components will be based on the merit of HTGR technology rather than attempting to adhere to traditional LWR rules.

A medium term time schedule for the design and construction of a prototype plant, commissioning and performance testing is proposed during the years 2002 and 2003. Pending the performance outcome of this plant and the current power demand, series production of 100 MWe units is foreseen.

UTILITY INTEREST

Load demand forecasting in South Africa includes a fossil-fired power plant under construction as well actioning the de-mothballing of a number of out-of-service boilers. In a worst case load growth scenario (3-4% load growth) the 20% reserve margin will be jeopardized by the year 2006. This suggests that serious consideration must be given to the immediate planning of additional power plant. Should ESKOM, South Africa's single largest utility, decide to construct a new large-scale coal plant, they would commit to a substantial capital intensive project with a construction lead time of 5-8 years. Economic viability dictates that such a plant be located in close proximity to the coal fields on the arid Highveld area. The latest plant under construction is fitted with dry cooling towers, hence adding to the enormous capital investment.

Keeping all of the abovementioned in mind, a logical solution would be to build lower capital cost intensive power plants, with 18 to 24 months' construction time and the freedom to be erected at coastal regions or as single plants in more remote areas eliminating the considerable costs associated with transmission. If the unit capital cost of such plants compare favorably with the bigger plants, then this would certainly become part of ESKOM's supply side options.

Indications are that modular, high temperature, helium-cooled, direct cycle nuclear power plants could be viewed as a strong contestant, hence the willingness to support a feasibility study to establish the PBMR viability.

South Africa, and ESKOM in particular recently completed studies on alternative electricity supply side options e.g. gas, hydro, wind and solar. Large, constant flowing rivers are lacking for producing hydro-electric power. Gas that must be imported from neighboring countries proved to be highly uneconomical. Wind, even at the coastal areas is insufficient for economical power production on large scale. The South African government, however, recently decided to spend a considerable amount of money on the supply of solar units to schools, clinics, and individual households, etc., in remote lying regions, such as Kwazulu-Natal.

Political perceptions

The politically sensitive nature of nuclear projects from a public acceptance point of view is well acknowledged. Open publicity right from the onset of the project should be handled by a public relations team, supported by the technical experts in co-operation with the client. This is largely possible due to the local need for electricity supply, the safety characteristics of the high temperature helium-cooled reactor and the economic advantages offered by the direct cycle power conversion. Support is found in a list of potential benefits, such as high local content, export development, desalination, exploitation of the existing skills base, minimal environmental impact, etc. The initial goal is to employ this beautiful technology to benefit the South African population at large. Mounting public awareness of the fact that the environmental impact of power sources derived from coal energy, is not negligible, contributes towards greater acceptance of nuclear as energy source.

Support will be sought from South Africa's Department of Minerals and Energy Affairs for the proposed techno-economic study.

TECHNO-ECONOMIC Study 1996

The decision to base a cost evaluation on a preliminary design rather than deriving it from costing models was derived from the fact that cost references are firstly, not direct cycle specific and secondly, deviates from existing design bases.

IST will compile a works proposal document identifying work packages to cover a broad spectrum of disciplines from the fuel acquisition right through to the civil structures. Indications of the limited funding available, will limit the depth of investigation. It is sincerely believed that expert involvement from all over the world in basic design, reviews, comments and costing, will however, enable arrival at a sound technical proposal with realistic cost estimates.

OWNER AND USER REQUIREMENT SPECIFICATION

The study is strongly driven by technical and financial objectives. ESKOM recently "published" the first draft of two PBMR-specific reports, namely:

- User's Requirement Specification (URS), and
- A Owners' Requirement Specification (ORS).

These documents specify, for example, reactor availability, reactor and target unit capital and operational costs.

The main and only real objective for this year's techno-economic study is to measure the results against the objectives as spelt out in the ORS/URS. Certainly a first prototype plant will cost more than the unit in series production, therefore the costing will be done for the design, prototype plant costs and for series production plants. The same applies for fuel.

!00 MWE REACTOR CONCEPT

The anticipated plant main characteristics are:

- Performance

Cycle	= Brayton direct cycle (recuperated)
Power	= 200 → 220 MWth; 100 MWe
Overall efficiency	= Better than 45%
Fuel	= Triso particles in 60 mm diameter spheres
Enrichment	< 20%
Power density	= 3 → 4 MW/m^3
Reactor Temperature	= 900 °C outlet, 604 °C inlet
Average burnup	= 130 MWd/kg_{HM}
Maximum allowable fuel temperature	= 1600 °C
Maximum He mass flow rate	= 130 kg/s at 7 MPa System Pressure
Pressure ratio	= 2.2 (2 stage compression with inter-cooler)
Fuelling	= Continuous
Defuelling	= Batch (once every 2-3 years)
Generator speed	= 3000 rpm
Turbo/compressor speed	= 12000 rpm
Decay heat removal	= Passive
Control System	= Non safety class. Industrial high standard with redundancy.

- Physical

Core height	= To be determined
Initial height	= 5,8 meters
Maximum height	= 9 meters
Core diameter	= 3,6 meters
Number of control rods	= 12
Control rod system	= Absorber rod + absorber chain
Support of rotational parts	= Magnetic
Building size	= L = 40m, W = 20m, H = 40m
Position	= Choice of depth based on economics /safety.

- Safety

No active safety system

Seismic design as required

Protection against external events (e.g. in case the reactor pit area above ground level, the confinement should be bunkered)
No off-site emergency plan (preliminary investigation indicates this to be possible).

Plant Layout

Indications are that the reactor, PCU, secondary systems, etc., most of the auxiliary control and instrumentation systems could be housed within a concrete structure of size 40mx20mx40m. The primary components could be housed in a protective reactor pit. One of the design packages should include the finalization of design criteria for these components and housing.

A manifold would be coupled to the reactor. The power conversion components are then coupled to this manifold or to primary pipework located inside the manifold.

The Reactor core and Fuel

The proposed reactor design would be based on the HTR-Modul reactor. It could simply be described as a pebble bed (sphere) core with graphite reflector, carbon insulation, steel core barrel and a steel pressure vessel. The main deviations from the Modul reactor are antcipated to be:

- A higher outlet temperature.
- Design provision must be made for a higher gas return temperature.
- Flow channels are proposed inside the carbon insulation.
- Control rod drive mechanisms mounted externally to the reactor pressure vessel.
- A separate shut down system could be excluded.
- Provision would be made for off-line batch defuelling and on-line *peú-a-peú* fuelling.
- Reactor vessel could be cooled by a cold helium leak stream from the PCU side.
- Core provided with nibs to enhance heat decay heat transfer to passive cooling system.
- Average core power density and average burnup anticipated to be higher.

Reactor isolation, reactor support, core internal packaging method, etc., will be similar to the Modul reactor. The fuel will be of the German design. The source will be available in the world and may eventually be partly manufactured in South Africa.

Power Conversion Unit (PCU) and Generator

The PCU main components, including the generator is anticipated to operate inside a 70 bar pressure boundary. This pressure boundary consists of a coupling vessel to the reactor (called the manifold) and two separate pressure vessel towers (4 bells). The turbo-compressors and intercooler will be housed in the tower nearest to the reactor while the power turbine, recuperator and pre-cooler are housed in the other tower.

Interesting characteristics (features) in the proposal are as follows:

- Separate turbo machinery shafts.
- Power turbine operates on low pressure, low temperature side.
- Differential pressure across the primary piping is low, especially the hottest pipes.
- Small leakage on piping allowed for.
- The pressure boundary is low temperature (can be left uninsulated or use cold He leak as coolant).
- Thermal expansion unconstrained.
- Differential thermal expansion e.g. on pipes can be addressed with bellows.
- Water in the system always at lower than system pressure during operation.
- Access to the first turbo compressor and generator relatively easy. Other components also accessible.
- Isolation between the reactor and PCU allows for enhanced access to PCU components.
- Size of the individual components makes manufacturing and handling easier.
- Testing and qualification of individual components is relatively simple.
- Easy replacement of components (especially in the prototype plant) is possible.
- Startup via a jet pump system. This allows for startup of the system in remote areas where the electricity grid is unavailable.

As is previously mentioned, all operating and accident scenarios may not be accounted for in the current engineering study.

Secondary systems

Of the many secondary systems, some brief comments are made of:

The passive heat removal system: We believe that a specific active residual heat removal system may not be required.

Defuelling system: Should be located externally to the reactor vessel. The defuelling chute must physically be as short as possible. The respective design proposals in Germany and China will be investigated.

Spent-fuel system: The use of an intermediate spent-fuel storage system consisting of multiple containers, cooled by natural air circulation is envisaged. Removal of the containers need to be investigated.

Primary loop clean-up system: Preliminary investigations suggest the use of an in-line filter. Active gasses will be adsorbed by dust particles and subsequent plate-out is anticipated on the cooled surfaces of the heat exchangers. A percentage of the gas will be filtered out in an anticipated bypass system. The presence of a gaseous waste storage system must still be investigated.

Control and Instrumentation

The control system is anticipated to be of typical high standard industrial type. Redundancy will be provided where required. Overall classification of the control system is anticipated to be non-safety graded.

LICENSING AND REGULATION

An overall licensing and regulatory basis appears to be lacking. This is an issue which requires in-depth resolution in the medium term. It is anticipated that the utility will undertake to develop in-house the regulatory guidelines.

Documentation indicating the licensing process as well as providing a basis for licensing (modified 10 CFR 50/52) must be put in place. Discussions with the local Council for Nuclear Safety (CNS) is to be scheduled on a continuous basis.

International contact will be established with relevant institutions and companies.

TECHNOLOGY

South Africa has for many years been involved in nuclear related projects. High as well as low enrichment plants were designed, built and operated. Fuel had been manufactured for the Koeberg Nuclear Power Plant and for the SAFARI materials test reactor. Isotopes, such as Molybdenum-99 are currently being produced and ongoing research and development work on laser enrichment technology forms part of the Atomic Energy Corporation's (AEC) present activities. A center of excellence on small turbines and small to large compressors resides within the AEC. Reactor studies have previously been conducted including small to medium size PWR power plants and a small *peú-a-peú* HTGR.

TIME SCALE

	← Design →			Construct			Commercial Use
Study		←Comp. Manufacturing→			← Comp. Testing →		→
1996	1997	1998	1999	2000	2001	2002	2003

THE ROLE OF THE IAEA IN GAS-COOLED REACTOR DEVELOPMENT AND APPLICATION

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Abstract

Within the Statute establishing the International Atomic Energy Agency there are several functions authorized for the Agency. One of these functions is "to encourage and assist research on, and development and practical application of, atomic energy for peaceful uses throughout the world...". The development of nuclear power is deemed an important application of this function. The representatives of Member States with national gas cooled reactor (GCR) programmes advise the Agency on its activities in the development and application of the GCR. The committee of leaders in GCR technology representing these Member States is the International Working Group on Gas Cooled Reactors (IWGGCR).

The activities carried out by the Agency under the frame of the IWGGCR include technical information exchange meetings and cooperative Coordinated Research Programmes. Within the technical information exchange meetings are Specialist Meetings to review progress on selected technology areas and Technical Committee Meetings and Workshops for more general participation. Consultancies and Advisory Group Meetings are convened to provide the Agency with advise on specific technical matters. The Coordinated Research Programmes (CRPs) established within the frame of the IWGGCR for the GCR programme include:

* Validation of Safety Related Physics Calculations for Low Enriched GCRs,
* Validation of Predictive Methods for Fuel and Fission Product Behaviour in GCRs,
* Heat Transport and Afterheat Heat Removal for GCRs under Accident Conditions, and
* Design and Evaluation of Heat Utilization Systems for the High Temperature Engineering Test Reactor.

This paper summarizes the role of the International Atomic Energy Agency in GCR technology development and application.

1. Introduction

The International Atomic Energy Agency (IAEA) has the function to "foster the exchange of scientific and technical information", and "encourage and assist research on, and development and practical application of, atomic energy for peaceful uses throughout the world".

The IAEA is advised on its activities in development and application of gas-cooled reactors by the International Working Group on Gas-Cooled Reactors (IWGGCR) which is a committee of leaders in national programmes in this technology. The IWGGCR meets periodically to serve as a global forum for information exchange and progress reports on the national programmes, to identify areas for collaboration and to advise the IAEA on its programme. This regular review is conducted in an open forum in which operating experience and development programmes are frankly discussed. Countries participating in the IWGGCR include Austria, China, France, Germany, Italy, Japan, the

Netherlands, Poland, the Russian Federation, Switzerland, the United Kingdom and the United States of America. In addition, the OECD-NEA and the European Union participate in the IWGGCR.

This paper describes the role of the IAEA in Gas-Cooled Reactor (GCR) technology development and application.

2. Background

Worldwide a large amount of experience has been accumulated during development, licensing, construction and operation of gas-cooled reactors. The experience forms a sound basis for programmes which are underway in several countries to develop advanced high temperature reactors for electric power generation and for process heat.

2.1. Summary of operating experience

In the United Kingdom approximately 937 reactor years of operating experience with carbon dioxide cooled reactors has been achieved[a]. Over 20% of the UK's total electricity is generated by its 20 Magnox and 14 AGR gas-cooled reactors, with the AGRs achieving a combined average annual load factor of 75.6% in 1994, the highest of all reactor types worldwide. This remarkable improvement relative to the earlier performance resulted from successful efforts by Nuclear Electric to reduce trip rates and outage times, to improve the refuelling procedures and to increase thermal efficiencies. However, no further GCRs are planned in the UK, and development work will be concentrated on further improvements in plant performance and life extension of existing plants.

[a] *based on IAEA PRIS data base and including the small {~50MW(e)} Calder Hall and Chapel Cross units.*

In France, about 200 reactor years of experience have been acquired through operation of eight Magnox-type reactors demonstrating the soundness, from a technical and safety point of view, of this reactor technology. However, the decision was made some time ago to concentrate on large pressurized water reactors, and the last of France's Magnox reactors, Bugey 1, was shutdown in 1994.

In Japan the 159 MW(e) Tokai-1 Magnox-type reactor continues to be a very successful plant.

The experience with the early helium cooled High Temperature Gas-cooled Reactors (HTGRs), the Dragon plant in the UK, the AVR in Germany and Peach Bottom in the USA was very satisfactory. The experience with the later HTGRs, Fort St. Vrain (330 MW(e)) in the USA and the THTR-300 (300 MW(e)) in Germany, was not entirely satisfactory. The problems which resulted in the shutdown of these plants were, however, not related to the basic reactor concept of helium cooling, and the use of graphite for neutron moderation and as a structural material, nor were they related to any safety concerns, but were primarily associated with technical and economic problems with first-of-a-kind systems and components.

2.2. Summary of national HTGR programmes

Active technology development programmes for HTGRs are proceeding in China, Japan and the Russian Federation.

In Japan an important milestone in development of gas-cooled reactors was reached in March 1991 with the start of construction of the High Temperature Engineering Test Reactor (HTTR) at the Oarai Research Establishment of the Japan Atomic Energy Research Institute (JAERI). This 30

MW(t) reactor will produce core outlet temperatures of 850°C at rated operation and 950°C at high temperature test operation. It will be the first nuclear reactor in the world to be connected to a high temperature process heat utilization system. Criticality is expected to be attained in 1998. The reactor will be utilized to establish basic technologies for advanced HTGRs, to demonstrate nuclear process heat application, and to serve as an irradiation test facility for research in high temperature technologies. The timely completion and successful operation of the HTTR and its heat utilization system will be major milestones in gas-cooled reactor development and in development of nuclear process heat applications.

In China, the Russian Federation and the USA development efforts for electricity producing systems concentrate on small modular HTGR designs with individual power ratings in the 80 to 280 MW(e) range. Strong emphasis is placed on achieving a high level of safety through reliance on inherent features and passive systems. Satisfying this objective forms the basis for the smaller power output of individual modules and for the reactor core configuration. Emphasis has also been placed on a maximum use of factory fabrication, as opposed to field construction, for better quality control and reduction in construction time.

A promising new approach to achieve economic advantage involves use of the modular HTGR with a gas turbine to achieve a highly efficient electric generating system. Recent advances in turbomachinery and heat exchanger technology have led to plant design and development activities in the USA and Russia, with the direct helium cycle as the ultimate goal. It is recognized that the unique features of the modular HTGRs will likely require prototype demonstration prior to design certification and commercialization. With the relatively small size of each power-producing module it is possible to contemplate such a demonstration with just one module, later expanding into a multi-module plant at the same site for commercial purposes. A Technical Committee Meeting on "Design and Development of GCRs with Closed Cycle Gas-Turbines" is scheduled for 30 October to 2 November 1995 at the Institute for Nuclear Technology, Tsinghua University in Beijing, China. A decision was recently made by the USA to focus on the ALWR concept and close out their GCR activities.

China's HTR development activities are focused on the 10 MW(th) Test Module HTR. Cconstruction of the HTR-10 Test Module began in late 1994 at the Institute of Nuclear Energy Technology of Tsinghua University in Beijing. This project will provide experience in design, construction and operation of an HTR. The test module is designed for a wide range of possible applications, for example, electricity, steam and district heat generation in the first phase, and process heat generation in the second phase.

In Germany a strong HTR technology programme was performed in the 1970s and 1980s, and an HTR design with a very high degree of safety has been developed both for electricity generation and for process heat applications. Inherent features and properties of HTRs are particularly conducive to achieving a nuclear technology that is "catastrophe free" and extensive research, development and demonstration activities have been conducted on key process heat plant components. The helium heated steam reformer, the helium/helium heat exchanger and the helium heated gas generator for coal refining have been successfully tested in pilot scale (e.g., 10 MW), and the AVR reactor has demonstrated operation at 950°C core outlet helium temperature.

In Switzerland, in the past, research activities for small HTR concepts including the gas-cooled district heating reactors have been conducted. Current HTR-related activities in Switzerland involve the PROTEUS critical experiments which are being conducted by an international team of researchers at the Paul Scherrer Institute in Villigen. Activities are underway in the Netherlands to assess the potential future role of modular HTRs as a highly safe technology for electric power generation. Other countries including Poland, Italy, Indonesia, and Israel have displayed interest in HTR technology and perform related assessments.

3. International Cooperation

The early development of nuclear power was conducted to a large extent on a national basis. However, for advanced reactors, international co-operation is playing a greater role, and the IAEA promotes international co-operation in advanced reactor development and application. Especially for designs incorporating innovative features, international co-operation can play an important role allowing a pooling of resources and expertise in areas of common interest to help to meet the high costs of development.

To support the IAEA's function of encouraging development and application of atomic energy for peaceful uses throughout the world, the IAEA's nuclear power programme promotes technical information exchange and co-operation between Member States with major reactor development programmes, offers assistance to Member States with an interest in exploratory or research programmes, and publishes reports on the current status of reactor development which are available to all Member States.

The activities carried out by the IAEA within the frame of the IWGGCR include technical information exchange meetings and co-operative Co-ordinated Research Programmes (CRPs). Small Specialists Meetings are convened to review progress on selected technology areas in which there is a mutual interest. For more general participation, larger Technical Committee Meetings, Symposia or Workshops are held. Further, the IWGGCR sometimes advises the IAEA to establish international co-operative research programmes in areas of common interest. These co-operative efforts are carried out through Co-ordinated Research Programmes (CRPs), are typically 3 to 6 years in duration, and often involve experimental activities. Such CRPs allow a sharing of efforts on an international basis and benefit from the experience and expertise of researchers from the participating institutes.

The IAEA's activities in gas-cooled reactor development focus on the four technical areas which are predicted to provide advanced HTGRs with a high degree of safety, but which must be proven. These technical areas are:

a) the safe neutron physics behaviour of the reactor core
b) reliance on ceramic coated fuel particles to retain fission products even under extreme accident conditions
c) the ability of the designs to dissipate decay heat by natural heat transport mechanisms, and
d) the safe behaviour of the fuel and reactor core under chemical attack (air or water ingress).

The first three are the subjects of Coordinated Research Programmes and the last was recently addressed in an information exchange meeting.

IAEA activities in HTGR applications focus on design and evaluation of heat utilization systems for the Japanese HTTR.

3.1. Co-ordinated Research Programmes (CRPs) in GCR development and application

3.1.1. CRP on Validation of Safety Related Physics Calculations for Low-enriched GCRs

To address core physics issues for advanced gas-cooled reactor designs, the IAEA established a CRP on Validation of Safety Related Physics Calculations for Low-enriched GCRs in 1990. At the initiation of this CRP the status of experimental data and code validation for gas-cooled reactors and the remaining needs were examined in detail at the IAEA Specialists Meeting [Ref. 1]. The objective of the CRP is to fill gaps in validation data for physics methods used for core design of advanced gas-cooled reactors fueled with low enriched uranium. Countries participating in this CRP include China, France, Japan, the Netherlands, Switzerland, Germany, the USA and the Russian Federation.

The main activities of the CRP are being carried out by a team of researchers within an international project at the PROTEUS critical experiment facility at the Paul Scherrer Institute, Villigen, Switzerland. Fuel for the experiments was provided by the KFA Research Center, Juelich, Germany, and initial criticality was achieved on July 7, 1992. Experiments are being conducted for graphite moderated LEU systems over a range of experimental parameters, such as carbon-to-uranium ratio, core height-to-diameter ratio, and simulated moisture ingress concentration, which have been determined by the participating countries as validation data needs. The Paul Scherrer Institute has been highly willing to incorporate experiments as defined by the several participating countries to provide results focused on their validation data needs. Key measurements being performed at PROTEUS which are providing validation data relevant to current advanced HTGR designs are summarized in Table 1. A summary of PROTEUS conditions is given in Table 2.

Table 1: Measurements at PROTEUS

* Shutdown rod worth
  - in core
  - in side reflector
* Effects of moisture ingress - for range of amount of moisture
  - on reactivity
  - on shutdown rod worth
* Critical loadings
* Reaction rate ratios (U-235, U-238, Pu-239)
* Neutron flux distribution

Table 2: PROTEUS Conditions

* UO_2 pebble fuel with 16.76% enrichment
* Core equivalent diameter = 1.25m
* Core H/D from 0.8 to 1.4
* C/U-235 from 5 630 to 11 120
* Water simulated by plastic inserts

Also data from the uranium fueled criticals at the Japanese VHTRC critical experiment facility on the temperature coefficient (to 200°C) of low enrichment uranium fuel have been provided by JAERI and analyzed by CRP participants. The results show that calculations of the temperature coefficient are generally accurate to within about 20 percent.

3.1.2. CRP on Validation of Predictive Methods for Fuel and Fission Product Behaviour in GCRs

The experience base for GCR fuel behaviour under accident conditions was reviewed at an IAEA Specialists Meeting in 1990 [Ref. 2], and a CRP on Validation of Predictive Methods for Fuel and Fission Product Behaviour in GCRs was initiated in 1993. Countries participating in this CRP include China, France, Japan, Poland, Germany, the USA and the Russian Federation. Within this CRP, participants are documenting the status of the experimental data base and predictive methods, cooperating in methods verification and validation and will identify and document the additional needs for methods development and experimental validation data.

Technical areas being addressed include:

* fuel performance during normal operation
* fuel performance during accidents (heatup)
  - non-oxidizing conditions
  - oxidizing conditions

- * fission product behaviour during normal operation
  - behaviour of gaseous and metallic fission products
  - behaviour of plateout
- * fission product behaviour during accident conditions
  - behaviour of gaseous and metallic fission products
  - re-entrainment of plateout
  - fission product behaviour in reactor building
- * performance of advanced fuels

3.1.3. CRP on Heat Transport and Afterheat Removal for GCRs under Accident Conditions

A CRP on Heat Transport and Afterheat Removal for GCRs under Accident Conditions also began in 1993 and the experience base at its initiation was reviewed in an IAEA Technical Committee Meeting [Ref. 3]. Countries participating in the CRP include China, France, Japan, Germany, the USA and the Russian Federation. The objective of this CRP is to establish sufficient experimental data at realistic conditions and validated analytical tools to confirm the predicted safe thermal response of advanced gas-cooled reactors during accidents. The scope includes experimental and analytical investigations of heat transport by natural convection, conduction and thermal radiation within the core and reactor vessel, and afterheat removal from the reactor. Code-to-code, and code-to-experiment benchmarks are being performed for verification and validation of the analytical methods. Assessments of sensitivities of predicted performance of heat transport systems to uncertainties in key parameters are also being investigated. Countries are participating in these benchmarks and experimental activities according to their own specific interests. Table 3 lists the benchmarks and cooperation in experiments included within the CRP.

Table 3: Benchmark Exercises and Cooperation in Experiments Included within CRP

	BENCHMARKS	COOPERATION IN EXPERIMENTS
■	Code-to-code (analyses of heatup accidents)	SANA-1
	VGM	SANA-2 pebble / prism - open topic
	GT-MHR	air / water RCCS - open topic
	HTTR [a]	
	HTR-10 [a]	
■	Code-to-experiment	
	HTTR RCCS mockup [a]	
	SANA-1 [a]	
	ST-1565	
	and others being considered	
■	Code-to-reactor	
	HTTR RCCS	
	(normal operation)	
	Startup/shutdown	
	HTR-10 RCCS	
	(normal operation)	

[a] *1995 activities*

3.1.4. Coordinated Research Programme in HTGR applications

To foster international cooperation in HTGR applications the IAEA's Division of Nuclear Power and the Division of Physics and Chemistry have established a CRP on Design and Evaluation of Heat Utilization Systems for the High Temperature Engineering Test Reactor (HTTR). The ultimate potential offered HTGRs derives from their unique ability to provide heat at high-temperatures (e.g., in the range from about 550°C to 1000°C) for endothermic chemical processes and, at 850°C and above, for highly efficient generation of electricity with gas turbine technology [Ref. 4]. Heat from HTGRs can be used for production of synthesis gas and/or hydrogen and methanol by steam-methane reforming, production of hydrogen by high temperature electrolysis of steam and by thermochemical splitting of water, production of methanol by steam or hydrogasification of coal, and for processes which demand lower temperatures, such as petroleum refining, seawater desalination, district heating, and generation of steam for heavy oil recovery and tar sand mining. If the heat demand is not in the immediate vicinity of the reactor, a chemical heat pipe could be developed as a high temperature heat transporter.

Several IAEA Member States are concerned about global environmental problems which result from burning fossil fuels. The application of nuclear process heat can make a significant contribution to resolve these problems. In order to select the most promising heat utilization system(s) to be demonstrated at the HTTR, some Member States wish to cooperate in the design and evaluation of potential HTTR heat utilization systems. Countries participating in this CRP include China, Israel, Germany, Russia, Indonesia, Japan and the USA. The processes being assessed are selected by CRP participants according to their own national interests depending on status of technology, economic potential, environmental considerations, and other factors.

The following are being examined:

- Steam reforming of methane for production of hydrogen and methanol
- CO_2 reforming of methane for production of hydrogen and methanol
- Thermochemical water splitting for hydrogen production
- High temperature electrolysis of steam for hydrogen production
- Gas turbine for electricity generation
- Combined coal conversion and steam generation

In addition, testing of advanced intermediate heat exchangers will be examined.

The CRP participants are collaborating by exchanging existing technical information on the technology of the heat utilization systems, by developing design concepts and by performing evaluations of candidate systems for potential demonstration with the HTTR.

Key tasks of the CRP are to:

a) Define the R&D needs remaining prior to coupling to the HTTR
b) Define the goal of the demonstration with the HTTR
c) Prepare design concepts for coupling selected systems to the HTTR and perform preliminary safety evaluations, and
d) Check licensability of selected systems under Japanese conditions.

Based on evaluations up to now on technology status, the first priority candidate systems to be connected to the HTTR are (1) steam (and/or CO_2) methane reforming system and (2) gas-turbine system. For other cnadidate systems the R&D shall be continued to bring them to the stage in their technology development when they will be considered feasible to be demonstrated at the HTTR.

More detailed information is included in a companion paper [Ref. 5].

3.2. Information exchange meetings (1993-1996)

3.2.1. GCR response under accidental air or water ingress

The IAEA Technical Committee Meeting on "Response of Fuel, Fuel Elements and Gas-cooled Reactor Cores under Accidental Air or Water Ingress Conditions" was hosted by the Institute for Nuclear Energy Technology (Tsinghua University, Beijing, China) in October 1993 [Ref. 6]. Some key conclusions from the Technical Committee are summarized in the following.

The response of gas cooled reactors to postulated air and water ingress accidents is highly design dependent and dependent upon the cause and sequence of events involved. Water ingress may be caused by tube ruptures inside the steam generator due to the higher pressure in the secondary loop. The core can only be affected if steam or water is transported from the steam generator to the reactor. Air ingress is possible only after a depressurization accident has already taken place and has to be looked at as an accident with a very low probability.

Considerable experimental data exists regarding behaviour of GCRs under air ingress conditions. These experiments have shown that self sustained reaction of reactor graphite with air does not occur below about 650°C and above this temperature there is a window of air flow rates: low flows supply insufficient oxidizing gas and fail to remove the reaction products, whereas convective cooling at high flows will overcome the chemical heating. Nuclear grade graphite is much more difficult to burn than coal, coke or charcoal because it has a higher thermal conductivity making it easier to dissipate the heat and because it does not contain impurities which catalyze the oxidation process.

Two serious accidents have occurred which have involved graphite combustion: Windscale (October 1957) and Chernobyl (April 1986). It is important to clearly understand these accident sequences, and the significant differences in the design of these reactors, compared to gas-cooled reactors, which use graphite as moderator and either helium or carbon-dioxide as coolant. Windscale was an air cooled, graphite moderated reactor fueled with uranium metal clad in aluminum. The accident was most likely triggered by a rapid rate of increase in nuclear heating (that was being carried out for a controlled release of the Wigner energy) which caused failure of the aluminum cladding. This exposed the uranium metal, which is extremely reactive, to the air coolant, and resulted in a uranium fire, which caused the graphite fire. Water was finally used to cool down the reactor after other efforts failed. Chernobyl was a water cooled, graphite moderated reactor. The rapid surge in nuclear power generation at Chernobyl resulted from a series of safety violations and core neutronic instabilities. Eventually liquid nitrogen was used to cool the burning debris. It must be emphasized that gas cooled reactors neither use air as coolant (as in Windscale) nor have core neutronic instabilities such as those of the Chernobyl reactor.

Safety examinations of German modular HTR design concepts are addressing even very hypothetical accidents such as the complete rupture of the coaxial hot gas duct. A large scale experiment, called NACOK, is being constructed at the KFA Research Center, Juelich, Germany to measure the natural convection of ingressing air and to provide data for validating theoretical models.

As a part of the safety review of the HTTR, extensive investigations have been carried out by JAERI of that reactor's response to air ingress accidents including rupture of the primary coaxial hot gas duct and the accident involving the rupture of a stand pipe attached to the top head closure of the reactor pressure vessel. Experimental and analytical investigations have shown that graphite structures would maintain their structural integrity because of the limited amount of oxygen within the volume of the containment which is available to oxidize graphite. Further, there is no possibility of detonation of the produced gases in the containment. Experimental test results showed that there is a large safety margin in the design of the core support posts.

JAERI has examined the response of the HTTR to a design basis accident involving rupture of a pipe in the pressurized water cooler. The ingress of water is sensed by the plant protection system instrumentation resulting in reactor scram and isolation of the pressurized water cooler. Analyses show that the amount of ingressed water is insufficient to result in opening of the primary system safety valves, and the auxiliary cooling system rapidly reduces the core temperatures thereby limiting the oxidation of the graphite structures to acceptable levels. Similar investigations have been conducted by INET for design basis accidents of the HTR-10 reactor assuming the rupture of one or two steam generator pipes.

The neutronic effects of moisture ingress on core reactivity and on control rod worth are being examined in Switzerland at the PROTEUS facility. Neutronic effects of water are simulated by inserting polyethylene (CH_2) rods into the core as this material has essentially the same hydrogen density as water. The effect of increasing amounts of "water" is first to increase the core reactivity to a maximum due to under moderation of the neutrons under normal conditions, followed by a reactivity decrease as neutron absorption by hydrogen becomes the dominating factor. Further, water addition into the core has the effect of reducing the worth of the shutdown rods. In the experiments to date, these effects have been well predicted, reflecting perhaps the mature state of reactor physics analysis methods.

To ensure the ultimate goal of a catastrophe-free nuclear energy technology, additional analyses of extreme hypothetical accident scenarios should be performed and, in parallel, methods for enhancing the passive corrosion protection of the graphite fuel elements and structures could be used. Experimental activities in Germany, China, Russia and Japan have shown that ceramic coatings can considerably increase the corrosion resistance of graphite. At the Technical University in Aachen and the KFA Research Center Jülich, Germany, a successful coating method has been developed which is a combination of silicon infiltration and slip casting methods to provide a SiC coating on the graphite. Corrosion tests have been conducted simulating accident conditions (massive water and air ingress) at temperatures to 1200°C. Future efforts are required to examine the behaviour of the ceramic coatings especially with neutron irradiation. Activities at INET have involved forming SiC coatings on graphite structures by exposing them to melted silicon. Oxidation experiments have shown very large reduction in oxidation rate compared to uncoated graphite. Other activities at INET have shown that addition of superfine SiC powder to the fuel element matrix graphite greatly reduces graphite oxidation because SiO_2 is formed by SiC-oxygen reaction thereby partly covering and isolating the graphite micropores from further corrosion. Demonstration of the high resistance to oxidation by air or water of SiC coating on graphite surfaces including successful tests on irradiated structures could result in advantages from a public acceptance point of view as well as a technical point of view for the future design of HTGRs.

The close examination of experience presented to the Technical Committee led to the conclusion that plant safety is not compromised for design basis accidents. Continued efforts to validate the predictive methods against experimental data are worthwhile. Protective coatings for fuel and graphite components which provide high corrosion resistance should continue to be developed and tested as these potentially could provide assurance of safety even for very extreme and hypothetical water or air ingress accident conditions.

3.2.2. Development status of modular HTGRs and their future role

The IAEA Technical Committee Meeting on "Development Status of Modular HTGRs and their Future Role" was hosted by the Netherlands Energy Research Foundation (ECN), Petten (the Netherlands) from 28 to 30 November 1994 on the occasion of the ECN workshop on the role of Modular High Temperature Reactors in the Netherlands, 30 November to 1 December 1994.

The Technical Committee Meeting was convened within the IAEA's Nuclear Power Programme on the recommendation of the IAEA's International Working Group on Gas-cooled Reactors (IWGGCRs). It was attended by participants from China, France, Germany, Indonesia, Japan, the Netherlands, Switzerland, Russia and the United Sates of America. The meeting reviewed the national and international status and activities of the following topics for high temperature reactors (HTRs):

* status of national GCR programmes and experience from operation of GCR's
* advanced HTR designs and predicted safety and economic performance
* future prospects for advanced HTRs and the role of national and international organizations in their development

Though considered an advanced type of nuclear power reactor, helium cooled, graphite moderated reactors have been under development for almost forty years. This Technical Committee Meeting was attended by experts from many countries in the nuclear power community, and represented a significant pooling of experience, technology development and aspirations. While the future role of helium cooled reactors cannot be stated with any certainty, this IAEA Technical Committee Meeting brought to focus the major technical issues, challenges and benefits affecting their future development and deployment.

3.2.3. 12th Meeting of IWGGCR

The 12th Meeting of the International Working Group on Gas-Cooled Reactors (IWGGCR) was hosted by the Netherlands Energy Research Foundation (ECN), Petten, the Netherlands on 2 December 1994 on the occasion of the IAEA Technical Committee Meeting on "Development Status of Modular HTGRs and their Future Role", from 28-30 November 1994 and the ECN workshop on "The Role of Modular HTRs in the Netherlands", 30 November - 1 December 1994. The meeting was attended by representatives from China, France, Germany, the Netherlands, Japan, Switzerland, the United Kingdom, the Russian Federation and the Nuclear Energy Agency of the OECD and by observers from Indonesia and the United States.

The IWGGCR welcomed the representative from the Netherlands to the Working Group as its newest official member.

The IWGGCR congratulated the Japanese Atomic Energy Research Institute (JAERI) on the good progress of the construction of the High Temperature Engineering Test Reactor (HTTR) at Oarai. The IWGGCR also congratulated the Institute of Nuclear Energy Technology (INET), Tsinghua University, Beijing on the start of construction of the HTR-10 Test Module at INET.

The meeting provided an international forum for information exchange between representatives of Member countries regarding their Gas-Cooled Reactor programmes. The members of the IWGGCR strongly felt that the present international cooperation conducted within the frame of the IWGGCR in the field of gas-cooled reactors is of benefit to their own national programmes and recommended that the Agency continue its information exchange actiities and cooperative research programmes in gas-cooled reactor development and application.

3.2.4. Graphite moderator life cycle technologies

Graphite has played an important role as a moderator and major structural component of nuclear reactors since the start of atomic energy programmes throughout the world. Currently there are many graphite moderated reactors in operation which will continue to produce power until well into the next century: also there are graphite moderated reactors currently under construction and others in the design stage.

The last IAEA Specialists Meeting on the status of graphite technology was convened in Tokai-mura, Japan in September 1991. Since that time considerable operating experience has been gained, and materials development and testing programs which are of international interest have been conducted. It is therefore considered appropriate for the international expertise in the nuclear graphite field to be brought together to exchange technical information on graphite lifecycle technologies.

The IAEA, following the recommendation of the International Working Group on Gas-cooled Reactors (IWGGCR), is planning to convene a Specialists Meeting on Graphite Moderator Lifecycle Technologies at the University of Bath, United Kingdom from 25-28 September 1995. A technical tour of an AGR reactor is also foreseen on 28 September, and a tour of the Windscale site is foreseen on 29 September.

The purpose of the meeting is to exchange information on the status of graphite development, on operation and safety procedures for existing and future graphite moderated reactors, to review experience on the influence of neutron irradiation and oxidizing conditions on key graphite properties and to exchange information useful for decommissioning activities. The meeting is planned within the frame of the International Working Group on Gas-cooled Reactors.

It is intended that the programme should involve all topics from the conception of the reactor design through the safe operation and monitoring of the core to the removal and safe disposal of the graphite cores at the end of life. The topics to be included are:

* status of national programmes in graphite technology
* carbon/carbon composites for in-core application
* core design
* core monitoring
* codes and standards
* graphite fuel element manufacture
* graphite property behaviour
* irradiation damage mechanisms
* radiolytic oxidation
* operation and safety procedures for graphite moderated cores
* seismic responses of graphite cores

3.2.5. Design and development of Gas-cooled Reactors with Closed Cycle Gas Turbine

The International Atomic Energy Agency is planning to convene a Technical Committee Meeting and Workshop on "Design and Development of Gas-cooled Reactors with Closed Cycle Gas Turbines" at the Institute of Nuclear Energy Technology, Tsinghua University, Beijing, China from 30 October to 2 November 1995.

The meeting is being convened within the frame of the IAEA's International Working Group for Gas-cooled Reactors (IWGGCR).

The purpose of the meeting is to provide the opportunity to review the status of design and technology development activities for high temperature gas-cooled reactors with closed cycle gas turbines (HTGR-GTs), and especially to identify development pathways which may take advantage of the opportunity for international cooperation on common technology elements.

Recent advances in turbomachinery and heat exchanger technology provide the potential for a quantum improvement in nuclear power generation economics by use of the HTGR with a closed cycle gas turbine. The HTGR-GT offers highly efficient generation of electrical power and a high degree of safety based on inherent features and passing systems. Enhanced international cooperation among national GCR programmes in common technology elements, or building blocks, for HTGRs with closed cycle gas turbines, could facilitate their development with overall reduced development costs. In addition to the common elements being addressed currently through IAEA Coordinated Research Programmes, the technical areas in which international cooperation could be beneficial include fabrication technology and qualification of the coated fuel particles, materials development and qualification, and development and testing of turbomachinery, magnetic bearings and heat exchangers.

The first day will consist of paper presentations on national and international activities on gas cooled reactors, and utility interest and economics of HTGR-GTs. This will be followed by two days of Workshop sessions on the following topics for HTGRs with closed cycle gas turbines:

a) power conversion
b) plant safety
c) fuel and fission product behaviour
d) materials

The Workshops will include technical paper presentations and discussions focusing on the status, needs, and proper development pathways in these technical areas. Reports will be drafted in the Workshops summarizing the status and development needs and especially identifying pathways for international cooperation in development and demonstration in common technology elements. The final day will involve presentations of reports by the Workshop chairmen to the Technical Commitee and discussion of these reports.

3.2.6. 13th meeting of IWGGCR

The 13th meeting of the IWGGCR will be convened in Spring of 1996 in Vienna. The topic for the second TCM to be convened in 1996 will be selected at this meeting.

3.3. Status report on GCR technology

At its 12th meeting the IWGGCR discussed the question of whether a new report on the status of GCR technology in 1995 should be prepared and issued. IAEA as an organization for promoting international cooperation and for providing a forum for exchange of information for advanced nuclear technologies offered coordination and publishing services for such a status report provided member countries of the IWGGCR support such activity and are willing to provide contributions about their national activities.

The last status report has been issued in 1990 and described mainly GCR designs under consideration in 1988/1989. In this report emphasis was put on technical design details and safety features. In the meantime program directions have changed in almost all member states. New developments have been initiated, others have been terminated.

In the UK significant progress has been made regarding technical performance and consequently economic figures of the AGRs. In Japan construction of the HTTR test reactor for high temperature applications has started and is proceeding on schedule. Process heat application possibilities are being prepared in an IAEA CRP. In China the decision to build a 10 MW HTR test reactor has been made and construction has started. The HTR program in the US has been modified and is now aiming at the development of a highly economic design of a modular HTGR with an integrated gas turbine. For the development and realization a cooperation agreement has been made with the Russian Federation. In the Netherlands HTR design evaluating activities have been launched within the PINK programme. In Germany, governed by strong antinuclear movements, the HTR program has been terminated, but significant know-how is available and HTR-useful R&D activities are going on.

Altogether, the working group expressed its opinion that the program redirections and the progress achieved in the last years together with very helpful contributions of IAEA within four CRPs are important and should be described in a new status report for distribution to IAEA member states. It was suspected that the new GCR achievements and the developments trends and tendencies are not sufficiently known in other interested countries. However, a next report describing the present status should also make clear that the HTR technology currently remains in a R&D status. Background and reasons for the delay of commercial HTR deployment should be included, the goals of present national strategies and their similarities, i.e. keeping open a very potential option for the future, should be elaborated.

The working group recommended that IAEA should take initiative for the preparation of a next version of a GCR status report. IAEA was willing to prepare an outline of a report for distribution to working group members for review and comments. The finally accepted outline should provide the basis for subsequent contributions of member states. An expanded outline has been prepared. The next step is to develop a first draft based on inputs from Member States. This is anticipated for early 1996.

3.4. Other forms of IAEA support

Several forms of IAEA support are also available for Member States interested in gas-cooled reactors but which do not have major development programmes. Upon official request, technical assistance can be arranged for developing countries for providing expert advice, training, fellowships and special equipment for research. This will assist developing countries to establish the expertise for incorporating advanced gas-cooled reactor technologies into their power generation programmes in the future.

4. Conclusions

Considerable gas-cooled reactor operating experience has been attained through operation of Magnox and AGR reactors, and the basic concept of helium-cooled graphite-moderated HTGRs has been technically proven with the Dragon plant in the UK, the AVR and THTR reactors in Germany and Peach Bottom and Fort St. Vrain in the USA. Construction is well underway on the HTTR engineering test reactor in Japan and completion and operation of the HTTR and its heat utilization system will be major milestones in gas-cooled reactor development and in development of nuclear process heat applications. Construction of a test module is planned to begin in 1994 in China. Further development efforts are on going in several countries including technology development for HTGRs with gas turbines for highly efficient generation of electricity, and future plants are predicted to attain a very high degree of safety through reliance on inherent features and passive systems.

IAEA programmes foster exchange of technical information and encourage cooperative research on gas-cooled reactors. Current IAEA activities focus on safety technology and heat utilization system technology. Especially for advanced reactors with innovative features, international cooperation can play an important role in their development and application.

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