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PROCEEDINGS OF
THE 1ST JAERI SYMPOSIUM ON HTGR TECHNOLOGIES

— DESIGN, LICENSING REQUIREMENTS AND SUPPORTING TECHNOLOGIES —

July 1990

Executive Committee of
The 1st JAERI Symposium on HTGR Technologies

日 本 原 子 力 研 究 所
Japan Atomic Energy Research Institute

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Proceedings of
The 1st JAERI Symposium on HTGR Technologies
- Design, Licensing Requirements and Supporting Technologies -

Executive Committee of
The 1st JAERI Symposium on HTGR Technologies
Japan Atomic Energy Research Institute
Oarai-machi, Higashiibaraki-gun, Ibaraki-ken

(Received June 8, 1990)

The Japan Atomic Energy Research Institute (JAERI) held the 1st JAERI Symposium on HTGR Technologies on March 19 and 20, 1990 at Tokyo Toranomon Pastoral In Central Tokyo, with support of the Atomic Energy Society of Japan on the occasion of the start of the construction of the High Temperature Engineering Test Reactor (HTTR), which is the first high temperature gas-cooled reactor (HTGR) in Japan.

In this symposium, the worldwide present status of research and development(R&D) of the HTGRs and the future perspectives of the HTGRs were discussed with 21 papers including 3 invited lectures, focusing on the basic strategy for development of HTGR, present status of HTGR design, key licensing safety issues and associated R&D of HTGRs.

About 240 participants attended the symposium from Japan, Bangladesh, Federal Republic of Germany, France, Indonesia, Italy, People's Republic of China, Switzerland, United Kingdom, United States of America and IAEA.

This report was edited as the Proceedings of the 1st JAERI Symposium on HTGR Technologies, - Design, Licensing Requirements and Supporting Technologies -, collecting the 21 papers presented in the Symposium.

Keywords: HTGR Technologies, HTGR Design, HTTR, Licensing Requirements, R&D, Licensing Safety Issue, Present Status of HTGR, Future Perspective of HTGR

第1回高温ガス炉技術国際シンポジウム報告
－設計・許認可上の重要項目及び研究開発－

日本原子力研究所
第1回高温ガス炉技術国際シンポジウム実行委員会

（1990年6月8日受理）

日本原子力研究所は，我が国初の高温ガス炉である高温工学試験研究炉（HTTR）の建設が決定されたのを機会に日本原子力学会の後援を得て，1990年3月19日，20日の2日間東京虎の門パストラルにおいて，第1回高温ガス炉技術国際シンポジウムを開催した。

本シンポジウムでは，各国の高温ガス炉開発の基本的計画及び設計の現状，許認可上の重要項目及び関連研究開発を中心に，招待講演3件を含む21件の論文が発表され，各国の高温ガス炉開発の現状と展望について討論された。また，参加者は海外からバングラデッシュ，西独，フランス，インドネシア，イタリア，中国，スイス，英国，米国，IAEAの計9ケ国，1国際機関及び日本からの約240名であった。

本報は，本シンポジウムで発表された計21件の論文を収録し，第1回高温ガス炉技術国際シンポジウムのプロシーディングとしてまとめたものである。

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Proceedings of

The 1st JAERI Symposium on HTGR Technologies

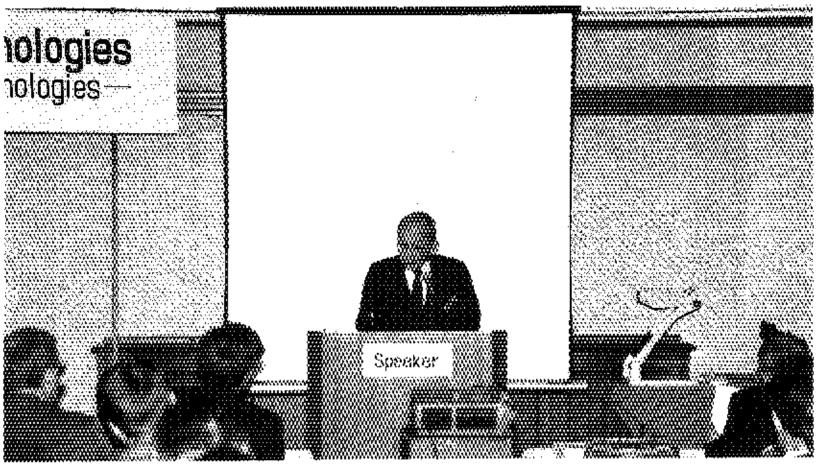
——— Design, Licensing Requirements and Supporting Technologies ———

March 19~20, 1990 at Toranomon Pastoral, Tokyo

Organized by The Japan Atomic Energy Research Institute
Supported by The Atomic Energy Society of Japan

Japan Atomic Energy Research Institute

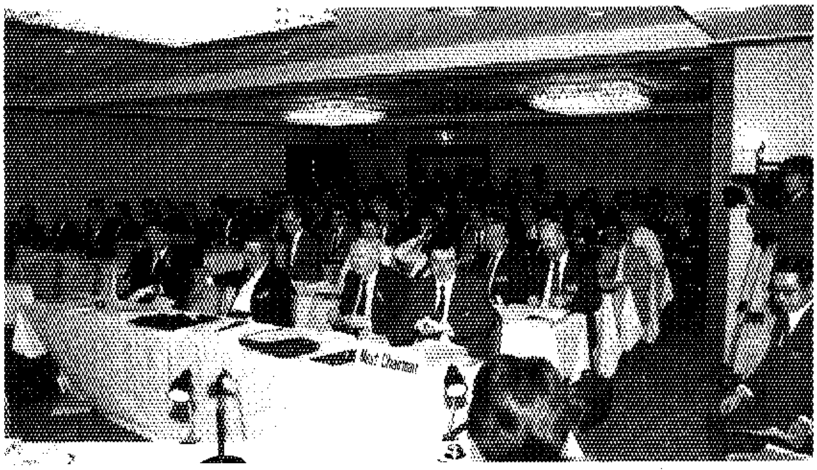
PHOTOGRAPHS OF THE SYMPOSIUM



① Mr. Y. IHARA, President of the JAERI presented opening address for the 1st JAERI Symposium on HTGR Technologies at 9:00 AM on March 19, 1990.

② Prof. T. MUKAIBO, Acting Chairman of Atomic Energy Commission of Japan presented the Keynote lecture on Perspectives on Utilization of Nuclear Energy and HTGR in Japan.

③ Prof. R. SCHULTEN, father of pebble bed HTGR presented an invited lecture on history, salient features and perspectives of HTGR.



④⑤ Many distinguished persons participated in the Symposium.

⑥ Exhibitions on HTGR R&D results such as full scale graphite block, support posts for HTTR were displayed in the room next to the meeting room. During coffee break time, participants discussed earnestly with the expositers on the R&D results.

⑦ Panel discussion was proceeded with distinguished panelists under an excellent chairmanship of Mr. H. MURATA, Advisor to the President of the JAERI.

⑧ Mr. K. SATO, Executive Director of the JAERI presented the closing address on March 20, 1990.

⑨ Welcome banquet was held in the evening of March 19, 1990.

⑩ A technical tour to the HTTR site, Oarai Research Establishment, JAERI was conducted on March 22, 1990.

MEMORIAL MESSAGES from participants

This has been a most successful Symposium, and Congratulations on the commencement of HTTR construction.
Neil Davis
UKAEA

For success of HTTR and HTGRs of the World.
Toyojiro Tachita
JAERI

The symposium has been most successful for exchanging views on both an organizational and personal level.
I wish the HTTR project every success.
John Wilson
AEA Technology
Harwell

For the prosperous future of HTGRs
佐野川好母
K. Sanokawa
JAERI

1) Congratulation for the perfect organization of this meeting!
2) Congratulation for having the ball now in Japan (the HTTR)! Kick it well - we all need to reach our common HTR-goal).
H. W. Hassel
CEC, NUKEM

Great zeal and efforts were shown by all the participants for the success of the Symposium and the promising HTTR project.
Yoshinori Ihara
伊原義徳
JAERI

Thank you very much for kind cooperation. S. Saito

Congratulations!
The Symposium and the overall JAERI Program for HTTR is Outstanding. We wish you success and look forward to increased cooperation.
Dan Mears
GCRA

This Symposium has been very enjoyable - and is attesting to all the progress on HTGR over the last 20 years and more.
Jim Scott
ORNL

Extremely interesting meeting and excellent organization.
A. Lennard

The 1st JAERI Symposium on HTGR Technologies

I hope very much JAERI's HTTR project will progress in good pace according to the programme and open the possible application of nuclear heat in the near future.
Hirumata

Having - or being here, that this 1. JAERI - HTR - Symposium is a real success and will successfully contribute to the promising Japanese approach in the HTR-development.
Louis Pellissier
KSA - PTH

I think the technology is very promising. Hope the prototype will be built quickly and successfully. Will help to keep the...
Wydble.
L. Novillo

With hearty thanks for your contributions to the Symposium
K. Sato, JAERI.
20 March '90

Congratulation for HTTR-decision and Hope for ceaseless HTR cooperation!
Nolst Ilsich, KFA, 20.3.90

Congratulation for the decision to build the new HTTR. D. Schlen.

Many thanks for your kind welcome.
D. BASTIEN. CEA FRANCE

A well organized and valuable Symposium. John Parker UK.

Gratulations and thanks to JAERI for organizing this conference and pushing HTGR development
J. Jehs

Congratulation to your decision to construct the HTTR and this very effective Symposium
J. Kupitz IAEA

祝贺吉高温气冷堆方面取得成就
希望中日两国在高温堆方面合作日益加强。 清华大学核研所 王大中

Congratulation!
I hope based on your experience a HTTR could someday be installed in one of the developing countries.
C. S. KARIM
BAEC
BANGLADESH

Congratulations to your HTTR. Finally another HTR project.
W. A. Simon
General Atomics

CONGRATULATIONS ON THE CONSTRUCTION OF HTTR. BEST WISHES TO JAERI AND CONTRACTORS FOR FULL SUCCESS. I VALUE AND LOOK FORWARD TO CONTINUING AND EXPANDING OUR COOPERATIVE EFFORTS.
A. C. Millunzi
U.S. DEPT. ENERGY

Congratulation to the JAERI scientist and my best wishes for your HTR plant.
F. Jaeger
IGNT. SUISSE

Once again it has been a special pleasure to be in Japan among colleagues. Thank You
H. Jones, ORNL, USA

I am pleased to be able to participate in this landmark conference and look forward to future HTGR interactions. Thank You
Gaye Bramblett GA, USA

Congratulations to JAERI for this very interesting meeting, preparing the future of HTGR
J. F. EVEYRAT
CEA / FRANCE

The meeting was informative and demonstrates a worldwide interest in the unique attributes of HTGR's. William Kerr
Consumers Power Co.
USA.

I am very grateful to be able to attend this symposium. For me it has been very interesting and useful.
Saunders

This symposium is very well planned and organized. It represented one of the most informative and enjoyable forums I have attended
Murrad D. Indonesia

The HTR symposium in Tokyo has elucidated the future direction in nuclear policy and program and the role of HTR in it.
This JAERI symposium is well organized. We are very happy.
I. SUBKI / BATAN.

I like to congratulate JAERI for the decision to construct HTTR. The meeting was for me an exciting experience in Tokyo
E. Arndt

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[Abbreviation of Affiliation]

AESJ	Atomic Energy Society of Japan
BAEC	Bangladesh Atomic Energy Commission, BANGLADESH
BMFT	Bundesministerium fuer Forshung und Technogie, FRG
CEA	Commissariat á l'Energie Atomique, FRANCE
CPC	Consumers Power Corporation, USA
DOE	Department of Energy, USA
GA	General Atomics, USA
GCRA	Gas-Cooled Reactor Associates, USA
HTR GmbH	Gesellschaft fuer Hochtemperaturreaktoren, FRG
IAEA	International Atomic Energy Agency
IGNT	Interessen an der Entwicklung Nuklearer Technologien, SWITZERLAND
INET	Institute of Nuclear Energy Technology, PRC
JAEC	Japan Atomic Energy Commission
JAERI	Japan Atomic Energy Research Institute
JEERI	Japan Energy and Economics Research Institute
KFA	Kernforschngsanlage Juelich GmbH, FRG
LASEN EPFL	Laboratoire de Systémes Energétiques Ecole Polytechnique Fédérale de Lausanne
NAEA	National Atomic Energy Agency, INDONESIA
ORNL	Oak-Ridge National Laboratory, USA
STA	Science and Technology Agency of Japan
TH, Aachen	Technische Hochschule Aachen, FRG
UKAEA	United Kingdom Atomic Energy Authority

PROGRAM OF THE SYMPOSIUM

March 19 (Monday)

8:30~ 9:00		Registration	
9:00~ 9:10		Opening Address	Y. IHARA (JAERI)
9:10~10:40	Session 1	Invited Lectures Chairman: T. IKUTA (JEERI)	
9:10~ 9:25		Perspectives on utilization of nuclear energy and HTGR in Japan	T. MUKAIBO (JAEC)
9:25~ 9:40		International aspects of HTGR development	J. KUPITZ (IAEA)
9:40~10:40		History, salient features and prospectives of HTGR	R. SCHULTEN (TH. Aachen)
10:40~11:00		Coffee Break	
11:00~18:00	Session 2	Basic Strategy for Development of HTGR and Present Status of HTGR Design Chairman: J. KUPITZ	
11:00~11:20		Current status and future plan of HTGR in Japan	K. IDA (STA)
11:20~11:40		Present status and future program of HTGR in the USA	A.C. MILLUNZI (DOE)
11:40~12:00		Utility/User requirements for and assessment of the MHTGR	W.E. KESSLER (CPC), D. MEARS (GCRA)
12:00~12:30		Experience in development and operation of HTGR in Germany and its prospects	E. BALTHESEN (KFA), H. DIEHL (BMFT)

12:30~14:20	Lunch
	Chairman: T. KONDO (JAERI)
14:20~14:40	Present status and development strategy of the HTGR program in PRC D. WANG (Tsinghua Univ.)
14:40~15:00	Present status and future program of HTR-research and development in Switzerland G. SARLOS (IGNT)
15:00~15:20	Technology, cost economics and other factors influencing selection of HTR in a developing country C.S. KARIM (BAEC)
15:20~15:40	Progress on HTR application's study in Indonesia I. SUBKI, M. DJOKOLELONO (NAEA)
15:40~16:00	Study of fission products redeposition by COMEDIE loop in the experimental SILOE reactor J.F. VEYRAT, E. BASTIEN (CEA)
16:00~16:20	Coffee Break
	Chairman: A.C. MILLUNZI
16:20~16:45	Design and safety consideration in Japanese High-Temperature Engineering Test Reactor (HTTR) S. SAITO (JAERI)
16:45~17:10	Modular High-Temperature Gas-Cooled Reactor design G. BRAMBLETT (GA)
17:10~17:35	Design details of HTR-Module and HTR-500 E. ARNDT (HTR GmbH)
18:00	Ajourn
18:30~20:00	Banquet
	Welcome Address T. FUKETA (JAERI) Y. MISHIMA (AESJ)

March 20 (Tuesday)

9:00~14:30	Session 3	Licensing Safety Issues and Associated R&D of HTGR Chairman: H. DIEHL
9:00~ 9:30		Key licensing safety issues and draft review results of HTTR Y. IBE (STA)
9:30~10:00		Licensing overview of the MHTGR A.C. MILLUNZI (DOE)
10:00~10:30		Licensing safety issues and results of HTR-Module safety concept review by independent experts I.A. WEISBRODT (HTR GmbH)
10:30~10:50		Coffee Break
		Chairman: D. WANG
10:50~11:30		Research and development related to licensing of HTTR T. YASUNO (JAERI)
11:30~13:30		Lunch
		Chairman: G. SARLOS
13:30~14:00		Research & development associated with licensing of MHTGR H. JONES (ORNL)
14:00~14:30		Research and development requirements before and beyond licensing HTR-Module N. KIRCH (KFA)
14:30~15:00		Coffee Break
15:00~17:00	Session 4	Panel Discussion - Future perspectives, public acceptance and international cooperation in HTGR - Chairman: H. MURATA (JAERI) Panelists: H. DIEHL (BMFT) A.C. MILLUNZI (DOE) S. AN (Tokai Univ.) K. SANOKAWA (JAERI) J. KUPITZ (IAEA)
17:00~17:10	Closing Address	K. SATO (JAERI)

Foreword

The First JAERI Symposium on HTGR Technologies was held in Tokyo, Japan on March 19-20, 1990, which was sponsored by the Japan Atomic Energy Research Institute (JAERI) with the support of the Atomic Energy Society of Japan in commemoration of the start of construction of the High Temperature Engineering Test Reactor (HTTR), the first HTGR in Japan.

The HTTR, developed by JAERI, is to be a high temperature gas cooled reactor with a power level of 30MW and a helium coolant temperature at the reactor core outlet of 850°C at the rated operation and 950°C at the high temperature test operation. The reactor is to be utilized for establishing and upgrading technology basis necessary for future HTGRs, serving at the same time as a potential tool for basic and innovative researches. The safety review of the HTTR by the government is presently under way and first criticality is expected in 1995.

The symposium was attended by 240 scientist and engineers including 27 participants from abroad from 9 countries and one international agency. We expected attendance of 3 participants from USSR, but unfortunately missed them with some troubles of issuance of visiting visa.

The symposium was addressed to review the national strategy of HTGR development, design, licensing requirements and supporting technologies of HTGR with keynote addresses by three distinguished speakers in Session 1. The presentations on the national strategy of HTGR development were made from 8 countries including developing countries, which provided the participants mutual acknowledgements on present progress and utilization plan of an HTGR. The design details, licensing requirements and supporting technologies were reported from three countries, i.e., Japan, U.S.A. and West Germany which have been strongly promoting the development of HTGR.

The symposium was concluded by a panel discussion on "Future perspectives, public acceptance and international cooperation in HTGR", which could not be included in the proceedings.

It is not possible to aknowledge individually all persons who contributed to the symposium. We highly appreciate kind suggestions, cooperation and assistances by the participants and the persons concerned forwarded directly and indirectly to the great success in the

symposium which was never expected at the early stage of the preparation for the symposium.

We are planning to hold subsequent symposia at 3-year intervals in Japan hereafter. We look forward to seeing successful development of HTGR in each country and also internationally.

Kazuo SATO
Chairman, Organizing Committee

Shinzo SAITO
Chairman, Executive Committee

OPENING ADDRESS

Yoshinori IHARA
President of JAERI

Good Morning, Ladies and Gentlemen,

On behalf of the Japan Atomic Energy Research Institute, I would like to express my greatest pleasure to welcome you all to the 1st JAERI symposium on HTGR technologies.

We JAERI, with the support of the Atomic Energy Society of Japan, have organized this symposium commemorating the commencement of construction of the High Temperature Engineering Test Reactor, HTTR, which is the first High Temperature Gas Cooled Reactor to be built in Japan.

In this symposium, we would expect as many as 30 foreign participants from 10 countries and I international organization.

We have carried out the design study on the HTGR along with the associated research and development for the past 20 years. In February 1989, the JAERI submitted safety analysis report of the HTTR to the government, and the safety review is underway smoothly.

We are planning to organize this type of symposium in Japan with three year intervals, keeping steps with the progress of our HTTR project in order to discuss the state-of-the-art HTGR technologies from the worldwide viewpoint. In this first symposium, we have organized several sessions including basic strategy and present design status of HTGR, safety issues and associated R&D for the licensing of HTGR and three invited lectures. Furthermore, the panel discussion on future perspective, public acceptance and international cooperation for HTGR is also organized in order to make this symposium most fruitful.

With respect to the research and development activities on nuclear energy, the JAERI is now performing 6 major research and development subjects, namely, Nuclear Safety, High Temperature Gas-Cooled Reactor, Nuclear Fusion, Radiation Applications, Nuclear Ships and Related Fundamental Studies. The research and development on HTGR in the JAERI started in late 1960s, aiming at the broader use of nuclear energy including the nuclear heat supply.

Now, as you might be well aware, worldwide anti-nuclear mood has been prevailing these days. The situation is almost the same here in

Japan. However, the problem of future shortage of energy resources has not been solved yet and the global greenhouse effect by burning fossil fuels and forests has become a new problem now.

Under these circumstances, responsibility of those engaged in nuclear energy is more important than ever. This problem could only be solved by the worldwide cooperative works among us.

The symposium was organized under these background. Needless to say, the HTGR gives superior performances, such as high inherent safety, high temperature heat supply, high thermal efficiency and high fuel burn up.

Although, the HTGR has not yet been the main stream for energy supply, it is our responsibility to make every effort to have these attractive technologies of HTGR well developed in the future.

Now, spring has come here in Japan, and the cherry blossoms are in the bloom. It is just a coincidence that this first symposium held in the season of fresh natural lives contributes for HTGR technologies to be in full bloom in the future and will become an important milestone for the history of HTGRs of the world.

In concluding my address, I sincerely hope that your kind and positive cooperation in the 1st JAERI Symposium on HTGR Technologies will give enough lead to the development of the world HTGRs.

Thank you for listening.

1. Invited Lectures

1.1 PERSPECTIVES ON UTILIZATION OF NUCLEAR ENERGY AND HTGR IN JAPAN

Takashi MUKAIBO
Atomic Energy Commission of Japan
JAPAN

As a country with almost no energy resources, it is expected in Japan that nuclear energy will bear an important role in coming years for the stable supply of energy.

The generation of energy by nuclear fission has become an industry of a considerably large scale. In Japan, the nuclear power is generating about one third of electricity and about ten times more around the world.

However, there are still a lot of technologies to be developed for the nuclear power system.

The improvement of efficiency of the conversion of heat to electricity is one of them.

As a kind of heat engine, it is at least worth-while to raise the operating temperature to improve the efficiency of the use of heat. The development of HTGR is along this direction.

Following the countries which started nuclear power programs earlier, Japan started R&D on HTGR in 1969. At that time, it was aimed at using the high temperature for the reduction of iron ore. The R&D on HTGR itself was undertaken by JAERI and the steel industries organized a R&D group for the handling and use of gas of temperatures around 1,000°C.

After a couple years of study, it was concluded that the reduction of iron ore by high temperature gas was not realistic mainly from economical view point and the research of steel industries mentioned above was abandoned. However, at JAERI, they continued such basic and engineering studies on HTGR as those on nuclear fuel and other materials, instrumentation; physical and other studies related to the design of the nuclear reactor, with some reduction of speed of development.

In recent years the slow down of the development of HTGR abroad was reported but there are still expectations on HTGR for the supply of heat as well as electricity in many countries. On the other hand, the cogeneration system of supplying heat and electricity starting from high temperature gas has been developing in recent years in and outside of the country.

Thus, in the latest long term plan of developing nuclear power in Japan in 1987, it was suggested to construct a high temperature gas cooled engineering test reactor for the purpose of accelerating the HTGR project, utilizing the knowledges and data so far obtained in the JAERI project.

JAERI is going to start its construction this year. The details of the reactor and the R&D programs will be described by other speakers in this symposium.

1.2 INTERNATIONAL ASPECTS OF HTGR DEVELOPMENT

Juergen KUPITZ
International Atomic Energy Agency
AUSTRIA

For about 30 years the HTGR has been under development in several countries, such as the Federal Republic of Germany, France, Japan, Switzerland, USSR, UK and USA. The technical feasibility and the advantages of this reactor line have not only been demonstrated by experimental results but also by the successful operation of a number of experimental and demonstration reactors.

While HTGRs have not yet been commercially deployed on a large scale, their potential to provide, besides electricity, high temperature process steam and process heat for various industrial applications has been, together with high safety margins, the continuous incentive for further development. During the last decade particular attention has been paid to HTGR module type reactors, which have emerged in the Federal Republic of Germany, USA and USSR. The specific advantages of this concept have contributed to a strong, renewed international interest in the HTGR for electricity generation and process steam production. The successful construction and operation of the HTTR in Japan will provide for important data and experience still necessary for the HTGR to produce high temperature nuclear process heat.

I. INTRODUCTION

The world's population has nearly doubled during the past third of the century and will continue to increase. Today's estimations forecast that at the end of this century about 6.2 billion people will live on this world and by 2020 even 7.8 billion. Besides many urgent needs, such as food, clothes and work, all people will need energy as an important prerequisite for socio-economic development in all parts of the world. Dependent on the geographical location and the level of industrialization, final energy is consumed in different forms: electricity, transport and in form of heat. It is important to note that in industrialized countries the major part of the primary energy is consumed in form of heat, i.e. district heat for hot water supply, space heating, process steam and process heat for various industrial complexes. Among many different energy supply options that countries may choose in order to meet their energy requirements the High Temperature Gas-cooled Reactor is one of the few options that provide a unique tool to produce energy in all forms, i.e. in form of electricity, low temperature district heat, process steam and high temperature process heat under environmentally acceptable conditions. Thus the development of the HTGR is an indispensable option for long-term energy supply.

II. HTGR-DEVELOPMENT

Technological developments for the High Temperature Gas-cooled Reactor (HTGR) are currently being carried out on a large scale, i.e. with programmes aiming at the construction of an HTGR plant, in Japan, the United States, the

Federal Republic of Germany and the USSR. Further HTGR specific work is being carried out in Switzerland, China and Poland.

Since about 10 years, the International Atomic Energy Agency has been sponsoring the International Working Group on Gas-cooled Reactors, in which other countries (Belgium, UK, France, Italy, Poland and Sweden) follow the development of HTGR technology in order to investigate the possible benefit for their national energy supply.

The HTGR concept has evolved from the carbon-dioxide-cooled and graphite moderated magnox reactors, that were built in the UK, France, Japan, Italy and Spain from the 1950s onwards. They were followed by the advanced gas-cooled reactors (AGR), which have been only constructed and operated in the UK. Experience from over 1000 reactor years of operation of these CO_2-cooled reactors comprises a very valuable database for the ongoing technology development and design of high temperature reactors.

The replacement of carbon dioxide by helium as coolant and the change to coated particle fuel embedded in graphite spheres or blocks without any metal cladding led to the design of the HTGR. The main objective of developing the HTGR has been to provide a nuclear heat source capable of producing steam conditions that are comparable to those attained in modern fossil-fixed plants and to provide in the medium term a tool for production of high temperature process heat for various industrial applications.

The long-term incentives for the development of the HTGR is in most countries still today its potential application as a heat source for various industrial purposes. Besides high efficient electricity generation even in arid areas and production of district heat and high quality process steam the HTGR is also capable to convert fossil fuels, such as coal, into other, clean energy carriers, such as methane and hydrogen, which do not pollute the environment and are convenient to use.

Dependent on national energy resources, strategies for energy utilization and prospectives for future energy consumptions the HTGR has been developed in different countries for various applications. Considering the long-term nature of the potential applications and the expected economic and environmental benefits, the development programmes in all countries are being carried out with substantial financial support.

III. MAIN DESIGN FEATURES OF HTGRS

HTGRs are characterized by the following main design features:

1. graphite moderator and reflector
2. helium as coolant
3. coated particle fuel
4. low-core power density

1. Graphite moderator and reflector

The core and reflector structure is composed of graphite, a material that has a high heat capacity and that sublimes at about 3600° C and retains adequate structural strength to above 2500°C. This contributes to large safety margins between normal operating temperatures and damage limits.

2. Helium as coolant

The primary coolant of the HTGR is helium, a non-condensable, chemically inert gas that does not contribute to or affect the nuclear chain reaction. Helium has low stored-energy content and remains in the gaseous phase under all conceivable operating conditions.

3. Coated particle fuel

HTGR fuel is not contained in fuel pins cladded by metal as is LWR and LMFBR fuel but is contained in fuel particles. The particles are about 0.2-0.6 mm in size and consist of a mixture of oxide or carbide uranium or thorium or uranium/thorium. For fission product retention one particle is enclosed by several coatings of ceramic material having high-temperature stability. The particles are homogeneously dispersed in a graphite matrix that is subsequently compressed to spherical fuel elements or, in the form of rods, filled into fuel channels of a multi-hole graphite block.

These particles remain intact and virtually retain fission products up to about 2000° C. They do not melt at a given temperature threshold but fail gradually under accident conditions, so a sudden release of fission products cannot occur.

4. Low core power density

The power density of an HTGR is about one order of magnitude less than that of an LWR and contributes in a major way to the high inherent safety of this type of reactor. Together with the high heat capacity of graphite in the core and reflector (the THTR-300 contains about 700 tons of graphite), it is ensured that reactor temperature transients in response to disturbances proceed very gradually. The slow thermal response provides for a forgiving reactor since the behaviour of the system is more predictable and more time is available to prevent transients from progressing into major accidents. Time is available to adjust the system or to take other corrective action.

IV. HTGR-PROGRAMMES

1. United States

The HTGR development in the USA has been a co-operative effort between the nuclear industry and utilities with governmental participation and co-funding. The HTGR programme started in the fifties and led to the construction and successful operation of the 40 MWe Peach Bottom Unit no.1 between 1967 and 1974 /Table 1/. In 1974, the 330 MWe Fort St. Vrain Nuclear Power Plant was put into operation. Fort St. Vrain experience has been mixed with excellent fuel performance and almost insignificant personnel radiation exposure, in contrast to the unique design of the helium circulators and their water-lubricated bearing system, which caused a reliability problem. High operating and fuel costs associated with a one-of-a-kind plant and cracks in the steam generators have resulted into the decision to shut down the plant in August 1989.

Following the Fort St. Vrain deployment, 10 large commercial HTGRs were ordered in the USA between 1971-74. However, these orders were terminated in the mid 1970s, due to the economic recession following the oil crisis. This recession led also to the suspension of other nuclear power projects.

In the following time, no new orders for nuclear plants were placed in the US. Later, in the wake of the TMI accident, a consensus emerged that a new approach was needed to overcome the concerns regarding nuclear power in tne USA. The nuclear power industry worked out general criteria and requirements for next generation reactors which included high safety potential for enhanced public confidence and investors' risk protection, improved economics and an easier licensing procedure. This consensus and the general trend in the US, to go to smaller reactor size units, led to a rigorous evaluation of HTGR design options, which than resulted into the selection of the Modular High Temperature Reactor. The MHTGR has become the reference concept in the US for the ongoing DOE/Industry design development programme, which is at present being carried out by the reactor companies, GA-Technologies, Combustion Engineering, the architect engineer Bechtel, Stone and Webster, the research centre ORNL and the utility group Gas-cooled Reactor Associates.

The MHTGR has a power rating of about 350 MWth, prismatic fuel, side by side arrangement of steel vessels for the annular reactor core and the helically-coiled steam generator which are connected by a coaxial cross duct /Table 2/. The MHTGR high safety potential is characterized by the incorporation of passive safety systems, such as the reactor cavity cooling system as a passive heat removal system with air as the ultimate heat sink.

The MHTGR design and technology development is in the preliminary design phase which is accompanied by licensing activities. The reference MHTGR plant configuration consists of four modules with a net output of 540 MWe. Recent very favourable comments on the MHTGR concept in the NRC Safety Evaluation Report have concluded that the HTGR offers qualitatively an overall enhancement of safety and that it meets the safety enhancement objective encouraged in the NRC Advanced Reactor Policy Statement.

A new push for the MHTGR in the US is expected from the plans of the government to construct a four reactor MHTGR plant for tritium production reactor at the Idaho Nuclear Engineering Laboratory (MHTGR-NPR). Since the MHTGR-NPR is a close variant of the civilian HTGR, results of technology development programmes may be applicable to both reactor designs.

Currently, a lead project feasibility study is being carried out by the prospective MHTGR vendor team of Bechtel, General Atomics and Siemens under the leadership of Consumers Power Company, along with additional support from Philadelphia Electric and GCRA. The study has developed a cost/risk sharing framework for the lead project deployment and a one-step licensing strategy. Renewed interest in the HTGR from utilities, vendors and licensing authorities indicate that this concept may have favourable prospects, which also led the DOE to significantly increase its funding level.

2. Federal Republic of Germany

The HTGR programme in the Federal Republic of Germany started in the mid fifties. A major milestone has been the successful construction between 1960-67 and 21 years of operation of the 15 MWe AVR experimental reactor in Jülich, which has provided confidence and experience in the pebble bed HTGR concept. The AVR has served as a large scale test bed for spherical fuel elements, intensiv studies for fission product behaviour and, in particular, the AVR demonstrated the high temperature capability of the HTGR along with its high safety potential.

Based on the AVR experience a 300 MWe pebble bed demonstration power plant, the THTR-300 was designed in the late 1960s. Construction began near Hamm in 1972 and the plant was handed over to the utilities in June 1987. Considerable delays and related cost increases were encountered due to requests for changes during the licensing process, which so far had been mainly adjusted to the Light Water Reactor. The THTR-300 has successfully demonstrated the HTGR pebble bed reactor concept with a prestressed concrete reactor pressure vessel. Since costs for construction, operation and decommissioning of a first-of-a-kind machine are always associated with a certain risk, a risk sharing contract was established between the Federal Government, the Government of the State North-Rhine Westfalia, and Hochtemperatur-Kernkraftwerk GmbH (HKG) as the owner and operating utility. Unsuccessful discussions about an increase of the funding limit to cover expected increase in decommissioning costs, additional losses due to increased licensing requirements and the possible interruption of the fuel supply for the plant led to the announcement by HKG that the THTR-300 would remain shutdown and plans for decommissioning would be initiated.

The HTGR programme in the Federal Republic of Germany has always been part of the energy programme with strong support by the government. The overall incentive to achieve high efficiency electricity generation with the possibility of cogenerating high quality process steam and to get a source for high temperature process heat has led to an HTGR design evolution process, from which two principal design concepts have emerged:

1. HTR-500 (550 MWe, PCRV) based on THTR-300 technology

   The HTR-500 developed by ABB/HRB was evaluated in 1983-84 by a utility group with the positiv conclusions that the concept was technical feasible, licensable and competitive in power costs with the much larger LWRs in the Federal Republic of Germany.

2. The HTR-module (80 MWe, steel vessel) as the first small modular reactor concept.

   The HTR Module developed by Siemens/Interatom was initially developed for industrial process heat applications, but is now also planned for electricty generation and, if requested, for cogeneration of process steam.

   In support of the intended achievement of a high temperature nuclear process heat source, several large test facilities were constructed and operated specifically designed to enhance coal gasification technologies and to test components for nuclear heat transfer. However, with regard to present low costs of fossil fuel, these investigations are being continued with less urgency than directly after the oil crisis 1973 when a special nuclear process heat project was established between nuclear and coal industries and the national research centre KFA-Jülich.

   In 1989, the German capabilities for HTGR design and technology development have been consolidated by the establishment of the HTR-GmbH, a company with partners from ABB/HRB and Siemens/Interatom. This new company is now co-ordinating the further development of HTGR designs as well as their international marketing.

3. Japan

Japan is a country with only very minor own fossil fuel resources and thus strongly depends on imports. The need for a timely development of nuclear power as an alternative energy source has always been emphasized as an important task to assure the country's long term energy supply. Taking into account that industries in Japan require large quantities of high temperature process heat and that future needs for coal gasification and thermochemical water splitting would need temperatures of about 950°C, Japan has been developing the HTGR.

The Japanese HTGR programme started in the late 1960s with preparatory design and research and development work for an experimental very high-temperature reactor (VHTR) performed by JAERI and supported by the Science and Technology Agency (STA). Applications studies of the HTGR for nuclear steelmaking were promoted by the Engineering Research Association of Nuclear Steelmaking (ERANS), a consortium of industrial companies supported by the Ministry of International Trade and Industry (MITI).

The study resulted into an outline for a pilot plant project for Nuclear Steel making connected to a 50 MWth heat source, the Very High Temperature Reactor (VHTR). However, depressions in the heavy industries and low costs of fossil fuel led to the plans to delay the actual application of nuclear steelmaking. In place of the VHTR project the High Temperature Engineering Test Reactor (HTTR) with a power output of 30 MWth was recommended and construction is commencing this year with first criticality in 1995.

The HTTR in combination with other major test facilities, such as the HENDEL loop will provide for the fundamental technology data basis necessary for the design of a later commercial high temperature nuclear process heat source capable to produce energy for a broad range of applications. The HTTR is currently the only construction project and is thus attracting significant international attention.

4. USSR

In the USSR an extensive HTGR programme was launched in the 1970s with the goal to produce process steam and heat for various industrial purposes. Large test facilities have been constructed mainly at the Kurchatov Institute in Moscow and several other institutes in Moscow, Charkov and Leningrad to perform experiments for gas dynamic studies, heat exchange, neutron physics, fuel testing and material qualification. The interest in the HTGR as passively safe reactor concept was intensified after the Chernobyl accident in 1986. Development and design work is centered around two reactor designs both using pebble bed cores. One is a small module type (VTR-M) with 200 MWth and the other is the VG-400, rated at 1060 MWth, an integrated design with PCRV. With the choice of the pebble bed fuel, close interaction were established with the Federal Republic of Germany in the framework of an agreement signed in 1988. Within this agreement discussions are taking place about the joint development and construction of the VG-M in Dimitrovgrad, USSR. The VG-M is planned to provide some HTGR experience and to be utilized as a test-facility for major components of the VG-400, which is in opposition to other HTGR countries, which are mostly advocating the module concept, supposed to become the main line in the USSR with regard to its substantial energy needs.

5. Switzerland

Switzerland has had a long history of involvement in HTGR technology. Already in the 1960s several Swiss industrial companies and the Paul Scherrer Institute participated in development programmes and provided engineering and components for several gas-cooled reactor projects. The Swiss HTGR activities have always been supported by the Government with the overall goal to provide the industry with a chance for supplying components for later construction projects. In this way considerable HTGR-specific know-how has been obtained.

In 1979, the IGNT (Schweizerische Interessengemeinschaft zur Wahrnehmung gemeinsamer Interessen an der Entwicklung Nuklearer Technologien) was established with partners from the Swiss nuclear components manufacturing industry, engineering companies and the research center PSI. The IGNT has a co-ordinating function for future HTGR work in Switzerland.

During the last years an interest emerged in small district heat reactors, which led to the design concept of the GHR-10. This HTGR is a 10 MWth heat-only reactor, which is based on pebble bed fuel and integrated into a PCRV. Its development is being carried out in a co-operative effort between Switzerland and the Federal Republic of Germany.

Currently, Switzerland is establishing an international project for reactor physics calculation for low enriched HTGRs at PSI. The overall objectives of this project are collection and review of data and validation of computer codes for improvement of reactor physics calculations and a further reduction of HTGR licensing uncertainties.

6. China

The People's Republic of China is a developing country, which has an urgent need for energy in all forms. China has a strong commitment to the LWR but has also expressed its interest in the modular HTGR for different applications including

- Electricity supply for remotely located areas
- Steam supply for chemical industries
- Heavy oil recovery

Since about 20 years, China has been performing own research and development activities covering mainly helium technology, PCRV experiments and coated particle fuel development. Work is being carried out at the Institute of Nuclear Energy Technology (INET) under the Ministry of Education, at the Beijing Institute for Nuclear Engineering and the South West Center for Reactor Engineering Research and Design in Chengdu, both under the new Ministry of Energy. Currently a feasibility study is being performed, which investigates the possibility of building a first demonstration plant in Chongqing, China's third largest city with a population of about 14 million people. This study is sponsored by the International Atomic Energy Agency.

The Chinese Government has emphasized the importance of HTGRs for long term energy supply and has included HTGRs into its High Technology Programme. Plans are also being discussed to build a small HTGR on the site of INET for experimental purposes.

V. HTGR - FEASIBILITY STUDIES

In order to study the technical and economic benefits and constraints associated with the introduction of HTGRs, several countries have performed feasibility studies. Within the framework of such studies, several topics are to be addressed including energy consumption and prediction of future energy utilization, economic analysis of different energy supply options, review of industrial capabilities for domestic participation, review of manpower availability, etc. Some studies have been supported by the IAEA within its Technical Assistance Project in form of expert missions, arrangement of fellowships and scientific visits.

Recent feasibility studies with IAEA assistance have been carried out for China, Egypt and the Republic of Korea. Further countries, such as Bangladesh, Indonesia and Turkey, have performed their studies on a bilateral basis with the vendor country.

The fact that HTGR feasibility studies are being performed indicates the international interest in this concept. The HTGR specific safety features, its broad application potential for electricity and heat production and the option for substantial domestic participation have been the main incentives of these countries to study the potential HTGR deployment.

VI. IAEA PROGRAMME ON HIGH TEMPERATURE REACTORS

The programme of the IAEA in advanced nuclear power technology promotes technical information exchange between Member States with major development programmes, offers assistance to Member States with interest in exploratory or research programmes, and publishes reports available to all Member States interested in the current status of development. For countries with active major HTGR programmes (e.g., participating in design, construction or operation of an HTGR), the IAEA programmes are co-ordinated by the International Working Group on Gas-cooled Reactors (IWGGCR). The IWG meets periodically to review the national programmes of its member countries, and to advise the IAEA on its technical programmes and activities in the field. This regular review is conducted in an open international forum in which current progress, problems and operating experience can be frankly discussed. Accordingly, this is a unique opportunity to share the lessons learned, since it brings together programme experts on a truly global basis, East-West and North-South.

For nuclear power to regain a position, in many countries, of providing an increasing fraction of generating capacity, critical attention is being paid to advanced reactor including the HTGR. Advanced reactors that are being developed for possible application in next generation of nuclear power systems include smaller sizes, simplified, standardized designs, improved economics and safety systems that rely more on natural laws to control the fission process and the decay heat removal.

A unique opportunity exists now to develop international user requirements for advanced nuclear power systems independent of a specific reactor type. The requirements will concern different areas including plant design, economics, performance and safety. The IAEA can play an important role by providing the opportunity for already existing national requirements to be discussed at the international level so that, as far as possible, internationally acceptable systems get developed. User requirements will have to be established in close co-operation with the public which may also contribute to allay many current public concerns on nuclear power.

VII. Summary and Conclusions

Increasing world energy needs demand that options be developed that generate energy in a safe, reliable and economic manner. Despite the past and present emotional controversy concerning its use, nuclear power is the most suitable and most adaptable energy alternative, for both, industrialized and developing countries, which can produce energy under environmentally acceptable conditions. In particular, the HTGR as a unique source for electricity generation, process steam and heat production may make an important contribution to the future world energy supply.

The technical feasibility and the advantages of the HTGR have been successfully demonstrated by experimental and demonstration reactors and today there is a significant interest in this concept. Several technological and institutional issues have still to be addressed for the further development and deployment of the HTGR. A suitable frame for these efforts may here to be provided by vendors, users and by governments. In this content, the successful construction and operation of the HTTR in Japan is a very important milestone for further technology improvements of the HTGR for high temperature process heat production.

Table 1. Main design data of HTGRs-Projects

	Dragon (UK)	Peach Bottom (USA)	AVR (Germany,F.R.)	FSV (USA)	THTR (Germany, F.R.)
Start/end of power generation	1966/1975	1967/1974	1968/1989	1976/1989	1985/1989
Power MWth/MWe	20/-	115/40	46/15	837/330	750/300
Fuel element	Cylinders	Cylinders	Spheres (pebble bed)	Hexagonal blocks	Spheres (pebble bed)
Helium temperature outlet (°C)	750	750	950	785	750
Helium pressure (bar)	20	25	11	48	40
Fuel composition	Thorium, uranium carbides	Thorium, ura nium carbides	Thorium, uranium oxides	Thorium, uranium carbides	Thorium, uranium oxides
Reactor vessel	Steel	Steel	Steel	PCRV	PCRV

9148p/vw/JK

Table 2 - Comparison of HTGR Concepts

Country	Japan	Fed. Rep. of Germany	Fed. Rep. of Germany	USA	USSR	USSR	Switzerland/Fed.Rep.of Germany
Concept	HTTR	HTR-Module	HTR-500	MHTGR	VTR-M	VG-400	GHR10
Thermal Power (MW)	30	200	1390	350	200	1060	10
Net electrical Power (MW)		80	550	139	80	280*	-
Average Core Power Density (MW/m^3)	2.5	3.0	6.6	5.9	3-4	6.9	2
Hot Helium Temperature (°C)	850/950	700	700	686	750/950	750/950	450
Helium Pressure (MPa)	4	6.6	5.5	6.4	5	5	1.5
Helium Flow Direction	down	down	down	down	down	down	down
Steam Temperature (oC)	-	530	530	538	540	535	135
Fuel Element Type	Prism. block	pebbles	pebbles	prim. block	pebbles	pebbles	pebbles
Fuel	LEU	LEU	LEU	LEU	LEU	LEU	LEU
Pressure Vessel Type	Steel	Steel	PCRV	Steel	Steel	PCRV	PCRV

* part of thermal power used for process heat production

1.3 HISTORY, SALIENT FEATURES AND PROSPECTIVES OF HTGR

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Abstract

The High Temperature Reactor belongs to the second generation of the use of nuclear power and presents new possibilities and characteristics: New, simplified safety technologies, the generation of process heat, combination loops with gas and steam turbines and the economical feasibility of small reactors.

During the last decades ceramic fuel elements with coated particles have been developed to applicability. In combination with helium as a coolant they allow temperatures of up to 1000°C while keeping the contamination of the primary loop to a minimum. This opens new areas of application for nuclear power exceeding the mere generation of electricity.

A new technology of nuclear safety has been discovered, which masters the dangers of all unintended changes in reactivity and core heatup due to decay heat. Even corrosion due to air or steam leaking into the primary loop can be avoided sufficiently. A new development of fuel elements with a coating of silicon carbide will avoid all possibilities of corrosion in a simple and transparent way. The objective is a security technology which is based only on the behavior and the properties of the fuel elements. The other components of the power plant should not have any function in the retainment of fission products. Any perceptible impact on the environment due to radioactivity can be ruled out in all cases. The new self driven safety concept simplifies the power plants by rendering certain components obsolete and by lowering the necessary standards for construction. This seems to be important for the acceptance and for the use of this technology in threshold and developing countries.

Nuclear power at high temperatures can help decrease the CO_2 problem because it makes the use of cogeneration more effective and can be used for oil drilling by steam injection. The High Temperature Reactor provides a future technology for the supply of heat for residential and industrial use and for the production of hydrogen or equivalent products from fossile fuels.

Due to the development of the fuel element and the SiC coating as a protection against air ingress into the primary loop the High Temperature Reactor can also be employed in a combination of a gas and steam turbine loop and thus increase its efficiency considerably.

Simplification of nuclear technology, mass production and the increase of efficiency provided by combination plants will improve the economic performance also for units with a rather low power output in the range of 200 to 500 MW_{el}, e.g. by coupling several units.

History, Salient Features and Prospectives of HTGR

The High Temperature reactor is a second generation nuclear system. According to the results of the development that has been performed in the last decades, the following properties and advantages of this type can be stated:/1/

With this system a new, mostly self-driven technology of safety has been created. This means that the fuel element itself is the prevalent component of safety. Due to this concept it will be possible to build simpler and therefore cheaper power plants. Moreover, the function of the reactor and its safety properties become highly transparent – a fact of great importance to the public.

Besides the generation of electricity the high temperatures permit the use of process heat. Particularly, this system is suitable for co-generation, which is important for heating of private houses and for producing energy for the industry. Nuclear process heat can be used to convert fossile energy resources into combustibles and fuels.

Furthermore, according to the state of development today it is visible, that gas turbines can be used in so-called combination loops, i.e. a combination of gas turbines and steam turbines. This leads to high degrees of efficiency which can contribute to lower prices for the generation of electricity.

Introduction and development of modular reactors lead to the profitable use of small reactors, which are important for the future reduction of CO_2 emissions. These small reactors are especially suitable for the infrastructures of threshold- and developing countries and for supplying industrial heat.

Thus, the HTR is a supplement of today's nuclear technology for the generation of electricity in small plants as well as a broadened application on the energy market.

The history of the High Temperature Reactor began at the end oft the fifties. At this time three important prototypes were built: The Peach Bottom Reactor, the AVR and the Dragon. The first and most important objective of this development was to achieve a high degree of efficiency for the generation of electricity, by using high temperatures. A combination of ceramic fuel elements and Helium as a coolant was to avoid any corrosion inside the reactor system.

An important milestone, or even the condition for the realisation of high temperatures, was the invention and development of coated particles, especially the buffer zone inside, which ensures mechanical decoupling of the fuel particle and its coating. The development of the coated particles is a uniquely successful combination of ceramic materials: graphite, which is used as a moderator, pyrolytic carbon and silicon carbide which constitute the coatings and finally the kernel of uranium oxide or carbide. Already in the seventies it became apparent, that with these particles a high burnup of more than 100,000 MWd/t could be reached, that maximum particle temperatures of at least 1250°C were possible and therefore a remarkably low contamination of primary loops of High Temperature Reactors could be acheived. The fuel element concept that was chosen permits the combination of fission and breeding materials, so the thorium fuel cycle could be tested as well as the cycle with low enriched uranium.

At this time the thorium based fuel cycle was investigated with special interest. It was found out, that conversion rates of 0.8 to 0.9 are possible, and therefore great amounts of uranium can be saved. This goal has become less important in the last ten years, since it became obvious, that the uranium recources which can be mined at low costs are big in relation to the demand within the next 50 years. The use of low enriched uranium and direct disposal of burnt out fuel elements will be so profitable in the next decades, that using the thorium cycle will not be neccessary for the time being. Fuel element refabrication is not profitable since the burnup is sufficiently high and the plutonium is used inside the reactor during one cycle.

In the seventies it was possible to increase the layout temperature of the AVR from 850°C to 950°C. Meanwhile the primary loop contamination did not increase, it rather decreased due to the improved quality of the fuel elements. This step proved that nuclear energy from high temperature reactors is also suitable for process heat. During the energy crisis of the early seventies it therefore appeared interesting to develop nuclear process heat in order to convert fossile energy sources, for example coal, into combustibles and fuels. At the time, these processes were looked upon as to be very profitable because of the high prices for crude oil. In the PNP project (Prototype for Nuclear Process heat) two prototypes for the conversion of coal with nuclear energy were built and operated. The materials developed in this project were suitable for helium temperatures of up to 950°C. As everyone knows, prices for crude oil have dropped dramatically since then and therefore the use of nuclear power for coal conversion is presently not profitable. Whithin the next years, however, with respect to the CO_2 problem, this process might well be applied profitably in one way or the other. Conventional processes for the conversion of fuels are by far defeated by nuclear heat, because the former are linked to a prohibitive production of CO_2.

Research and development for nuclear process heat also included the development of the ADAM/EVA sytem for the transport of latent energy over long distances. For this purpose a prototype plant was built and successfully operated. In particular it was demonstrated in this plant that a conversion of methane with water into hydrogen and carbon monoxide could be achieved by nuclear heat. Especially this technique can gain, whithin the next decades, great importance for the generation of fuels from methane, since it is one of the key technologies for the reduction of CO_2 emissions.

Steam injection for enhanced oil recovery was also investigated in this project. The only nuclear system that can provide the steam qualities required is the HTR. This process for the recovery of tertiary oil bears two advantages compared to conventional steam injection. On one side, oil production increases by above 40%, on the other side, environmental problems of the combustion of oil, especially the increased emission of CO_2, are avoided. According to recent results tertiary oil recovery can be applied profitably as soon as the price for crude oil reaches the limit of 20$/barrel.

In the seventies the use of gas turbines with HTR's was analysed in a bigger study. At the time, two important problems appeared: The contamination of the gas turbine which would have complicated its maintenance, and the accidental air ingress into the reactor. At the time these problems were unsolvable, but due to recent developments, this technology seems to be fesable in the near future.

In the seventies and eighties the technology of medium power High Temperature Reactors was demonstrated by the construction and operation of the Fort St.Vrain reactor and the THTR. These systems proved to comply with all physical and technical demands, especially the expected safety properties could be demonstrated. In particular, contamination in these systems showed to be extremely low, like in the small units. As this conference will proceed there will be further reports on this topic. Unfortunately, the long term test phase of this reactor was discontinued – for mainly political reasons, as in the case of the SNR – when, after two and a half years of operation, the reactor had generated three billions of kilowatthours of electricity.

In the last years the german industry developed the concept of a reactor with a power of 500 MW_{el} based on the experience of the construction and operation of the THTR 300. At the same time, the modular system with a thermal power of 200 MW was planned, based on the design of the AVR. Siemens and ABB have founded a common company to present these plants on the future market.

The operating experience of the AVR permits a new safety concept for HTRs, as it is now realized in the Modular HTR. Lately a safety evaluation for the module has been reviewed by experts and the commission for reactor safety, which draws positive conclusions for all normal layout accidents. At the same time, hypothetical accidents were thoroughly investigated in further studies /2/. I would like to present the results of these studies exemplarily in my further explanations.

In the case of the module, the overall amount of releasable reactivity is roughly 8%. It is remarkable and of decisive importance that this addition of reactivity can be compensated by the system's negative temperature coefficient, without exceeding the acceptable maximum accident temperature of 1600°C. Thus, the question for conceivable reactivity accidents could be sufficiently answered with this statement. The investigations, however, also study reactivity accidents connected to core heatup accidents. All conceivable events for full load, part load, for the hot critical reactor without load and, at last, for the cold reactor without load were studied.

All in all it shows, that even with a release of all conceivable amounts of reactivity maximum fuel element temperatures of 1600°C are never reached.

The increase of reactivity in the case of a water and steam ingress has been thoroughly studied. To keep these increases small, the moderation ratio was adapted in such a way that the maximum increase of reactivity is not greater than the one caused by an unintended withdrawal of the absorber rods, wich is mastered by the system. The moderation ratio was chosen to ensure that even if the control rods are not operated in the case of a water and steam ingress, a temperature of 1300°C is not exceeded. The ingress of water and steam into the cold reactor has been studied, too. It has been stated that in this case a maximum temperature of 500°C is reached.

The ample studies wich were performed so far show that under all operational and accidental conditions reactivity accidents are excluded and that even combined with the breakdown of the main heat absorber they cannot lead to undue temperatures. One can therefore state that by reactivity accidents there is no danger of a release of radioactive materials.

The extensive studies concerning fuel element heating experiments wich were performed by Schenk and Nabielek /3/, show that for a representative batch of burnt fuel elements, heated to a temperature of 1600°C the release rates for the most important fission products, namely cesium, strontium and iodine are lower then 10^{-5}. Moreover it has to be stated, that these high temperatures, during core heatup accidents of the module, are reached only in 1% of the fuel elements, and this only during approximately 60h of accident time. These data permit the conclusion that during a core heatup accident of a modular reactor only small amounts of radioactive materials are released, without causing serious damage to the environment. The experience from fuel element heatup studies that have been performed so far also shows, that with an increase of temperature to 1700°C the fission product release within the first 100 hrs is only 3 to 4 times bigger. Only at temperatures of more than 1800°C the release rates for the most important fission products, cesium and strontium, increase substantially. This clarifies that the design with a layout temperature of 1600°C is based on conservative assumptions. Also, for the calculation of heat conductivity in the core during heatup accidents pessimistic data have been assumed.

An extensive study by Fricke, /4/, investigated maximum accident temperatures during core heatups, depending on the power of the reactor (fig. 1).

It becomes evident, that for an increase of reactor power in the range between $200MW_{th}$ and $300MW_{th}$ the increase of maximum temperatures is basically linear. It is remarkable that this relation applies for the inlet / outlet temperature spans between 250°C and 700°C as well as between 300°C and 900°C. An extensive analysis shows that the maximum temperature depends almost only on the power density and the geometry of the reactor. Furthermore, the power peak factor is of great importance. This phenomenon is understandable if it is taken into account that the heat conductivity inside the pebble bed increases considerably with increasing temperature and therefore the removal of decay heat starts earlier and more intensely with higher temperatures.

According to the state of development today an increase of power from $200MW_{th}$ to $250MW_{th}$ seems to be possible as soon as adequate operating experiences are available.

For this transition two factors may be decisive: The complete use of reserves and the use of an AVR-type two-zone refuelling.

The use of silicon carbide as a coating for the spherical fuel elements wich is mentioned below can provide a further reduction of the release of radioactive fission products at higher temperatures. Albeit this contribution can not yet be calculated due to a lack of experience. However, there is a good chance for an increase of the possible power of $200MW_{th}$ by increasing the allowable maximum temperature, if further knowledge on the behavior of coated fuel elements becomes available.

Extensive studies on the effects of an ingress of water, steam or air were neccessary.

The evaluation procedure conducted by experts and the Commission for Reactor Safety, which was completed recently, showed for all cases within the layout of the plant, that considerable damage due to fission product release are ruled out. Furthermore, all conceivable accidents, which surpass the range of layout (hypothetical accidents) were studied in detail. Greater damages to the primary loop with extremely fast depressurization followed by an ingress of air, as well as an ingress of the complete amount of water into the primary loop were studied also. The generation of explosive gases like CO and H_2 was calculated. Conceivable explosions of gases inside the reactor building do not cause any intolerable damages.

All these events which surpass the layout range are alleviated by the fact that they occur at a low speeds and that no considerable release of fission products takes place in the first 20hrs. Interventions inside the most important parts of the reactor building are possible without surpassing the allowable radiation doses, so simple counter-measures can be taken, for example in the case of an air flow through the reactor.

The evaluations by experts and the Commission for Reactor Safety which are mentioned above point out that the module as it is presented today has a matured safety concept, which does not seem to need or be worth further improvement with respect to damages to the environment. Nevertheless the intended protection of the fuel elements with SiC coatings is attractive because of the simplification of the reactor and the transparency of the safety concept.

According to studies conducted by the Ceramtec company a coating of reaction bound silicon carbide with a thickness of approximately .3mm can be applied to the spherical fuel elements using only common commercial methods. Corrosion tests in a surrounding of air have so far been performed at temperatures of up to 1600°C. They show positive results and a remarkable resistance against corrosion. The active mechanism consists of the fact that due to the surplus of silicon inside the layers self-healing layers of silicon dioxide are formed, which will presumably avoid any corrosion with air up to the melting point of silicon dioxide at 1800°C. The resistance to temperature shocks was found to be sufficient, as well (fig.2). The fast cooling of the fuel elements from 1400°C down to normal temperature by applying water did not cause any changes in the appearence of fuel elements and layers after it had been repeated ten times.

According to commercial experiences the strength of the layers is in the range of super alloys up to a temperature of 1400°C (fig.3). Some simple falling experiments show that the usual refuelling technique with a falling hight of 1m can be maintained.

A first irradiation experiment for the full dose and reactor operating temperatures is to be carried out this year. The behavior of graphite and reaction bound silicon carbide is largely known. Thus, this test serves mainly to prove the behavior of the bond between the graphite surface of the fuel element and the reaction bound layer of silicon carbide.

The additional use of fuel elements coated with silicon carbide can be useful for the modular concept in two ways.

The tranparency of the safety concept becomes even better. It is obvious to anyone that corrosion accidents including their effects can be eliminated. Consequently, the fuel element actually is the safety component for all accidents. The complicated evaluations of accident behavior can largely be avoided.

The system will be simplified in some areas, leading to a reduction of building costs. In particular, the complicated gas purification equipment, the machinery for the control of water ingress into the primary loop and some redundancies become obsolete. We hope that this will lead to a substantial simplification of the power plant, so it will be suitable not only for highly qualified industrialized countries, but also easier to build and operate in countries with a lower level of technical development.

This measure also renders possible the use of combination plants with gas and steam turbine loops.

The fuel cycle of the HTR with an enrichment of 7% is not suitable for the production of explosives. The uranium that is produced is denaturated to an extent that makes it useless for arms production. Furthermore it has to be mentioned, that the high portion of Pu-238 would require constant cooling of this material.

Branching off the fuel elements which show extremely low burnup and irradiation requires the removal of at least 200,000 fuel elements during 500 days of full load. This could be easily controlled and never be done by terrorists, simply because of the transport problem /5/.

The refabrication of such elements is complicated by the fact that it leads to the production of a sol-gel, which is well known to be hard to control in refabrication processes.

Furthermore one may mention the success of the international Non - Proliferation Treaty, which has been very effective so far. All nuclear weapons are produced by national organisations, and there is even a great chance for decreasing their numbers or even banning them in the future.

Todays economic calculations for Modular HTRs show that the costs of electricity produced by a module with two reactors are 30% to 40% higher compared to a big light water reactor with $1300MW_{el}$. A power plant with four modules would lead to an increase of 10% to 20%. Double arrays with four modules each and a total power of $640MW_{el}$ can produce electricity at equal costs.

The simplification of the equipment based on the use of reaction bound and sintered SiC coated fuel elements will provide a simplification of plants and a decrease of electricity costs. Estimations foresee that a plant with two modules will only be 10% to 20% more expensive than a big LWR, thus a plant with 4 modules will be competitive.

These analyses and estimates do not take into account the possible reductions of specific costs due to increased power and the use of reserves so far implemented.

Fuel elements with SiC layers also permit using the combined loop with gas and steam turbines, since the turbine breakdown followed by an air ingress can be controlled. The use of the combination loop provides an increase of efficiecy by 1.3. This can be shown based on today's gas-heated combination power plants, which employ a similar process of gas and steam turbines. According to our expectations a power plant with two modules using this process could be competitive and a plant with four modules could produce electricity at lower costs than an LWR.

Mass production of the Modular HTR can lead to a further reduction of costs. Assumed that a total power of 5 to 10 GW is to be realized in a series of modules, 50 to 100 modules would be required. In this case the costs for building and components would be reduced. Presumably, a competitive instrument which is attractive for the use in threshold and developing countries also due to its simplicity, can be developed based on two modules.

The worldwide reduction of CO_2 emissions is the most important task for the future nuclear technology. The HTR, especially the modular HTR, provides important conditions for a solution of this kind.

- The fuel element can fulfil the decisive safety function in these systems, while other components loose their importance with respect to safety. The SiC-coated fuel elements will be the only crucial safety component in these reactors. With this concept it will be possible to build and operate reactors which rule out any damage to the environment.
- This concept provides an understandable and transparent safety technology, which is also easier to understand for non-technicians.
- Simplifications and cost reductions are possible, so electricity can be produced with smaller plants at competitive prices.
- The introduction of combined gas and steam turbines, already established in gas power plants, is possible with the Modular HTR. One can expect that this technology will provide the lowest costs for nuclear electricity production in the next decades.
- At last, it should be remembered that not only the production of electricity, but also the generation of heat for the future market is of great importance for the reduction

of CO_2. The small HTR can contribute especially to this task.

As well as the Module, the HTR 500 is an interesting plant for the future electricity market. To some extent, the safety properties of the Module can be directly transferred to the HTR 500. This applies especially to reactivity accidents including a leakage of steam into the primary circuit. The use of a prestressed concrete vessel provides an extremely safe containment. Extensive studies at KFA have proved that the PCV remains stable under all circumstances, including hypothetical accidents. It has been demonstrated that at the temperatures which are reached during hypothetical core heatups the PCV is heated only at its inner surface and that the rest of the PCV is not influenced by the increased temperatures to a great extent. The biggest leaks which can open in the PCV can be assumed to be smaller than $30cm^2$, corresponding to refuelling and fuel element withdrawal tubes. Therefore any severe effects of corrosion due to an ingress of air into the primary loop after depressurization is ruled out.

In the last years it has also been stated, that the graphite structures inside the HTR have a high capacity for the retainment of almost all fission products, exept iodine. Since the accidents of the HTR 500 proceed at very low speeds, accident control requires only simple means. The release of fission products starts only after 15hrs, so interventions inside the reactor building are possible under all circumstances.

The HTR 500 is designed and planned according to the technology of the THTR. This technology has already been agreed upon under german nuclear laws.

Both concepts, the HTR-Module and the HTR 500 provide an excellent starting position for the future power economy.

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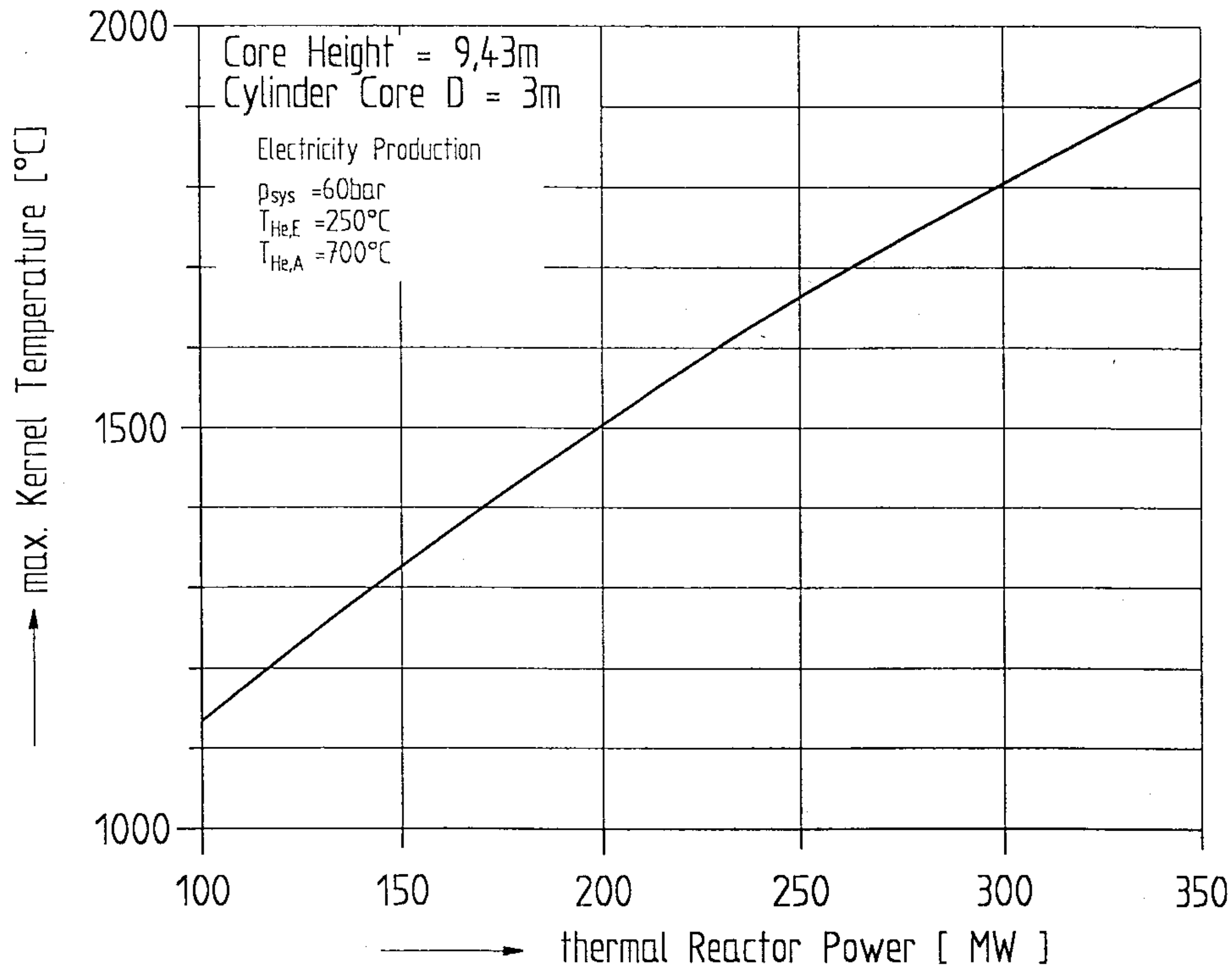

Fig. 1: Maximum Accident Temperature Depending on the Power of the Module

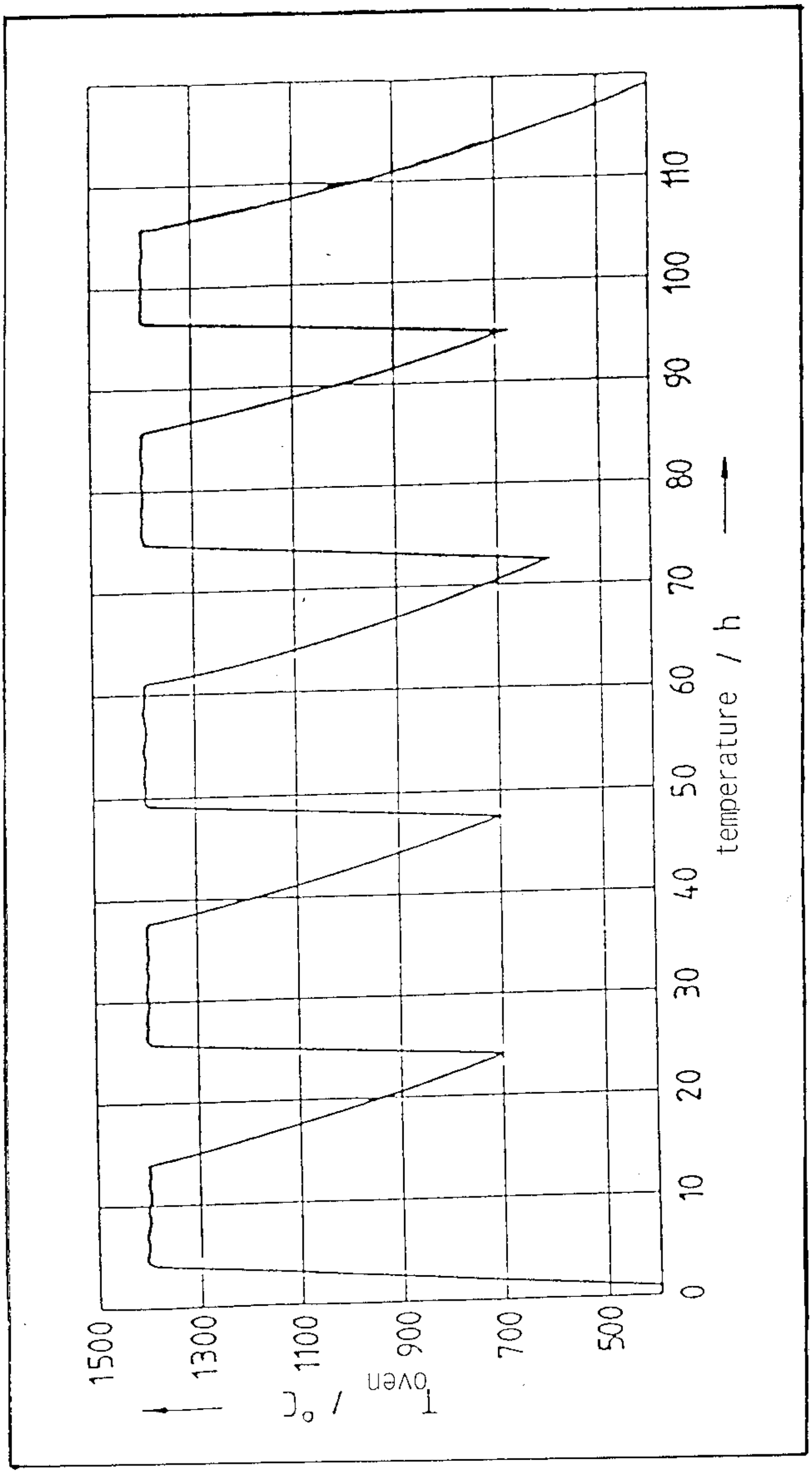

Fig. 2: Temperature-Shock Tests of SiC-Layers

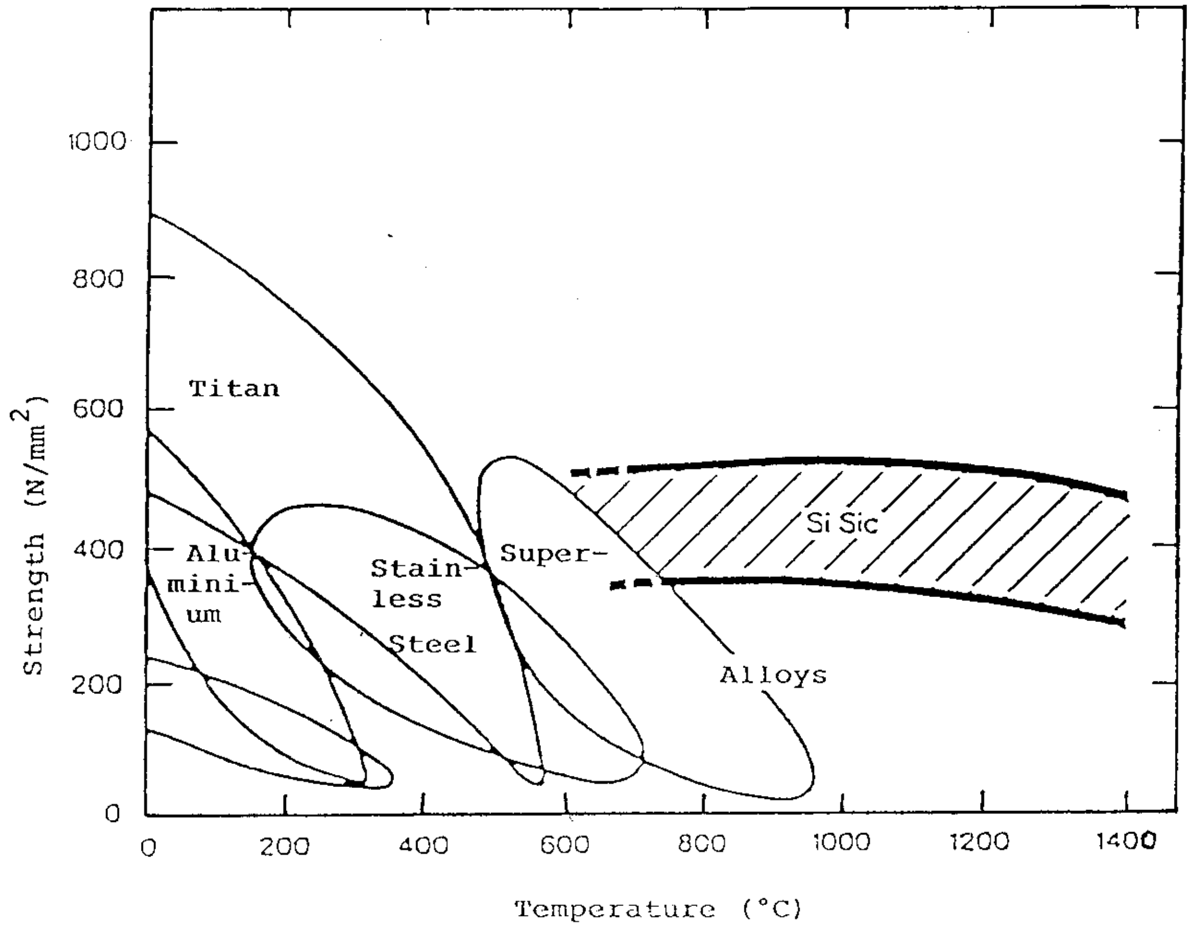

Fig. 3: Strength of Different Materials Depending on Temperature

2. Basic Strategy for Development of HTGR and Present Status of HTGR Design

2.1 CURRENT STATUS AND FUTURE PLAN OF HTGR IN JAPAN

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ABSTRACT

A high Temperature Gas-Cooled Reactor (HTGR) has excellent characteristics such as producing heat at higher temperature, higher inherent safety and higher fuel burnup, and is considered as one of the most promising nuclear reactors to improve the economy and to extend the application of nuclear energy. Therefore, Japan has been involved since 1969 in R&D for a Multipurpose Very High Temperature Gas-Cooled Reactor (VHTR) in collaboration with associated organizations and institutions. In 1987, the Japanese Atomic Energy Commission (JAEC) issued the revision of Long-Term Program for Development and Utilization of Nuclear Energy, recommending that Japan should proceed with the development of more advanced new technologies for the future, in parallel with the existing nuclear system. Then, the Japan Atomic Energy Research Institute (JAERI) decided to construct the High Temperature Engineering Test Reactor (HTTR) instead of the experimental VHTR in order to establish and upgrade the technology basis for an HTGR, serving at the same time as a potential tool for new and innovative basic research.

In order to establish and upgrade HTGR technology basis systematically and efficiently, and also to carry out innovative basic researches on high temperature technologies, Japan will carry out necessary R&D in cooperation with the JAERI, universities, national research institutes as well as industries. In addition, in order to promote the R&D on HTGR more efficiently, Japan will promote the existing international cooperation with the research organizations in foreign countries.

1. Introduction

A high temperature gas-cooled reactor (HTGR) would give superior performance with safety features; it would supply high temperature heat and have high thermal efficiency, inherent safety characteristics and high fuel burn up. The development of HTGR is important for broadening nuclear energy utilization. With these factors as the background, Japan has been involved since 1969 in research and development (R&D) for a Multipurpose Very High Temperature Gas-Cooled Reactor, in collaboration with associated organizations and institutions, that is, the Japan Atomic Energy Research Institute (JAERI) initiated R&D for HTGR in 1969. Japanese Atomic Energy Commission (JAEC) took up the concept of developing a multipurpose HTGR of 1000°C reactor outlet temperature first, in the "Long-Term Program for Development and Utilization of Nuclear Energy" in 1972, with the supports of private industries.

A big project of R&D in the total system of a pilot plant and element technologies had proceeded with the aim of establishing a high temperature process heat utilization system for steel making over 6 years. In parallel with this, the JAERI undertook the development of the reactor system. The project was, however, abandoned, judging that the process heat application could not be realized on a commercial basis in the steel making industrues in the near future.

In 1980, the JAERI began to detailed design for an experimental HTGR and the related research, the need for which was stressed in the long term program in 1978, taking over the existing R&D results. But the circumstances on the development of the experimental HTGR had changed adversely in the following years in terms of the nation's finances, oil prices and demand for process heat, although the technical foundation for building the experimental reactor had been established in the JAERI.

In June 1987, the JAEC issued the revision of the "Long-Term Program for Development and Utilization of Nuclear Energy", recommending that Japan should proceed with the development of more advanced new technologies for the future, in parallel with the existing nuclear system. Table 1 shows the long-term program of nuclear energy and the role of the HTTR in Japan. The long-term program emphasized that the HTGR, which has excellent characteristics such as producing heat of higher temperature, higher inherent safety and higher fuel burnup, is considered as one of the most promising nuclear reactors to improve the economy and to extend the application of nuclear energy. Therefore, it is recognized that the promotion of the R&D on HTGR is significant for the development of nuclear energy in Japan, although the social evaluation of HTGR has changed. It is essentially important for our country, which has limited energy resources, to make efforts to obtain more reliable energy supply by the extended use of high temperature heat from nuclear reactors. Hence, efforts are to be continuously devoted to establish and upgrade HTGR technology basis, keeping up technologies and human resources accumulated so far. It is also expected that making basic researches at high temperature using the HTGR will contribute to the innovative technologies in the future. Then, the construction of the High Temperature Engineering Test Reactor (HTTR) was decided instead of the experimental HTGR. The HTTR aims at establishing and upgrading the technology basis necessary for HTGRs, serving at the same time as a potential tool for new and innovative basic researches.

Figure 1 shows the bird's view of reactor building. The reactor vessel and primary cooling system are contained in the containment vessel which is installed under the ground level.

Figure 2 shows a schematic schedule of the HTTR program. The safety review of the HTTR by the government is presently underway. Construction is scheduled to start in 1990 and first criticality is expected in 1995.

The project of HTTR, which will be the first high temperature gas-cooled reactor in Japan, has been thus identified by the JAEC as a leading project in nuclear energy development.

2. Future Plan and Organization for HTGR Research and Development

The JAERI should take the initiative in performing the R&D program on HTGR in Japan systematically and efficiently in cooperation with universities and national research institutes as well as with industries.

As a leading organization of the R&D on HTGR in Japan, the JAERI is to construct and operate the HTTR, carry out the R&D to establish and upgrade HTGR technology basis, and conduct various innovative basic researches on high temperature technologies such as advanced ceramics and fusion reactor materials. In addition, it is necessary to promote active reception of researchers in the JAERI from industries, universities and national research institutes as well as from foreign organizations, and vice versa. It is also desirable to conduct the R&D on specific subjects flexibly in a joint research organization by researchers both inside and outside of the JAERI.

Universities are to conduct positively basic researches on the HTGR technologies and innovative researches on high temperature technologies.

For example, basic researches on the following items are expected to be carried out using the HTTR : Nuclear physics and engineering tests, development of nuclear instrumentation system at high temperature, researches on high conversion technology employing thorium fuel, high temperature irradiation tests on large sized heat resistant materials, graphites and ceramics in simulated reactor environment, various irradiation tests with installed instruments, radiation chemistry and researches of nuclear heat utilization such as thermochemical hydrogen production.

National research institutes are expected to conduct innovative basic researches mainly in the fields of material science and technology such as heat resistant materials for the HTGR and on high temperature irradiation behavior of ceramics as well as heat resistant materials.

Industries are to evaluate and study the trend of future demand and economics of HTGR from the standpoint of future program of HTGR technologies. They are also expected to upgrade basic technologies of HTGR and to participate the test reactor program to utilize the results of R&D.

A Research Association on HTGR Plant which is a non-governmental organization, was set up in 1985 in order to review and survey on the various aspects of the HTGR plants and to study viability of the HTGR power plant in Japan. The association has discussed on the clarification of the mechanisms of the inherent characteristics of the passive safety and on the deployment of a scenario on introduction of HTGR plant in Japan.

3. Role of HTTR in R&D for advanced HTGRs

The R&D subjects for the future HTGRs are described below, which are summarized in Table 2.

(1) Establishment of technology basis on HTGR

In order to establish the technology basis on HTGR, the R&D subjects which should be performed through the design, construction and operation of the HTTR are as follows:

a) Characteristics of reactor system

Confirmation of the fluid dynamic characteristics in the reactor vessel and the helium gas leakage characteristics from the system.

b) Nuclear and thermal-hydraulic characteristics of the core

Confirmation of the nuclear and thermal-hydraulic characteristics through the criticality test, start-up test and normal operation.

c) Plant dynamics, operation and control
Confirmation of the plant dynamics and plant control characteristics of parallel loading system of the intermediate helium/helium heat exchanger and pressurized water cooler, systematically.

d) Overall performance of high temperature components
Confirmation of the overall performances of the high temperature material and high temperature components such as gas circulators, valves, pipings, etc.

e) FP release behavior and related items
Evaluation of the FP release behavior such FP release from fuel, FP plate out, shielding performance, etc.

f) Operation and maintenance of the plant and long-term continuous operation
Accomplishment of the long-term continuous operation and accumulation experience of the plant operation and maintenance through the operation, mainenance and repair.

(2) Upgrading present HTGR technologies

The R&D subjects for upgrading present HTGR technologies which improve HTGR performance and its economy are upgrading of basic technologies and core performance, establishment of high temperature nuclear heat application technology, improvement of thermal efficiency and achievement of passive safety (core cooling only with passive components) technology. The HTTR should be utilized for these R&D.

1) Basic technologies

As for coated particle fuel, it is important to improve FP retaining capability under long term high temperature operation such as high power density and high burn up. By the irradiation experiments using the OGL-1 and other test facilities, strength and chemical stability under high irradiation dose and at high temperature over 1400°C are to be clarified.

As for graphite materials and graphite/carbon composites used for fuel elements and reflectors, their higher strength and dimensional stability at high temperature and under high irradiation dose are of importance. Irradiation experiments on several types of graphite materials using the HTTR are to be conducted to obtain chemical and physical properties, strength, corrosion properties, etc.

As for heat resistant materials used for liner materials for high temperature pipings and heat transfer tubes of helium/helium intermediate heat exchangers (IHXs), materials and structural tests on modified Ni base alloys, Ni-Cr-W super alloys, etc. are to be conducted. As for high temperature components such as IHXs made of these developed materials, it is necessary to verify the structural integrity by overall functional tests using the Helium Engineering Demonstration Loop (HENDEL).

2) Core performance

As for improvement of core performance, reactor core exit coolant temperature 950°C, core power density from 5 to 10W/cm^3, fuel burn up about 100GWD/T are expected as targets of an advanced core of the HTTR. The advances of the HTTR which is made of fuel elements, graphite components and control rods developed by basic technology test, is to be tested to clarify and improve its performance.

(3) High temperature nuclear heat application technology and improvement of thermal efficiency

It is important and necessary to establish high temperature nuclear heat process application technology through the R&D. Recently, the greenhouse effect and climate change on the earth has become real concerns internationally. Since the contribution of fossil fuels and felling of wood are the major source of CO_2, the utilization of nuclear heat should help to solve the problems an potential alternative of energy source.

In addition to the thermo-chemical hydrogen production, steam reforming of methane and high temperature electrolysis of steam, the Iodine-Sulfur (IS) process has been studied for the thermo-chemical hydrogen production. The bench scale apparatus for the IS process has been installed in the JAERI for cycle demonstration of the process and is under test operations. A fundamental study on material selection of solid electrolytes and ceramics electrodes for application to the electrolysis of steam has been carried out by measurements of electrical conductivity and phase stability.

Pilot plant tests of heat utilization system using developed material and components is to be conducted. Then, the verification test by supplying high temperature heat of about 900°C is to be performed by connecting the system to the HTTR through the IHX in near future.

(4) Innovative basic researches on high temperature technologies

1) R&D on nuclear fusion reactor

a. As for blanket structural materials of nuclear fusion reactors, irradiation test on integrity of large welded structure is to be conducted at high temperature.

b. Tests on continuous recovery of released tritium are to be conducted by irradiating tritium breeding target, Li_2O, $LiAlO_2$ and etc., at high temperature.

2) R&D on radiation chemisty

Basic researches on thermal decomposition o plastics, pitch, tar, etc. under irradiation are to be carried out.

3) Others

Various basic researches on high temperature technologies such as high temperature irradiation tests of large sized specimens with installed instruments, are to be carried out, using the HTTR which has irradiation regions of 10 to 30cm in diameter, at temperature from 400 to 1100°C.

4. Prospects of international cooperation

In order to promote the R&D on HTGR more efficiently, Japan has international cooperation with the research organizations mainly in Federal Republic of Germany (FRG), United States of America (USA), People's Republic of China (PRC) and IAEA as shown in Fig. 3.

Since 1979, the collaboration between the JAERI and the Juelich Nuclear Research Center (KFA) in the FRG has been carried out in the area of fuel, graphite, metals, instrumentation, components and safety. In 1988, the KFA-JAERI agreement has been revised to include joint experiments at the AVR reactor. Since May 1988, the JAERI has participated in the safety related experiments of the AVR investigation program such as LOCA simulation tests.

The information exchange has been carried out on the HENDEL and the KVK of GHT/Interatom since 1984, and on the HTTR and the HTR-Module of Interatom since 1988.

The JAERI has also executed the R&D on HTGR in cooperation with the U.S. Department of Energy since 1985. Joint experiments on fission chamber and fuel were started in March 1986. Collaborative testing of graphite materials and metals were started in May 1987. Information exchange on the MHTGR and the HTTR was started in September 1987.

The JAERI and the Institute of Nuclear Energy Technology (INET) in the PRC have exchanged general informations since 1986.

Japan has also contributed to the meeting of international working group for Gas Cooled Reactor organized by the IAEA.

The main features of the R&D on HTGR in the FRG and the USA are the demonstration of excellent functions and safety and the improvement in the economy of small and medium sized modular HTGRs for power generation. In the FRG, development of basic technologies such as high temperature components and process heat technologies for direct utilization of nuclear process heat are in progress. The PRC takes a great interest in the HTGRs.

On the other hand, the main objective of the R&D on HTGR in Japan is to establish and upgrade HTGR technology basis so as to develop a reactor system appropriate for direct utilization of nuclear process heat using the HTTR. Various irradiation experiments, demonstration tests on nuclear process heat utilization using the HTTR are the main features of Japanese HTGR program.

Table 3 illustrates the future perspective of international cooperation in Japan.

For the efficient prosecution of the R&D on HTGR in Japan, it is important to introduce foreign experiences and technologies and to promote international cooperation by offering Japanese technologies. As for future prospects of international cooperation, the following are envisaged.

1) Based on the domestic technologies and the results of R&D, it is essential to establish technology basis by ourselves through the construction and operation of the HTTR. It is also necessary to make effective utilization of foreign technologies under international cooperation with advanced countries such as the FRG and the USA. In additio, Japan considers that it is appropriate to upgrade the HTGR technologies based on our own technologies and to offer them overseas.
2) The HTTR can be used for irradiation tests of large sized specimens at high temperature and also for an international iradiation test facility for HTGR fuels, graphites, heat resistant materials, ceramics, etc. Moreover, the HTTR is a unique research facility from the viewpoint of supplying high temperature heat of about

900°C at the exit of the intermediate heat exchanger and is expected to be served as an international joint research facility of thermal utilization system.

3) If the HTGR technology basis is established, small and medium sized reactors with inherent safety peculiar to the HTGR are to be realized. Japan is, therefore, willing to transfer the results of R&D to other countries, which need them, under the agreement of the appropriate R&D international cooperation.

5. Conclusion

The HTGR has a superior performance supplying high temperature heat and having high thermal efficiency, inherent safety characteristics and high fuel burn up. To promote the HTGR program in Japan under the recognition of present status of circumstances around the HTGR, the HTTR progrm has been initiated in 1987 according to the Long-Term Program for Development and Utilization of Nuclear Energy, which has recommended that Japan should proceed with the development of more advanced new technologies for the future. As the leading organization of the R&D on HTGR, the JAERI will play a key role to construct and operate the HTTR, carry out the R&D to establish and upgrade HTGR technology basis in cooperation not only with universities, industries, national research institutes in Japan but also with the research organizations in foreign countries such as the FRG, PRC, USA, IAEA, etc.

Furthermore, the HTTR should be most utilized for upgrading present HTGR technologies, conducting innovative basic research on high temperature technologies, establishing high temperature nuclear heat application technology and improving thermal efficiency.

Table 1 The long-term program of nuclear energy and the role of the HTTR in Japan

·The Long-Term Program of Nuclear Energy in Japan

revised in June 1987 by the Japanese Atomic Energy Commission

·Necessity for development of more advanced technology along with the existing nuclear system
·HTGR is one of most promising nuclear reactors

·Shortage of energy in future
·Extended use of high temperature
·Basic research at high temperature for innovative technologies

Construction of HTTR

·Establishing and upgrading the HTGR technology basis
·Potential tool for new and innovative basic researches

Table 2 HTGR R&D Program

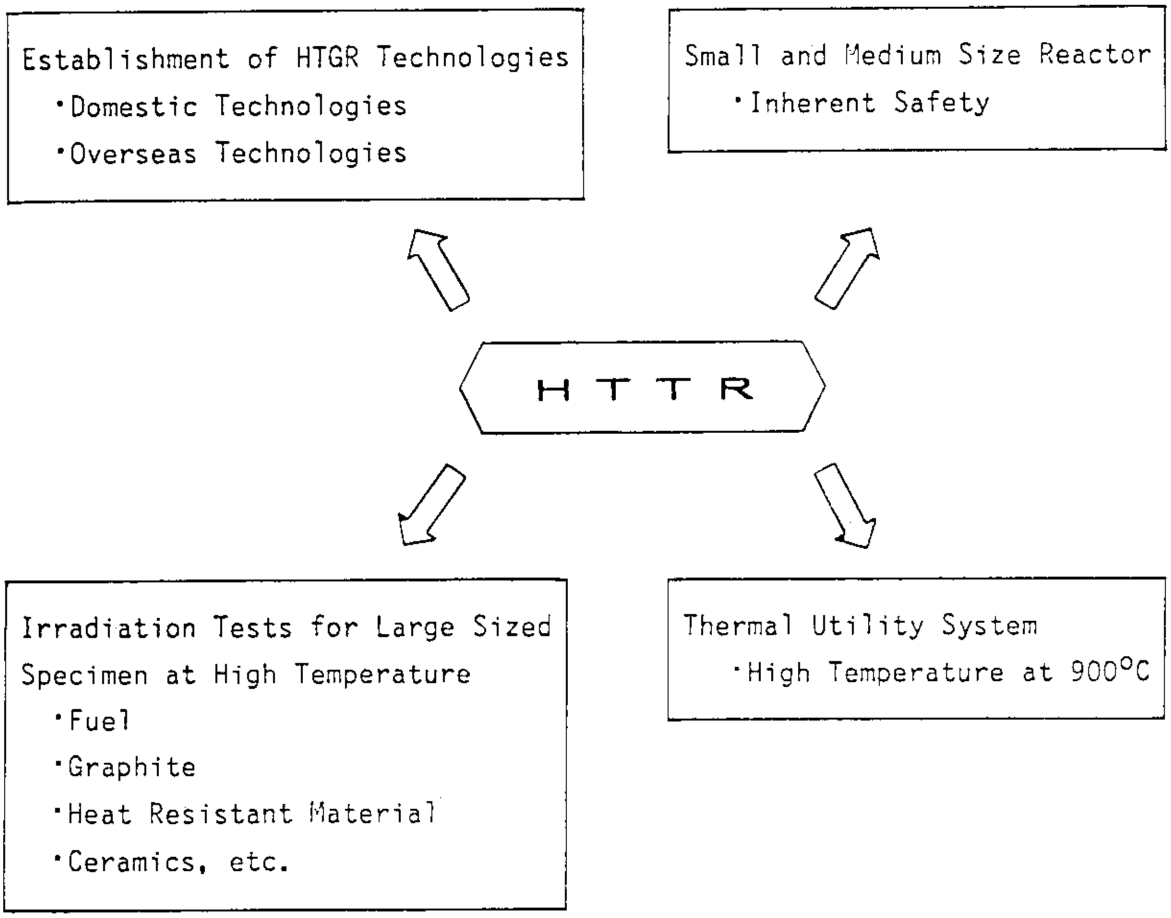
(1) HTGR Technology Basis
- (a) Fundamental Tests on Reactor System
- (b) Nuclear and Thermal-Hydraulic Characteristics of Reactor Core
- (c) Plant Dynamics, Operation and Control
- (d) Overall Performance of High-temperature Components
- (e) FP Release Behavior and Related Items
- (f) Operation and Maintenance of the Plant and Long-term Continuous Operation

(2) Upgrading Present HTGR Technologies
- (a) Basic Technologies
  - · Coated Particle Fuel
  - · Fuel Element
  - · Graphite Materials
  - · Heat Resistant Materials for Control Rod
  - · Heat Resistant Materials for Hot Gas Piping and High-temperature Components
- (b) Core Performance
- (c) High-temperature Nuclear Heat Application Technology and Improvement of Thermal Efficiency
- (d) Passive Safety Technology

(3) Innovative basic Research on High-temperature Technologies
- (a) R&D on Nuclear Fusion Reactor
- (b) R&D on Radiation Chemistry
- (c) Others

Table 3 Future perspective of international cooperation in Japan

The Role of HTTR in Future Cooperation Items



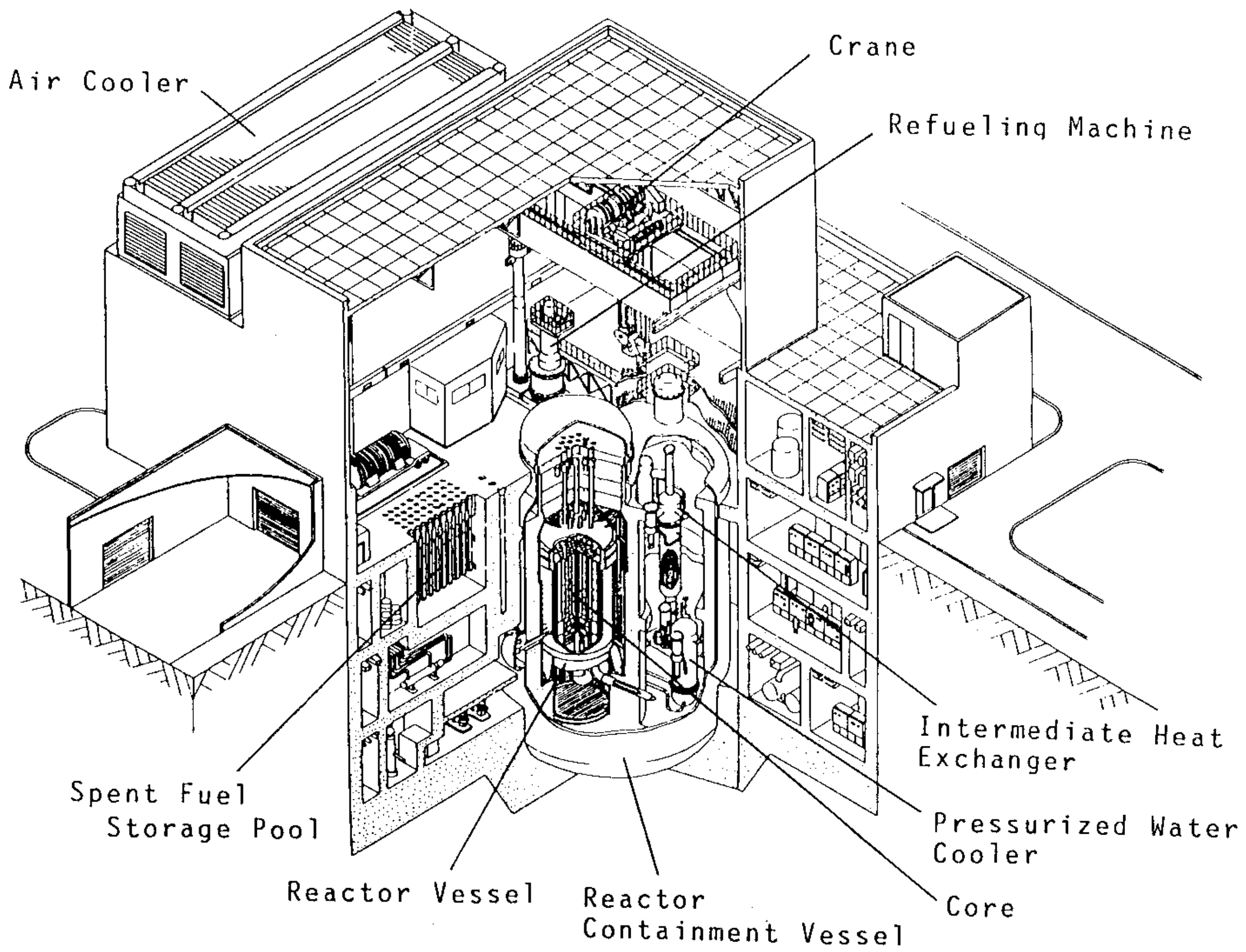


Fig. 1 Bird's eye view of HTTR reactor building

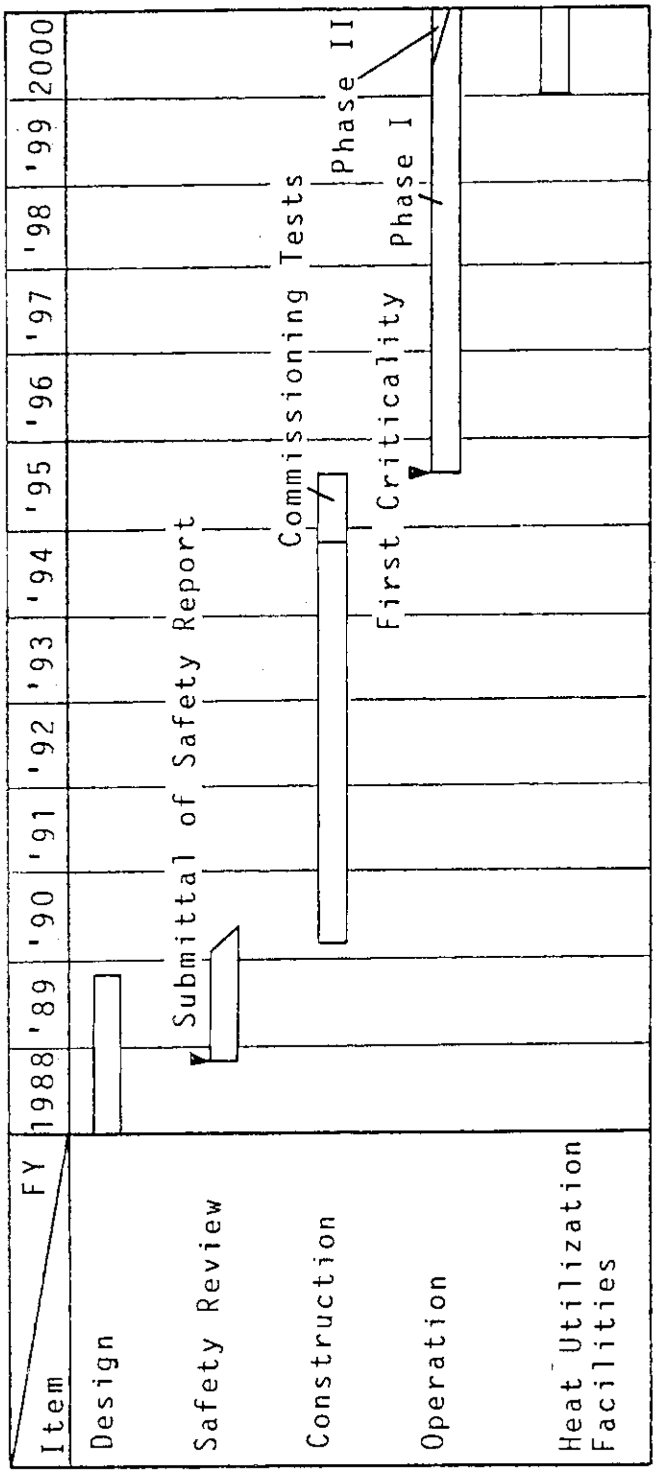


Fig. 2 Schedule of the HTTR program

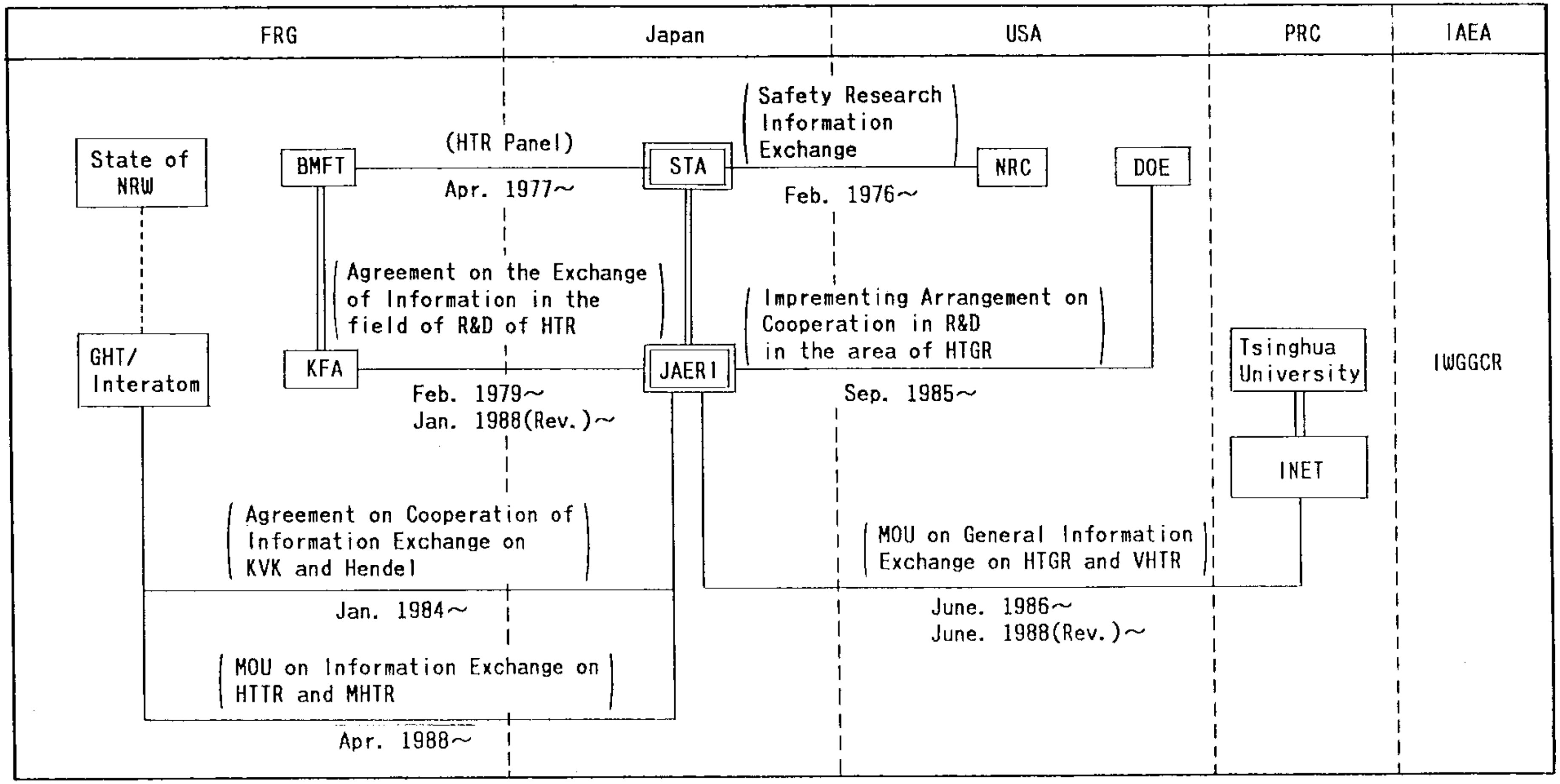


Fig. 3 Scheme of international cooperation

2.2 PRESENT STATUS AND FUTURE PROGRAM OF HTGR IN THE USA

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Department of Energy
USA

Materials used with overhead projector in the oral presentation in the Symposium are listed here because a full paper for oral presentation and a mamus-cript for issuing the proceedings of the Symposium were not submitted by the author to the Executive Committee of the Symposium in due time for issuing the proceedings of the Symposium.

US-DOE MHTGR PROGRAM

PROGRAM OBJECTIVE

DEVELOP HTGRs FOR BROAD RANGE OF APPLICATIONS IN SUPPORT OF COMMERCIAL/USER INTERESTS IN SAFETY AND HIGHER TEMPERATURE CHARACTERISTICS OF THESE PLANTS

[DIV510.JOYCE]73 3 23-JUN-88

US-DOE MHTGR PROGRAM

HTGR PROGRAM PARTICIPANTS

- NUCLEAR ISLAND ENGINEERING
  - GENERAL ATOMICS
  - COMBUSTION ENGINEERING, INC.
  - BECHTEL GROUP, INC.
- TECHNOLOGY
  - OAK RIDGE NATIONAL LABORATORY
  - IDAHO NATIONAL ENGINEERING LABORATORY
- UTILITY/USER
  - GAS-COOLED REACTOR ASSOCIATES
  - ELECTRIC POWER RESEARCH INSTITUTE
- BALANCE OF PLANT
  - STONE AND WEBSTER ENGINEERING, CORP.
  - COMBUSTION ENGINEERING, INC.

[DIV510.JOYCE]287 16 11-JUL-88

SUMMARY OF AREAS CONSIDERED IN NRC REVIEW OF MHTGR

AREAS REVIEWED

- FUEL DESIGN
- REACTOR PHYSICS
- REACTOR VESSEL
- PASSIVE HEAT REMOVAL SYSTEMS
- SAFETY ANALYSIS
- HEAT TRANSPORT EQUIPMENT
- COMPONENTS OF THE PRIMARY SYSTEM BOUNDARY
- INSTRUMENTATION
- CONTROL
- ELECTRICAL SYSTEMS
- SELECTED AUXILIARY SYSTEMS
- OCCUPATIONAL EXPOSURE
- HUMAN FACTORS
- SAFEGUARDS AND SECURITY
- SOME BALANCE OF PLANT ITEMS

AREAS DEFERRED

- SEISMIC DESIGN
- RADIOACTIVE WASTE HANDLING SYSTEMS
- MECHANICAL EQUIPMENT DESIGN
- STRUCTURAL GRAPHITE COMPONENTS
- MODELING OF FISSION PRODUCT TRANSPORT
- NUCLEAR DESIGN
- PHENOMENA INVOLVING CHEMICAL PROCESSES
- FLUID FLOW DESIGN
- REACTOR INTERNALS
- VESSEL SYSTEM AND SUBSYSTEMS
- HEAT TRANSPORT SYSTEM AND SUBSYSTEMS
- SHUTDOWN COOLING SYSTEM AND SUBSYSTEMS
- REACTOR CAVITY COOLING SYSTEM
- REACTOR BUILDING
- PLANT PROTECTION AND INSTRUMENTATION SYSTEM
- PLANT CONTROL, DATA, AND INSTRUMENTATION SYSTEM
- MISCELLANEOUS CONTROL AND INSTRUMENTATION GROUP
- ELECTRICAL SYSTEMS
- SERVICE SYSTEMS
- STEAM AND ENERGY CONVERSION SYSTEMS
- OPERATIONAL RADIONUCLIDE CONTROL
- OCCUPATIONAL RADIATION PROTECTION
- EMERGENCY PREPAREDNESS
- ROLE OF OPERATORS
- SAFEGUARDS AND SECURITY
- PROTOTYPE PLANT TESTING
- SAFETY ANALYSIS
- TECHNICAL SPECIFICATIONS AND ADMINISTRATIVE CONTROLS
- QUALITY ASSURANCE

LISTING OF MAJOR AREAS REQUIRING SUPPORTING ANALYSIS, RESEARCH, OR TESTING

FUEL DESIGN
- FUEL PERFORMANCE MODELS
- FUEL PERFORMANCE STATISTICS FROM LABORATORY TESTING
- MANUFACTURING QUALITY CONTROL
- FUEL PERFORMANCE UNDER ACCIDENT CONDITIONS
- EFFECTS OF FUEL COMPOSITION ON PERFORMANCE
- EFFECTS OF EXTERNAL CHEMICAL ATTACK ON FUEL PERFORMANCE

NUCLEAR DESIGN
- METHODS AND DATA VALIDATION
- UNCERTAINTIES IN NEGATIVE TEMPERATURE COEFFICIENT OF REACTIVITY
- CONTROL MATERIALS

THERMAL AND FLUID FLOW DESIGN
- CORE FLOW DISTRIBUTION
- NOT STREAKS
- DIFFERENTIAL PRESSURES AND SHEAR FORCES DURING DEPRESSURIZATION EVENTS
- FLOW-INDUCED VIBRATION ON CONTROL ROD GUIDE TUBES

REACTOR INTERNALS
- SEISMIC DESIGN AND FRAGILITY DATA
- IN-SERVICE DETERIORATION OF MATERIALS

VESSEL SYSTEM
- ASME AND STAFF APPROVAL FOR ELEVATED TEMPERATURE SERVICE
- CATASTROPHIC FAILURE PROBABILITY
- NEUTRON IRRADIATION EFFECTS
- SEISMIC DESIGN, INCLUDING SUPPORT SYSTEM

REACTOR CAVITY COOLING SYSTEM AND REACTOR CAVITY
- HEAT TRANSPORT DESIGN
  - VESSEL HOT SPOTS
  - IN-VESSEL CONDUCTION
  - EMISSIVITIES
  - EFFECT OF WATER VAPOR
- REPAIR AND RECOVERY
- MODELING CONSERVATISM AND SENSITIVITIES TO UNCERTAINTIES
- SEISMIC DESIGN AND FRAGILITY DATA
- REACTOR CAVITY TEMPERATURES
- DUCT AND CHIMNEY DESIGN
- HEAT TRANSMISSION TO THE EARTH

RADIONUCLIDE CONTROL (SOURCE TERM, FISSION PRODUCT TRANSPORT)
- ASSUMPTIONS AND MODEL FOR BACK CALCULATION FROM SITE BOUNDARY

OPERATIONS
- ADVANCED CONTROL SYSTEM DEVELOPMENT
- HUMAN FACTORS ANALYSIS
- CREW SIZE AND TRAINING
- WORKER EXPOSURE DURING MAINTENANCE

PROTOTYPE PLANT TESTING

US-DOE MHTGR PROGRAM

SCHEDULE FOR RTDP SECTION REVIEW

CATEGORY	RTDP SECTION	SECTION TO NRC
BASE TECHNOLOGY		
– FUEL & FISSION PRODUCTS	6	10/90
– GRAPHITE	7	8/90
– METALS	8	8/90
– CONTROL MATERIALS	9	TBD
– PHYSICS	11	4/90
SYSTEM/COMPONENT TECHNOLOGY	10	
– RCCS		6/90
– OTHER		TBD

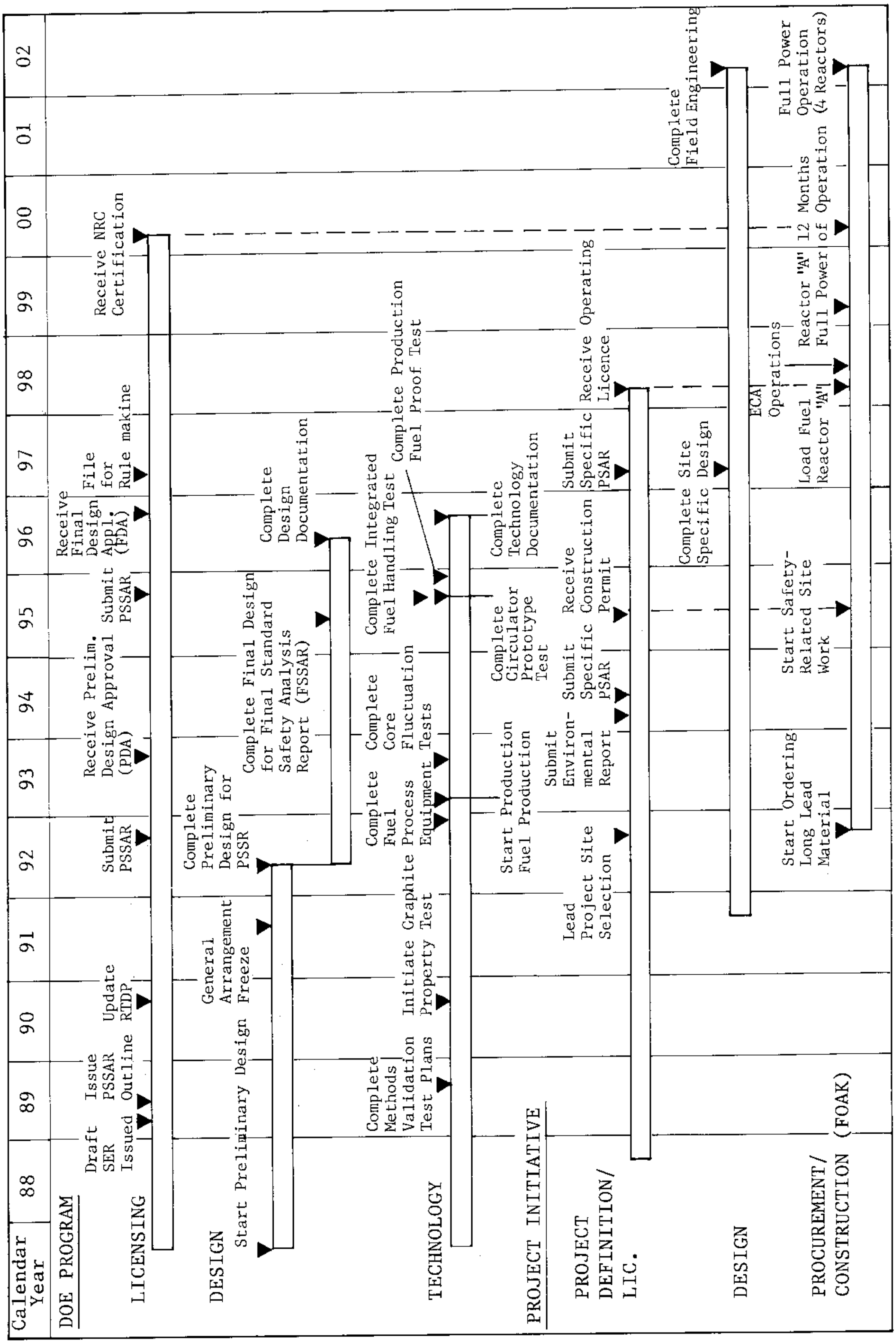
U.S. DOE MHTGR PROGRAM

STATUS OF CODE INQUIRY

- o CODE INQUIRY SUBMITTED TO ASME CODE COMMITTEE - 11/87
- o CONTINUOUS INTERACTION WITH ASME CODE COMMITTEE AT QUARTERLY MEETINGS - 87-90
- o REVISED DRAFT OF INQUIRY/PROPOSED REPLY SUBMITTED TO ASME CODE COMMITTEES - 2/90
- o MATERIALS TEST PROGRAMS COMPLETED BY ORNL AND C-E (MML) - 3/90
- o DETERMINE SIGNIFICANCE OF CREEP DAMAGE DUE TO RELAXATION & DEVELOP RESPONSE - 3-4/90
- o SUBMIT RESPONSE TO QUESTION ON CREEP RELAXATION TO ASME CODE COMMITTEES - 5/90
- o FINAL MATERIALS PROPERTIES DATA PACKAGE IN PREPARATION, TO BE SUBMITTED TO ASME CODE COMMITTEES - 5/90

(INFORMATION PROVIDED WILL SATISFY ALL THAT IS NECESSARY FOR CODE COMMITTEE DECISION BY 5/90)

MHTGR MASTER SCHEDULE



2.3 UTILITY/USER REQUIREMENTS FOR AND ASSESSMENT OF MHTGR

William E. KESSLER
Consumers Power Company
Luther D. MEARS · Harold L. GOTSCHALL
Gas-Cooled Reactor Associates
USA

This paper describes the approach used by Gas-Cooled Reactor Associates (GCRA) in developing Utility/User Requirements for the Modular High Temperature Gas-Cooled Reactor (MHTGR). As representatives of the U.S. utility/user industry, it is GCRA's goal that the MHTGR concept be established as an attractive nuclear option for safe, economic energy supply with limited ownership risks. Commercially deployed MHTGR systems should then compete favorably in a mixed-fuel economy with options using fossil, other nuclear and other non-fossil sources.

To achieve this goal, the design of the MHTGR plant must address the strains experienced by the U.S. industrial infrastructure during deployment of the first generation of nuclear plants. Indeed, it is GCRA's intent to utilize the characteristics of MHTGR technology for the development of a nuclear alternative that poses regulatory, financial and operational demands on the Owner/Operator that are, in aggregate, comparable to those encountered with non-nuclear options.

The dominant risks faced by U.S. owner/operators with current nuclear plants derive from their operational complexity and the degree of regulatory involvement in virtually all aspects of operations. The MHTGR's approach to passive safety provides the technical basis for dramatic simplification of overall plant licensing and operations and, thus the opportunity for reducing the risks of nuclear plant ownership.

The paper describes utility participation through GCRA in the MHTGR Program and the rationale for the selection of key requirements in the context of a business risk management philosophy. It also provides a summary assessment of the current design against key top level requirements.

1. INTRODUCTION

The traditional utility role in supplying electricity includes generation planning, procurement, licensing, operation, maintenance and decommissioning of power plants. In conducting these functions, the utility must cope with uncertainties in external factors such as load growth, interest rates and the regulatory climate, as well as internal demands on finances, management and staff. Nuclear generation has been particularly vulnerable due to its capital intensive nature and the intensity of safety regulation. In recent years, the nuclear option has been foreclosed in the U.S. because of the risks derived from these uncertainties.

In February, 1984, the Office of Technology Assessment (OTA) issued a report entitled "Nuclear Power in an Age of Uncertainty" (Reference 1). The opening paragraph in the Overview and Findings section states:

> "Without significant changes in the technology, management and level of public acceptance, nuclear power in the United States is unlikely to be expanded in this century beyond the reactors already under construction. Currently, nuclear power plants present too many financial risks as a result of uncertainties in electric demand growth, high capital costs, operating problems, increasing regulatory requirements, and growing public opposition."

Although six years have passed since issuance of the OTA report, the technical, social and management issues it characterizes represent an ongoing source of business risk that remain of great concern to the industry. At issue are the processes required of industry to assure public health and safety to the satisfaction of regulatory authorities. Today, all of the advanced reactor concepts under development in the U.S. seek to avoid the complex and exacting demands that have been placed on industry to license plants currently in operation. The Modular High Temperature Gas-Cooled Reactor

(MHTGR) concept's approach to passive safety offers the prospect of dramatic simplification in overall plant licensing and operations.

Prior to issuance of the OTA report cited above, some utilities had indicated their view that fundamental changes in the approach to nuclear power plant design should be evaluated. In the fall of 1983, Pacific Gas and Electric (PG&E) Company sponsored an "Evaluation of Small Modular High Temperature Gas Cooled Reactors Applied to Electricity Generation" (Reference 2). This study exposed U.S. utilities to the MHTGR safety concept for the first time. The MHTGR safety concept is based on the retention of fission products within the ceramic fuel during both normal and accident conditions by refractory material coatings surrounding the fuel and without reliance on operator action or A-C powered equipment to provide shutdown and emergency cooling. An outgrowth of the PG&E study was the philosophical framework for Utility/User Requirements for the MHTGR: Definition of a nuclear option offering investment returns and business risk management potential on a par with coal-fired alternatives.

In April of 1984, the HTGR Program was redirected to a systematic assessment of multiple HTGR options. In September of 1985, that process culminated in selection of a reference MHTGR plant configuration consisting of four 350 MWt reactor modules supplying steam to two turbine generators (see References 3 or 4 for a description of the design). Today, the development of the reference plant design and supporting technology is continuing after extensive review of the conceptual design by the Nuclear Regulatory Commission (NRC). Independent analyses by NRC contractors support expectations that the MHTGR safety concept will limit the consequences of nuclear accidents to extremely low levels.

As depicted in Figure 1, the design has been developed in response to Utility/User Requirements established by GCRA (Reference 5) along with other top-level requirements from the DOE and the NRC plus relevant industry codes and standards. The MHTGR is distinguished from other nuclear options in that the safety concept permits a substantial shift in regulatory emphasis from plant design, construction, operation and maintenance activities to fuel fabrication. It is this aspect of MHTGR development that offers the potential for a nuclear option approaching cost/risk parity with coal-fired alternatives.

2. UTILITY/USER PARTICIPATION IN MHTGR DEVELOPMENT

Market forces usually control the introduction of new technologies, and development costs are amortized over predicted sales of the product. In most cases, transactions occur with sufficient frequency for the requirements of the buyer, seller and regulator to be sorted out in the give and take of the market so that a stable infrastructure for risk-sharing evolves with time. Because of the length of time, large costs and complex institutional commitments required for nuclear power, design evolution through the trial and error of the market place is especially difficult. Substantial Owner/Operator involvement from the early stages of design development will increase the likelihood of commercial success.

Within the U.S. MHTGR Program, vendor/supplier interests are represented through a team of industrial participants, primarily funded by the DOE, while GCRA, which is privately funded by utilities, has primary responsibility for infusing the perspective of the Owner/Operator. This arrangement fosters normal buyer/seller viewpoints within the Program while interactions with the NRC permit each to gauge the implications of an emerging regulatory environment on the future market. In doing so, the Program positions both buyer and seller interests to identify and address their risks as groundwork for a Project initiative in response to market opportunities.

GCRA's principal programmatic interaction with the MHTGR Program is through 1) the development of Utility/User Design Requirements and the related assessments of the design against such requirements and 2) the identification of Utility/User Priorities to guide resource allocations for ongoing development programs and assessments of such programs for their overall effectiveness. Each of these activities is conducted with the intent of minimizing risks to the End-User in the areas of plant procurement, licensing, operation, maintenance and decommissioning; i.e., the relevant areas of the traditional Utility/User's role in supplying electricity.

An essential element of bringing the Owner/Operator perspective to the Program is the participation of experienced utility management and senior technical people in these activities. Major design decisions for the MHTGR plant are reviewed with utility personnel, along with the results of competitive economic analyses and the status of NRC interactions. In addition, GCRA utility personnel participate in meetings with the management of vendor/supplier companies and the DOE to assess Program strategy and progress toward disposition of open areas.

As the design of the MHTGR has progressed, more comprehensive utility technical assessments have been undertaken. At the completion of Conceptual Design, GCRA conducted a Utility Design Review Workshop (Reference 6) with emphasis on operations and maintenance activities. The panel was composed of senior technical representatives, and led to the recommendation that a continuing review of the design by utility representatives experienced in power plant operation and maintenance be established. A Utility Working Group was subsequently organized to deal with these and other issues related to operations and maintenance in an ongoing effort. These activities are complemented by privately funded site specific studies for individual utilities. GCRA staff assimilates the findings of utility studies and reviews, and provides DOE and Program participants with the utility perspective on Program issues.

The following section describes the rationale for the key Utility/User Requirements for the MHTGR that are most pertinent to major utility industry issues. It should be noted that the process of requirements development is dynamic and requires an ongoing utility participation to assure appropriate interpretation and implementation.

3. THE RISK MANAGEMENT RATIONALE FOR KEY UTILITY/USER REQUIREMENTS FOR THE MHTGR

The fundamental difference in business risk between nuclear and fossil generation options is the extent of regulation imposed to protect public health and safety. This not only results in the requirement for considerable additional equipment but also in a complex framework of operating and operability limits as conditions for continued operation. It has been the industry's experience that the processes developed to assure the health and safety of the public to the satisfaction of regulatory authorities represent a major source of risk to capital and operation and maintenance (O&M) costs and to plant availability throughout the life of the facility.

Further, the discipline and vigilance required for the "exacting" chore of establishing and continuously verifying the ongoing safety of the plant, in accordance with regulations, distinguishes the organizational requirements and corporate cultures of nuclear from fossil power plants. These risks have often translated into a high level of stress for utility personnel and, in extreme cases, have resulted in major reorganizations within the corporate structure.

As an example of the difference in risk, the nuclear option is the only energy supply alternative for which formally approved and periodically exercised plans for the rapid sheltering and evacuation of the offsite public are legally required as a prerequisite to power operation (10CFR50.47). A quantum step toward eliminating this risk difference is embodied in the Utility/User requirements through specifying that NRC and EPA criteria for protection of the public be met at the plant Exclusion Area Boundary (EAB) of 425 meters without consideration of sheltering or evacuation. Satisfying this requirement is made possible by the characteristics of the MHTGR design and the integrity of the coated fuel particle.

As another example, the NRC grants an operating licensing on the basis of demonstrated financial and technical capability and an acceptable plant safety analysis. The philosophy behind the regulatory process is one of assuring that the plant is operated and maintained such that the assumptions of the safety analysis (equipment performance, operator action, etc.) are valid throughout the life of the facility. When a system is identified as safety-related for the purposes of regulation, it automatically becomes burdened with the multiplicity of risks that arise from assuring and documenting that the requisite level of reliability has been provided in each step of development from design through power generation. The Utility/User requirements call for the avoidance of reliance on complex active systems for the attainment of safety goals, which minimizes safety-related equipment whose functions are subject to operational licensing requirements. Again, the MHTGR meets these requirements by the characteristics of the design and the utilization of naturally-occurring processes resulting in plant simplifications to the extent that regulatory compliance is not dependent on operator action or the performance of AC-powered equipment. In the absence of offsite consequences and given long time intervals to obtain assistance in the event of an accident, staff training demands are minimized while automated control devices and operator action to avoid damage to equipment (protection of the owner's investment) can be employed much as they are in a non-nuclear plant. It is expected that these characteristics of the MHTGR will permit a substantial reduction in operational licensing requirements relative to current nuclear plants.

This same absence of offsite consequences should allow the potential for close-in siting to generation load centers as well as the potential for close proximity siting to industrial parks for cogeneration/process heat applications. Coupled with the goal that the plant should have a busbar cost advantage of at least 10% relative to a comparably sized state-of-the-art coal plant alternative, while posing comparable levels of institutional and financial risk, the MHTGR should result in a nuclear option that is more compatible with the needs of the majority of U.S. utilities in terms of size, complexity and operational demands.

The foregoing captures the essence of the risk management philosophy in Utility/User Requirements for the MHTGR. The following sections extend this rationale to 1) generation planning, 2) procurement and licensing and 3) operation and maintenance.

3.1 GENERATION PLANNING

The generation planning function within utilities forecasts changes in demand for electricity and identifies generation options to meet those demands. On a broad scale, utilities within a generation region strive for a portfolio of supply options as a hedge against availability and/or price escalation in any one sector, while providing an appropriate mix of base load, intermediate and peaking capability. In the past, a mixed-fuel economy consisting of fossil, nuclear and other energy sources was considered a rational objective. The practical choices consisted then, as now, of coal, nuclear, oil and gas; with others (hydro, wind, solar, geothermal, etc.) representing viable options only under unique geographic circumstances.

The major source of risk in generation planning is changes in the general economy which effect load growth and the cost of money. While this risk is by no means unique to the nuclear option, the capital intensive nature and longer lead times of nuclear plants result in comparatively greater financial exposure in the event of schedule extensions or delays. It follows that the larger the unit being delayed, the greater the financial exposure. To address this concern, one of the requirements for the MHTGR was that the reference plant should produce a nominal electric output of 550 MWe. This requirement evolved from GCRA surveys which showed the size range of highest interest for future capacity additions to be in the range of 400-700 MWe. In addition, it is required that features be incorporated to facilitate incremental capacity additions associated with single or multiple reactor modules. Taken as a whole, these provisions result in smaller units with shorter deployment schedules and opportunities for adaptive decisions in the event of unforeseen circumstances.

3.2 PLANT PROCUREMENT AND LICENSING

Having made the decision to add nuclear capacity, Utility/Users must address procurement and licensing risks.

The construction of the current generation of nuclear plants resulted in a great diversity of plant designs. While several larger utilities developed A-E capability in-house, the usual situation involved the utility choosing between four reactor vendors and about twenty A-Es and/or constructors. In addition, many utilities incorporated design features in accordance with their individual preferences. These combinations resulted in wide differences in design and construction practices at a time when the regulatory process was still evolving. Except for the early fixed price, turnkey plants, the procurement and licensing risks were borne almost exclusively by the Owner.

Utility/User Requirements for the MHTGR address these issues by requiring NRC design certification of the Nuclear Island. With the deployment of the initial Lead Project and its related testing and safety demonstrability capability, early certification thereafter is expected. The Nuclear Island, containing the reactor modules and the safety-related buildings and equipment, is within the purview of the NRC. It is required that the Nuclear Island be physically separated from the Energy Conversion Area which includes the turbine generators and ancillary non-safety-related facilities. It is further required that the reactor modules accommodate a spectrum of turbine plant configurations in addition to the reference configuration which consists of four reactors supplying steam to two turbine generators. With all siting considerations, including seismic design criteria, the design of the Nuclear Island (and the rest of the reference plant) is intended to accommodate approximately 85% of potential domestic U.S. power plant sites.

The intent of these requirements is to minimize licensing risks through certification of a Nuclear Island and to nurture the emergence of a vendor entity that would construct the Nuclear Island on a turnkey basis. The option is preserved for the Owner to select and procure the Energy Conversion Area according to its capabilities and risk preferences. It remains the Owner's responsibility to license the site and staff for operation.

3.3 PLANT OPERATION AND MAINTENANCE

Operation and maintenance (O&M) costs for current generation nuclear plants are high and continuing to rise. Based on GCRA survey data, the staff on recent nuclear plants is approximately 4 or 5 times larger than that required to operate a modern coal plant of comparable size. Regulator and industry requirements for the current generation of nuclear plants have generated costly documentation and monitoring programs requiring ever increasing staffs. In addition, data from the Institute of Nuclear Power Operations (INPO) show that the equivalent availability of operating plants hovered around 61% from 1980 to 1987 (Reference 7). This same period, during which the average cost of production (fuel and O&M costs) rose by 128% (Reference 8), corresponded with a concerted industry effort to improve performance. The effort appears to be paying off in that the U.S. Council on Energy Awareness (USCEA) reported that in 1988 the average capacity factor of U.S. nuclear plants was five percentage points higher than in 1987 (Reference 9). The experience with poor performance and increasing O&M costs is attributed to the complex and exacting nature of current plants, which leads to equally complicated and detailed operational licensing requirements and staff training demands. In the recent past, increased staff and extended outages were frequently needed for plants to be backfitted with hardware required for compliance with evolving regulations. As noted earlier, there is a wide range between the best and worst performers.

These risks are addressed by requirements that lead to simplification of the plant and the use of equipment familiar to owners of fossil facilities wherever possible. There is less risk to the Utility/User if economic plant operation, maintenance and performance can be achieved from a design based on industry average expectations of staff and equipment. Accordingly, the MHTGR is required to deliver 80% equivalent availability* with maximum use of practices consistent with modern fossil-fired facilities and standard

*Equivalent Availability as defined for the MHTGR is the same as the capacity factor of a plant with a 100% load factor.

"off the shelf" components and materials. The intent of these requirements is to define a nuclear plant option that will yield an equivalent availability of 80% with staff and training requirements approaching those of a contemporary coal plant.

As noted previously, an outgrowth of the requirement on public health and safety is a design insensitive to operator error and the performance of AC-powered equipment. These characteristics distinguish the MHTGR from other nuclear options and support a goal for plant staff of 258 personnel or less. Reliance on naturally-occurring processes is expected to translate into an overall simplification of plant operations and a marked reduction in operational licensing requirements with a corresponding decrease in redundant, highly documented and heavily staffed nuclear programs needed for current nuclear plants. However, it will likely be necessary for industry to mount an intensive program; sustained throughout design development and Lead, Replica and follow-on projects; for regulatory provisions appropriate to the MHTGR to be internalized in the industry (e.g., NRC, INPO, ANI) and reflected in lower operating costs.

Requirements for the MHTGR also address some concerns unique to nuclear facilities relative to plant life extension and component replaceability. With regard to plant life extension, longer plant life reduces the number of siting and decommissioning proceedings for the same generation capacity. While the Atomic Energy Act limits the operating license to a period not to exceed forty years, a number of nuclear plant owners are finding that many plant components will have a high residual value at the expiration of the initial operating license. MHTGR requirements specify designing for a forty-year service life (based on a conservatively projected duty cycle), with identification of and provisions for acquisition of data needed to support an application for renewal of the operating licensing prior to its expiration.

As a contingency for unexpected equipment failure, provisions are required for the removal and replacement of components, including those within the primary coolant boundary. These provisions are considered to be prudent for any power plant, and they minimize investment in "life-of-the-plant" components based on speculative data for the long-term duty cycle and material properties. This approach also provides a high likelihood that the service life of a well maintained and conservatively operated plant can be extended.

An additional measure of investment protection during the operating life of the plant has been taken with requirements that 1) limit expected plant unavailability due to long outages (those greater than six months), including those not expected to occur in an individual plant's lifetime, to less than one percent and, 2) limit the annual risk of property damage to the annual premium for commercially available property damage insurance. The intent of these requirements is to limit the Owner's risk of major accidents or equipment failure, including those that could lead to plant write-off, to a level that is on par with other energy options and consistent with available commercial practices.

Utility/User Requirements also limit the frequency of exceedance of design limits associated with safety related equipment. This requirement recognizes the financial implications to the User industry of a challenge to the licensing basis of the certified MHTGR. The intent is to provide margin relative to the safety analysis upon which the operating license is based and thereby minimize the risk of a regulatory shutdown of multiple MHTGR plants.

4. SUMMARY ASSESSMENT

4.1 SAFETY AND LICENSING

The NRC staff, led by the Advanced Reactors Group, have conducted a sustained activity to establish a policy framework for licensing advanced reactors. Their process has included extensive interactions within the NRC staff, and with the Advisory Committee on Reactor Safeguards (ACRS) and the Commissioners. The NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants was promulgated in final form in 1986 following an earlier draft publication for public review and comment. Additional efforts directed toward interpretation and implementation of the policy statement have involved senior independent reviewers including a former Chairman of the NRC, as well as continuing interactions with the ACRS and the Commissioners. This well documented activity resulted in the development of a NRC staff recommendation to the Commissioners on specific key policy issues in mid 1988. Final disposition of key policy issues is expected to be achieved in 1990.

The MHTGR licensing review has been conducted within this evolving advanced reactor licensing policy framework. Reference 10 presents a summary of this review prepared by GCRA staff. Following a number of early interactions, the review began in earnest with the submittal of the Preliminary Safety Information Document in 1986 shortly after the promulgation of the advanced reactor policy statement. Document submittals specifically in support of the review totalled approximately four thousand pages. The initial submittal addressed event frequency ranges down to 5×10^{-7}/year, and to a more limited extent to 10^{-8}/year. During the course of the review, the NRC staff identified two successively more challenging sets of events for analysis by the designers without consideration of minimum frequency limits. In addition, the NRC staff contracted for independent analyses of accident response, focusing on the most challenging events. The results of the independent analyses as documented in a draft Safety Evaluation Report (SER) were generally in excellent agreement with the results of the designers, and confirmed the benign response characteristics and the relative simplicity of the analysis of the MHTGR.

The doses at the 425 meter site boundary for all events were below EPA minimum dose levels for initiation of protective action. In fact, the worst case whole body dose for an individual remaining at the site boundary for the entire 30 day event period was about equal to natural background. While recognizing that uncertainties in safety response remain (as they will for any design even after extended operating experience, although to a substantially lesser degree), the results of the extensive review of the MHTGR amount to a resounding validation of the design at its current state of development. The final outcome of the preapplication MHTGR licensing review awaits the completion of a final SER.

4.2 OPERATION AND MAINTENANCE

In assessing operation and maintenance costs, updated estimates of operation and maintenance (O&M) costs for the MHTGR were developed during 1989 through the GCRA Utility Working Group. Preliminary costs for the the Lead, Replica and N^{th} MHTGR Plants are summarized in Table 1.

Projected staffing levels for the Lead Plant shown in Table 1 embody a substantial reduction relative to current nuclear experience because of the safety concept and benign nature of the plant design. In addition, highly automated plant control and information management systems, as well as the opportunity to treat much of the plant in a "not of regulatory concern" manner, contribute to a reduced staff size.

Data shown for the MHTGR are based on the expectation that future work will verify that the MHTGR attributes translate into an overall simplification of plant O&M. It is also of interest that, while the onsite staff is projected to decrease from 375 to 301 personnel between the Lead and N^{th} Plants, major contributors to further reduction in O&M costs are attributed to reductions in NRC fees and liability and property damage insurance premiums.

The Utility Working Group noted the following conclusions in the course of their work:

- The MHTGR's potential for lowering the cost of plant operations and for reducing the economic risks associated with nuclear plant operational licensing requirements to a level approaching those of alternative fossil plants represents a major incentive for its deployment.

- Upon verification of MHTGR safety related attributes, significant savings may be associated with reductions in liability and property damage insurance premiums.

- It will be necessary for industry to mount an intensive program sustained throughout design development and Lead, Replica and follow-on projects for regulatory provisions appropriate to the MHTGR to be internalized in the industry (e.g., NRC, INPO, ANI) and reflected in lower operating costs.

Staffing levels for the N^{th} Plant projected in this study exceed the goal of 258 personnel stated in Utility/User Requirements. Continuing attention to the underlying factors that drive staffing levels (e.g., operational licensing requirements) is needed to establish the bases for further reductions.

4.3 AVAILABILITY

An overall equivalent availability requirement of 80% has been established for the plant design with total plant downtime allowances for either scheduled or unscheduled outages not to exceed 10%. Assessments of both scheduled and unscheduled plant outages have been conducted by the Program participants with results summarized below.

- <u>Scheduled Outage Assessment</u> - This study provided the planned outage schedule over the life of the plant as dictated by reactor refueling and turbine-generator inspections and maintenance requirements. The major impact of the assessment was identification of the need for additional spent fuel storage capacity to meet the need for refueling two reactors in series, a necessity if the planned outage design allotted hours are to be met. The assessment resulted in an estimated scheduled outage rate of 12%.

- <u>Unscheduled Outage Assessment</u> - This study provided an initial assessment of unplanned outage hours considering only the major equipment items based on industry experienced mean time to failure and mean time to repair for similar equipment. The results showed an estimated forced outage rate of 8.9%.

Although the results of these two studies project an overall equivalent availability for the current plant design of ~79%, slightly short of the requirement, it is believed that the equivalent availability requirement is achievable provided reliability and maintenance considerations are adequately addressed as the design proceeds.

It is worthwhile noting, in regard to just the basic availability of the four-module MHTGR plant, that the utilization of four separate modules to fulfill the total plant power output requirement results in the greater ability to produce power at some level at all times, i.e. a higher basic availability than a plant with a single reactor and turbine with the same output. This should translate into reduced reserve requirements on the overall utility system to cover for unscheduled outages.

4.4 POWER GENERATION COSTS

Throughout the MHTGR development effort, costs have been developed to design, construct and operate MHTGR power plants along with a comparison of these costs with relevant alternatives, primarily modern coal plants. The costs are developed by the design participants in general conformance with the U.S. Department of Energy cost estimating guidelines for advanced nuclear technologies. GCRA has the programmatic lead for coordinating cost estimating groundrules and assumptions, developing the owner's cost and O&M costs, integrating the cost estimates and performing the cost assessments. The annual updated cost assessment is nearing completion, but for the present, last year's results must suffice (Reference 11).

Summary results are presented in Figures 2 and 3. Figure 2 presents the MHTGR equilibrium plant busbar cost advantage versus a comparably sized modern coal fired plant at 1785 $/kw ('88$ and total costs, including AFUDC) over a range of coal prices and real escalation rates for coal. For the reference coal values of $1.55/MBtu and 1.2% real escalation rate, the busbar (30 year levelized) cost advantage is approximately 8%, just short of the 10% goal.

Figure 3 presents the MHTGR busbar cost estimates over a range of plant sizes (number of modules) versus a comparable range of coal plant sizes using reference coal price projections. Data points for gas/oil combined cycle plants are also included as they are the prevalent near-term option in the U.S., due to the current low prices of gas and oil. As noted from Figure 3, the MHTGR's cost competitiveness versus comparably sized fossil plants suffers further for the 2x1 configuration (270 MWe plant) and is clearly lost for the 1x1 configuration (135 MWe plant). This effect is driven primarily by the higher fixed costs associated with the MHTGR plants.

At present, there is added uncertainty in the cost competitiveness of the MHTGR due to recent softness in coal prices and major variances in recent and projected coal plant capital costs. On the other hand, counter uncertainties exist due to the projected tighter restrictions on emissions from all fossil fired plants. In addition, estimates are becoming available for mid-size advanced light water reactor plants that must be carefully evaluated.

While the MHTGR offers much promise as a user friendly nuclear alternative, design and cost improvement efforts must be intensified to assure that it meets its cost competitiveness goal.

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TABLE 1

SUMMARY OF O&M COST DATA FOR MHTGR PLANTS (89$)

	LEAD PLANT	REPLICA PLANT	N^{th} PLANT
PLANT PERFORMANCE			
Net Plant Rating, MWe	540	540	540
Base Capacity Factor, %	77	78.5	80
Net Generation, mkwh/yr	3642	3713	3784
ONSITE STAFF	375	342	301
POWER GENERATION COSTS			
Salary and Payroll Taxes, $M/yr	16.7	15.3	13.3
Maintenance Materials			
Fixed, $M/yr	2.1	1.6	1.6
Variable, $M/yr	0.7	0.5	0.5
Subtotal, $M/yr	2.8	2.1	2.1
Supplies and Expenses			
Fixed, $M/yr	2.8	2.8	2.8
Variable, $M/yr	0.4	0.4	0.4
Subtotal, $M/yr	3.2	3.2	3.2
Offsite/Central Support, $M/yr	1.9	5.0	5.0
Subtotal, Power Generation Costs			
Fixed, $M/yr	23.4	24.7	22.7
Variable, $M/yr	1.1	0.9	0.9
Subtotal, $M/yr	24.5	25.6	23.6
ADMINISTRATIVE AND GENERAL COSTS			
Pensions and Benefits, $M/yr	3.9	3.6	3.1
Nuclear Regulatory Fees, $M/yr	5.7	1.9	1.9
Liability Insurance, $M/yr	0.5	0.5	0.3
Property Insurance, $M/yr	4.0	4.0	2.0
Replacement Power Insurance, $M/yr	0.0	0.5	0.5
Other A&G Expenses	3.7	3.8	3.5
Subtotal, $M/yr	17.8	14.3	11.3
TOTAL O&M COSTS			
Fixed, $M/yr	41.2	39.1	34.1
Variable, $M/yr	1.1	0.9	0.9
Total, $M/yr	42.3	40.0	35.0
Total, mills/kwh	11.6	10.8	9.3

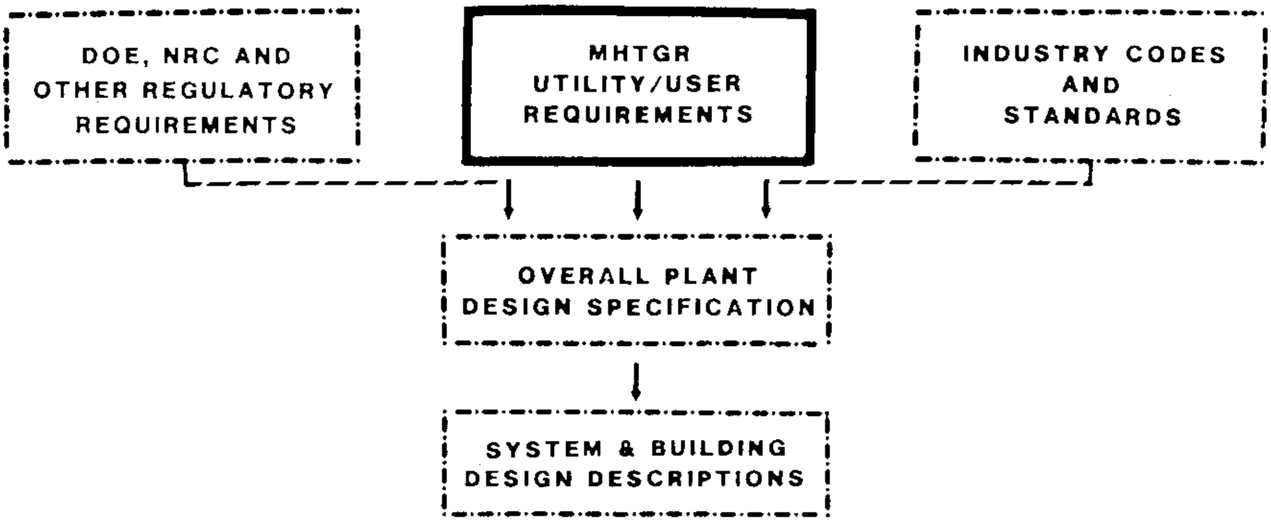

FIGURE 1

IMPLEMENTATION OF UTILITY/USER
REQUIREMENTS FOR THE MHTGR

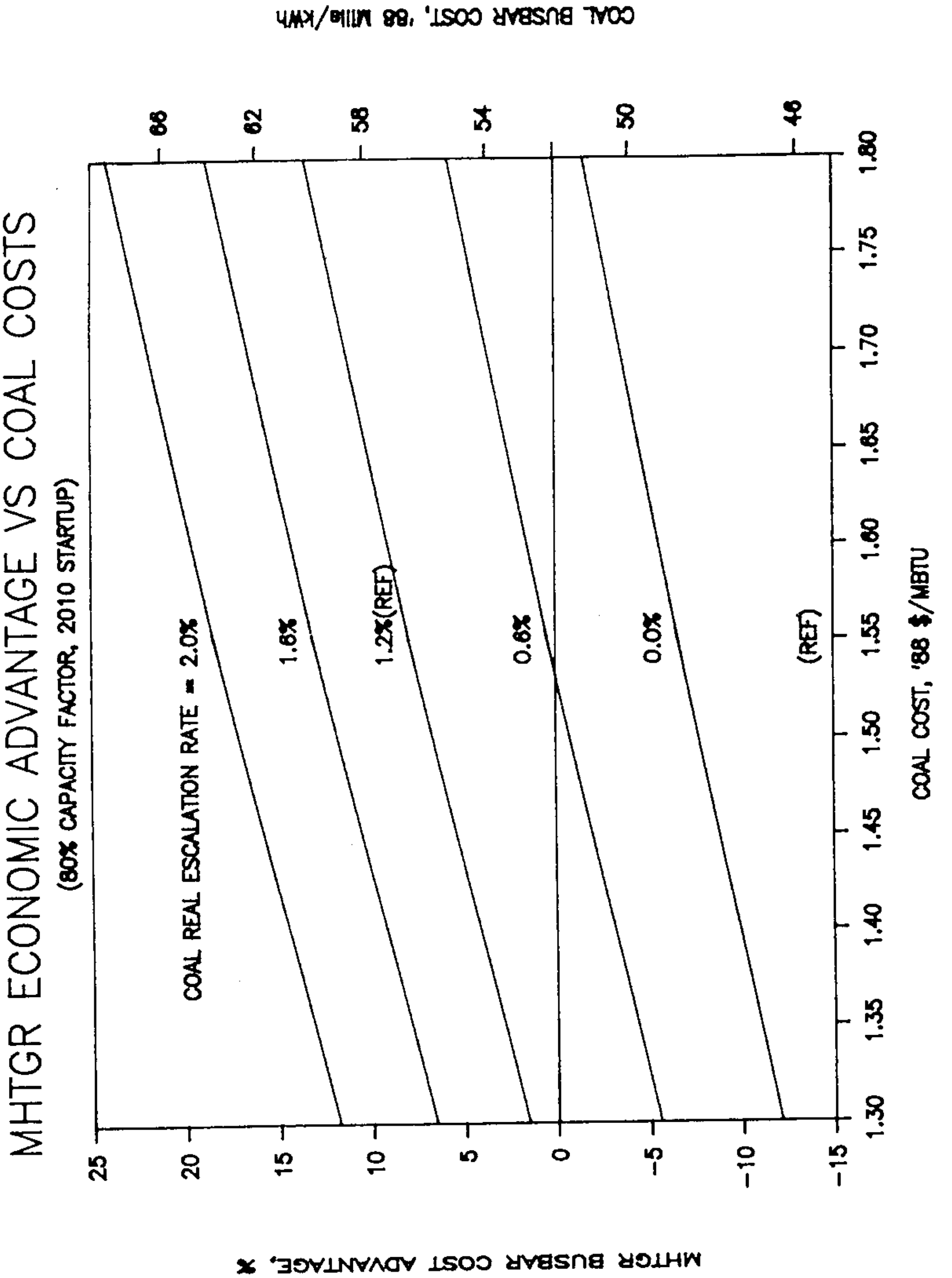
MHTGR ECONOMIC ADVANTAGE VS COAL COSTS
(80% CAPACITY FACTOR, 2010 STARTUP)
COAL BUSBAR COST, '88 Mills/kWh
66
62
58
54
50
46
COAL REAL ESCALATION RATE = 2.0%
1.6%
1.2%(REF)
0.6%
0.0%
(REF)
25
20
15
10
5
0
-5
-10
-15
MHTGR BUSBAR COST ADVANTAGE, %
1.30
1.35
1.40
1.45
1.50
1.55
1.60
1.65
1.70
1.75
1.80
COAL COST, '88 $/MBTU

FIGURE 2

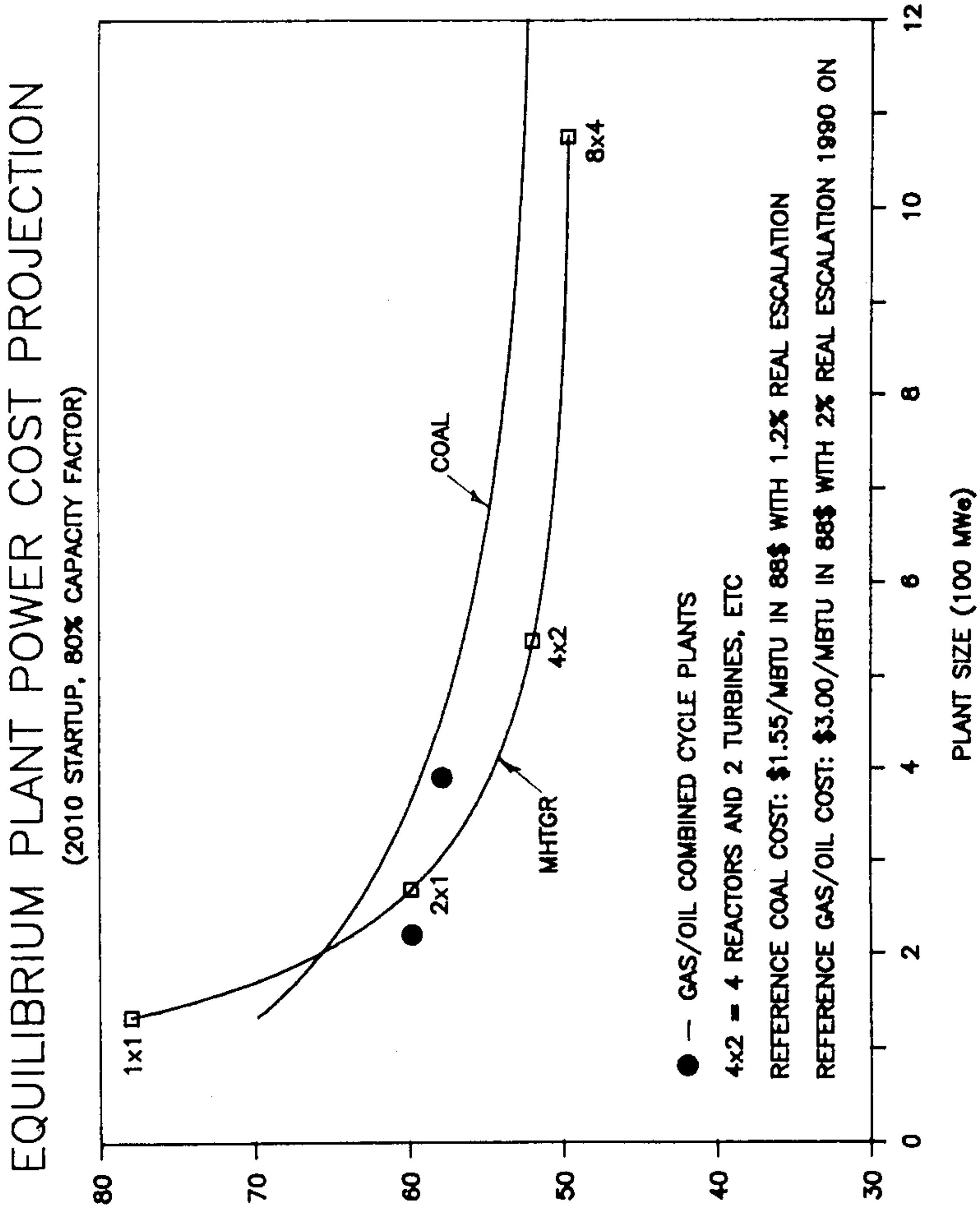
EQUILIBRIUM PLANT POWER COST PROJECTION
(2010 STARTUP, 80% CAPACITY FACTOR)
COAL
MHTGR
1x1
2x1
4x2
8x4
● – GAS/OIL COMBINED CYCLE PLANTS
4x2 = 4 REACTORS AND 2 TURBINES, ETC
REFERENCE COAL COST: $1.55/MBTU IN 88$ WITH 1.2% REAL ESCALATION
REFERENCE GAS/OIL COST: $3.00/MBTU IN 88$ WITH 2% REAL ESCALATION 1990 ON
PLANT SIZE (100 MWe)
80
70
60
50
40
30
0
2
4
6
8
10
12

FIGURE 3

2.4 EXPERIENCE GAINED WITH HIGH TEMPERATURE REACTOR AND THEIR PROSPECTS

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1. Experience gained with the development and operation of High Temperature Reactors

The Federal Republic of Germany can look back on a major programme on the development of high temperature reactors. Experience has been gained over the last three decades with the construction of the pilot plant pebble bed reactor AVR and the demonstration reactor THTR 300 as well as with an extensive R+D programme.

At the end of the fifties the programme was launched shortly after nuclear activities for the peaceful uses of atomic energy were permitted in the Federal Republic of Germany. In concurrence with similar programmes in other countries the main motivation was to provide a highly economic power supply. In the sixties the AVR was built, successfully taken into operation and the THTR was designed as the next step.

The German programme in cooperation with Switzerland, concentrated on advanced power plant concepts utilizing gas turbine technology in parallel to the start of THTR construction.

During the construction of the THTR, the German HTR programme approached a critical period owing to:

- the increase of nuclear licensing requirements,
- construction delays and high incremental costs for the completion of the demonstration plant THTR 300
- the lack of a concept for a THTR follow-up plant acceptable to the utilities at that time,

no chance for the realization of an HTR gas turbine or a process heat plant as THTR successor on a private basis.

These impacts created a crisis for the still young, still developing, not yet commercially adapted technology.

A general technology reassessment was carried out, which resulted in a unanimous decision on the completion of the THTR 300 and continuation of the programme. The basis, however, was a shifting of the responsibilities for further activities and initiatives from the government to the private sector.

The users/utilities gave orders to the reactor industry to develop conceptual designs for THTR follow-up plants on a steam cycle basis in contracts without public involvement.

The Federal Government expressed willingness to continue to support further, in particular safety-related, R+D activities, provided the private industry will assume full responsibility for THTR follow-up plants including process heat plants without public involvement.

At present, orders for plants cannot be expected for some years to come also because there is sufficient capacity to meet the demand for electricity and in view of very low prices and no supply shortage expectations for conventional primary energy carriers.

Subsequently, taking limited budget resources into account, the reactor industry was forced to practise concentration, which was accomplished by the foundation of a joint Siemens-ABB daughter company called HTR-GmbH and by staff reduction. The state-owned research centre, KFA, is reducing its HTR-related activities in the framework of reorientation to new non-nuclear research areas.

In spite of this background, very broad experience has been acquired, which now has to be utilized for the further technical improvement of the future plant concepts, for the full utilization of the safety potential and especially for the improvement of the economic features.

1.1 Experience gained with the AVR

First power plant experience was obtained with the small, 15 MWel pebble bed reactor AVR, a power plant and research facility. Located at the research centre KFA Jülich, it was operated from 1967 to 1988 following a 5-year construction period. The total electricity generation was 1,670 GWh. The plant behaviour and fuel quality improvement enabled the gas outlet temperature to be increased stepwise from 750° to 950°, without increasing the coolant gas activity of less than 25 Ci. The continuous, trouble-free operation at this high temperature level generated new programmes for the investigation of very ambitious process heat applications such as coal gasification and steam reformer technology. Moreover, an extensive programme for the development and qualification of high temperature metallic materials, as well as of heat exchanging components, was launched.

Another achievement of AVR was availability in a demonstration phase in 1975 amounting to 92%, thus demonstrating the successful performance of the fuel handling system and the continuous fuel reload scheme. Later the possibility of a change in the fuel cycle during reactor operation was demonstrated by the continuous replacement of high-enriched uranium/thorium fuel by low-enriched uranium fuel without thorium, which was not only proof of the pebble bed system's flexibility, but also important from the aspect of expected high-enriched uranium supply restrictions following INFCE procedures.

The robustness of the plant and of components was demonstrated after a water ingress accident in 1978 from a 3 mm^2 leak in a final superheater tube. 27 tons of water flooded the lower parts of the core, the fuel extraction tube and the blowers. After steam generator repair by tube plugging and drying procedures, the operation continued without any consequences.

An important task of AVR was to serve as a test facility for fuel qualification - not only for the fuel development represented by more than 20 different fuel designs, but also for the qualification of THTR reference fuel and the advanced low-enriched fuel for follow-up plants, thus providing the data base and creating confidence in the new safety concepts which are based on the fuel quality (1600°C concept). In the last few years operation was centred almost exclusively on tasks relating to safety features, in particular to the performance of a simulated loss-of-coolant accident, on core physics, fission product release and plate-out and dust behaviour. The results of these tests significantly increased confidence in the design and safety features of small HTRs like the HTR module, i.e. by verification of models and codes for passive decay heat removal.

Component and incore inspection shortly before closedown showed no indication of any age-induced wear-out.

At the end of 1988 the plant was finally shut down. A cost-benefit analysis made it difficult to justify further operation. The non-commercial costs were rather high and the expected additional information from the plant was limited. The plant, at present still ready for full operation, is now waiting for the decommissioning licence. A post-operation examination programme has been prepared.

1.2 Experience gained with the THTR 300

The THTR 300 nuclear power plant (THTR 300) with a rated electrical power output of 300 MW is the second high temperature reactor plant with a pebble bed core in the Federal Republic of Germany.

Nuclear commissioning started in August 1983 with loading of the core. On September 13, 1983 the reactor reached a self-sustaining nuclear chain reaction for the first time. The hot functional tests began one year later. After a very detailed test programme on different power levels 100% power was achieved on September 23, 1986. Before handover to the owner company on June 1, 1987, operation of the power plant in accordance with the terms of the supply contract was demonstrated.
Up to September 1988 the THTR 300 had generated 2.89 billion kWh.

1.2.1 Plant concept and plant design

The physical design of the reactor core and the large pebble bed reactor as a whole had been verified. This applies not only to the reactivity balances but also to the dynamics and the thermodynamics of the reactor core. There were deviations in the flow behaviour of the actual pebble bed from the results of model experiments which could have been compensated by adapting the fuel circulating strategy for further operation, but which certainly require detailed investigation and better understanding with a view to future pebble bed cores.

The mechanical behaviour of the freely insertable incore rods corresponded to the predicted behaviour based on experiments. Even under extremely unfavourable conditions all 42 incore rods could be inserted into the pebble bed, reliably putting the reactor into a long-term cold, subcritical condition.

Already in the commissioning phase, an increased number of damaged spherical fuel elements appeared. Analysis work indicated that this was caused by core rod testing beyond specified operation instructions. During further operation the number of damaged elements decreased. The primary coolant gas activity did not change, from which coated particle integrity can be concluded.

The fuel elements behaved as specified. This applied in particular to the activity release rates. The special safety characteristics of the high temperature pebble bed reactor have been confirmed by tests and during power operation in an outstanding way.
The fuel circulating system had proved its functional capability in the course of continuous operation and by circulating 1.3 million spherical elements up to July 1988. Improvements regarding its control and coping with damaged spheres had been successfully implemented.
During commissioning the spheres circulation rate was reduced with increasing reactor power due to an unfavourable routing of the coolant bypass flow through the damaged spheres separator. With remote handled repair, the flow conditions were improved and the required spheres circulation rate was met.

The thermodynamics of the primary system corresponded to the design within the limits set by the best possible calculations and measurements.
The reactor components - pre-stressed concrete reactor vessel, shutdown rods, steam generators, helium circulators - also

behaved in accordance with the design.

1.2.2 Operating properties

The operating data of the power plant correspond to the design data. The warranted net power output of 296 MW was reached precisely. The heat rate of the secondary system was better than warranted. The thermal efficiency of the power plant was thus 40.2%. This corresponds to the data of modern conventional power plants.

The control response of the THTR had been optimized so that the requirements of the operator company regarding load changes and frequency back-up control could be met perfectly. Contrary to other nuclear power plants, the THTR 300 could be operated in the load-following mode (weekly load cycle) without jeopardizing the fuel elements.
It has been demonstrated that dismantling and repair work at the primary system can be performed without major effort.
Time availability was 61.7% in 1987 and 52.4% in 1988. Experience with in-service inspections has shown that there is still a considerable potential for further reducing the scheduled downtimes.
In fall 1988, during a planned shutdown in order to perform scheduled in-service inspections, several broken bolt heads from bolts fixing the cover plates on the insulation in the hot gas ducts were found. Detailed investigations revealed that this was caused by reduction of the ductility of the bolt material by exposure to thermal neutrons, together with an overloading of the bolts by secondary forces from different thermal expansion behaviour of the 18-layer metal foil insulation.
Redundant bolt-fixing of cover plates prevented the structure from deterioration. There was, however, unanimous agreement that the damage had no safety relevance.

The plant has now been shut down since the end of September 1988. At present preparatory work for the decommissioning licence is being carried out. The reasons for the shutdown are not related to technical or safety aspects. They are rather hidden in the complexity of the project organization, the limited financial capabilities of the utility shareholders, open end cost figures for decommissioning and dismantling of the plant.

1.3 Experience gained with Research and Development

Beside the experience derived from plant operation, the broad R+D programme brought significant additional progress.
A few examples of major achievements for utilization in THTR following-up plants are:

- Production-scale fuel manufacturing process development and qualification of the reference fuel is completed and proven - a most important prerequisite for the new safety concepts.
- Improved new concrete qualities for the reactor vessel have been developed.
- The feasibility of magnetic bearings for blowers has been demonstrated, enabling the industry to perform a full-size blower demonstration test.
- Significant progress has been made in establishing HTR design criteria.
- Promising tests are under way to demonstrate the feasibility of welded transitions from austenitic primary component materials to ferritic secondary circuit materials under specific HTR conditions.
- Special design measures for thermal expansion compensation of different core structures, as well as for hot gas mixing, have been successfully tested.
- Thermal core bottom structure movements for large HTRs and core-reflector interactions have been experimentally investigated, supplemented by respective code verifications.
- The core behaviour under earthquake conditions has been analysed in a large test facility and showed very satisfactory behaviour.
- The develcpment of non-lubricated ball-bearings for long-term operation in HTR conditions resulted in good progress.
- In addition to the excellent characteristics of the fuel elements in operating conditions, their qualification for direct final disposal has been proved.

With the intention of covering the specific requirements of the process heat market, appropriate components have been developed and tested on large scales:

- hot gas ducts including hot gas valves for 950/900°C and helium.
- 2 He/He heat exchanger modifications (helix and U-tube type) have been tested with a capacity of 10 MW.
- By investigating heat supply direct to chemically reacting gases, 2 helium-heated steam reformer modifications composed of original-size reformer tubes have been tested successfully in HTR conditions.
- R&D for special components was accompanied by a comprehensive materials testing programme for the relevant alloys.

2. The future of the HTR development

The AVR and the THTR 300 are two impressive engineering achievements. We must, however, be quite clear in our minds about the fact that these technological successes have not led to a commercial breakthrough on the market. The high temperature reactor is overshadowed by its big brother, the light water reactor. All the nuclear reactors in commercial operation in the Federal Republic of Germany are light water reactors. The LWR also leads worldwide and its dominant position has recently been further consolidated.

The existence of this big fleet of light water reactors, whose operation is both economical and reliable, poses major obstacles in the way of the entry on the market of all new reactor types.

On the other hand, we must also remember that,in the 1980s, nuclear energy development stagnated; This situation has changed with the beginning of the 1990s as regards energy policy and environmental policy. If during the last decade of this century power stations in the Federal Republic of Germany have to be replaced, then this will be an opportunity for the revaluation of the different reactor lines.

2.1 The general conditions governing energy policy and environmental policy

Whereas during the 1970s energy policy - and also energy research - was coloured by the notion that the energy reserves available worldwide would one day be depleted, the most important limiting factor today is the environment. We must, however, realize that both the assumed shortage of resources and also increasing environmental problems exercise only a very indirect influence on investment decisions. The first consideration for enterprises is the profitability of their investments.
With regard to the development of the high temperature reactor, it is my impression that this aspect is all too frequently pushed aside because the highlight is on the elegant technological solutions and the outstanding physical properties of this reactor type.

This approach does not lead to success. A new reactor design must be economically more efficient in its sphere of application than alternative reactor lines, both nuclear and conventional. At the same time, this advantage as regards economic efficiency must be far more than marginal. Indeed, it must be quite considerable in order to justify the investment risk involved in introducing a new reactor line.

Let us think back to the beginning of the development of nuclear engineering. At that time, it was hoped to produce electrical power at practically no cost. As we all know, these expectations have not been fully fulfilled, but I mention this because we should continue to bear in mind just how high the expectations were which triggered off the major advance in nuclear energy development in the 1960s.

Much of this dynamism has been lost. Formerly outstanding economic advantages, which were not doubted for a moment, have dwindled. They must be proved anew and are being regarded with scepticism. A study carried out within the framework of the OECD in 1989 on the subject of "Projected Costs of Generating Electricity" shows clearly that the utilization of nuclear energy is not economically efficient in all the Member States.

The observation of a high safety standard, which rules out any possibility of nuclear disasters and a threat to the population, constitutes a basic prerequisite for the application of nuclear engineering.

The challenge of providing a markedly less expensive supply of energy while at the same time fulfilling the safety prerequisite constitutes the problem confronting the high temperature reactor - it is the challenge which will have to be met if this reactor type is to continue to exist.

I think it is necessary to face this challenge if this technology is to become commercially successful.

A strategy aiming to find a niche in one or the other special case in which a high temperature reactor offers advantages as a result of the combination of particular circumstances is doomed to failure because this is a technology which is, after all, expensive and which involves a great deal of preliminary work and investment.

2.1.2 Fields of application for high temperature reactors

Two features characterize the high temperature reactor:

- Owing to its safety engineering characteristics, this reactor can also be constructed in small units without the loss of too much economic efficiency
- Owing to the high temperature of the cooling gas, the reactor lends itself to co-generation, and the process steam can be used for industrial purposes.

Both characteristics are complementary: industrial power stations and combined heat and power stations for the public power supply require smaller units than big light water reactors with a capacity of 1,300 MW. The HTR's different safety performance, perhaps even greater safety, can make it easier for the public to accept a site for such a reactor in the vicinity of residential areas, and can also facilitate the integration of such reactors in an industrial plant in full compliance with safety requirements.

The decoupling of heat from a power station entails the loss of electrical power. The worse the steam conditions are, the greater the loss of electrical power. A light water reactor is therefore put at a greater disadvantage than a high temperature reactor. District heating systems are already being decoupled from light water reactors.

Despite the many technological and physical advantages of the high temperature reactor in this respect, it must also reveal economic advantages. These have yet to be proved and demonstrated.

With regard to industrial process heat, the light water reactor is no competitor for a simple technical reason: its temperature level is too low. If nuclear energy is to be used to this end, then the high temperature reactor is needed.

An analysis of the industrial processes for which high temperature reactors could be used to advantage has shown that three sectors can be considered:

- refineries
- steam-flooding of heavy oil deposits
- aluminium production works

The high temperature reactor is economically efficient in special cases, but not generally. In the case of the above-mentioned applications, practically 100% availability of the energy supply is required in order to make full use of the capacity of the production facilities, which cost thousands of millions of marks. For this reason, the high temperature reactor will probably not be used in this connection for the time being, but later on following the demonstration of its efficiency elsewhere.

2.2 Fields for technological development

In the Federal Republic of Germany, the high temperature reactor technology is being advanced in the following three product lines:

- the HTR 500
- the modular reactor with a capacity of 200 MW thermal

- the heat and power-generating reactor with a capacity of 10 or 20 MW thermal.

In this connection, the modular reactor attracts great attention because the HTR-specific advantages are clearly revealed.

The development in each case has matured to the extent that the next stage of construction would be to build a prototype facility. So far, however, no operator could be found for such a facility. The research and development activities have therefore been designed in such a way that they can be used in the main for all the three product lines mentioned above.

I should now like to speak about several priorities of the development activities.

2.2.1 Safety

The development activities give priority to the continued improvement of the safety of the reactor system. Unlike light water reactors, for which it was often necessary to integrate subsequent safety requirements in existing technology, the high temperature reactor can build on the experience acquired in the course of many years with regard to reactor safety.

In view of the high degree of inherent safety, the focus is on the performance of the entire facility, which cannot be checked on an experimental basis. For this reason, particularly pains-taking investigations are required. In addition, probabilistic safety analyses are also necessary.
The Technical Inspection Service as the adviser for the licensing authorities drew up an expert opinion on the modular HTR reactor, independent of the site envisaged. Apart from demonstrating the capability of the reactor of receiving a license, the expert opinion was to provide proposals for a further improvement of its safety performance.

The safety analysis will also cover the scenarios for serious accidents, in order to be able to demonstrate to the population the reactor's safety in the event of such accidents.

In addition to ensuring the safety of the population, efforts are being directed to further reducing the reactor staff's already low level of radiation exposure.

The safety of the high temperature reactor depends decisively on the fuel element. In this connection, further improvement of the good retention capacity of the particles and the corrosion resistance of the fuel elements is considered possible. The aim is to develop a corrosion-resistant fuel element, which will not be damaged even by serious accidents involving the ingress of water and air of high temperatures. A start has been made on the initial measures for a corresponding developing programme and steps for its extension have been introduced.

2.2.2 Raising the temperature

The HTR is distinguished from other reactor designs not only by the safety-related aspects, but also by the fact that it operates at a high temperature. During the course of HTR development, the original expectations for temperatures above 1,000°C had to be modified chiefly on account of the limited resistance of high temperature materials. Raising the temperature continues to be a goal with a view to extending the range of applications for the HTR.

2.2.3 Economic efficiency

I have already emphasized the importance of the economic efficiency of the HTR for its future. This must be reflected in the relevant development activities. The disadvantage that worldwide not one commercial HTR is on the order books should - turned into an advantage - have the result that technological concepts will not be prematurely shelved unnecessarily.

Economic advantages may be found in other core configurations or, perhaps, also in another fuel element. The advantage of low cost final storage with corrosion-resistant fuel elements should be used to the full.

Perhaps a direct-cycle plant equipped with a helium turbine will be feasible if the fuel element has a very high retention capacity for fission products.

The feedback for the safety concepts will have to be studied. If the systems are not overloaded, there will be conservative characteristics which can be reduced.

This is another field offering a great deal of scope for theoretical and experimental studies.

2.2.4 Final storage

The changed energy supply situation worldwide and low prices have triggered off a discussion on the economic efficiency of the reprocessing of nuclear fuel as opposed to direct final storage. In view of this situation, the higher burn-up rate which can be achieved with the high temperature reactor and the suitability of the ceramic fuel elements for final storage acquire a new importance. This may result in a more favourable estimate of the fuel cycle costs for high temperature reactors.

Studies are being conducted on fuel element behaviour during storage operations at the repository as well as on long-time behaviour.

The decommissioning of the AVR and the THTR 300 will yield practical experience with the deloading of the pebble-bed core as well as with the transport and interim storage of fuel elements.

2.2.5 General activities

The high temperature reactor technology is still in the process of development. Apart from the feasibility of the construction of an HTR, which has been demonstrated in the Federal Republic of Germany and is now being demonstrated in Japan, there are still several possibilities for improving the design application of advanced materials and improved components.

2.2.6 International cooperation

Close international contacts have been established between the institutions, the firms and also the scientists and engineers who are working on high temperature reactor technology. This is clearly reflected by this conference. This international cooperation is both sound and necessary. It strengthens the know-how basis considerably and stabilizes HTR development worldwide.

We must, however, not disregard the fact that the high temperature reactor technology is now fully developed and has left the domain of science. In the 1990s, high temperature reactors will have to be constructed on a commercial basis if they are to continue to exist as a reactor line.

An event such as this conference should also be used to sound out the ways in which the relevant international cooperation can help us to advance towards the attainment of this goal.

2.5 PRESENT STATUS AND DEVELOPMENT STRATEGY OF THE HTGR PROGRAM IN PRC

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ABSTRACT

This paper summarizes the present status and main progress of chinese HTGR R&D program. It is being carried out in the INET and the relevant institutions and covered in the national high technology research and development program. Briefly introduces the technical features of the 10MW HTR-Test Module which will be a joint project with Siemens-Interatom, KFA Juelich. Some preliminary results and prospects of HTGR application in the heavy oil recovery and petrochemical complex are also described. At last, this paper discusses the role and position of HTGR in the future energy system and its development strategy in China.

1. The Status of HTGR Program

The research and development program of HTGR in China began in mid seventies. In the first phase begining from 1974, the target was to develop a 100MWt HTGR thorium thermal breeder with two-Zone pebble bed core. In parallel with the design work, a series of research and development work for HTGR key technologies and components were carried out. But this project was stopped in end of seventies due to various reasons mainly in financial problem.

The second phase of HTGR research and development program was in period of the Sixth Five-year plan (1981-1985). The State Science and Technology Commission gave support to continue some basic technology development and to investigate the possibilities of nuclear process steam and heat for industry application.

The present research and development program on HTGR is covered in the high technology research and development program since 1987. Considering the development of nuclear energy in the next century in China, it is necessary to develop the advanced reactors which have inherent safety, economical viability and highly fuel utilization. The high temperature gas cooled reactor is the one of these advanced reactors to be developed for future applications.

The main topics and progress of HTGR program are as following:

1) HTR fuel technology development:

Two types of Coated Particle have been studied. One is BISO CP with 200μ carbonized resin-high enriched U(O,C) kernel, the other is TRISO CP with 500μ low enriched UO_2 kernel.

External gelation process is used for kernel preparation. Laboratory scale facilities have been set up, kernels which meet the design specification can be made now.

Coating is carried out in a fluidized bed. Pre-irradiation characterization shows that the quality of the CP is satisfying. Irradiation qualification is just started. First screen test has been carried out in the reactor of IAE, Beijing. The main parameters of the test are as following.

nth	1.01×10^{14}n/sec·cm^2
temperature	<300°C
burn up	0.39% fima, 0.78% fima

Post-irradiation annealing experiment at 1100°C shows that the fission product release rate R/B of I and Xe is promising. Specification and reference test will be started on the basis of screen test later on.

The R&D of fabrication technology of spherical fuel element was started in 1987. A fuel element manufacture facilities on laboratory scale was set up. Last year we concentrated our effort on:

- Raw material, especially natural flake graphite.
- Metrix preparation
- Over-coating technique
- Forming technique
- Heat treatment of green fuel element

2) Graphite, metallic material and thermal isolated materials development.

- Investigation of the characteristics of the domestic nuclear graphite.
- Corrosion test of graphite samples in air and different conditions.
- To develop the heat resisting alloys for the steam generator tube and other structural applications.
- Investigation of the ceramic thermal isolated materials.

3) Helium technology and components development

- A high temperature helium test loop has been designed, the helium heaters have thermal output of 1MW and heat helium up to temperature 750°C in the first operation phase. Part of the components are in fabrication and preparation.
- A small helium purification test loop has been set up and operated in last year.

4) Charging and discharging technique for spherical fuel elements. A full dimensional test facility of 10MWt HTR-Test Module for verification of components and investigation of the pebble bed flow behaviour has been designed. This test facility is expected to be set up in 1990.

5) Research work on the thorium-uranium fuel cycle and HTGR fuel reprocessing technique. This is a long-tern development program based on use of thorium resource in the future. The main research work and progress in recent years as follows.

- Simulation of flow of fuel ball in the fixed bed burner with naphthalene, the burner and test loop had been designed at set up, burning experiment with single ball has been performed. The experiment shows that burning rate is about $0.09 g/cm^2 \cdot hr$.
- The double-roll fuel particle crusher, the secondary burner, the system of CO_2 and O_2, the pneumatically transportation system as well as a zigzag column have been designed and manufactured.
- The dissolution of ThO_2 kernel was studied in the closed pressurized vessel with Thorex reagent. A dissolution facility for cold experiment with output of 0.5kg/day was designed.
- The single cycle solvent extraction process with acid feed solution has been chosen. The laboratory experiments have been carried out, thorium and uranium are recovered to more than 99.90%. The decontamination factors are $>10^3$.

6) Application studies of the modular HTGR for different potential users e.g. heavy oil field, petrochemical complex and for local power plant. The preliminary results have been carried out, the details will be discussed later.

2. The Project of 10MW HTR-Test Module

In order to introduce and develop the modular HTR technique in practice, 10MW Test Module will be built at the site of INET in the north-west of Beijing. This is a joint project which collaborated among the Institute of Nuclear Energy Technology (INET), PRC, Siemens-Interatom GmbH and the Nuclear Research Center, Juelich (KFA), FRG.

The main object of the Test Module Reactor is to verify and demonstrate the Unique features of HTR-Module on a real nuclear test facility. Therefore, the aims for the Test Module have been defined jointly as follows:

- The test Module will be designed for a wide range of possible applications, e.g. electricity, steam and district heat generation in the first operation phase and process heat generation, methane reforming in the second phase.
- The relevant components can be tested and proven at nominal conditions, e.g. graphite core structures, steam generator, belium blower and fuel handling facility.
- Verification of the inherent safety features of the HTR-Module such as negative temperature coefficient of reactivity, temperature limitation due to passive decay heat removal and limitation of power excursion due to water ingress.
- The test Module is capable to withstand extremely high core temperature, so that fuel element mass-test could be carried out for nominal reactor conditions at temperature up to 1600°C.

The conceptual design of the 10MW Test Module was carried out jointly by Interatom and INET in 1988. The main data of the Test Module are as follows:

Maximum thermal power	20 MW
Normal thermal power	10 MW
Primary helium pressure	30 bar
Helium temperature	250°C/700°C
Secondary steam pressure	35 bar
Life steam temperature	435°C
Power density	2 MW/m^3
Core diameter	190 cm
Average core height	176 cm
Height/diameter radio	0.93
Number of fuel element	27,000
Heavy metal content	5 g/fuel element
Average burn-up	80,000 MWd/t
Fuel element incore time	1078 EFPD
Loading scheme	OTTO

Figures 1 and 2 show the vertical cross section of the reactor and the primary circuit of the Test Module. It has the following important design features:

The reactor core and steam generator are housed in two separate steel vessels and positioned side by side in a staggered arrangement. The graphite cylinder core which holds the pebble bed has a diameter of 1900mm and a height of approx. 2200mm. This is adequate for the required volume of the active core which amounts to $5m^3$.

The new fuel elements are charged from the top of the core via five charging tubes. The fuel elements are removed from the core-bottom via a fuel element discharge tube with an inner diameter 500mm.

Decay heat is removable by surface coolers outside the reactor vessel. The surface cooler system is subdivided into two trains. It is sufficient to dissipate decay heat by means of passive heat transfer mechanisms to the simple surface coolers.

The primary helium pressure of 30 bar is chosen, the inlet and outlet temperature of the core are 250°C and 700°C. A secondary pressure of 35 bar and a life steam temperature of 435°C are chosen so that the small standard industrial turbines can be used.

The average thermal power of the Test Module plant is set to be 10MW. In order to enhance the experimental flexibility the maximum thermal power output is set to be 20MW.

Some variants of the secondary heat sink have been evaluated. As sample Fig. 3 shows the heat flow diagram for district heating and electricity generation with the maximum thermal power of 20MW.

Figure 4 shows the time schedule of the whole cooperation programme. The 10MW Test Module will take 5 years for design, construction, installation and commissioning.

3. The Application Studies of HTGR in Heavy Oil Recovery and Chemical Industry

1) Heavy oil recovery application:

The heavy oil reserve is relatively rich in China, since begining of '80, heavy oil recovery by injecting steam had been developed in several oil fields in order to increase the crude oil production. But conventional thermal recovery of heavy oil needs a great quantity of high temperature steam with high pressure. About 30-40% of produced crude oil would be consumed to supply such amount of injecting steam.

In some pilot areas of thermal recovery, the injecting steam with temperature 355°C and pressure 170 bar is produced from the small oil-fired boilers. Therefore, using HTR instead of oil-fired boilers can save a great amount of crude oil.

For the investigations on the use of the HTR in heavy oil recovery. The Shanjasi section of Shengli oil field has been selected to serve as reference case for this study. The initial OIP in Shanjasi section is 66×10^6 tons, but the exploration is not finished and 100×10^6 tons of OIP is expected. The crude oil is very heavy and does not flow at reservoir conditions. Therefore, the heavy oil has been extracted by the steam-soak process since October 1984. The production planning aims at an output of 1 million tons heavy oil per year with subsequent upgrading in a special refinery.

The main aims of the application study are to find out whether

- the physical properties of the Shanjasi reservoir are suited for a continuous steam injection, i.e. oil recovery by steam drive process.
- the nuclear steam is economic compared to conventional steam generated by oil-fired boilers.

The evaluation of the physical properties of Shenjasi reservoir with respect to the steam drive process was a major effort of the study. For calculation of the reservoir properties a numerical simulation computer code (NUMSIP model) has been developed by INET. Base on the results of calculation and evaluation, a option of using 2 HTR with total thermal output 400MW for steam and electricity generation was proposed.

As it is shown in Fig. 5 the oil production capacity of 1 million tons per year (about 20,000 bbl/d) may last up to 13 years with steam soak and partly steam drive. Then steam soak will decrease, and the production by steam drive will be remained for other 20 years. It means in total a duration of about 33 years with a nearly continuous oil production of 0.5 million tons per year. According to preliminary investigations a ratio of 4 tons steam to 1 ton oil can be expected. With this boundary condition a steam production of about 2 million tons per year (about 6000t/d=250t/h) is needed. This steam amount can be generated by one HTR-module with a capacity of 200MW thermal. Another HTR-Module is necessary for electricity generation to meet the electricity demand in oil-field area. Therefore, 2×HTR-module plant is proposed as a energy source for the Shanjasi heavy oil-field. Figure 6 represents the flow scheme for such 2×HTR-module plant.

It is proposed to interconnect the feedwater steam circuits of both HTR-modules to get a cogeneration plant generating injection steam for heavy oil recovery and electricity for the Shengli grid. Each steam generator generates about 77kg/s live steam with a pressure of 190 bar and temperature of 530°C. From there the steam is distributed to injection wells and the turbine. The electrical output is about 75MW. The steam leaving the turbine at different extractions is partly used for the preheaters to form a part of the feedwater. The other part of feedwater (with 70kg/s) is fed from outside via a water treatment station.

The economic assessment is performed on the basis of the dynamic cost calculation. The most important results of this application study are as following:

- Recoverable oil by steam drive using HTR nuclear steam supply system: approx. 15 million tons in the total period of about 30 years.
- Average oil production by steam drive: approx. 0.5 million tons per year.
- Additionally recoverable oil due to oil substitution by nuclear energy: approx. 4 million tons (25×10^6bbl)
- 75MWe of electricity could be supplied to the Shengli-oilfield grid. The electricity production amounts to 600 million KWh per year which is equivalent to an oil consumption of about 140,000t/a.
- Electricity produced by the HTR-module plant will have the near same price (levelized cost) as produced by an oil-fired power plant.

The application study of use HTR in heavy oil recovery has been carried out jointly by INET, Beijing and KFA, Juelich, and supported by Shengli oilfield Co. in Chinese side and Siemens KWU/Interatom in German side. It is well understood the Shanjasi HTR application study is a preliminary evaluation in the technical and economical possibilities. But it can be regarded as an example and other heavy oil field with similar properties may also be possible candidates for HTR application project.

2) Chemical industry application:

The similar investigation has also carried out for use of the HTGR in Chemical industry. The Yangshan Petrochemical complex located in Sough-West of Beijing in a distance of about 55km is selected as a reference user for this application study.

In recent years, the annual energy consumption for supplying steam, process heat and part of electricity in this huge petrochemical complex is in rank of 1.2 Million tons of oil. Considering use of an HTGR nuclear energy source instead of the conventional oil fired boilers seems to be attractive and beneficial, due to saving oil consumption is an important policy at moment and in the future in China. Total requirement of steam in different pressure and temperature ranges is approx. 730t/h in Summer and 1650t/h in Winter. The Steam parameters are 118bar/525°C, 47-50bar/450°C, 34-39bar/350°C, 8-13bar/280°C, and 3bar/133°C. The way of steam supply is mainly by steam-electricity combination production (oil fired co-generation plant). Consumption of electricity was about 1.2 billion kWh in the year of 1987, total electricity supply capacity in this area should be upto 120MWe.

Considering the steam demand and the electric power requirement (see above) at least a 4 HTR-Module Co-generation plant is necessary with the following main parameters:

- Thermal power output 4×200Mwth=800Mwth.
- Live-steam massflow approx. 4×250t/h=1000t/h(about 278kg/s)
- Live-steam pressure/temperature: 190bar/530°C.

Four HTR-Modules supply the steam for operating three back pressure turbines and for supplying four process steam systems. Total electric generator output is 139MWe, the parameters of process steam systems are: 118bar, 500°C, 30.24t/h; 48bar, 450°C, 73.08t/h; 36bar, 350°C, 310t/h and 10bar, 280°C, 500t/h, total process steam massflow is about 915t/h.

The tertiary systems for all of the four process steam systems are separated by heat exchangers from the secondary circuit so that radioactivity possibly contained in the secondary circuit is not able to pass through to the process steam of the petrochemical plants.

This application study has been carried out by the joint work between chinese side (INET and YSPC) and German side (KFA, juelich and Interatom GmbH). More detailed economic and safety studies for application of HTR-Module in the Chemical industry will be continued.

4. The Development Strategy of the HTGR in China

The important role of HTGR, in particular modular type HTGR, in the future energy system of China is based on its inherent safety futures, fuel cycle flexibility and highly fuel utilization, potential for high efficiency electricity generation or co-generation ofi process steam and process heat for broad industry users. Therefore, HTGR will plaÿ three important roles in the future energy system, they are:

- To provide a pollution-free heat source up to temperature 950°C for large energy-consuming centers to improve environmental quality and to save fossil-fuel consumption.
- To provide the electricity with high efficiency in special regions where shortage of energy sources, difficulty of transportation and serious environmental pollution.
- To provide the possibilities of use of thorium resources from a long-term point of view.

The HTGR development and application in China can be considered to divide into three stages:

- The first stage, up to the end of this century:

The main goal of this stage is to develop the key technologies of the modular HTGR and to construct a HTR-Test Module reactor. Meanwhile we expect to construct a pilot modular HTGR plant if the international cooperation is favourable and the domestic financing is possible.

- The second stage, at beginning of next century:

The goal will be to promote the commercial modular HTGR applications. In this period, the HTGR which has helium temperature 750°C can be used as co-generation plant to provide process heat and electricity for the enhanced oil recovery, petrochemical complexes and large energy-consuming industries. The potential market of HTGR in this field will be tremendous. According to the energy forecast, the annual heavy crude oil production will reach 30 Million tons and the annual production capacity of the total refinaries will reach 200 Million tons around the year of 2020. The energy demand of these two industrial fields provides huge potential market of at least 36 GW (thermal), it needs about 180 HTR-Module for generating process steam and electricity.

- The third stage, up to mid of next century.

The process heat generating HTGR (with temperature 950°C) will be spreaded in commercial market in this period. The main application might be the use of the high temperature process heat of HTGR for coal gasification and liquefaction to substitute part of coal consumption and to produce the clean and high calorific value gaseous and liquid fuel.

This would result to partially solve the shortage problem of the liquid fuel (e.g. crude oil) supply in that time. According to estimation of the energy demand in the year 2050, the total demand of liquid fuel (crude oil etc.) reaches to 460-530 Million tons, but the annual crude oil production decreases to 120 Million tons, therefore, the shortage of 350-400 Million tons of liquid fuel should be solved. The coal gasification and liquefaction will be a important way to fill in the gaps between the liquid fuel demand and the natural crude oil supply. If use of HTGR process heat to convert coal and produce 400 Million tons (equivalent crude oil) of synthetic fuel, it needs about 470 Gw (thermal) of nuclear heat. Meanwhile, nuclear heat-coal convention can save a tremendous amount of coal compare with the conventional coal gasification and liquefaction. The high temperature nuclear process heat for coal gasification can only be provided from HTGR, therefore, this application of HTGR could not be substituted by other reactor types.

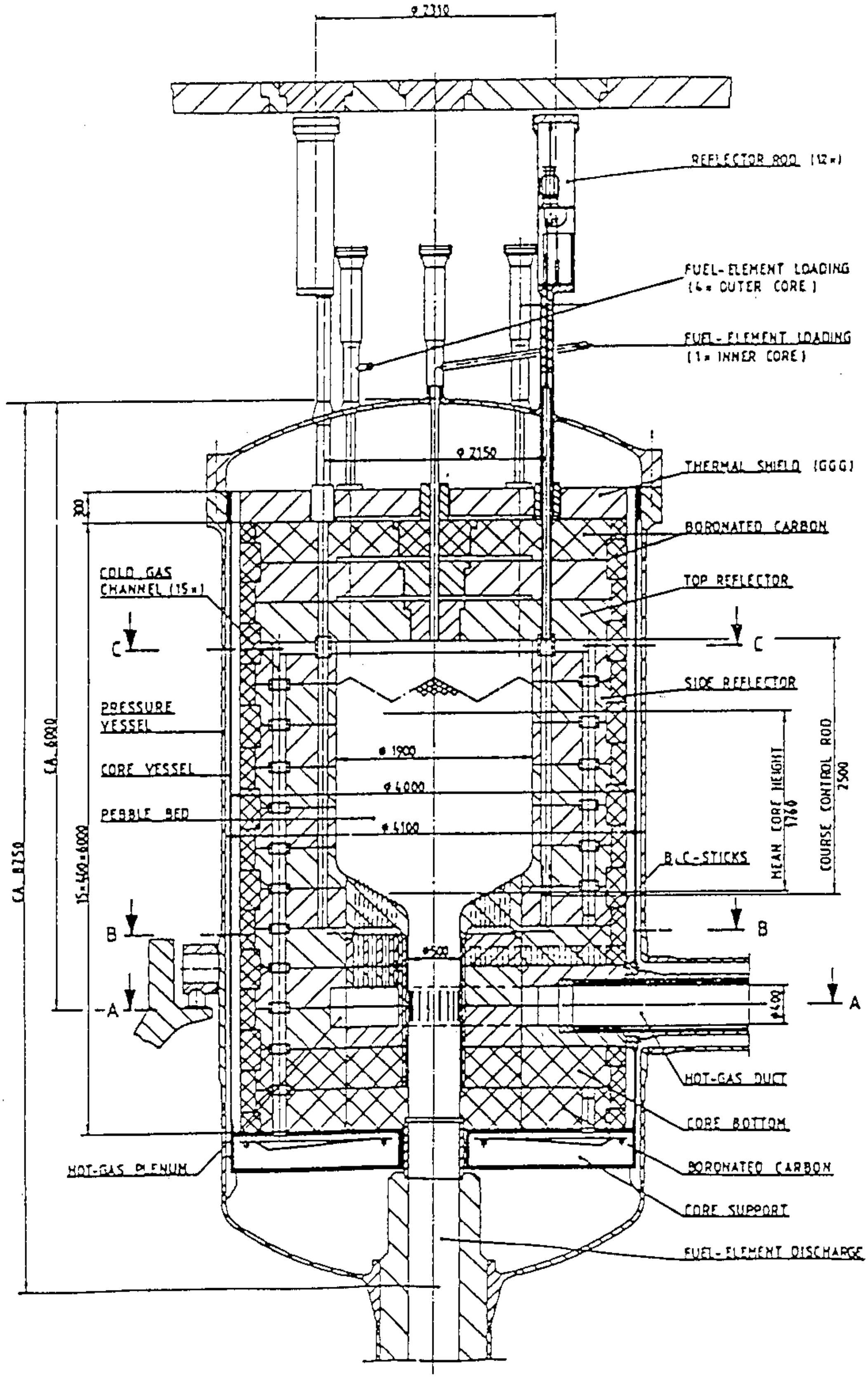

FIG. 1 VERTICAL CROSS SECTION OF THE TEST MODULE

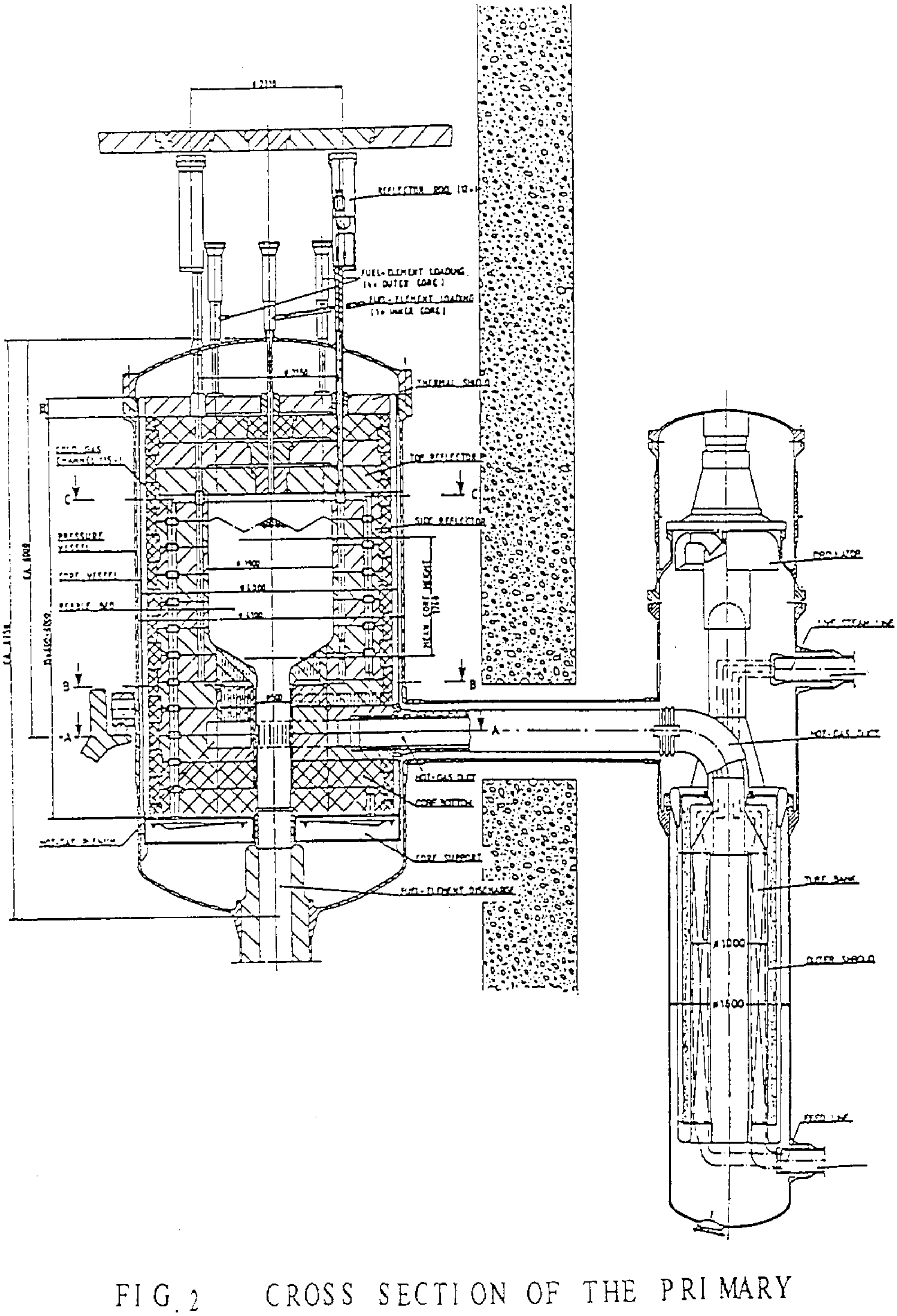

FIG. 2　CROSS SECTION OF THE PRIMARY CIRCUIT OF THE TEST MODULE

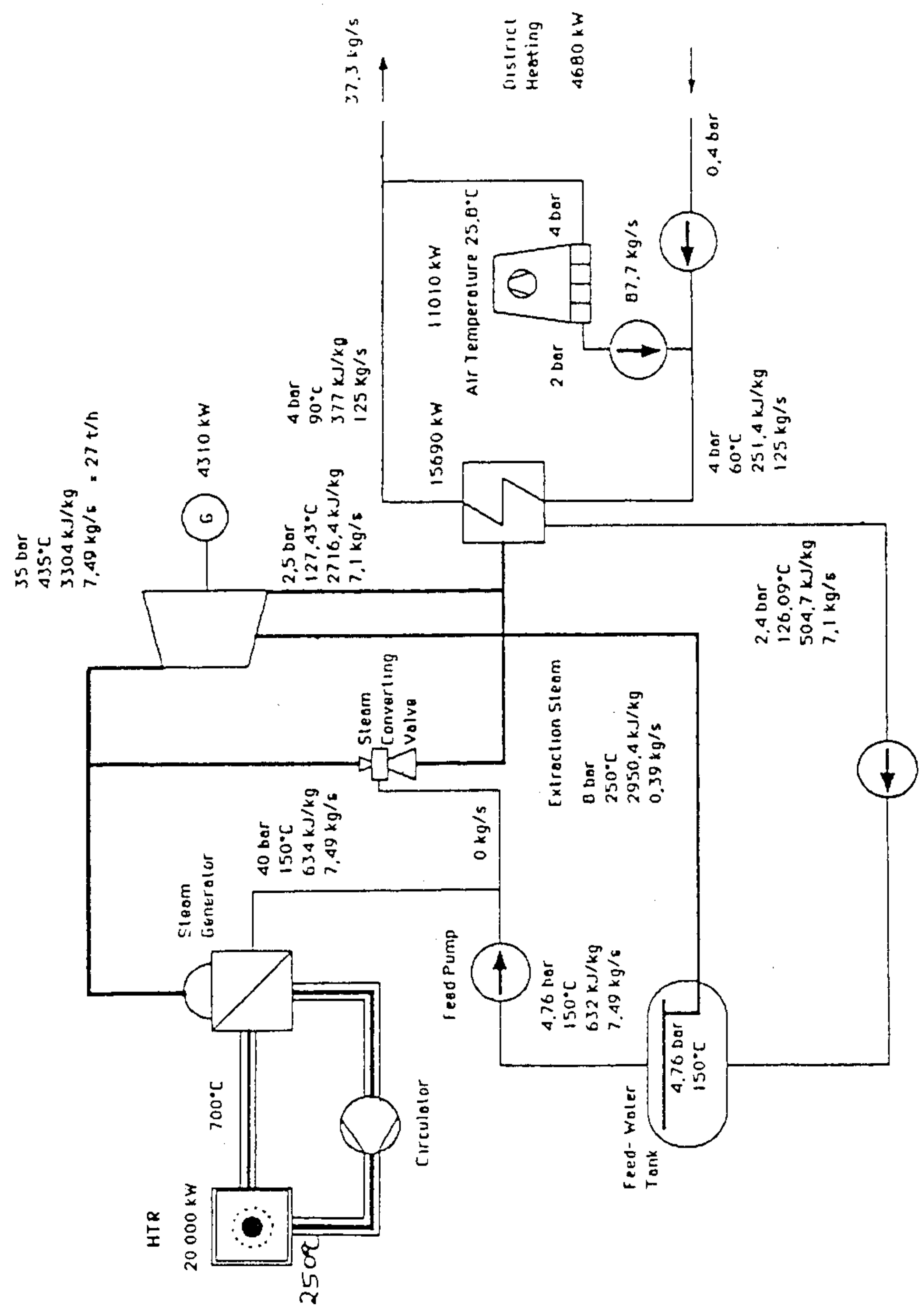

FIG. 3 HEAT FLOW DIAGRAM
- DISTRICT HEATING AND POWER GENERATION

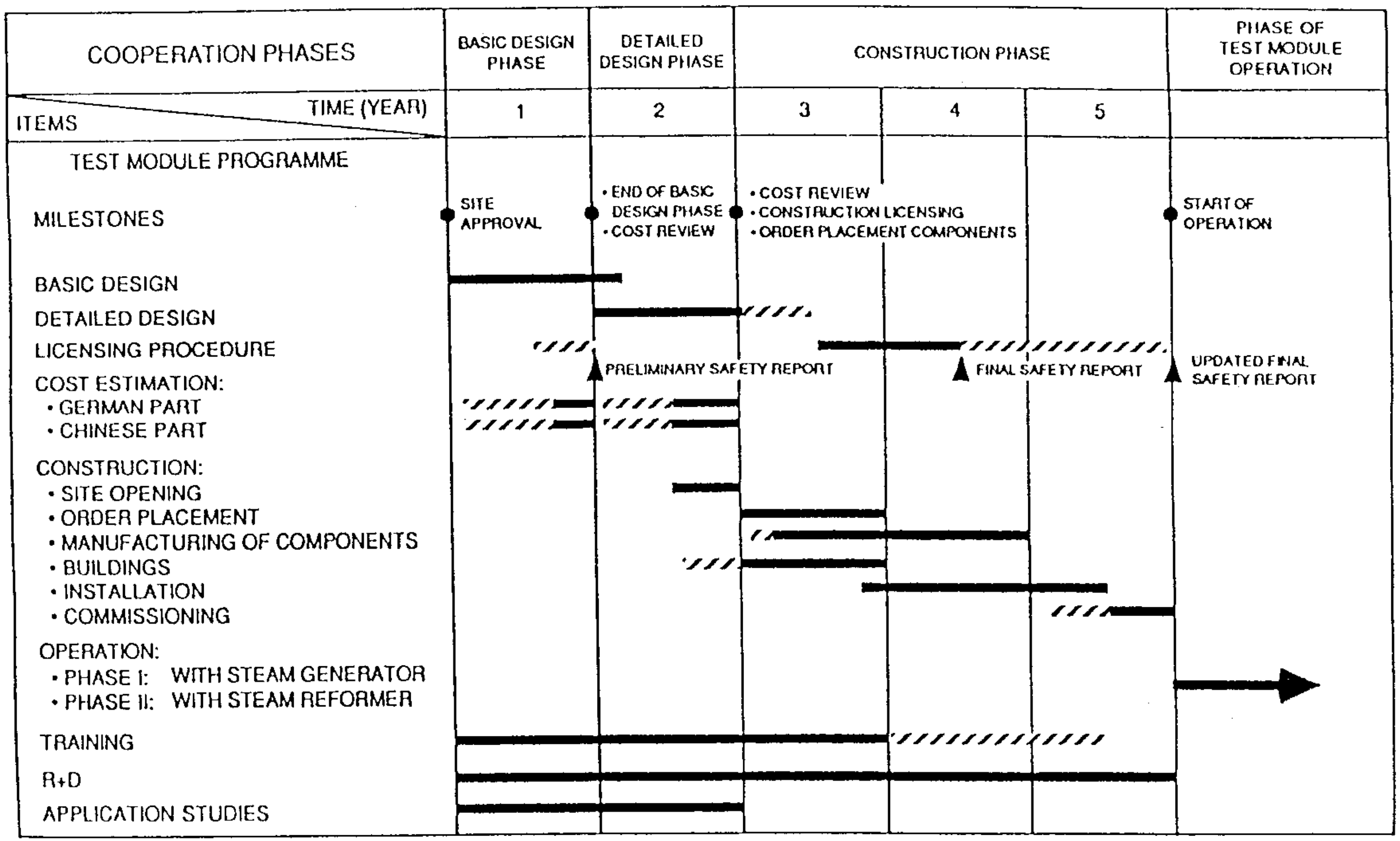

FIG. 4 OVERALL TIME SCHEDULE TEST MODULE CHINA

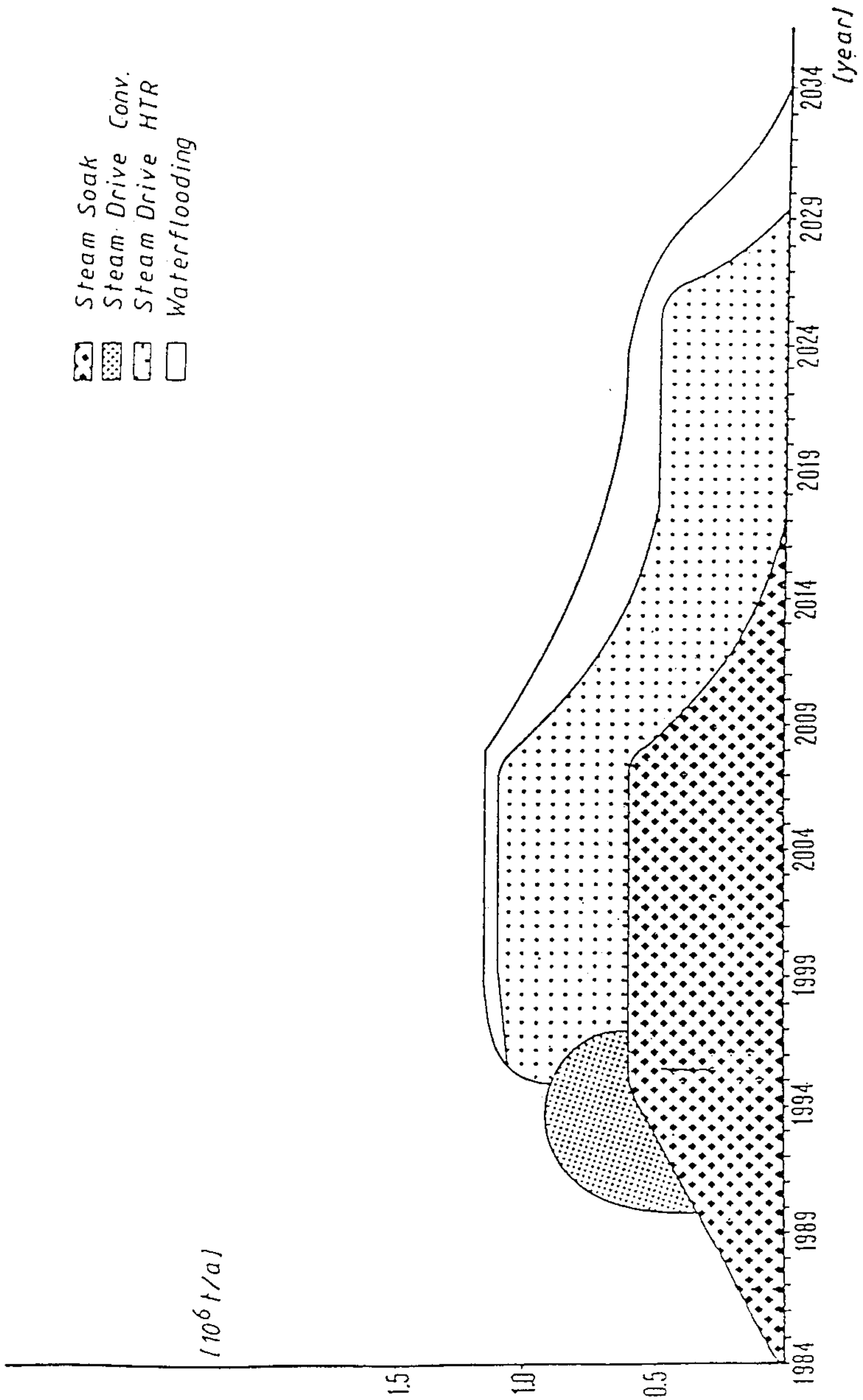

FIG.5 PRODUCTION SCHEDULE AT SHANJASI

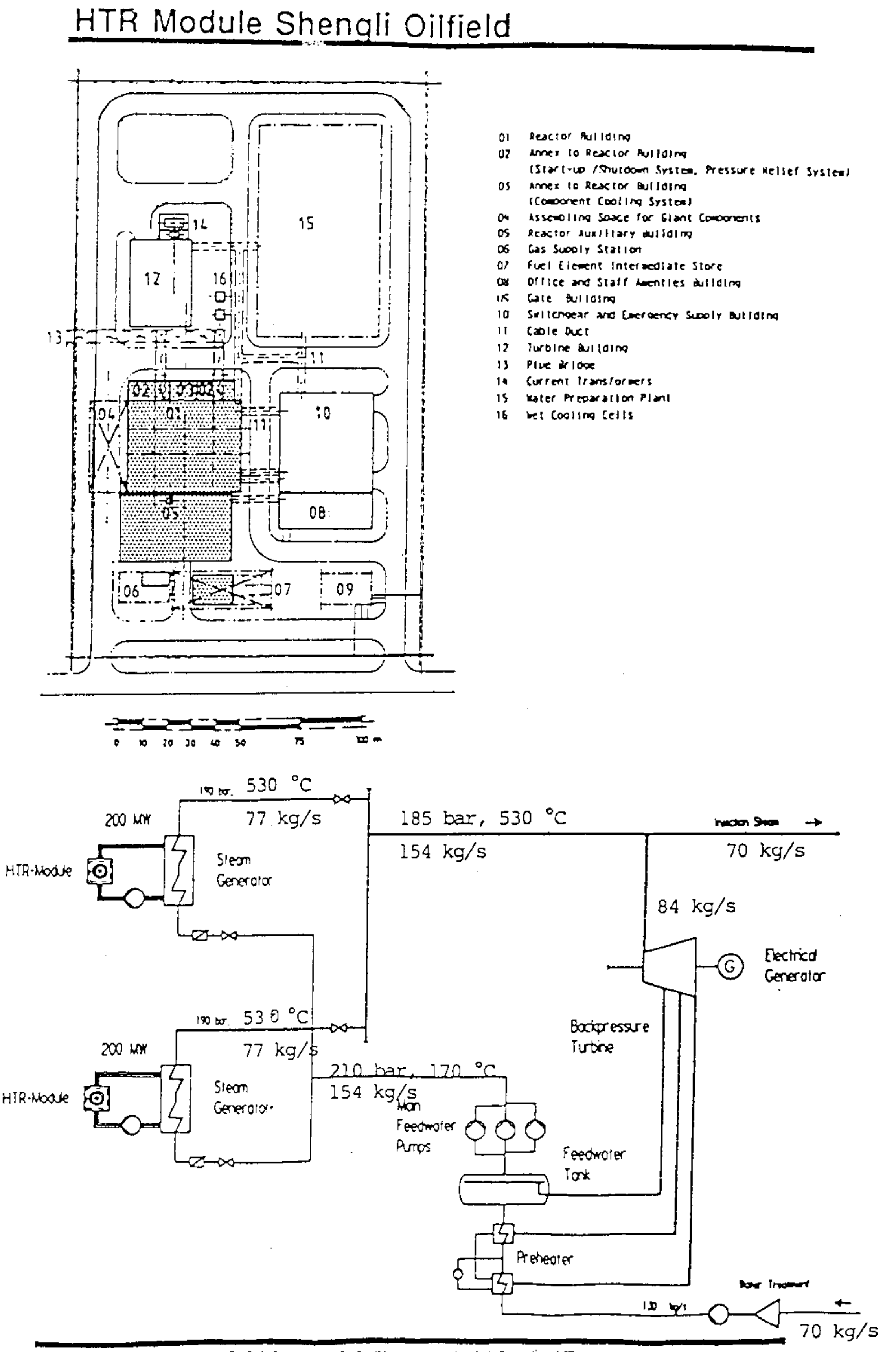

FIG. 6 HTR-2 MODULE SITE PLAN AND FLOW SCHEME OF POWER AND STEAM COGENERATION

2.6 PRESENT STATUS AND FUTURE PROGRAM OF HTR-RESEARCH AND DEVELOPMENT IN SWITZERLAND

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Interessen an der Entwicklung Nuklearer Technologien
SWITZERLAND

1. General Status of HTR R&D in Switzerland.

Research, Development and Design of Gas-cooled Reactors have already a long and well accepted tradition in Switzerland. These activities began with the design and construction of an experimental gas-cooled power station of 30 MW by the Swiss industry in the early sixties.

However, it was soon recognized, that for a small country like Switzerland, nothing but the cooperation within international projects opens a field for successful activities in the domain of nuclear energy. Therefore Swiss industry restricted themselves to the development and the supply of reactor systems and components only. Through extensive research and the participation in various international projects such as AGR, AVR, FSV, THTR, HHT, HTR etc., it was possible to gain valuable know-how, which qualified the Swiss industry as reliable manufacturers of reactor components and enabled them to render all kinds of engineering services for nuclear power plants.

Basic research for reactor physics and for reactor safety, is mainly concentrated at the Swiss Federal Institute for Reactor Research (PSI) and at the technical Universities.

Since the beginning of the german pebble-bed type high temperature reactor development, Swiss companies and institutions were participating at the R&D. This led finally to the supply of components for the THTR, e.g. the steam generators. A loosely formed group of industrial and engineering companies as well as the PSI took an active part in the followup-project, the high temperature reactor with helium turbine called HHT, where a considerable number of components were developed and designed in Switzerland. This project has been laid aside due to the great efforts in R & D which would have been neccesary for a final design.

In order to utilize the existing vast experience, the companies concerned and the PSI agreed in 1979 to join as partners and established the "Swiss Community for the Development of Nuclear Technology" called IGNT. Partners of IGNT are ABB, Baden; Sulzer, Winterthur; B+G, Lausanne; EWI, Zürich; Colenco, Baden; and the PSI. IGNT represents and coordinates the

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activities of the partners , especially in the field of the high temperature gas-cooled reactors.

As a consequence of this partnership, the members of IGNT engaged themselves in the development of the HTR-Technology in general and particularly in the design of the HTR-500. Another project which was pursued in Switzerland over the recent years, is the gascooled heating reactor, a small pebble-bed HTR of 10 to 20 MW for district heating purposes. In 1988 work on this project has been terminated for political reasons, but will probably be taken up in 1990/91.

2. Activities of Swiss Industry and Engineering Firms in HTR Technology

2.1 ABB ASEA BROWN BOVERI AG.

ABB as a main contractor for power stations, follows and develops their own line of plant control- and protection-systems, to ensure safe operation and controllability.

Modern protection and control systems make multiple use of microprocessors and have superseded in many applications the old hardwired systems. But before introducing into nuclear installations, they have to prove and demonstrate the same standard of reliability as the controls which are in operation in existing power plants today.

The development of a micro-processor based, bus-oriented control system for

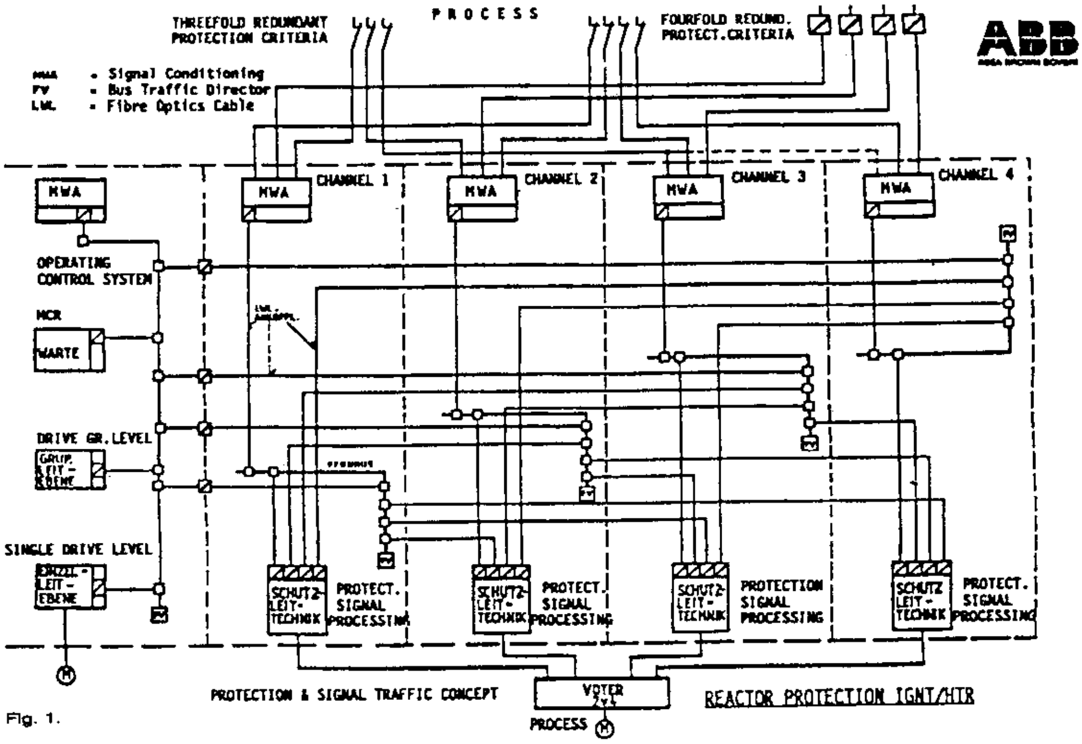

Fig. 1.

reactor protection needs still a considerable amount of research. The final requirements for such a protection system have been laid down and specified in accordance with the rules of the German Kerntechnischer Ausschuss (KTA) e.g design criteria, tripping concept, safety criteria and safety equipment.

2.1.1 Protection system concept

The reactor protection system consists of four redundant and physically separated channels, which permit the signal-traffic between different stations. Each channel is equipped with its individual bus for signal conditioning, signal processing and for communication with other channels. Each station is directly coupled to the bus via a fibre optic and the signal processing units are connected to all other channel-buses by a seperate fibre optic. Interaction is thus prevented but the entire signal information is available in each channel. Fig.1.

A protection initiation signal is produced in each channel and released by a fail-safe unit. The protection system detects the self-indicating active faults and those of the non self-indicating passive faults. Automatic and periodic test procedures are integrated in the protection system, which control soft- and hardware. Each channel supervises the correct performance of the own test procedure and the one of another channel. The four-channel system allows therefore the repair or the replacement of modules without causing a breakdown of the system. The hierarchical system of selfcontrol is shown in Fig. 2.

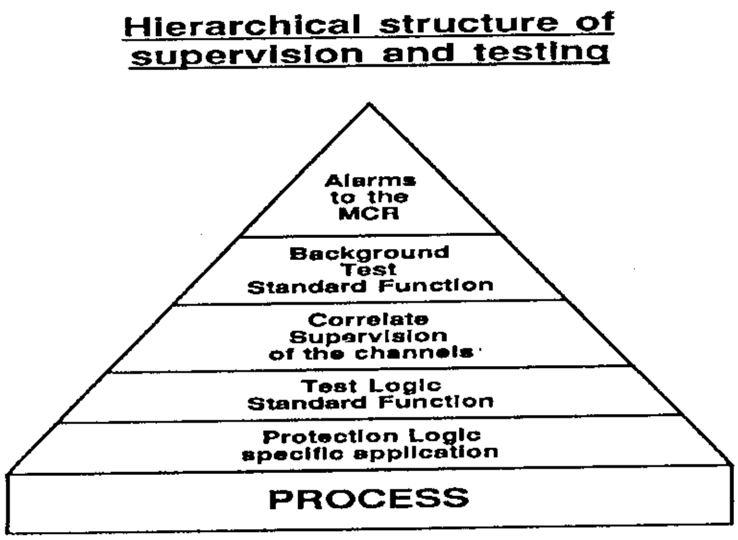

Fig. 2.

Reliability and redundancy tests are carried out in different steps. The first step is the individual soft- and hardware tests, which is followed by a second phase, where an entire simulation system is built up. The last step is then the introduction in an existing plant.

2.2 BONNARD & GARDEL

The engineering company Bonnard & Gardel has been involved since many years in the design and develpment of prestressed concrete reactor vessels.

Since 1974, Bonnard & Gardel has contributed to the development of the PCRV's for HTGR-projects in close cooperation with several institutes of the Technical University Lausanne (EPFL). A 1:20 scale model test with a multi-cavity pressure vessel was perfomed and basic knowledge gained on the behaviour of the vessel, by applying a steadily increasing internal pressure, starting from the elastic state of the concrete until rupture. Good agreement has been obtained between the results of the calculation and the tests. Fig. 3, 4.

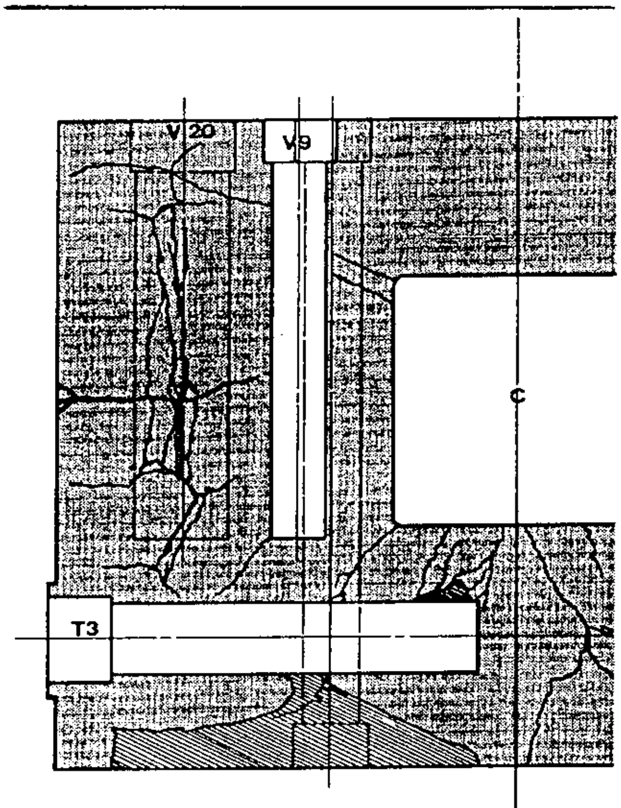

Fig. 3. Crosssection of the concrete testblock after rupture

These codes, are based on a three-dimensional finite element method and have since been continuously improved. They allow to calculate the static and dynamic layout of pressure vessels under all kinds of operation conditions and the analysis of the transition state from pure plastic behaviour up to rupture. Crack-propagation as well as the change of the concrete-properties with time, i.e. shrinking, creep and relaxation may also be taken into account. These cumputer codes are still further developed and extended modules are incorporated taking in account the latest results from the material science research.

A substantial part of the research and design work is devoted to the closures of the PCRV caverns with large diameters. First tests were made with a closure in the shape of a thick inserted concrete shell wedged in the vessel by inclined steel castings. The testmodel was exposed to both load and thermal cycling. Apart from the static behaviour of the component, the dynamic behaviour of these castings have also been examined. The diameter of the model-cavern was 1 m (Fig. 5.). The results so far show that the system is well suited to support the imposed load. It is planned to increase the load up to rupture and to determine the safety factor.

At present the layout and design of a new test section to test larger diameters of concrete closures is under way. These tests will be carried out in cooperation with Sulzer Bros. Ltd.

2.3 COLENCO AG.

Colenco a subsidiary of Motor Columbus is actively involved in HTR R&D work, with the emphasis on the structural analysis of the graphite core. There are two main fields of interest, the core bottom and the side reflector.

The description of the behaviour of the core bottom under the core load, temperature changes and under different operation and failure conditions, is of great importance for the design. The relative movement of the graphite blocks under the described conditions determines the distribution and occurrence of gap-widths between the blocks. Radial applied spring forces have to prevent excessive movements. Fig. 6.

A further task is to calculate the side reflector loading caused by the pebble-bed during control rod insertion. Research is necessary to develop continuum-

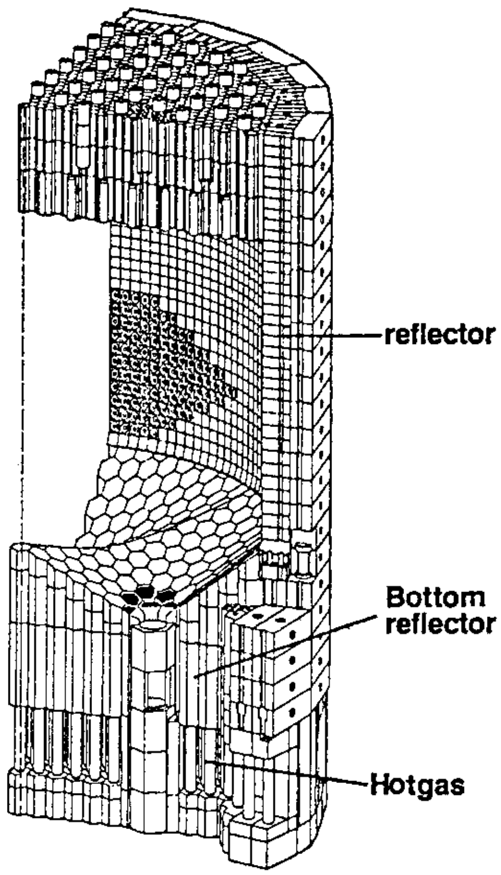

Fig. 6. Top, side, bottom-reflector

mechanical material-laws, which are able to describe the force flow through the pebble bed and to the side reflector. Up to now, this extremely complex problem has been solved by model tests, only.

2.4 ELEKTROWATT ENGINEERING SERVICES Ldt.

The contribution of Elektrowatt to the HTR-programme was the development of a calculation procedure to predict the seismic behaviour of complex structures and specially the pebble bed core. Initially calculations of the dynamic behaviour of the whole core structure have been performed.

Further work has been concentrated on the pebble bed itself. Simple pebble tests have been made to determine the material-laws between two pebbles and in particular the static and dynamic stiffness laws, friction factors, plastic behaviour etc. These results formed the basis for the analytical calculation of a pebble bed with a limited number of discrete pebbles. Subsequently a material-law could be derived which describes the pebble bed as a non linear but elastic continuum.

The next step, the development of a non-linear computer code for the dynamic calculation of the pebble bed is in progress. This work has been delegated to the institute for Informatics at the ETH Zürich and is nearly completed.

2.5 SULZER BROS. Ltd.

Since the early 1950's Sulzer has been involved in the development and the design of nuclear power plant components, based on 140 years of experience in the field of the classical fossil-fired boilers. The first nuclear steam generator for a gas-cooled reactor has been built in 1961. By now eleven gas-cooled reactors are equipped with Sulzer steam generators, amongst them are those for the high temperatur reactor THTR-300. A 10 MW intermediate heat exchanger for experimental operation at 950 °C was built for the nuclear process heat production and coal gasification. Details of the different applications are shown in Table 1.

Station	Country/ Year of Contract	Reactor Power (MWe)	Gas	Tube Shape	Manufactured by	Status of Plant 1989
Lucens	CH 1961	7	CO2	Helix	Sulzer Bros.	dismantled
Oldbury	GB 1962	2 x 280	CO2	Serpent	John Thompson (L)	operating
St.Laurent 1/2	F 1964/66	480/515	CO2	Serpent	CCM Sulzer Stein/Alsthom	operating
Dungeness "B"	GB 1965	2 x 600	CO2	Serpent	Int.Comb. Ltd.(L)	operating
Fort St.Vrain*	USA 1965	330	He	Helix	Stearns-Rogers	operating
Vandellos	E 1968	515	CO2	Serpent	CCM Sulzer Stein/Alsthom	operating
EL-4 Monts d'Arrée	F 1968	80	CO2	Helix	CCM Sulzer	decom- missioned
THTR Schmehausen	FRG 1972	300	He	Helix	CCM Sulzer Sulzer Bros.	operating**
GCFR (GBRA)	B	1200	He	Helix	-	project**
Delmarva*	USA	700	He	Helix	-	project**
HTR-500	FRG	550	He	Helix	N.N.	project SG
HHV Jülich	FRG 1973	Test Plant	He	Helix	Sulzer Bros.	dismant. HX
HHT (direct cycle)	FRG	600	He	Straight	-	project**
KVK Bensberg	FRG 1981	Test Plant	He	Helix	Steinmüller (L) Sulzer Bros.	
HTR-500 (CAHE)	FRG	550	He	Helix	N.N.	project
PNP (process heat)	FRG	500/170 (therm.)	He	Helix	N.N.	project**

* together with General Atomic **project discontinued (L)Licensee

STEAM GENERATORS (SG) AND HEAT EXCHANGERS (HX) FOR GAS-COOLED REACTORS Tab. 1.

2.5.1 Steam generator for the THTR-300:

The THTR steam generator stands as a typical example for the helix type design and will be dealt with here in some detail.

2.5.1.1 Design

The flowdirection of the coolant gas and the arrangement of the steam generators around the core (entails) hot gas entering at the bottom and leaving it at the top. Arranging the helix-bundle in an cross-counterflow, the water flows and evaporates in the downward direction. The steam generator works on the once-through principle. Feedwater entering a given tube is preheated, evaporated and superheated in the same tube. Eighty high pressure tubes are grouped in two, 88 reheater tubes in groupes of eight. All water and live steam subcollectors are led through the double steel closure at the top. (Fig. 7).

The heating surface consists of an assembly of helically coiled tubes arranged in concentric cylinders, giving a compact circular heat exchanger. This allows fabrication in the shops and to transport the whole units to the site where they are mounted in the vertical position within the concrete pressure vessel. (Fig. 8)

2.5.1.2 R&D Support

It is obvious that for this type of reactor component a great amount of supporting R&D is necessary. The increasing technological demand for advanced designs requires quick analytical support as well as immediate access to experimental facilities. Over the last two decades a number of research programs were initiated to solve specific problems for steam generators and heat exchangers. Some of these may be mentionned to show the compexity of the problems the designer has to solve.

Gasmixing: In a steam generator bundle the nominal gas temperature distribution may be disturbed by various influences, e.g. hot streaks, manufacturing tolerances, plugged tubes etc. To predict temperature distribution, the knowledge of gas mixing effects is required. Therefore tests were performed to gain a better understanding of eddy diffusivity effects and to support analytical methods applied in designing helical steam generators.

Downward flow evaporation (DFE): With the THTR already operating in the downflow evaporation mode, experimental investigation is continued to provide more information on flow behaviour, in particular at very low flow rates during start up and shut down procedures. The test program is jointly performed with the PSI.

Flow induced vibrations must be known in order to avoid damage. Extensive tests and theoretical analyses have been conducted on single tubes and on straight and helical tube bundles.

The bimetallic weld (BMW) marks the transition from ferritic to austenitic material. Numerous BMW service failures in fossil fired boilers indicate a potential weakness of this weld. Many tests have been conducted to prove the welds. Shock test series and thermo cycling have reached more than 40'000 hrs. and 25'000 cycles without failure, figures which exceeds largely the actual service conditions.

Creep fatigue: Combined effects of creep and fatigue are determined by stress analysis. Although such tests do not yield very scientific results, they serve to enhance the confidence in the chosen design.

Surface protection: Tubes must be supported and allowed some movement. This raises questions of friction, sticking and wear. Conditions are aggravated by the chemical inert helium, since helpful microoxidation at the points of contact does not take place. The helical tubes in the THTR steam generator are protected by a sleeve and wedge assembly. Their surfaces are treated to withstand fretting and wear conditions. Fig. 9.

Integrated expansion modulus: One major design problem is the accomodation of thermal expansion in a high temperature heat exchanger. In this new design, the movements can be absorbed in a special section of the heat exchanger bundle, where the tubes are supported in such a way that the upper and lower bundle part can move relative to each other.

Fabrication tests for large helical bundles: With regard to the increased attention helical type heat exchangers have gained, fabrication test were performed to demonstrate the feasibility of coiling and threading large helices to the required dimensions for steam generators or other heat exchangers.

2.5.1.3 Operation

A comparison between predicted and measured gas and water/steam temperatures along the high pressure bundle showed good agreement and confirmed the thermohydraulic layout.

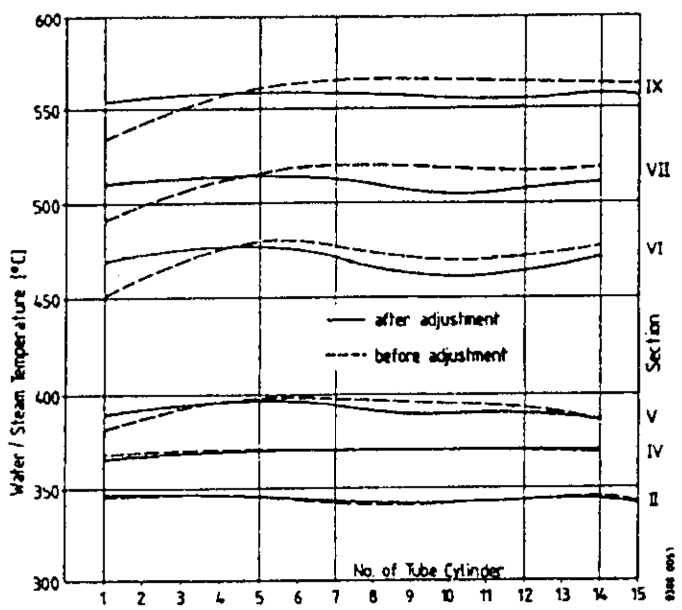

Figure 2 Radial Temperature Distribution at 6 Levels.

Flow stability, also extremely constant, was achieved by fixing an orifice in each steam generator tube and an adjustable throttle in each of the above mentionned subsystems of two high pressure tubes. During comissioning certain throttles were readjusted and a more uniform radial steam temperature profile was achieved. Fig. 10 shows the temperature profiles before and after adjustments at full load and at six different levels.

The six steam generators for the THTR showed excellent performance although they have been operated during the comissioning of the reactor, for quite a while far away from the nominal layout point. In any case the helix type design with DFE has proven the high standard of performance, which is necessary for a nuclear power plant.

3. Activities of Research Institutes and Universities in HTR-R&D

3.1 SWISS FEDERAL RESEARCH INSTITUTE, PSI

Since a number of years PSI has changed his policy on reactor research and today the emphasis is to resolve more generic problems of the HTR technology, predominantly for HTR safety. In this contex research work is performed namely in the fields of Material Science, Thermohydraulics and Reactor Physics.

3.1.1 Material Technology

In Material Technology research work is performed to obtain basic knowledge and data of fracture mechanics of different high temperature materials. Of particular interest is the gas-side corrosion and protecting layers against fretting and wear of structural material.

A smaller activity is related to understand gas-side corrosion of ceramic materials preferably graphite. Tested are different protecting layers which should withstand corrosion, wear and stresses.

3.1.2 Thermohydraulics

The main activity is related to gain basic knowledge of 'down flow evaporation' DFE at very low mass flow rates e.g. startup and shutdown conditions. The layout and design of helix type steam generators are directly influenced by possible density wave oscillations occuring at these flow conditions. As aconsequence these oscillations may induce low-cycle-fatigue and tube vibrations. Basic knowledge of these phenomena will be used to validate computer codes for layout and design. Fig. 11.

3.1.3 Reactor Physics

Although HTRs have been extensively investigated in the past, the shift towards low enrichments and away from the mixed thorium/uranium fuel cycle as well as the introduction of new core materials (e.g. hafnium as burnable poison) reveals a lack of experimental data against which design and safety evaluation procedures can be validated. In addition, some effects such as reactivity increase caused by water ingress are more important in the smaller HTRs of current interest.

The possibility of steam generator or liner cooling system leaks necessitates the consideration of accidental water ingress in HTRs. Most graphite moderated HTR systems are significantly undermoderated for reasons relating to fuel cycle economics (the conversion ratio increases in undermoderated systems so that less fissile material needs to be supplied).

This means that these systems will gain reactivity as moderator is added to the core. Arguments based upon the volume of the reactor core as compared with the entire primary cooling circuit and the amount of water contained in the steam generators and liner cooling systems are used to limit the amount of water that must be considered in accident analyses to a small fraction of the maximum void space in an HTR core (about 2.5% in the case studied by Hübel and Lohnert).

Pebble-bed HTRs usually have a strong reflector effect because of the undermoderated core with nearly 40% void coupled with a high density graphite reflector.The high thermal neutron flux in the reflector compared with the adjacent core region enhances the worth of control rods located in the reflector regions. One of the effects of water ingress is to reduce the worth of the reflector and hence of any control rods or poison ball reserve shutdown sytems (KLAK) located in the reflector regions. Similar considerations apply to the modular prismatic block HTR designs which usually have a relatively small core diameter in order to minimize the maximum fuel temperatures under accident conditions.

The reactivity effects caused by water ingress into a small undermoderated HTR core can be quite large. The data shown in Fig. 12 were calculated for one of the experimental configurations using 16.7% enriched LEU AVR fuel pebbles proposed for the PROTEUS critical facility. A moderator-to-fuel pebble ratio of 1-to-2 (C/U and C/U-235 atom ratios of 950 and 5630, respectively) and a pebble filling factor of 0.6046 were assumed. The borated steel safety control rods were located 8 cm from the boundary of a 1206 mm diameter core in a 1000 mm thick radial graphite reflector.

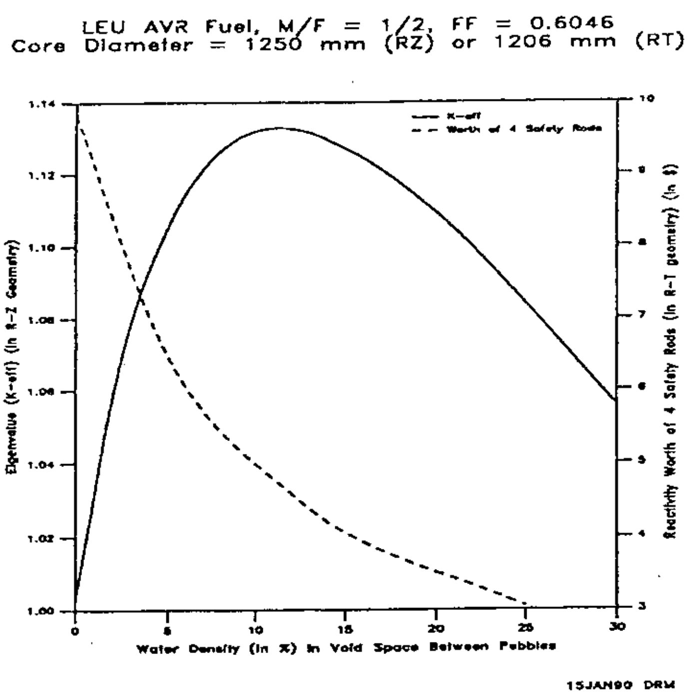

Figure 3: Eigenvalue and Safety Rod Bank Reactivity Versus Water Density (LEU AVR I M/F = 1/2, FF = 0.6046, Core H/D (RZ) = 120/125 cm, Core D (RT) = 120.6 cm)

The only previous water ingress experiments that have been performed for pebble-bed reactor systems are the high-enriched U-Th (HEU) experiments at Graz and KFA, Jülich. The Graz experiments used a relatively small amount of HEU fuel in a heterogeneous externally driven system whereas the KFA, Jülich experiments used HEU pebble-bed fuel in a subcritical assembly with thin graphite reflectors and are thus not very useful in assessing the accuracy of water ingress calculations for low-enriched U (LEU) pebble-bed systems of representative size and neutron leakage.

In order to cover this domain with experimental data and reduce the design and licensing uncertainties for small- and medium-sized helium-cooled reactors using low-enriched uranium (LEU) and graphite high temperature fuel, a series of critical experiments in the zero-power reactor facility PROTEUS are planned.

The main objectives of the new experiments are to provide first-of-a-kind, high quality experimental data on:

1) the criticality of simple, easy-to-interpret, single core region, low-enriched uranium (LEU), high temperature reactor (HTR) systems for several moderator-to-fuel ratios and several lattice geometries;

2) the changes in reactivity, neutron balance components and control rod effectiveness caused by water ingress into this type of reactor; and

3) the effects of the boron and/or hafnium absorbers that are used to modify the reactivity and the power distributions in typical HTR systems.

The new experiments have been authorized within Switzerland and work on the design and licensing of the modified PROTEUS critical facility is now in progress. The Swiss contribution to the international LEU HTR experimental program consists of the facility construction, licensing and operating costs as well as a considerable portion of the scientific staff. The Federal Republic of Germany contribution of the LEU pebble bed HTR fuel needed for the initial experiments as well as significant scientific support has been essential in the planning of these experiments. Work on the necessary international contractural and safeguards arrangements for the transfer of the fuel to PSI has been initiated. The HTR critical experiments are presently scheduled to begin early in 1991.

The experiments have been accepted as an International Atomic Energy Agency (IAEA) coordinated research program (CRP) entitled "Validation of Safety Related Reactor Physics Calculations for Low Enriched HTR's" in the framework of the Agency's Gas Cooled Reactor Working Group. In addition to the basic Swiss and Federal Republic of Germany cooperation, the Soviet Union, the United States of America, the People's Republic of China and Japan have decided to participate and will delegate some of the scientific specialists necessary to plan, execute and analyze these experiments and insure that they are relevant and cost effective with respect to the various gas cooled reactor national programs. Some other countries may also join in. The cooperating international partners met at PSI in October 1989, to begin to define the details of the initial experimental program.

As a result of this meeting and based upon the detailed PSI calculations for alternative facility design layouts have been carried out in the earlier part of the year. It has been decided to begin the first phase of the experiments with a 125 cm diameter pebble bed core and the lowest possible moderator-to-fuel pebble ratio of about 1-to-2 (C/U atom ratio of about 950/1). (Fig. 13) The experiments will continue with other moderator-to-fuel pebble ratios and deterministic, in addition to random, pebble bed lattice geometries so as to provide an adequate integral data base for the validation of LEU HTR design calculations for both pebble bed and block type HTR designs.

A new safety report for performance of such LEU HTR experiments in the PROTEUS critical facility is near completion. Following submission of the safety report to the Swiss licensing authorities, the LEU HTR PROTEUS activities planned for 1990 include:

1) detailed experimental planning in cooperation with our international partners;

2) completion of the design drawings and procurement of long lead items (after approval of the safety report); and finalization of fuel transport arrangements.

3.2 TECHNICAL UNIVERSITIES ETHZ, EPFL.

3.2.1 Physical properties of concrete

Physical and thermal properties of concrete are of general interest in particular to qualify suitable types of concrete for all those applications where higher temperatures occur.

The purpose of the tests carried out at the Institute for material science at the ETH, Zürich is, to receive basic data for gravel- and basalt-concrete, which allows to calculate and predict the temperature distribution in the pressure vessel walls during operation. Mea-sured were the thermal con-ductivity as a function of the moist content in the concrete, the pozzolanic reaction and creep and shrinkage properties at temperature up to 300 °C. Specimen with a representative size are now fabricated to finalize these tests. (Fig.14)

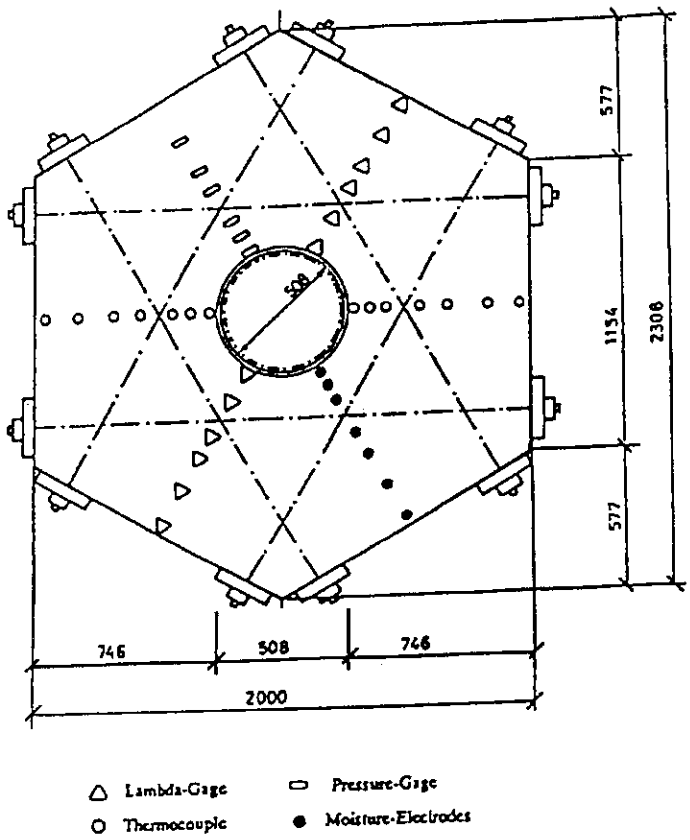

Fig.14 PLAN VIEW OF THE MODEL VESSEL

In addition measurements to determine the residual strength of concrete after thermal cycling has been carried out at the Institut de Static et Structures at the EPFL, Lausanne

4. Future Activities

When we are considering the present world-wide unsatisfactory situation in the nuclear area, we will be faced immediately with the question "what are the driving forces which justifies further research and development in the field of High Temperature Reactor Systems?".

Each machine or technical system has, within the technical utilization and lifetime, a positive evolution and there is no plausible reason why this should not be the same for nuclear reactor systems also. Therefore there will always be necessary improvements in design and for safety, which call for generic activities in the field of research and development leading to new and/or advanced systems.

Advanced systems in terms of nuclear reactors are reactors having considerable safety-margins due to selfregulating inherent and passive system characteristics, with high system efficiencies an therefore producing less nuclear waste and less dissipating waste heat to the invironment.

High Temperature Reactors (HTR) are advanced systems as they fulfill these requirements. As a consequence HTR-technology attracts world-wide considerable interest, although there is for political and economical reasons no demand for nuclear reactors at the moment.

Nevertheless the demand of electricity and district heating is steadily increasing in Switzerland; the electricity by about 2 - 3% per annum. This means that the demand for electricity will in any case call for new nuclear reactors to be built in the future. This however gives new and advanced reactor concepts the opportunity to be considered for these power plants.

The chances for the high temperature reactors are intact, when the HTR-community is able to show the advantages of the system and to prove that there are reliable and licensable designs. Therefore IGNT has signed new contracts for cooperation with Germany on the HTR-500 development and has ensured funding for future research by the governement.

Lausanne, Würenlingen, 12.2.1990.

This report does not include Figs. 4, 5 and Figs. 7~13.

2.7 TECHNOLOGY, COST ECONOMICS AND OTHER FACTORS OF HTR AS RELEVANT TO A DEVELOPING COUNTRY

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BANGLADESH

1. Global Energy Balance

Energy is a pre-requisite for attainment of economic development of any country. Average per capita consumption of primary fuels in the developing countries is about one-twentieth of that of the developed countries. A look at the global energy balance would indicate that 23 industrialized countries accounted for about 51% of the estimated 351 Exa Joules of primary energy consumed globally in 1987[1]. In the same year these countries consumed 56% of the estimated 10,100 TWh of electricity generated globally (Tables 1 & 2). Though it is very difficult to suggest a rigid correlation between the growth rates of energy and GDP, it is established that energy co-efficient, which is the ratio of the growth rates of energy and economy, should be higher for the developing countries then the developed ones. The present low level of consumption of energy and electricity in developing countries, therefore, necessitates a fast growth rate in demand for energy in order to attain a sustainable growth in economy.

The realization of finiteness of reservesof fossil fuel has been given due consideration in all countries since the energy crisis. The developed countries, having the advantage of their technological and infrastructural bases, were more successful in solving the problem as compared to the countries with limited energy resources and weaker technological base.

A look at the global energy-mix would indicate that it is not only conservation or efficiency in demand management that helped tackle the situation. In fact, alternate sources of energy, especially nuclear power, played a pivotal role in moderating the effect of the crisis. For example, 1794.4 Twh of electricity was produced by 429 nuclear power plants of different countries in 1988. This is roughly equal to the total electricity production of the world in the year 1975[2]. It is estimated that had the same amount of electricity been produced by burning oil, it would have needed an additional supply of 504 million tons of oil in that year. This amount, equivalent to 10 million barrels per day, is almost double the crude produced by Saudi Arabia or 72% of the oil produced by countries in the Middle East in that year. On the other hand, the same amount of electricity would have needed an additional supply of 750 million tons of coal in that year. This amount may be compared with the global production of 3400 million tons of coal in 1988, out of which only about 340 million tons was available for export (Table 1). This additional demand would have exerted certain influence on supply and price structure of the fossil fuels and thus could eventually lead to an energy crisis of a different type or could aggravate the situation of energy crisis.

1.2 Projections of Global Energy Demands

Growth in demand for primary energy resources of the world was phenomenal during the period from the end of the Second World war and up to the 1970's. Cheap and abundant supply of fossil fuels, especially of crude oil, triggered this unprecedented growth in demand for energy. It is estimated that in the 50's and the 60's most of the countries, and especially in the developed ones, energy demand grew at rates in excess of 5% per annum. This trend was, however, slowed down following the energy crisis through measures of austerity, manifested in innovation and adaptation of various techniques of energy conservation and demand management. Even then, the growth in demand has been far from being static and various projections indicate substantial increase in next 2-3 decades and even beyond that time horizon.

Different international organizations projected global demand for primary energy resources keeping in view the limitations on the reserves and supply of fuels. Some of these forecasts are given in Table 3.

It is seen from the Table that IEA/OECD, WEC and IIASA are almost unanimous in forecasting an increase in total energy demand from about 7,000 MTOE in 1980 to 10,000-12,000 MTOE in 2000 AD; also, a further increase is forecast for the period 2000-2010 AD and for 2010-2020 AD. These reflect what the related organizations consider to be low energy scenarios, with only a moderate increase in per capita energy use in the industrialized countries and considerable increase in energy conservation and in the efficiency of energy end-use is assumed[3].

The two forecasts/projections for electric energy in Table 3 suggest that the increase in the demand for electric energy will be even more pronounced than the increase in the demand for primary energy. This conclusion is supported by experience of all countries even after the energy crisis, that the growth in demand for electricity was higher than those of GDP and primary fuels. The projections on demands may be compared with the global consumption in 1988 of primary fuels, which was about 8250 MTOE and electricity production, which was 10,100 TWH.

Presently, the so-called concept of "Low-Energy Path" is being advocated for future development of energy sector. The concern is motivated by the hazards of green house effect from increasing burning of fossil fuel. In historical perspective it is seen that higher growth in energy is desirable for maintaining sustainable development of developing countries. As such the "Low-energy path" could be disastrous for such countries. The developed countries, with four-fold per capita energy consumption of the average of developing countries, are contributing towards the pollution. It is expected that it could be easier for them to adopt a path of low energy growth. The developing countries, on the other hand, must try to increase their energy consumption. The future of the world would depend on the success of these two categories of countries in attaining these goals.

1.3 Energy and Environment

One of the most important and difficult part of the problem of energy supply is its relationship with environment. A realistic global demand-supply balancing of energy must take into consideration the environmental consequences of energy resources. Coal, for example, is considered by many to be a potential resource, technology is available for its extraction at reasonable costs, and that it can be adapted to the immediate needs of a variety of users, especially in production of electric power. However, adverse effects of burning coal, which may vary widely in geographical scale may be extremely harmful to the environment. At one extreme is the possibility that Carbon dioxide from burning of Coal could accumulate in the atmosphere and produce an increase in the mean temperature of the earth. At the other extreme are the site specific and local problems like disposal of ash, transportation and effects of effluents like Nitrogen, Sulfur and their compounds. Presently there is a convergence of opinion that the burning of fossil fuels could conceivably produce a permanent warming of the atmosphere, accompanied by even larger change in temperature differential between the poles and the equator. It has been estimated that from the end of the 19th Century until 1975 the CO_2 concentration in the atmosphere increased approximately by 18% (from about 280 to 330 ppm). The amount of fossil fuel burnt during the same period paralleled the increase in the CO_2 content[4].

It is often suggested that electric power production sector is a small contributor to the 'Green House' gases. However, electricity production contributes about 25% of the world's CO_2 emission at the present time. Nuclear and hydro-electricity, the only two technologies immune from discharge of CO_2, account for about 37% of electricity produced globally. Thus, in fact, 63% of electricity generated by fossil fuels contribute a quarter of the total concentration of CO_2 in the atmosphere[3]. If alternatives to fossil fuels for electricity generation is not adapted, then concentration of CO_2 in the future will assume an alarming proportion. Thus from environmental consideration hydro-electricity and nuclear power are the most benign forms of technology for production of electricity. Unfortunately, hydro-potentials are limited by natural constraints, which leaves the mankind with practically one alternative, namely nuclear option, in meeting the ever increasing demand for energy.

2. SMPR Study and Size of New Generation Plants

Project Initiation Study of the IAEA on Small and Medium Sized Power Reactors (SMPR) was started in 1983. So far it has been possible to identify potential suppliers and to clarify certain preliminary aspects of cost. Information on cost are too sketchy to provide a basis for further analysis or to facilitate decision making by the prospective buyers.

Bangladesh Atomic Energy Commission (BAEC) recently completed a Technical, Economic and Financial Feasibility Study on the Rooppur Nuclear Power Project. Two Consulting firms were commissioned for this purpose. A team of BAEC personnel participated in the study as the local counterparts of the Consultants.

As a part of selection of technology for the project all potential manufacturers of SMPR, which were identified in the SMPR study of the IAEA, were requested to provide information according to a proforma. Criteria for selection and the relative weightage for each of the criterion were set jointly by the Consultants and BAEC. Modular HTR was identified as one of the technologies suitable for the project on the basis of the overall rating of this technology.

The potential market for SMPR is mainly limited to the developing countries. In spite of the fact that such plants have the obvious disadvantage of Économy of Scale', some developing countries could select such reactors during the period 1990-2000 simply because the projected growth of grids would not allow integration of a larger plant in foreseeable future. Interest of potential suppliers in developing or promoting a new design will depend on the size of the potential market of such reactors.

The demand for electricity especially in the developing countries, which was suppressed in the past by resource constraints and by the uncertainty in price and availability of fuel, has again started growing steadily. Growth rate of energy in such countries are usually high due to the present low level of electricity generation, contribution of suppressed demand and for the need for developing energy sector in keeping with the economic goals. For example, at a 10% growth in demand (which is not uncommon in many developing countries) the total grid capacity could grow by 60% in five years or by 150% in ten years. Thus a country with a installed capacity of 2000MW could accommodate a 600MW nuclear power plant in year 2000 A.D. (using the rule of thumb that the largest plant in the grid should be around 10% of the total grid size).

It has not yet been possible to start implementation of the first SMPR. If the construction time, including time for design, safety evaluation, negotiations, etc. is about 8 years, then many countries could attain the grid capability to incorporate a larger reactor (600MWe, for example) by the turn of the century. This may substantially shrink the size of potential market for the SMPR's. Prospects of SMPR's, especially in the context of the developing countries should be evaluated keeping such limitations in view. If the transitional phase (time gap within which the grid size can reach a level when a 600MWe plant could become acceptable) is not very significant, then such countries would prefer a larger plant (600MWe) because of cost-economic advantages and for other factors, like availability of a proper reference plant, experience with operation, diversified sources of fuel, services and spares. Unfortunately, associated disadvantages of SMPR's are quite prominent, especially regarding cost-economics.

3. Relative Advantages and Disadvantages of HTR

Sailent features of relative advantages and disadvantages of HTR over other types of reactors according to the above selection criteria utilized in the feasibility study of Bangladesh are as follows[5,6].

3.1 Safety

The HTR designs available now or likely to be ready for export in foreseeable future have negative void coefficient. These designs are reported to incorporate passive safety systems based on physical and chemical features of the reactor, independent of systems like ECCS. It is envisaged that decay heat could be transferred to reactor vessel through radiation in the event of a design basis accident. In this case the maximum temperature is not expected to exceed certain safe level, thereby ensuring a margin to the melting point of uranium. This advantage is reported to be confirmed by rig tests and other experiments. Such calculations and results should be verified to ascertain the adequacy of the mode of heat transfer and effects, if any, on the structural materials in the event of a design basis accident. If such verification can confirm the claimed "inherent safety" of the design, then this particular advantage can be considered as a redeeming feature of HTR.

3.2 Fuel Supply

HTRs use coated particles in their fuel elements, which could be advantageous in retaining radioactivity within the fuel elements. Failure of such fuel is less likely due to the absence of separate cladding material. However, the sources of supply of fuel are limited and the fabrication appears to be complicated. On-load refueling is an advantage of HTRs with pebble bed fuel, but sampling and withdrawal of spent fuel elements could be complicated operation. These factors have to be considered duly in the process of selection of technology.

As in the case of larger and conventional nuclear power plants, the back end of the fuel cycle in identified as one of the uncertainties of the new generation plants. High burn-up of certain types of such plants could be encouraging from the point of view of safeguards, but discouraging for the supplier to reprocess spent fuel. The importing developing country will have to arrange for disposal and management of entire spent fuel elements. Disposal of such fuel elements, for example the pebble bed fuel elements, need to be investigated.

3.3 Ease of Construction

Some of the HTRs in the SMPR range are modular in nature, which necessitates substantial shop fabrication. This ensures better control over quality and project schedule. However, the scope of local work and transfer of technology could be reduced to a certain extent as compared with other reactors. Ways and means have to be found to improve upon this situation, probably through involvement of buyer's personnel in different phases like detailed designing, shop fabrication, QA & QC programs etc. Potential buyers can also participate in international studies on development of HTR through the auspices of IAEA and other international forums. This would not only ensure a proper mode of technology transfer, but also help incorporate requirements of developing countries in such deisngs.

3.4 Special Materials

HTRs use helium gas, which is inert, efficient as coolant, does not change phase and largely precludes possibilities of chemical interference with the properties of structural materials. Inventory and annual make-up requirements do not possibly justify the establishment of helium plant in the buyer country. Sources of supply on commercial basis are known to be diversified, which may be considered an advantage over the alternative like heavy water. Graphite in the reactor core act as an additional heat sink. However, the accidental situation of ingress of water from steam generator into the core need careful examination in the process of design evaluation. The alternate design consideration of placing the steam generator at a level lower than the core can reduce the possibility and quantity of ingress, nevertheless the possible consequences can not be ignored totally.

3.5 Waste disposal

The problem with reprocessing of spent fuel is more or less the same for any type of power reactor and the related technology have not been finalized conclusively. However, structural stability and probability of failure of pebble bed fuel elements may be considered carefully on the basis of available statistics. In general such fuel elements are considered to be capable to retain structural integrity under varied operating conditions. On-site storage of fuel elements spent over the entire life of the plant need to be kept in view in case of a contingency of non-availability of reprocessing opportunities in future. For a HTR module with a total capacity of 300MWe, the total space for on site storage of fuel over the life time of the plant is estimated to be about 2 hectares.

3.6 Reference Plant and Operating Experience

Inadequacy of operating experience and non existence of a proper reference plant for HTRs may be considered to be a disadvantage. This has to be traded off with its other advantages notably safety. Operat-g ing experience with existing HTRs, though small in extent, can be used as a basis for assuming acceptability of the concept in general and the relevant components in particular.

3.7 Licensing

It is needless to point out that it is difficult for a developing country to carry out a full scale review of safety of the selected nuclear power plant. It is therefore felt useful that the manufacturer obtains a 'non-site-specific-license' on the proposed design from the organization responsible for safety review and licensing in the country of the Vendor. The task of the local regulatory organization will then be reduced to review safety analysis of the design as relevant to the site for the plant.

3.8 Modular Concept

Most of the HTR's are small in size (80 to 150MWe). Thus from cost economic point of view, a large size turbine is usually envisaged to serve 2-4 such modules, giving a plant size of 300-400MWe. Load following capabilities of such a plant need to be assessed properly. Also loss of load or a probable failure in the secondary loop and their effects on individual module should be analyzed to ascertain safety of the modules and also of the total plant under such simulated conditions.

Other advantages and disadvantages of HTR are more related to the particular Vendor to be selected for a developing country. On the basis of the criteria considered above, it can be assumed that a HTR could be an option for a developing country as its introductory nuclear power plant. Cost economics and factors and uncertainties that may influence the selection of technology are described in the subsequent paragraphs.

4. Capital Cost of Plants with HTR

It is frequently argued that the smaller plants are penalized due the economy of scale. The pertinent question is what is the basis for the scale-down factor and what should be considered as the reference plant for determining such scale-down factors. For example, specific cost (\$/KW) of a new design of 300MWe PWR can be determined on the basis of the specific cost of a larger PWR. But the same scale-down factor is largely inapplicable for another new concept, for example, a modular HTR. The latter reactor does not have the elaborate Emergency Core Cooling Systems along with all the required redundancy. If this cost is excluded, then it can be expected that the specific cost of a modular HTR could actually be even lower or at least comparable to the specific cost of a larger PWR. In practice, however, it is seen that the effect of passive safety system is not at all reflected in the quoted specific cost of the latter type of reactor.

It is understood that any Vendor has to make substantial investment in developing a new concept and in designing a reactor on that basis. This is also natural that this entire cost should not be passed on to the buyer of the first plant of this type, rather this cost should be distributed among at least 4-5 buyers of the particular design. If this is not done than the new generation plants can not compete with any of the conventional alternatives.

4.1 Cost Estimates

Experience with design, construction and implementation of modular HTRs is not significant. Therefore, it could be difficult for a Vendor to estimate the cost in spite of its wide experience with other types of reactors. The uncertainties in buyers' countries usually tempt the Vendor in quoting an extremely high cost in order to cover all the risks associated with such uncertainties. Moreover, the size of the potential market is not clear, which could influence the supplier to include all its direct, and indirect cost of R&D on the first buyer. Moreover, many basic R&D programs have been financed by Governments of the developed countries. Also, in case of modular plant with 3-4 units serving a signal turbine, the residual design cost is already expected to be shared by at least these 3-4 first units. In case of a larger plant order for 3-4 repeat reactors are more difficult to obtain. It is believed that a supplier does not usually invest in design of a new concept unless it has preliminary indication about a number of prospective buyers, which means that the supplier should set a target to recover its extra R&D cost for development of a modular HTR from 9-16 units and the estimate for the plant should be prepared accordingly. It can, therefore, be assumed that the specific cost of a mocular HTR should be at the lower end of the estimates of all SMPRs and close to larger LWRs being built presently. Shop fabrication, reduction of costs at site, reduction in soft-ware cost etc. factors also add to the above justification.

An analysis of cost figures for different nuclear plants built in Europe[5,6] show that the specific cost of plants of size 1000Me lie within a range of \$1200-1700/KW (Fig.1). Using a scaling factor of 0.2, the specific cost of a HTR could be 27% higher, which means that the specific cost of a modular HTR should be in the range of \$1520-2160/KWh. A mean value of the order of \$1850/KWh could be considered as the realistic specific cost of a modular HTR. Efforts need to be made to construct a plant at this target specific cost.

The fuel cycle cost of HTRs, which is completely within the control of the supplier is also reported to be high. This is difficult to justify, because the average burn-up of HTR fuel is of the order of 3-4 times higher than the standard PWR fuel, yet the fuel cost per KWh is about 40% higher for the HTR fuel. Such fuel is already in use for a number of years, so that the higher cost is also hardly justifiable. If the HTR has to compete with its alternative, then the fuel cost must be brought down at least to the level of PWR fuel.

5. Cost Economics

Cost economic performance of a modular HTR of size 320MWe was evaluated under the local conditions in Bangladesh. For this purpose a station to station comparison among the alternatives like, modular HTR, oil-fired plant, coal-fired plant and gas-fired plant was performed under different conditions. Indicators like Levelized Generation Cost, Cost Benefit Ratio, Net Present Worth and Economic Internal Rates of Return were compared among the above options.

Economic ground rules used in the economic evaluation of nuclear option with four 80MWe modular HTR, Coal-fired and gas-fired plants can be seen in Table 4. It may be mentioned that the case of oil-fired plant has not been shown separately assuming that cost figures would be comparable to that of gas-fired plant.

A proper cost economic evaluation of the nuclear project is rendered difficult due to uncertainties in its cost estimation as described in the preceding paragraph. Therefore, four cases of nuclear plant cost have been considered (Table 4).

So far as the alternatives are concerned, the greatest difficulty is associated with the uncertainties prices of various fossil fuels. At the present time, prices of both coal and oil are very low and it is very difficult to make any long term projection on the future trend of their prices in the international market. As such the following prices, as suggested in the Feasibility Study for Bangladesh[5] are used in the calculations:

Coal $49/Ton delivered at site; Oil $100/Ton; Gas $2.64/GJ

5.1 Results of Calculations

Results of the calculations of Levelized Generation Cost of various alteratives under the assumed conditions[4] at a capacity factor of 80% are shown in Table 5. Table 6-10 show the results of sensitivity calculations in the following ranges of different parameters:

Table 6 Increase in capital costs by 15%

Table 7 Decrease in capital costs by 15%

Table 8 Variation in capacity factor (50% - 100%)

Table 9 Variation in discount range (8% - 16%)

Table 10 Variation in rate of escalation of fuels (0% - 4%)

These calculations show that levelized generation cost of a 320MWe HTR, built at a specific cost of $1850/KW with fuel cost at 8 mills/KWh at a discount rate of 10% is about 49 mills/KWh. This assumes a life time average plant factor of 80%. This is marginally cheaper than that of a coal-fired plant with Flue Gas Cleaning system and coal price of $45/tonne, for which the generation cost is 57.48 mills/KWh, or a oil-fired plant at \$15/barrel of crude, for which the generation cost is 54.23 mills/KWh or a gas-fired steam turbine at \$2.90/1000 cft. of natural gas, for which generation cost is 54 mills/KWh. The interest rate assumed for these calculations is 6.5% with a repayment schedule of 15 years with a grace period of 5 years for construction[5,6]. These preliminary figures show that even for the target specific cost of \$1850/KW, the nuclear plant with HTR looses the economic advantage over the conventional alternatives if the cost overrun exceeds 10% and fuel price reaches 10 mills/KWh. Importance of these representative calculations is manifested in the fact that the decision of potential financiers and also the government could be influenced by the economic advantage of the nuclear option over its conventional alternative, especially if the indigenous primary energy resources is underpriced or subsidized for power generation (as is done in Bangladesh, where natural gas is sold at US 80 cent per 1000 cft. for power generation.).

Figure 2 shows the comparison between the nuclear with HTR option with coal-fired plant with flue gas cleaning system at different price of coal and different average life time plant factor. This shows that the advantage of nuclear option decreases with decreasing plant factor, increasing plant cost and decreasing price of coal.

Economic Internal Rate of Return, Net Present Worth at 10% discount and Benefit Cost Ratio of different scenarios of the nuclear option with HTR as compared to the coal and gas-fired plants are shown in tables 11 and 12.

These representative results of calculations reveal that the advantages of the nuclear option over its alternatives are limited to the nuclear cost scenario N4, namely for the case when the specific cost is close to \$1850/KW installed.

6. Financing

Financing is the biggest hurdle in the way of implementation of a nuclear power project. A possible solution is to split the supplies into packages and to diversify sources of supply accordingly so that the burden of financing is shared by a number of suppliers. An indicative splitting could be as follows:

Nuclear island	40%
Turbine	20%
Electrical equipment	10%
Intake channel	4%
Commissioning, erection	6%
Civil works	20%
	100%

Assuming an expenditure curve covering six years (one year for designing and five years for construction) with six equal installments of payment, the yearly burden on any single financing source will be between 0.6% to 6.7% of total plant cost. For a specific cost of US $1850/KW this will mean that the yearly financial requirement for implementation of the project will be US $100 million to be shared tentatively according to the break-up shown above. It is felt that the item excepting the nuclear island and civil works could be covered to a great extent by suppliers' credits. Part of the cost of the nuclear island could be mobilized from the supplier's country and the rest of it could be raised through commercial loans. The local currency cost may be provided by the buyer country. In this context it will be very helpful if a Joint Venture Company is formed for the project with the local and expatriate partners with participation in equity. Such a scheme will help enhance credibility of the project and increase confidence of the financiers in its viability. The Government of the buyer's country may provide guarantee on repatriation of income of the equity share holders, debt servicing and all other payments in foreign currency.

Prospects for financing nuclear power projects in developing countries are far from being conducive. Provisions of international and regional development financing institutions and various consensus discriminate financing for nuclear projects. For example, according to the OECD Consensus, total financing from its Member states can not exceed 85% of the total export volume. This means that the importing country must provide the local expenses (say 20% of the cost), 15% of the foreign exchanges and also the interest during construction (say about 25-30% of total cost depending on loan conditions). This means that the importing developing country must arrange for at least 45% of total cost (including IDC).

Conditions for obtaining credit insurance for such projects are also very complex. Theoretically the total supply could be split into packages in order to distribute the burden of financing among multiple sources. However, even in case of a split package financing it is very difficult to obtain the first commitment of funds from any of the suppliers.

Developing countries have to depend on multiple sources for all development activities. It is often seen that commitment to install a nuclear power project can directly or indirectly affect inflow of funds for other sectors.

Under these circumstances it is becoming more and more difficult to arrange finance for implementing nuclear power projects in the developing countries, like Bangladesh[7]. Solution to this problem would need intensive interaction with the World Bank, the Asian Development Bank and other regional financing institutions, export credit organizations, regional Consensus, like the OECD and also various credit insurance organizations. It is felt that many of the above kinds of organizations are not certain about the appropriateness of nuclear power in the context of a developing country with limited indigenous primary energy resources, safety, reliability, capability of such countries in operation of maintenance of such plants, cost-economics and other matters. An action oriented action program, led by the IAEA, may be initiated in finding possible solutions to the stated problems. The proposed action program could address itself to clarifying above questions and in convincing such organizations in providing finance for implementing nuclear power projects. It may be mentioned in this context that at present there exists a global consensus on the need to reduce environmental pollution and energy related activities feature prominently in releasing into atmosphere a substantial quantity of so called 'Green House' gases. Nuclear power is an appropriate proposition in reducing environmental pollution and hence there is a pressing need to encourage this source of energy, especially in developing countries having scarcity of indigenous fuels.

7. Conclusions

HTR could be a technology useful to many developing countries provided its economic performance is proved to be better than its alternatives. Logically the specific cost should be lower than other SMPRs and there is no reason why such a plant could not be built at a specific cost stated earlier. The technology, especially with respect to safety and licensing has certain points that need further clarification. Issuance of a non-site specific license/design clearance in the manufacturer's country could add to the confidence in the buyer country. The codes, standards and guides applicable to HTRs are not as exhaustive as for LWRs, especially the PWRs. It is true, a few HTRs have been licensed and are operating in the developed countries and some codes, guides and standards could be in existence in those countries. These need to be compiled and new ones developed to ensure their completeness. This could be achieved only through international cooperation. International/collaborative studies may also be carried out on certain technical aspects like the applicability and adequacy of the passive safety systems in the event of a design basis accident, independent confirmation that in the worst case of such a situation the maximum fuel temperature will remain far bellow its melting point, temporary storage and final disposal of waste and spent fuel, possibly in collaboration with a third country willing to cooperate in this respect, effect on the core in case of water ingress from the steam generator, analysis of different cases of accidents, exchange of design information, standardization of fuel elements, etc. In parallel, the suppliers should realistically firm up their cost estimation and also the fuel cycle cost. International cooperation in solving the problem of financing is yet another area, which needs urgent attention.

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Table 1: Global Balance of fossil fuels (Coal & Oil) 1987 [1]

Region	CRUDE OIL (MBD)			COAL (Trillion BTU)		
	Production	Import	Export	Production	Import	Export
North America	12.43	5.04	2.11	21,720	2,884	509
Central and South America	3.70	1.68	1.40	590	228	385
Western Europe	4.03	8.65	2.60	8,970	466	3,639
Eastern Europe & USSR	12.01	2.04	2.70	25,440	1,471	957
Middle East	12.94	0.53	9.80			93
Africa	5.23	0.59	3.82	4,330	1,019	114
Far East & Oceania	5.97	5.91	2.03	29,310	4,058	4,429
World Total	56.31	24.47	24.47	90,360	10,126	10,126
Total in EJ	126.88	55.14	55.14	95.33	10.67	10.67
Total in MTOE	2971	1291	1291	223	25	25

Table 2: World Net Electricity Generation by Type (1987) [1]

(In trillion watt hours)

Region	Thermal	Hydro	Nuclear	Others	Total
North America	2076.4	584.3	528.2	16.4	3205.2
Central & South America	118.9	300.2	7.0	7.8	433.9
Western Europe	1001.2	488.2	620.1	8.7	2118.2
Eastern Europe and USSR	1588.4	246.6	229.9	-	2044.9
Middle East	156.4	8.8	-	-	165.3
Africa	189.9	42.8	6.3	0.4	239.3
Far East & Oceania	1287.5	359.7	251.4	6.9	1905.5
Total	6398.7	2030.7	1642.7	40.2	10112.3
Percentage	63.3%	20.1%	16.2%	0.4%	100.0%

Table 3: Future Global Energy Demand [3]

	1980	1990	2000	2010	2020
A. Primary Energy Demand(MTOE)					
1. IEA/OECD (1982)	6,900	8,230 8,750	10,500 12,100		
2. CEC (1986)	7,270		10,800		
3. IIASA (1985)	6,800	8,000	9,900		
4. WEC (1986)	7,700	9,400	11,100	3,300	15,500
5. Goldenberg(1985) et. al	7,800				8,400
B. Developing Countries					
1. IEA/OECD	950	1,410 1,620	2,320 2,840		
2. CEC	1,100		2,270		
3. WEC	1,950		3,500		5,000
4. Goldenberg et. al	2,220				4,400
C. World Electricity Demand (Twh)					
1. IIASA	8,100	11,000	16,200	17,500	
2. Goldenberg et al	8,150			15,600	

IEA - International Energy Agency
CEC - Commission of the European Community
WEC - World Energy Commission
IIASA- International Institute for Applied Systems Analysis

Table 4: Assumptions for cost-economic analysis [5]

1. Cost

	HTR				Coal	Gas
	N1	N2	N3	N4		
Construction cost ($M)	1229	1040	875	590	450	245
O & M						
Fixed ($M)	13.5	13.5	11.3	13.5	5.74	2.43
Variable (mills/Kwh)					4.26	1.82
2. Construction Period (Years)	6	6	5	5	5	4.5
3. Economic Life (Years)	30	30	30	30	25	25
4. Discount Rate	10 %					
5. Average Plant Factor	80 %					
6. Size of plants	320 MWe					
7. Average Interest Rate	6.5 % (IDC capitalized)					

TABLE 5: COMPARISON AMONG ALTERNATIVES

DISCOUNT RATE = 10%

(FIGURES IN MILLS/KWH)

ALTERNATIVES	CAP.CH.	O&M	FUEL	DECOM.	TOTAL
1.1. NUCLEAR COST SCENARIO N1	75.41	6.13	11.17	0.31	93.02
1.2. NUCLEAR COST SCENARIO N2	63.82	6.13	11.17	0.31	81.42
1.3. NUCLEAR COST SCENARIO N3	53.69	5.16	9.12	0.26	68.23
1.4. NUCLEAR COST SCENARIO N4	34.68	6.13	8.00	0.31	49.12
2.1. COAL WITH FGC	27.61	6.87	23.01	0.00	57.48
2.2. COAL WITHOUT FGC	21.24	3.43	23.01	0.00	47.68
3. OIL	15.99	2.92	35.32	0.00	54.23
4.1. GAS/STEAM	15.08	2.92	36.17	0.00	54.18
4.2. GAS/GT	6.79	4.20	50.92	0.00	61.91

TABLE 6: COMPARISON AMONG ALTERNATIVES
(CAPITAL COST INCREASED BY 15%)

1.1. NUCLEAR COST SCENARIO N1	86.72	6.13	11.17	0.31	104.33
1.2. NUCLEAR COST SCENARIO N2	73.39	6.13	11.17	0.31	91.00
1.3. NUCLEAR COST SCENARIO N3	61.74	5.16	9.12	0.26	76.29
1.4. NUCLEAR COST SCENARIO N4	39.89	6.13	8.00	0.31	54.33
2.1. COAL WITH FGC	31.75	6.87	23.01	0.00	61.62
2.2. COAL WITHOUT FGC	24.42	3.43	23.01	0.00	50.86
3. OIL	18.39	2.92	35.32	0.00	56.63
4.1. GAS/STEAM	17.34	2.92	36.17	0.00	56.44
4.2 GAS/GT	7.81	4.20	50.92	0.00	62.93

TABLE 7: COMPARISON AMONG ALTERNATIVES
(CAPITAL COST DECREASED BY 15%)

1.1. NUCLEAR COST SCENARIO N1	64.10	6.13	11.17	0.31	81.71
1.2. NUCLEAR COST SCENARIO N2	54.24	6.13	11.17	0.31	71.85
1.3. NUCLEAR COST SCENARIO N3	45.64	5.16	9.12	0.26	60.18
1.4. NUCLEAR COST SCENARIO N4	29.48	6.13	8.00	0.31	43.92
2.1. COAL WITH FGC	23.47	6.87	23.01	0.00	53.34
2.2. COAL WITHOUT FGC	18.05	3.43	23.01	0.00	44.49
3. OIL	13.59	2.92	35.32	0.00	51.83
4.1. GAS/STEAM	12.82	2.92	36.17	0.00	51.91
4.2. GAS/GT	5.77	4.20	50.92	0.00	60.89

TABLE 8: COMPARISON AMONG ALTERNATIVES
(EFFECT OF VARIATION ON CAPACITY FACTOR)

ALTERNATIVES	CAPACITY. FACTOR %	CAP.CH.	O&M	FUEL	DECOM.	TOTAL
		(FIGURES IN MILLS/KWH)				
1.1. NUCLEAR COST SCENARIO N1	50.00%	120.66	6.13	11.17	0.31	138.27
1.2. NUCLEAR COST SCENARIO N2	50.00%	102.10	6.13	11.17	0.31	119.71
1.3. NUCLEAR COST SCENARIO N3	50.00%	85.91	5.27	9.12	0.26	100.55
1.4. NUCLEAR COST SCENARIO N4	50.00%	55.49	6.13	8.00	0.31	69.93
2.1. COAL WITH FGC	50.00%	44.17	6.87	23.01	0.00	74.00
2.2. COAL WITHOUT FGC	50.00%	33.98	3.43	23.01	0.00	60.42
3. OIL	50 00%	25.59	2.92	35.32	0.00	63.83
4.1. GAS/STEAM	50.00%	24.13	2.92	36.17	0.00	63.23
1.1. NUCLEAR COST SCENARIO N1	60.00%	100.55	6.13	11.17	0.31	118.16
1.2. NUCLEAR COST SCENARIO N2	60.00%	85.09	6.13	11.17	0.31	102.70
1.3. NUCLEAR COST SCENARIO N3	60.00%	71.59	5.27	9.12	0.26	86.24
1.4. NUCLEAR COST SCENARIO N4	60.00%	46.25	6.13	8.00	0.31	60.68
2.1. COAL WITH FGC	60.00%	36.81	6.87	23.01	0.00	66.68
2.2. COAL WITHOUT FGC	60.00%	28.31	3.43	23.01	0.00	54.76
3. OIL	60.00%	21.32	2.92	35.32	0.00	59.20
4.1. GAS/STEAM	60.00%	20.11	2.92	36.17	0.00	59.20
1.1. NUCLEAR COST SCENARIO N1	70.00%	86.19	6.13	11.17	0.31	103.79
1.2. NUCLEAR COST SCENARIO N2	70.00%	72.93	6.13	11.17	0.31	90.54
1.3. NUCLEAR COST SCENARIO N3	70.00%	61.36	5.27	9.12	0.26	76.01
1.4. NUCLEAR COST SCENARIO N4	70.00%	39.64	6.13	8.00	0.31	54.08
2.1. COAL WITH FGC	70.00%	31.55	6.87	23.01	0.00	61.42
2.2. COAL WITHOUT FGC	70.00%	24.27	3.43	23.01	0.00	50.71
3. OIL	70.00%	18.28	2.92	35.32	0.00	56.52
4.1. GAS/STEAM	70.00%	17.24	2.92	36.17	0.00	56.33
1.1. NUCLEAR COST SCENARIO N1	80.00%	75.41	6.13	11.17	0.31	93.02
1.2. NUCLEAR COST SCENARIO N2	80.00%	63.82	6.13	11.17	0.31	81.42
1.3. NUCLEAR COST SCENARIO N3	80.00%	53.69	5.27	9.12	0.26	68.34
1.4. NUCLEAR COST SCENARIO N4	80.00%	34.68	6.13	8.00	0.31	49.12
2.1. COAL WITH FGC	80.00%	27.61	6.87	23.01	0.00	57.48
2.2. COAL WTHOUT FGC	80.00%	21.24	3.43	23.01	0.00	47.68
3. OIL	80.00%	15.99	2.92	35.32	0.00	54.23
4.1. GAS/STEAM	80.00%	15.08	2.92	36.17	0.00	54.18
1.1. NUCLEAR COST SCENARIO N1	90.00%	67.03	6.13	11.17	0.31	84.64
1.2. NUCLEAR COST SCENARIO N2	90.00%	56.72	6.13	11.17	0.31	74.33
1.3. NUCLEAR COST SCENARIO N3	90.00%	47.73	5.27	9.12	0.26	62.37
1.4. NUCLEAR COST SCENARIO N4	90.00%	30.83	6.13	8.00	0.31	45.27
2.1. COAL WITH FGC	90.00%	24.54	6.87	23.01	0.00	54.41
2.2. COAL WITHOUT FGC	90.00%	18.88	3.43	23.01	0.00	45.32
3. OIL	90.00%	14.21	2.92	35.32	0.00	52.45
4.1. GAS/STEAM	90.00%	13.41	2.92	36.17	0.00	52.50
1.1. NUCLEAR COST SCENARIO N1	100.00%	60.33	6.13	11.17	0.31	77.94
1.2. NUCLEAR COST SCENARIO N2	100.00%	51.05	6.13	11.17	0.31	68.66
1.3. NUCLEAR COST SCENARIO N3	100.00%	42.95	5.27	9.12	0.26	57.60
1.4. NUCLEAR COST SCENARIO N4	100.00%	27.75	6.13	8.00	0.31	42.19
2.1. COAL WITH FGC	100.00%	22.09	6.87	23.01	0.00	51.96
2.2. COAL WITHOUT FGC	100.00%	16.99	3.43	23.01	0.00	43.43
3. OIL	100.00%	12.79	2.92	35.32	0.00	51.03
4.1. GAS/STEAM	100.00%	12.06	2.92	36.17	0.00	51.16

TABLE 9: COMPARISON AMONG ALTERNATIVES
(VARIATION IN DISCOUNT RATE)

ALTERNATIVES	DISCOUNT RATE	CAP.CH.	O&M	FUEL	DECOM.	TOTAL
				(MILLS/KWH)		
1.1. NUCLEAR COST SCENARIO N1	8.00%	60.14	6.13	11.17	0.45	77.89
1.2. NUCLEAR COST SCENARIO N2	8.00%	50.89	6.13	11.17	0.45	68.64
1.3. NUCLEAR COST SCENARIO N3	8.00%	42.82	5.16	9.12	0.38	57.48
1.4. NUCLEAR COST SCENARIO N4	8.00%	27.91	6.13	8.00	0.45	42.49
2.1. COAL WITH FGC	8.00%	22.74	6.87	23.25	0.00	52.85
2.2. COAL WITHOUT FGC	8.00%	17.49	3.43	23.25	0.00	44.17
3. OIL	8 00%	13.17	2.92	36.08	0.00	52.17
4.1. GAS/STEAM	8.00%	12.42	2.92	36.95	0.00	52.30
4.2. GAS/GT	8.00%	5.82	4.20	52.02	0.00	62.04
1.1. NUCLEAR COST SCENARIO N1	10.00%	75.41	6.13	11.17	0.31	93.02
1.2. NUCLEAR COST SCENARIO N2	10.00%	63.82	6.13	11.17	0.31	81.42
1.3. NUCLEAR COST SCENARIO N3	10.00%	53.69	5.16	9.12	0.26	68.23
1.4. NUCLEAR COST SCENARIO N4	10.00%	34.68	6.13	8.00	0.31	49.12
2.1. COAL WITH FGC	10.00%	27.61	6.87	23.01	0.00	57.48
2.2. COAL WITHOUT FGC	10.00%	21.24	3.43	23.01	0.00	47.68
3. OIL	10.00%	15.99	2.92	35.32	0.00	54.23
4.1. GAS/STEAM	10.00%	15.08	2.92	36.17	0.00	54.18
4.2. GAS/GT	10.00%	6.79	4.20	50.92	0.00	61.91
1.1. NUCLEAR COST SCENARIO N1	12.00%	92.68	6.13	11.17	0.21	110.19
1.2. NUCLEAR COST SCENARIO N2	12.00%	78.43	6.13	11.17	0.21	95.94
1.3. NUCLEAR COST SCENARIO N3	12.00%	65.99	5.16	9.12	0.18	80.44
1.4. NUCLEAR COST SCENARIO N4	12.00%	42.24	6.13	8.00	0.21	56.58
2.1. COAL WITH FGC	12.00%	33.04	6.87	22.80	0.00	62.71
2.2. COAL WITHOUT FGC	12.00%	25.42	3.43	22.80	0.00	51.65
3. OIL	12.00%	19.14	2.92	34.66	0.00	56.73
4.1. GAS/STEAM	12.00%	18.05	2.92	35.50	0.00	56.48
4.2. GAS/GT	12.00%	7.83	4.20	49.98	0.00	62.01
1.1. NUCLEAR COST SCENARIO N1	14.00%	111.97	6.13	11.17	0.14	129.41
1.2. NUCLEAR COST SCENARIO N2	14.00%	94.75	6.13	11.17	0.14	112.19
1.3. NUCLEAR COST SCENARIO N3	14.00%	79.72	5.16	9.12	0.12	94.12
1.4. NUCLEAR COST SCENARIO N4	14.00%	50.55	6.13	8.00	0.00	64.68
2.1. COAL WITH FGC	14.00%	39.05	6.87	22.62	0.00	68.54
2.2. COAL WTHOUT FGC	14.00%	30.04	3.43	22.62	0.00	56.09
3. OIL	14.00%	22.62	2.92	34.11	0.00	59.65
4.1. GAS/STEAM	14.00%	21.33	2.92	34.94	0.00	59.19
4.2. GAS/GT	14.00%	8.93	4.20	49.18	0.00	62.31
1.1. NUCLEAR COST SCENARIO N1	16.00%	133.32	6.13	11.17	0.10	150.71
1.2. NUCLEAR COST SCENARIO N2	16.00%	112.82	6.13	11.17	0.10	130.21
1.3. NUCLEAR COST SCENARIO N3	16.00%	94.92	5.16	9.12	0.08	109.28
1.4. NUCLEAR COST SCENARIO N4	16.00%	59.62	6.13	8.00	0.10	73.85
2.1. COAL WITH FGC	16.00%	45.64	6.87	22.47	0.00	74.97
2.2. COAL WITHOUT FGC	16.00%	35.10	3.43	22.47	0.00	61.01
3. OIL	16.00%	26.43	2.92	33.64	0.00	63.00
4.1. GAS/STEAM	16.00%	24.93	2.92	34.46	0.00	62.31
4.2. GAS/GT	16.00%	10.09	4.20	48.50	0.00	62.79

TABLE 10: COMPARISON AMONG ALTERNATIVES
(VARIATION IN ESCALATION OF FUEL PRICE)

10% DISCOUNT

(FIGURES IN MILLS/KWH)

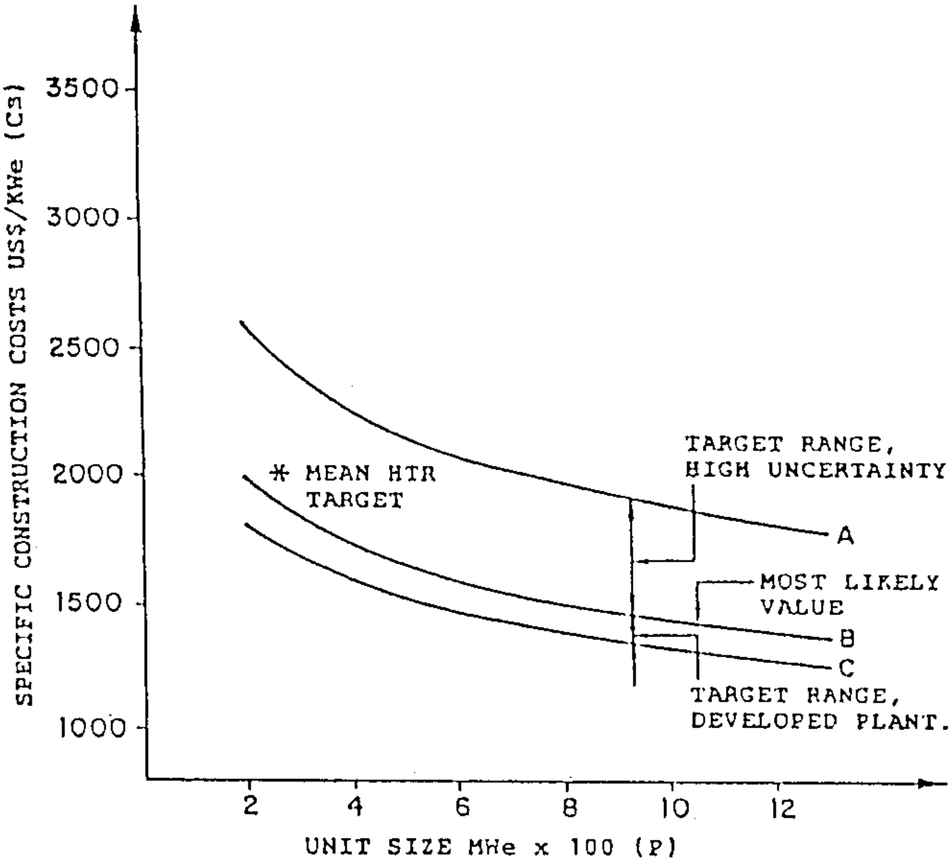
ALTERNATIVES	ESCALATION %	CAP.CH.	O&M	FUEL	DECOM.	TOTAL
1.1. NUCLEAR COST SCENARIO N1	0.00%	75.41	6.13	11.17	0.31	93.02
1.2. NUCLEAR COST SCENARIO N2	0.00%	63.82	6.13	11.17	0.31	81.42
1.3. NUCLEAR COST SCENARIO N3	0.00%	53.69	5.16	9.12	0.26	68.23
1.4. NUCLEAR COST SCENARIO N4	0.00%	34.68	6.13	8.00	0.31	49.12
2.1. COAL WITH FGC	0.00%	27.61	6.87	19.54	0.00	54.01
2.2. COAL WITHOUT FGC	0.00%	21.24	3.43	19.54	0.00	44.20
3. OIL	0.00%	15.99	2.92	25.37	0.00	44.28
4.1. GAS/STEAM	0.00%	15.08	2.92	25.98	0.00	43.98
1.1. NUCLEAR COST SCENARIO N1	1.00%	75.41	6.13	13.16	0.31	95.01
1.2. NUCLEAR COST SCENARIO N2	1.00%	63.82	6.13	13.16	0.31	83.41
1.3. NUCLEAR COST SCENARIO N3	1.00%	53.69	5.16	10.74	0.26	69.85
1.4. NUCLEAR COST SCENARIO N4	1.00%	34.68	6.13	9.42	0.31	50.54
2.1. COAL WITH FGC	1.00%	27.61	6.87	23.01	0.00	57.48
2.2. COAL WITHOUT FGC	1.00%	21.24	3.43	23.01	0.00	47.68
3. OIL	1.00%	15.99	2.92	29.87	0.00	48.79
4.1. GAS/STEAM	1.00%	15.08	2.92	30.60	0.00	48.60
1.1. NUCLEAR COST SCENARIO N1	2.00%	75.41	6.13	15.55	0.31	97.40
1.2. NUCLEAR COST SCENARIO N2	2.00%	63.82	6.13	15.55	0.31	85.81
1.3. NUCLEAR COST SCENARIO N3	2.00%	53.69	5.16	12.70	0.26	71.81
1.4. NUCLEAR COST SCENARIO N4	2.00%	34.68	6.13	11.14	0.31	52.26
2.1. COAL WITH FGC	2.00%	27.61	6.87	27.20	0.00	61.67
2.2. COAL WITHOUT FGC	2.00%	21.24	3.43	27.20	0.00	51.87
3. OIL	2.00%	15.99	2.92	35.32	0.00	54.23
4.1. GAS/STEAM	2.00%	15.08	2.92	36.17	0.00	54.18
1.1. NUCLEAR COST SCENARIO N1	3.00%	75.41	6.13	18.46	0.31	100.31
1.2. NUCLEAR COST SCENARIO N2	3.00%	63.82	6.13	18.46	0.31	88.71
1.3. NUCLEAR COST SCENARIO N3	3.00%	53.69	5.16	15.07	0.26	74.18
1.4. NUCLEAR COST SCENARIO N4	3.00%	34.68	6.13	13.22	0.31	54.34
2.1. COAL WITH FGC	3.00%	27.61	6.87	32.29	0.00	66.76
2.2. COAL WTHOUT FGC	3.00%	21.24	3.43	32.29	0.00	56.96
3. OIL	3.00%	15.99	2.92	41.92	0.00	60.83
4.1. GAS/STEAM	3.00%	15.08	2.92	42.94	0.00	60.94
1.1. NUCLEAR COST SCENARIO N1	4.00%	75.41	6.13	22.00	0.31	103.86
1.2. NUCLEAR COST SCENARIO N2	4.00%	63.82	6.13	22.00	0.31	92.26
1.3. NUCLEAR COST SCENARIO N3	4.00%	53.69	5.16	17.97	0.26	77.08
1.4. NUCLEAR COST SCENARIO N4	4.00%	34.68	6.13	15.76	0.31	56.88
2.1. COAL WITH FGC	4.00%	27.61	6.87	38.48	0.00	72.96
2.2. COAL WITHOUT FGC	4.00%	21.24	3.43	38.48	0.00	63.15
3. OIL	4.00%	15.99	2.92	49.97	0.00	68.88
4.1. GAS/STEAM	4.00%	15.08	2.92	51.18	0.00	69.18

Table 11 Economic Comparison of HTR with Coal

Indicators	N1 / Coal	N2 / Coal	N3 / Coal	N4 / Coal
Net Present Worth 10% Discount (in $ Million)	-417	-54	-126	105
Benefit- Cost Ratio	0.618	0.943	0.840	1.166
Economic Internal Rate of Return	0.97%	8.9%	6.26%	16.89%

Table 12 Economic Comparison of HTR with Gas

Indicators	N1 / Coal	N2 / Coal	N3 / Coal	N4 / Coal
Net Present Worth 10% Discount (in $ Million)	-455	-319	-189	56
Benefit- Cost Ratio	0.582	0.665	0.770	1.088
Economic Internal Rate of Return	3.09%	4.38%	6.14%	11.86%



LEGEND:

1. Curves A, B, C are derived from analysis of historic costs, presented as targets for LWRs in Europe (1987)

   A. Target range, high uncertainty

   B. Most likely value [Cs $p^{-0.2}$]

   C. Target range, developed plant

2. * - Target HTR specific cost

FIGURE 1

TARGET CONSTRUCTION COST OF HTR BASED ON LWR COST IN EUROPE

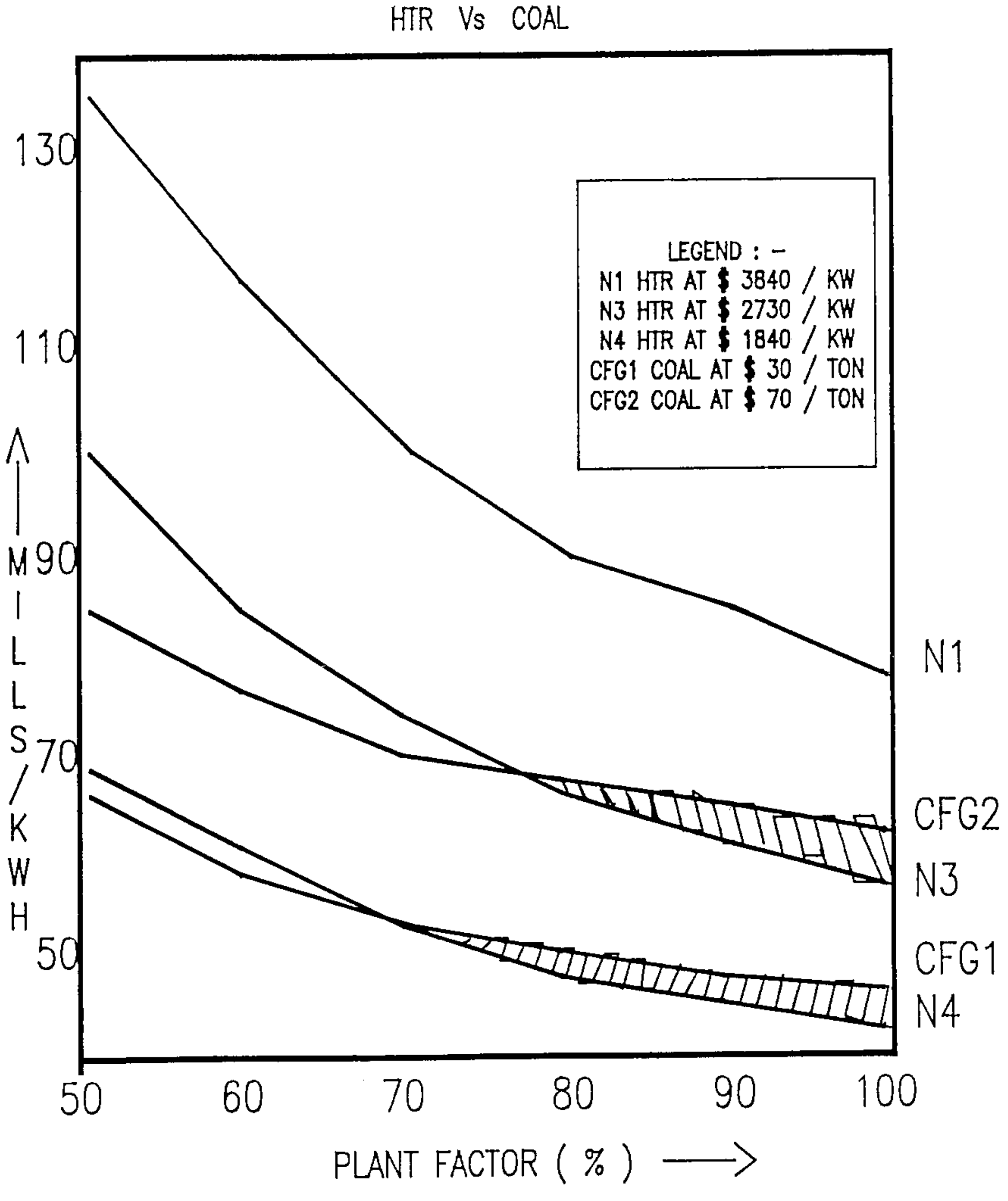


FIG. 2 COMPARISON BETWEEN OPTIONS

2.8 PROGRESS ON HTR APPLICATION'S STUDY IN INDONESIA

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ABSTRACT

Study on the application of HTRs for the enhanced oil recovery in the Duri oil field (Sumatra, Indonesia) was performed in 1986/1987. The economic and technological advantages over crude burning option were identified. Crude oil prices, HTR capital costs, discout rates, company's income structure represented dominant parameters. Further sensitivity calculations on important economic parameters were obtained to reflect the condition of 1988.

This nuclear option was also incorporated in the energy planning study for the whole Indonesia using the MARKAL model, and resulted the conditions of its applicability. The scenarios chosen in this MARKAL study were high and low GDP growth rate, whereas the criteria chosen were the minimum cost with and without a predetermined policy of reduced domestic use of oil. In the high scenario the HTRs as well as the natural gas options could not compete against the low cost boilers with crude-oil fuel. But in the case of reduced domestic oil use the HTRs came out to supplement the crude-burning boilers starting in the sixth five year plan (1994 - 1999), even earlier than the natural gas option.

The authors further discuss the industrial environment, in relation with the regional development, the possible local participation, as well as the plan to materialize the merits of this novel application.

1. Introduction

A prefeasibility study of the applicaton of HTRs in Indonesia, namely for the heavy oil recovery in the Duri oil field, was completed and reported to the Indonesian Government. The study was performed under the cooperation between KWU, Interatom of the Federal Republic of

Germany and BATAN, BPPT, MIGAS, LEMIGAS, PERTAMINA of Indonesia. From the Duri steam flood requirement 4 units of 4 HTR 200-MWt modules were exercised, and positive results were obtained. Various advantages, economical and technological, on the use of nuclear steam supply systems over conventional crude-burning ones were identified (1).

Later an economic study of only one unit of 4 HTR 200-MWt module introduction, resulting in a similar positive results, was also presented by KWR (2). Whilst uncertainties associated with assumptions still have to be verified through a comprehensive feasibility study, some issues are hindering the decision to proceed. Among these are short term practice of oil production sharing contracts, actual low oil price, optimistic view on the availability of other energy sources, institutional and safety aspects (3).

The 1988 oil price situation and its declining tendency did not support the (micro-economic) profitability of this nuclear alternative. Further if the burned crude has the price of the well-head cost and the field electricity is provided by the field associated gas or by the crude in the same well-head cost manner, it is very difficult for any other energy source to compete (4).

2. Energy Resources of Sumatra Riau Province

Sumatra is a large island among 13 thousand islands of Indonesia. The resion is 0.47 million square kilometers, compared to 2.0 millions of Indonesian land and to 5.2 millions of total Indonesian land and sea. Out of to-day's Indonesian population, which is 179 millions, 37 millions (or 20%) live in seven Provinces of Sumatra Island.

One of the Provinces is Riau, borderning with the Malaysia and Singapore, and it covers a small part of the eastern coast of Sumatra as well as archipelagoes spreading in the Malacca straight, on the Natuna and the South China Seas. This province, as most other Sumatra provinces are, is endowed with large mines and energy resources. Tin, bauxite, oil and gas are the products of Riau province, which area totals to 94.5 thousand square kilometers.

The Caltex Pacific Indonesia, operating a number of oil fields in this area, contributes the largest share of the whole Indonesian crude oil production. One of the oil fields is the Duri field, covering a productive area of about 100 square kilometers. Since the crude has high average gravity and viscosity (22 API, 120 cp), its exploitation requires enhanced oil recovery using the steam flooding, which had become the subject of this nuclear application study. Refineries are located on the coast (Dumai and Sungai Pakning), and are connected with pipeline system from the oil fields.

The gas reserve in the Natuna islands represents the largest in Indonesia, although the exploitation needs special treatment due to substantial content of carbon dioxide, Tin is mined in the Karimun, Kundur and Singkep islands, as bauxite in the Bintan island.

3. The Prefeasibility Study

The fact that producing crude by consuming one fifth out of the product has yet crossed the conservation concept of the energy resources, it also becomes an environmental issue later as well as an economic one whenever the oil price escalates consistently. Meanwhile studies for fuel alternatives, namely coal from the same island or gas from the Natuna islands, showed higher in cost as well as in capital.

The prefeasibility study executed jointly by KWU, Interatom, BATAN, BPPT, MIGAS, Lemigas, and Pertamina was aimed to judge whethr a nuclear alternative using HTR-Module is a viably economic solution and whethr a further comprehensive feasibility study ought to be performed.

It was assumed that several units of HTR-Module are installed for cogeneration of required injection steam and electricity consumed for production processes as well as for the oil complex needs.

In general the cogenerating plants shall serve for the whole lifetime of the oil field as the steam generator and as electricity source utilizing a back-pressure tubrine. But as soon as the oil field stops producing crude, the back-pressure turbine can be converted into a full condensing one, yielding electricity for the rest of the reactor life. This idea was thought to be very sound but it means that the plant shall be optimized for this future purpose, too.

Furthermore the nuclear alternative needs a substantial lead time of 6-8 years while the conventional (crude-burning) steam generation is proceeding steadily in the Duri fields. This means that the later the HTR is introduced the less unit number is required.

4. Conclusion of 4×4 HTR-Module Study

The technical concept, based on the proven KWU LWR-technology and the AVR experiences, is well established. Its safety concept is based on inherent physical properties, which enable the reactor to shut down the nuclear reaction and to remove the decay heat without relying on active engineering safety systems.

Compared to conventional steam supply by using only oil fired steam generators, the HTR-Module application for the Duri project, will increase the total government take from the year 2001 to 2010 by 10.2 billions US$, at 2% p.a. real oil price increase. This corresponds to an increase of about 63%, from 16.2 billions US$ (for the conventional alternative) to 26.4 billions US$ (for alternative 4×4 HTR-Module power plants). This additional "government take" will have considerable advantages for the national economic growth. For Caltex, it will increase their revenues, too.

Another important aspects from the national economic point of view is that approx. 17 millions bbl/y of crude can be substituted by nuclear fuel. Taking 26 US$/bbl real oil price (1987 US$) or correspondingly 40 $/bbl current in the year 2000, the resulting additional foreign currency amounts to about 700 millions of current US$/y.

The alternative fuels to crude and nuclear energy could be coal or natural gas from Natuna gas field. The required equivalent amount of coal would be about 5 million tons per year. Considering appropriate environmental conditions, the HTR-Module alternative offers by far more advantages compared to coal. Natuna gas can be considered as a competitive fuel to Duri crude. But up to now, no adequate information for comparison investigation is available. The date of possible introduction of a first HTR-Module power plant depends on the already running oil field development. Further conventional steam generators may be put on order and go into operation. This means that at a certain date the investment in conventional facilities might be so large that a decision for an HTR-Module alternative could be too late.

The minimum time for start of HTR-Module operation would be about 7 years:

- 1 year for feasibility study including determination of site data and clarification of financing,
- 2 years of site dependent preplanning,
- 4 years of plant construction.

That means, if the first HTR-Module plant should go into operation in 1995, it would be necessary to take the decision early enough, and the year 1988 was recommended.

The maximum level of estimated production capacity of the Duri oil field, which will be reached in the coming 10-12 years, requires approx. 4000-5000 t/h steam and 140 MW electricity. Generating steam and electricity the 4×4 HTR-Module plants can save 90-100 million tons of crude oil in 40 years. Further additional crude could be substituted if in the adjacent oil industry its electricity requirement is supplied from these HTR-Module power plants. The substituted oil will be available for increasing the export capacity or for domestic demand. Producing 300 MW electricity in total, the 4×4 HTR-Module will give a surplus of 160 MW which can be fed into the nearby grid, where the demand is increasing, too.

With the decision to introduce this HTR-Module technology, Indonesia has the opportunity to take the advantages of the corresponding technology transfer. In the project implementation and plant construction, it is expected that up to 40% of the overall investment can be supplied domestically.

Additional aspect is that the Indonesian industry jointly with KWU/ Interatom may get the opportunity, to have an access to the Asian market for the HTR technology. The HTR-Module offers also additional applications, such as steam and electricity cogeneration in refineries, petrochemical complexes and other industries.

Based on the statement mentioned before and the jointly elaborated results, considerations can now be taken whether or not this technology should be introduced in Indonesia. If a positive decision for a project by the respective authorities of Indonesia is considered, thereafter a feasibility study should be initiated and the respective financing should be secured. The feasibility study should include items such as:

- determination of site data required for the HTR-Module plant;
- optimization of field development plan adjusted to HTR-Module application;
- plant concept including involvement of Indonesia suppliers;
- organizational set up of the owners/operators group;
- financial planning.

5. Introduction of 1×4 HTR-M

The input data for the investment covers one unit of four HTR reactor modules, one unit of water treatment plant, owners cost/common facilities, steam distribution systems and various operating & maintenance costs. The difference to the former analysis is that a certain sale price of steam and of electricity has been assumed. In other words, having assumed steam and electricity prices, economic merits of 1×4 HTR-M are calculated.

The investigations are performed on the basis of current price. A 4% p.a. inflation rate and 2 DM/US$ exchange rate are assumed. Sensitivities in oil price and currency exchange rate variations have also been performed.

For economic evaluations the most relevant criteria calculated are net cash flow project operation, internal rate of return (IRR), net present value (NPV) of the net cash flow, payout time.

The feasibility and profitability of this project, like other projects in the oil industry, depend essentially on the oil price development. It was assumed also that oil price used is 26 US$/bbl (real) in the year 2000 and 31 US$/bbl in the year 2015 (upper boundary). These values correspond with the chosen escalation in the study: 1, 2 and 3% p.a. oil price increase in real terms.

The analyses show that the payout time is 12-13 years. The resulting net cash flow, including interest during construction, indicates that in the first years (including 4 construction years) equity is needed in order to assure the finaciability of the project. In the period after 2000, the net cash flow curve shows very positive figure. The influence of DM/US$ exchange rate variations is insignificant.

The internal rate of return on investment lies between 14 to 20% at 1 and 3% real oil price increases. An IRR on investment of approximately 10% p.a. for the power plant utility business, respectively the electricity supply sector, has been considered as quite reasonable and is generally accepted. Considering an IRR of 15% p.a. as acceptable, the presented result shows promising prospects.

Hindering Issues

The nuclear alternative, according to the study, is justified in the long term (more than 20 years) assuming an escalation of oil price. While current practice in the product sharing contract refers to period of about 15 years, the prospect in more than 15 years is beyond the company's interest, especially when the actual low price of oil at present does not justify the usually accepted forecast of price escalation in the near term.

High capital cost of HTR modules compared to crude burning mobile steam generators is the next argument. The question is who shall invest and bear the risk additionally to the existing running project.

Another variant is that the oil company will purchase required steam and electricity to a nuclear company on site. This nuclear company will construct an NPP and apply for licenses, and will take care that all the steam and electricity produced are well absorbed by the oil production process and by the surrounding community continuously.

Furthermore the current arrangement in the production sharing contract includes incentives in which the oil company enjoys, that make nuclear alternative poses disadvantages due to the large capital investment. Among these disadvantages are the following:

- Investment credit, i.e. a substantial part of investment, is paid from the revenue to the company annually as an incentive of using new technology.

- Depreciation is accounted with the double declining balance, combined with the straight-line method.

- Internal rate of return shall be in the range of 20-25%.

The figures of proven oil reserves range between 6.6×10^9 barrels to 9.5×10^9 barrels. In both figures the Duri heavy oil is included. The rates at which the proven reserves are increased by new discoveries depend on the exploration expenditure. Exploration has to be stimulated to maintain an adequate production level. But since 1982 the actual exploration and development expenditure have declined steadily. Exploration activities in more remote and offshore areas have posed increasing cost in the past, therefore to be optimistic it is likely that only with increasing total exploration expenditure a sufficient amount of new discoveries can be obtained in the long run.

According to the MARKAL study, it was predicted that today proven reserves would be depleted about the year 2007 in case of the high figure even with the reduced domestic oil consumption strategy. On the other hand, the natural gas alternative for the Duri is sought to use the Natuna gas, being even expensive (about 5 $/MSCF) and capital intensive (1000 km long), although from the total proven reserve point the amount is very large (50.58 TCF). At present the gas alternative investigation is being proposed before stepping into further HTR study.

If ultimately this Duri HTR project is executed, it is the first in the world that an oil company in a developing country becomes the host for nuclear reactors. It implies that large efforts to convince various institutions shall be carried out especially in the safety aspects of the HTR.

6. Concluding Remarks

The preliminary feasibility study on application of HTR for the Duri oil field was successfully performed under the cooperation between KWU, Interatom of the Federal Republic of Germany and BATAN, MIGAS, LEMIGAS, Pertamina of Indonesia. Various advantages, economic and technological, on the use of nuclear steam supply systems (operating in the cogeneration mode) over conventional crude-burning ones were identified.

Uncertainties associated with assumptions still have to be verified through a comprehensive feasibility study, as thereby some issues are yet to be deliberated. Among these are short term practice of oil producttion sharing contract, actual low oil price, optimistic view on the availability of other energy resources, institutional and safety aspects.

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2.9 STUDY OF FISSION PRODUCTS REDEPOSITION BY COMEDIE LOOP IN THE EXPERIMENTAL SILOE REACTOR

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CEA has a long national and international experience in the field of experimental irradiations. It is operating two experimental reactors OSIRIS (Saclay/France) and SILOE (Grenoble/France), in which a large number of devices and loops about the different reactors systems used through the world are irradiated. For HTGR Studies, a COMEDIE helium loop has been built to study the release and deposition of fission products from the fuel.

A modification of this loop, to be able to do in situ blow down tests to study fission products lift off, is in progress.

Around the loop itself, continuous fission gas analysis is feasible in a special equipped Fission Gas Analysis Laboratory. Post-Irradiation Examinations can be conducted in hot cells, located close to the reactor itself.

1) SILOE EXPERIMENTAL REACTOR

The table 1 gives the main characteristics of this experimental reactor.

Table 1

SILOE REACTOR	
General Characteristics	• Open core pool -type reactor • Power rating : 35 MW • Reflector : water and beryllium • Core : 93 % enriched U_5 standard fuel elements • Available fluxes : - thermal neutrons : 4.10^{14} n.cm^{-2}.s^{-1} - fast neutrons, energy greater than 0.1 MeV : $4.5.10^{14}$ n.cm^{-2}.s^{-1} • Four horizontal neutron beams - maximum flux of thermal neutrons : 10^9 n.cm^{-2}.s^{-1} • Material testing : 75 % • Fundamental research : 20 % • Isotopes production and miscellaneous : 5 %

Table 1 (Suite)

SILOE REACTOR	
Experimental Facilities	• Direct access to the core from the pool • Irradiation locations with very high fast neutron flux in the reactor core • Possibility of irradiation of long fuel rods • Movable devices mechanisms allowing continuous adjustment of irradiation power • Core shape adjustable to experiments
The Hot Cell	• Dimensions : 4,50 x 2,00 x 3,50 m • Direct underwater access from the irradiation pool • Special equipment for observation (periscope and television camera), dismantling of capsules For post - irradiation examinations : - the irradiated materials are transferred to specialized hot Laboratories
Computer controlled data acquisition and processing (reactor and experiments parameters)	Data acquisition and processing systems enable : • slow rate acquisition of analog measurements, as well as fast rate as high as 20 bursts per second of 140 measurements • real time acquisition and processing • automatic control and regulation • visualization of process output on curve plotters and cathode ray tube displays • log book edition

2) COMEDIE LOOP

The COMEDIE loop is a fuel irradiation loop for High Temperature Gas cooled Reactor systems experiments.

Table 2

- Operating Pressure : 30 to 70 bars
- Operating Temperature : 400° to 800° C
- Flow of Helium : 40 g/s
- Available diameter in the irradiation zone : 70 mm
- Available height : 500 mm
- Maximum unperturbed flux :
  - thermal neutrons : $3.0.10^{14}$ $n.cm^{-2}.s^{-1}$
  - fast neutrons : $3.0.10^{13}$ $n.cm^{-2}.s^{-1}$
  - gamma heating : 4.5 wg^{-1}
- Gas temperature :
  - at the exit to the in pile section : 300 to 800° C
  - in the blower : 300° to 550° C
- Gas composition : helium with the addition of CO, CO_2, H_2, CH_4 and H_2O.

A synoptic view of the loop is shown on diagram 1.

2.1. Applications

The loop is designed for the study of the release, migration, and deposition of fission products formed in high temperature gas cooled reactor fuel elements.

- irradiation of part of a fuel block under normal operating conditions,
- reirradiation of a fuel element already irradiated in another facility,
- release of fission products by the fuel element, ie migration of fission products out the particle, then the matrix graphite and finally through the block into the gas,
- deposition of solid fission products, ie entrainment and deposition of fission products on the metallic tubes or on other components,
- contamination of specimens by the fission products carried by the gas and study of decontamination,
- corrosion of materials in presence of fission products carried by the gas.

2.2. Description

- The entire flow of gas crosses successively the irradiation section, the deposition section and the filter ;
- The gas is propelled by a gas bearing blower ;
- The gas flow can be adjusted from 0 to 40 $g.s^{-1}$.

The loop includes :

• a section for the irradiation of the fuel element,
• a section for the deposition of condensable fission products ; both are located in the reactor pool,
• an out of pile part with heater, blower, cooler and control panels.

A filter placed immediately after the deposition section allows the distribution of fission products in the circuit to be determined.

2.2.1. Fuel irradiation section

The loop is of dimensions to be able to irradiate FORT ST-VRAIN type or AVR type fuel element.

The irradiation section occupies position 76 on the SILOE grid and can be displaced 230 mm by a mechanical moving system, in the reactor pool.

The irradiation section is limited by a tube separating the outward and return flow of the gas.

The irradiated loading is centered in the separating tube. The upper part of the loading ensures the biological shielding of the plug, the dynamics of the gas flow, and the connections of the instrumentation.

It is possible to remove and replace the loading using SILOE's hot cell, and so to re-irradiate fuel.

• Useful diameter : 70 mm,
• Useful height corresponding to the zone of greatest flux : 500 mm
• Temperature : 300° to 800° C,
• Gas flow rate : 0 to 40 $g.s^{-1}$.

The irradiation section can be displaced to achieve fuel power cycling.

Temperature and flow variations are feasible.

2.2.2. Deposition section

The deposition section includes three bundles of tubes. The number of tubes in each bundle is a function of their diameter. Length of the tubes : 2 800 mm.

The cooling of the tubes is uniformly ensured along the whole length by the outgoing gas. Thermocouples measure the temperature at several points. Speed is adjustable.

In a bundle, the tubes are identical, but could differ between bundles. It is thus possible to simultaneously study the behaviour of three types of tubes, placed in identical thermal and hydraulic conditions.

Analysis of the deposits formed in the tubes can be made :

- during stops in the irradiation by gamma scaning on the in pile bench
- at the end of the experiment after dismantling the clusters.

2.2.3. Out of pile part

It includes :

Cooler

This is a tube exchanger cooled by industrial water. A by-pass valve enables adjustment of the exit temperature of the gas.

Blower

This is a hydrodynamic blower supplied by a frequency changing unit. The minimal temperature of the walls in contact with the gas is 120° C.

The characteristics of the operating point maximum are :

- P inlet : 70 bars
- T inlet : 550° C
- Speed : 11.850 rpm
- ΔP : 1 bar
- Flowrate : 60 g.s^{-1}.

Heater : power 32 Kw;

Auxiliary Purification section

The circuit is sized to treat 1 to 5 % of the flow of the principal circuit, i.e. 0.6 to 2 g.s^{-1}. The gas passes through three successive stages, each containing two traps, one in operation and the other on standby or under regeneration. The traps of the first stage are filled with a product oxidizing the hydrogen and the carbon dioxide at a temperature of 250° C.

The traps in the second stage contain 4 Å molecular sieve at room temperature. The filter traps most of the water and carbon dioxide.

The last stage traps contain active charcoal maintained at - 195° C by immersion in liquid nitrogen. All the gaseous impurities contained in the helium are trapped, particularly fission gases.

The molecular sieve and active charcoal traps are lead shielded.

Other auxiliaries :

Analysis

Eight sampling points are situated throughout the circuits to analyse the gases permanently present in the helium by chromatography.

Three sampling points out of eight also enable an analysis of fission gases by gamma spectrometry.

Analysis of the humidity of the gas is made with special cells.

Additives

The addition of impurities in the helium is carried out at the blower exit by means of very high pressure systems.

It is possible to add :

- carbon dioxide and monoxide
- methane
- hydrogen
- water vapor and liquid water.

Various supply systems : helium, vacuum, etc.

Control panels with electrical supply of heaters blower, automatic monitory of the loop, alarms.

Data acquisition systems for the loop : computers and recorders.

3. COMPLEMENTARY FACILITIES

To study the evolution of a nuclear fuel, it is particularly interesting to follow the life of the fission products formed in it, as well as their migration kinetics.

To complete the COMEDIE loop, two facilities are available close connected to SILOE Reactor

- Fission gas Analysis Laboratory
- Hot Cells for Post Irradiation Examinations.

3.1. Fission Gas Analysis Laboratory

This is a unique Laboratory for fission gas analysis located in the SILOE reactor building and operated by CEA/Fuel Behaviour Study Dpt.

Measurements are made of the instantaneous release of stable and radioactive xenon and krypton emitted by a nuclear fuel operating under power and temperature conditions representative of those of a plant.

Drawing of the fission gas towards the analyser instruments is carried out by continuous or discontinuous sweeping with purified helium.

The loop for sampling and continuous analysis of gas is shown in fig. 2.

3.2. Hot Cells Post-Irradiation Examinations

The main possibilities are :

- dismantling
- non destructive examinations :
  . radiography, visual examinations, dimensional measurements, gamma scanning, specific weight, sampling of fission gas and free volume measurements, etc.
- destructive examinations :
  . tensile tests, hardness tests, etc.
  . metallographic examinations :
  In the HTR field, special examinations can be performed on the particles and fuel block : free volume, micrographic examinations, diffusion rate of fission products.

3.3. In pile gamma spectrometry

A high precision gamma-scanning bench, fully automatized and immersed in the reactor pool, allows irradiated rigs to be analysed as early as one our after an irradiation cycle.

The in pile section is disconnected for transfer to the bench.

By displacement, we determine :

- the repartition of fission products in the loading of the irradiation section ;
- the deposits formed in each bundle of tubes of the deposition section.

4) EXPERIMENTAL PROGRAMS

Essential problems which can be studied in COMEDIE loop relate to release migration and deposition of fission products on the surfaces, both in normal operation and lift off in the case of speed variation in tubes of the heat exchanger.

In the case of normal operation, in steady state conditions, COMEDIE loop gives the possibility to know the total balance of the fission products :

- knowledge of the fission products source, to create a significant fission products source, bare or failed particles can be included in the fuel block.

- knowledge of the deposited fission products.

- knowledge of the fission products composition of the flowing gas.

The effect of a blow down experiment on the deposited fission products is very interesting to be studied. As the deposition section is made of 3 clusters of several tubes, some of them are the reference for the deposits, and by special mechanisms, it is possible to create in situ blow down effect in the others. In this case, deposited fission products are partly swept away. The balance of deposited and drawn fission products is possible.

COMEDIE loop is now in progress for a refurbishment in the frame of a DOE/MMES Contract. The loop will be modified, so as to be able after an irradiation period to create fission products deposition on well qualified tubes of the deposition section, to blow helium through some of these tubes and recover fission products on a special filter located at the head of the loop. These experiments can be done with several parameters :

- shear ratio
- quality and material of tubes
- temperatures
- water content and impurities in the gas.

This program is foreseen on a period to 1993.

With the availability of the COMEDIE loop in SILOE Reactor and supporting facilities, CEA is in possession of a very highly capable facility to conduct irradiation programs in the field of HTGR Technology. On the other hand, CEA has reached a very high level in the knowledge of irradiation technology from years, in the field of all reactors systems by its technical means and the qualification of the staffs. CEA is so able to offer interesting possibilities to study HTGR problems under irradiation.

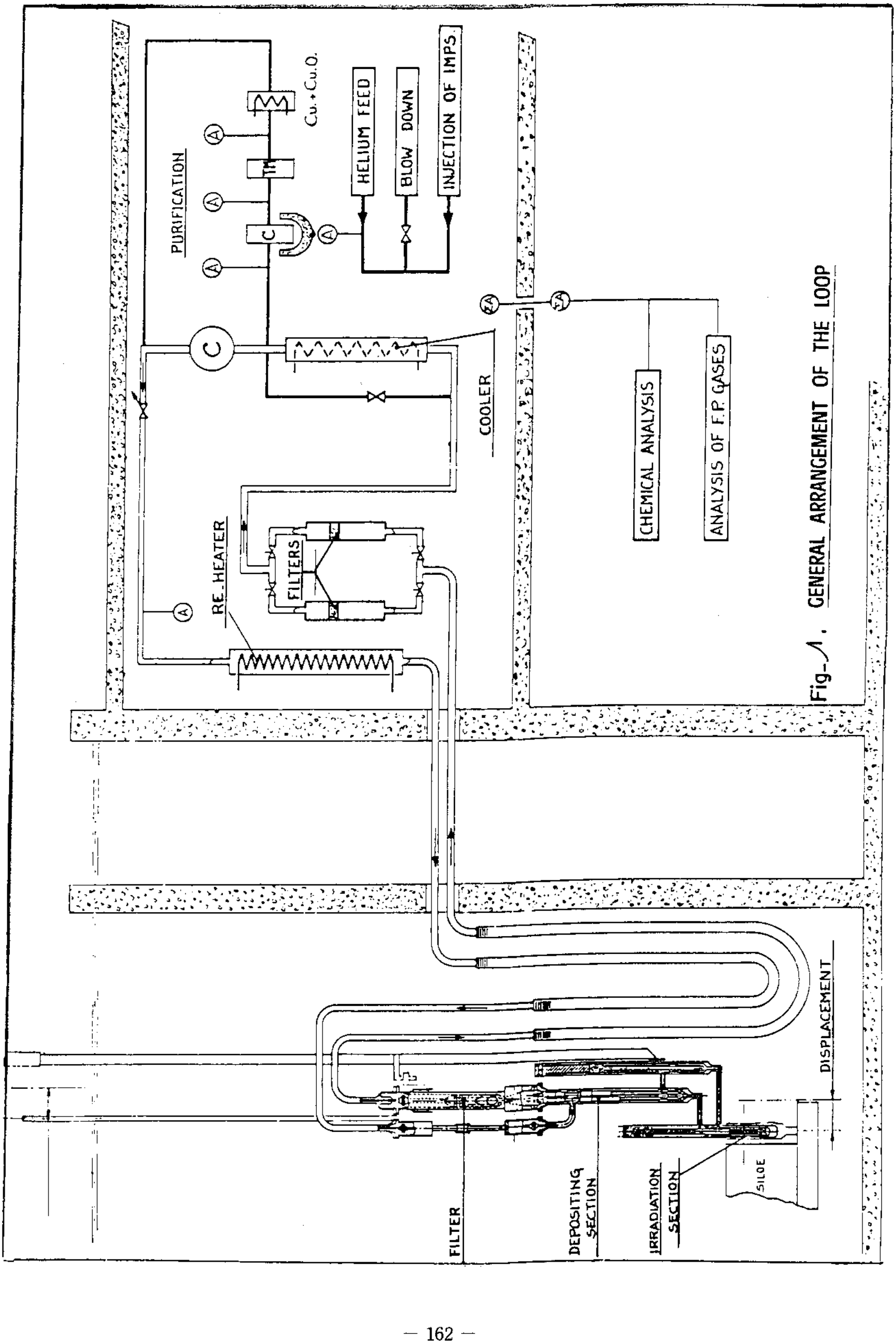

Fig_1. GENERAL ARRANGEMENT OF THE LOOP

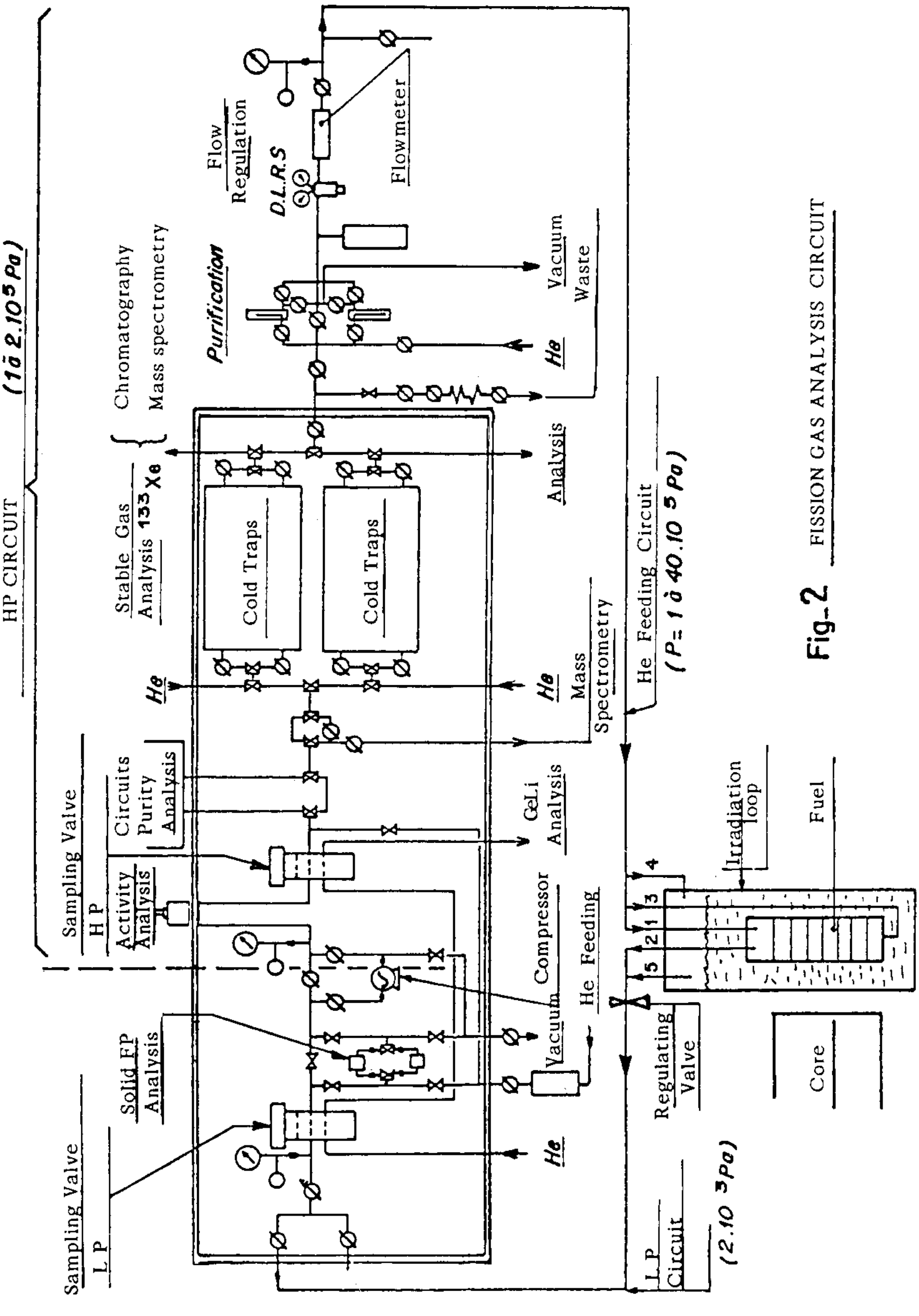

Fig.-2 FISSION GAS ANALYSIS CIRCUIT

2.10 DESIGN AND SAFETY CONSIDERATIONS IN THE JAPANESE HIGH-TEMPERATURE ENGINEERING TEST REACTOR (HTTR)

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ABSTRACT

The HTTR is a high temperature gas cooled test reactor with thermal output of 30MW and maximum reactor outlet coolant temperature of 950°C, aiming at establishing and upgrading the technology basis for advanced HTGRs and performing various innovative high temperature basic researches.

This paper describes the design of the HTTR specifically focussing on design considerations for achievement of 950°C outlet gas temperature and safety considerations in the design. The paper also describes briefly safety evaluation of the HTTR and its construction schedule.

The HTTR is the first HTGR in Japan and will be the first HTGR in the world which provides high temperature gas higher than 800°C outside the reactor vessel. Therefore, various considerations were taken in the design to achieve reactor outlet coolant temperature of 950°C in the HTTR.

Due considerations were made how to minimize the maximum fuel temperature, how to reduce the coolant flow rate ineffective to cool the fuel, how to minimize thermal and irradiation stresses of graphite block, how to resolve the problems relevant to high temperature and high pressure in the primary cooling system design, and so on.

Several important safety considerations for fuels, reactor shut-down system, back-up reactor cooling system, multiple barriers against FP release, etc., were also adopted in the design of the HTTR.

These elaborate considerations in the design of the HTTR together with extensive research and development make it possible to produce and take out very high temperature helium gas up to 950°C outside the reactor vessel in the HTTR.

The safety review of the HTTR by the government is presently underway. Construction is scheduled to start in 1990 and first criticality is expected in 1995.

1. Introduction

The High Temperature Engineering Test Reactor (HTTR) aims at establishing and upgrading the technology basis necessary for an HTGR, serving at the same time as a potential tool for new and innovative basic researches. The HTTR has a prismatic block type fuel core with 30MW thermal output and outlet coolant temperature of 850°C at rated operation and 950°C at high temperature test operation.

Since 1969, the JAERI has carried out R&D works on block type fuel, high temperature alloy, high temperature in-core instrumentations, high temperature components, reactor physics, heat transfer and fluid dynamics, FP plate-out, etc., in order to construct the HTTR which can supply high temperature helium gas up to 950°C at the outlet of the reactor vessel for the first time in the world.

The design of the HTTR with the reactor outlet coolant temperature of 950°C has been realized with many considerations together with adoption of the results of these R&D works. On the other hand, there was no guideline of safety design for an HTGR in Japan in the early stage of HTTR design although there were detailed guidelines of safety design and safety evaluation for an LWR. Then, we internally set up the safety design principles for the HTTR discussing with the specialists inside and outside the JAERI and regulatory authorities.

The detailed design of the HTTR connected with the specific considerations for achievement of coolant temperature of 950°C in the reactor outlet and safety design principles for the HTTR together with some results of safety evaluation are described in the paper.

2. Design of the HTTR with Consideration for Achievement of Coolant Temperature of 950°C

2.1 Outline of HTTR design

The HTTR consists of a reactor pressure vessel (RPV), a primary cooling loop with an intermediate helium-helium heat exchanger (IHX) and a pressurized water cooler (PWC) in parallel, an auxiliary cooling system (ACS), reactor vessel cooling system (VCS) and related components. The RPV is 13.2m in height and 5.5m in inner diameter, and contains the 30MWt core and reactor internal structures as shown in Fig.1. A major specification of the HTTR is listed in Table 1. The reactor building contains a containment vessel, auxiliary systems for cooling systems, ventilation and air conditioning systems, a reactor control room, fuel handling and storage facilities and so on.

2.2 Core

The reactor core is graphite moderated and cooled by helium gas, and prismatic fuel elements in the form of hexagonal blocks are used. The active core consists of 30 fuel columns and 7 control rod guide columns as shown in Fig.2, where each column is 5 blocks (2.9m) high. The active core of 2.3m in diameter is surrounded by replaceable reflector composed of a layer of hexagonal graphite blocks. The permanent reflector surrounds the replaceable reflector and consists of large polygonal graphite blocks fixed by core restraint devices. Each hexagonal graphite block made of the domestic IG-110 has three dowels on the top and three associated sockets in the bottom, and the blocks are fixed with dowel-socket method. The reactor core is cooled by downward helium gas whose temperature is 395°C at the reactor inlet.

Reactivity control is provided by control rods, which are individually supported by mechanisms located in the stand-pipes connected to the hemispherical top head of the RPV and inserted into the fuel region and replaceable reflector region. The reactor shutdown from a high temperature condition is made by the insertion of 9 pairs of control rods in the reflector region at first, and then 7 pairs of control rods in the fuel region are inserted in the condition of core temperature lower than 850°C. Back-up shutdown capability is provided by insertion of boron-carbide/graphite pellets into separate holes in the control rod guide blocks. Refueling is accomplished under a depressurized condition during reactor shutdown.

In order to achieve high reactor outlet coolant temperature, it is important to keep the maximum fuel temperature as low as possible to maintain fuel intactness. To minimize the maximum fuel temperature, the optimization of fuel zoning in the axial direction as well as in the radial direction and reduction of coolant flow ineffective to direct cooling of the fuel were made in the design.

As concerns the achievement of uniform radial fuel temperature distribution, a four-region radial fuel zoning in which the higher enriched uranium fuel is provided in the outer zone is determined. Since there is a large temperature rise of the coolant of 450-550°C in the core, it is better to have an axial power distribution which is inversely proportional to the axial temperature distribution of the coolant rather than to establish a uniform power distribution in order to achieve a uniform axial fuel temperature distribution. Therefore, the fuels with different enrichment are also used in the axial direction to satisfy the above-mentioned requirement. The axial distributions of power and fuel temperature are shown in Fig.3.

As for the reduction of coolant flow ineffective to cool the fuel, such as leakage flow, cross flow and bypass flow, the following considerations are taken in the design.

1)Leakage flow through the gaps between permanent reflector blocks was reduced by narrowing gaps and adopting seal mechanism of seal elements and key elements in the gaps. Leakage flow through the gaps between hot plenum blocks was reduced by the newly developed triangular graphite seal elements.

2)Cross flow was reduced by making total core pressure drop as low as possible in coolant flow design and by scraping out the graphite block to diminish cross flow gap width.

3)The reduction of bypass flow was made by narrowing gaps between fuel blocks and between top shield blocks.

As the result, effective core flow rate as high as 88% of total flow has been achieved. Major nuclear and thermal-hydraulic characteristics are tabulated in Table 2. Reactivity power coefficient is largely negative and each reactivity temperature coefficient is also negative.

2.3 Fuel

A fuel block which keeps fuel rods in it is a pin-in-block type hexagonal block of 58cm in height, 36cm in across flats as shown in Fig.4.

The fuel consists of coated particles of low enriched uranium oxide whose average uranium enrichment is 6% and kernel diameter is 600μm. The particles are bonded together with graphite powder in fuel compacts, which are contained in a graphite sleeve to form a fuel rod. The fuel rods are contained within vertical holes of 4.1cm in diameter of the graphite blocks. Helium gas flows downward through the gap between the vertical hole and a fuel rod to remove the heat produced by fission and gamma heating.

In order to establish the design for the outlet helium gas of 950°C in the HTTR, the optimization of fuel channel radius, fuel pitch and number of fuel rods in the graphite block with respect to thermal and irradiation stresses was necessary. The irradiation stress is mainly caused by the dimensional decrease of graphite due to irradiation. The detailed stress analysis was performed for fuel blocks which have a viscoelastic behavior under thermal and irradiation-induced loads. In order to attain the required lifetime of fuel blocks, the following approaches were taken:

1)optimization of fuel channel radius and relative arrangement of channels in order to minimize the shutdown stress of blocks,

2)reduction of temperature difference between coolant flowing in the channels and coolant flowing in the block-to-block gap, and

3)elimination of fuel channels which give unallowable stress in 33 pin fuel block. This resulted in deleting two out of three channels in the corners of the fuel block adjacent to the replaceable reflector.

2.4 Reactor cooling systems

The reactor cooling systems are composed of a primary cooling system (PCS), an ACS and two VCSs as shown in Fig.5.

The primary cooling circuit of the PCS is separated into two lines outside the RPV. The heated helium gas is cooled by a 10MW He-He IHX in one line or cooled directly by a 20MW PWC in the other line. The heat in finally removed by an air cooler in both lines although another PWC is necessary after IHX in the first line. The demonstration plant for process heat utilization will be connected at the secondary line of the IHX in future.

In order to maintain the integrity of the reactor cooling system against the severe high pressure and high temperature condition various considerations are taken in the design.

A co-axial double piping system as shown in Fig.6 is adopted in the primary circuit. Cold gas of about 400°C flows the annular space between the inner and outer pipes toward reactor inlet and hot gas from the reactor flows inside the inner pipe. Temperature of the inner pipe is almost the same as that of the outer pipe because the inside of the inner pipe is thermally insulated by Kaowool and lined with Hastelloy-XR, which is a specifically developed heat resisting super alloy with high corrosion resistivity at high temperature. The pressure boundary is formed by the outer pipe, and the inner pipe is designed only to withstand the pressure difference between the inside and the outside of the inner pipe.

The IHX is a vertical helical coil counter flow type heat exchanger as shown in Fig.7. Primary coolant flows on the shell side with secondary coolant flowing on the tube side. Then, heat transfer tubes in the IHX form the primary coolant pressure boundary under the high temperature condition. In order to maintain the integrity of the tubes, mechanical loading on the tubes is reduced by minimizing pressure difference between the inside and the outside of the tubes. The pressure difference in a normal operation is only 0.1MPa. In addition, Hastelloy-XR is used at high temperature structures such as heat transfer tubes, tube sheets, tube support structures, a center pipe and a liner. The outer vessel of the IHX acting as the pressure boundary is designed to be cooled by colder helium gas which flows in the annular channel between the inner vessel and the outer, and heat transfer tube bundles and tube sheets are arranged axially symmetrical to reduce the thermal stresses.

Furthermore, the provision to get a sufficient mixing of hot helium gas in the plenum region inside the RPV is made in order to reduce hot streak in the primary coolant loop and the heat exchanger.

The coolant coming from the core is mixed in the 7 hot plenum blocks, then, the coolant from the 7 hot plenum blocks is mixed into one flow in the hot plenum as shown in Fig.8. There is, however, temperature difference of about 100°C in maximum among the flows from the 7 hot plenum blocks. Based on the extensive experiments, a T-type junction is adopted instead of a Y-type junction in the hot plenum blocks and a mixing promoting plate is provided in the hot plenum for sufficient mixing of the coolant. These provisions are estimated to reduce the hot streak from about 60°C to 6°C.

The ACS is operated when there is a trouble in the PCS but flow path in the primary cooling circuit is still kept. Both VCSs are operated at 100% flow rate during normal operation in order to cool biological shield around the RPV, and they serve to cool the RPV and core in the accident like pipe break of the PCS in which the flow path in the primary cooling circuit is not kept. Therefore, the PCS is nonsafety class but the ACS and the VCS are safety class.

2.5 Irradiation capability

Many possible irradiation regions are reserved in the HTTR to be served as an irradiation test reactor in order to promote innovative high temperature basic researches as well as to irradiate fuels and materials for HTGR. Namely, there are four regions in the core and many regions in the replaceable reflector and the permanent reflector as shown in Fig.9. Specific irradiation capabilities in the HTTR are to be able to irradiate a large-sized sample at elevated temperature although maximum thermal and fast neutron fluxes are an order of $10^{17} n/m^2 \cdot sec$.

3. Specific Considerations in Reactor Safety Design

3.1 Major safety design principles

The JAERI has set up following safety design principles for the HTTR referring the safety design criteria for LWRs taking into account the inherent safety characteristics of an HTGR and design requirements as a test reactor.

1)Coated particle fuel shall not fail during normal reactor operation and an anticipated operational occurrence. The maximum fuel temperature including systematic and random uncertainties shall not exceed 1600°C even in an anticipated operational occurrence.

2)A reactor shall be shut down safely and reliably from any operation condition with control rod system. A backup reactor shutdown system independent of the control rod system shall be provided.

3)A severe accident resulted by control rod ejection shall be avoided.

4)The residual core heat after reactor shutdown shall be removed safely and reliably for any anticipated operational occurrences and accidents.

5)A containment vessel shall be provided to prevent FP release and excessive air ingress into the core in case of depressurization accident.

6)The pressure of water in the secondary water cooling system shall be controlled to be lower than that of primary helium gas to prevent large amount of water ingress into the core in case of rupture of a heat tube in a pressurized water cooler.

7)The pressure of helium gas in the secondary helium cooling system shall be controlled to be slightly higher than that of primary helium gas to prevent FP leakage from the primary system to the secondary system through crack of a tube in the IHX.

8)The pressure-resisting and heat-resisting functions of the structures where high pressure and high temperature coolant is contained are separated in order to reduce mechanical loads on high temperature metal structures.

3.2 Limitation of fuel failure

In order to reduce FP release from the core, the maximum fraction of defected fuel during fabrication including contamination in coating and compact matrices is limited to be less than 0.2% and no fuel failure is allowed during normal operation and anticipated operational occurrences. The failure mechanisms of the fuel considered during operation are

1)palladium attack on sillicon-carbide layer,

2)kernel migration, and

3)burst of coating layer with excess FP gas pressure.

From the R&D works in the JAERI, it became clear that failures of coated particle fuel caused by the above-mentioned mechanisms could be kept out practically by limiting fuel temperature and maximum burnup lower than 1600^{o}C and 4% FIMA, respectively.

3.3 Stand-pipe fixing device

A stand-pipe fixing device is provided at the top of stand-pipes as shown in Fig.10 in which control rod drive mechanisms are installed, in order to prevent stand-pipe together with control rods from jumping up and adding large excess reactivity in the core in case of stand-pipe rupture accident. The stand-pipe fixing device also restricts leakage gas flow rate in the accident to prevent control rods floating up by the gas flow.

3.4 Residual heat removal systems

The PCS removes residual heat in the core in normal reactor shut-down condition. Besides the PCS, the HTTR has two other residual heat removal systems which are an ACS and a VCS. The flow diagram of the ACS and the VCS are shown in Fig.11.

(1)Auxiliary cooling system

The ACS automatically starts up when the reactor is scrammed in the accident in which the core cooling by forced circulation of coolant is possible, while the PCS is stopped. The ACS is a safety system with redundant dynamic components such as gas circulators, water pumps and valves which are also backed up with emergency power supply.

The residual heat of reactor core can be removed by the VCS without the ACS. The ACS, however, is needed from the viewpoint of operational flexibility, as the VCS takes long time to cool down the core.

(2)Vessel cooling system

The VCS is used as residual heat removal system when the forced circulation in the primary cooling circuit cannot be maintained due to the rupture of the inner pipe or both pipes in the co-axial double piping. The VCS is also a safety system equipped with two independent complete sets which are backed up with emergency power supply.

3.5 Multiple barriers against FP release

The HTTR has multiple barriers against FP release into the environment, namely, fuel coatings, reactor coolant pressure boundary, containment vessel and reactor building. Most of the high temperature gas-cooled reactors being designed in other countries have also these barriers except containment vessel. Therefore, the description is limited to the necessity of a containment vessel in the HTTR in the section.

The functions of the containment vessel in the HTTR are

1)to contain FP, and

2)to limit amount of air which possibly reacts with graphite in the reactor core in an accident.

There is no sufficient barrier against FP release for the accident of primary pipe rupture which cannot be excluded from the HTTR safety evaluations, if there is no reactor containment vessel.

The HTTR has a steel containment vessel inside its reactor building and the reactor building serves as a confinement which is called "service area". The service area is maintained at slightly negative pressure to the environment by ventilation systems in both normal operation and accident condition. The off-site radiation dose in such an accident as pipe rupture in the PCS is remarkably reduced by the containment vessel together with the confinement.

Furthermore, in the accident of primary pipe rupture, no effective countermeasure to limit the amount of air ingress to the reactor core is possible except for the containment vessel. The amount of oxidation of graphite in the reactor core is limited to very low level in the HTTR with a containment vessel.

4.Safety Evaluation

The safety evaluation of the HTTR has been made taking account of inherent safety as an HTGR and specific design features of the HTTR. The category of the events to be evaluated is based on "Examination Guide for Safety Evaluation of Light Water Nuclear Power Reactor Facilities". That is, in confirming the adequacy of basic design policy of reactor facilities, it is necessary to evaluate abnormal conditions, namely "Anticipated Operational Occurrence", and then to evaluate the events beyond the scope of "Anticipated Operational Occurrence", that is "Accidents". On the other hand, it is necessary to evaluate "Major Accidents" and "Hypothetical Accidents" to judge the appropriateness of reactor siting condition based on the "Guideline for Reactor Site Evaluation".

As for anticipated operational occurrences, were evaluated those events which include conditions beyond normal reactor operation resulting from a single failure or malfunction, or a single operator error anticipated to occur during the life time of the reactor facility such as control rod withdrawal, decrease of coolant flow, loss of electric power and so on. As for accidents, those events which are beyond anticipated operational occurrences and, though the frequency of those occurrence is smaller, should be postulated from the viewpoint of the possibility of release of radioactivity from the reactor facility were evaluated. Table 3 summarizes the selected events which are included in these two categories. The analytical result of each of these events was evaluated in accordance with the acceptance criteria which were established for these two categories for the HTTR, respectively. The adequacy of the design of the HTTR was confirmed through these safety evaluation.

The analytical result of the depressurization accident caused by rupture of the double concentric pipes in the PCS is shown in Fig.12 as an example of accident analyses. Maximum fuel temperature once decreases due to reactor scram, but, it increases again because the heat removal by the VCS is not enough to continue to decrease the fuel temperature. The peak temperature, however, is lower than the initial. On the other hand, RPV temperature and pressure in the containment rise higher than the initial values, respectively, but, still lower than the limits.

According to the Japanese guideline for site evaluation, it is required to assume the occurrence of a serious accident which is unlikely to occur from a technical standpoint in order to evaluate potential risk, that is, hypothetical accident.

Although the complete fuel failure can not be postulated by the mechanistic evaluation of a serious accident in the HTTR, 100% of FP release from the core is assumed in the evaluation of radiation exposure to the public in the hypothetical accident according to the guideline. Even in such a severe assumption, the effective dose equivalent is 6.3mSv, which is lower than the limit of 0.25Sv by two orders.

5.Afterword

The HTTR is a high temperature gas cooled test reactor which has various aims and operational modes, while it can provide at the same time very high temperature coolant up to 950°C at the outlet of the RPV for the first time in the world. The JAERI has carefully thought out the safety design principles for it as written in this paper reflecting the results of R&D works in HTGR technology and experiences of LWR designing and operation.

The JAERI submitted the safety analysis report of the HTTR to the Science and Technology Agency (STA) for safety review by the Government on February 10, 1989. The safety review by the STA was completed on December, 1989 and Nuclear Safety Commission is currently reviewing the draft safety evaluation report transmitted from the STA. The installation permit will be issued in summer 1990 and it will take about five years for the construction of the HTTR plant as shown in Fig.13. The first reactor criticality will be attained in 1995.

Table 1 Specification of the HTTR

Thermal power	30MW
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 cooling loop	1
Heat removal	IHX and PWC (parallel loaded)
Primary coolant pressure	4MPa
Containment type	Steel containment
Plant lifetime	20 years

Table 2 Major nuclear and thermal-hydraulic specification

Nuclear	
Excess reactivity	15%Δk
Uranium enrichment	3 - 10wt%
Average	6wt%
Power peaking factor	
Radial	1.1
Axial	1.7
Fuel burn-up (average)	22GWd/t
Reactivity coefficients	
Fuel temperature coefficient	$-(1.5 \text{ to } 4.6)\times10^{-5}\Delta k/k/{}^\circ C$
Moderator temperature coefficient	$(-17.1 \text{ to } 0.99)\times10^{-5}\Delta k/k/{}^\circ C$
Power coefficient	$-(2.4 \text{ to } 4.0)\times10^{-5}\Delta k/k/MW$
Prompt neutron lifetime	0.67 - 0.70ms
Effective delayed neutron fraction	0.0047 - 0.0065
Themal-hydraulic	
Total coolant flow rate	10.2kg/s (950°C operation)
Core pressure drop	6 - 9kPa
Effective core coolant flow rate	88%
Maximum fuel temperature	1492°C

Table 3(1) Selected events as anticipated operational occurrences

(1)Abnormal change of reactivity or power distribution in the core
- Abnormal control rod withdrawal from subcritical condition
- Abnormal control rod withdrawal at power operation

(2)Abnormal change of heat generation or removal in the core
- Decrease in primary coolant flow rate
- Increase in primary coolant flow rate
- Decrease in heat removal in the secondary cooling system
- Increase in heat removal in the secondary cooling system
- Loss of off-site electric power

(3)Transient during the irradiation tests
- Reactivity addition by the move of irradiation specimen
- Insulation degradation of irradiation capsule

(4)Transients during safety demonstration test
- Decrease in primary coolant flow rate during the gas circulator trip test

Table 3(2) Selected events as accidents

(1)Increase in reactivity in the core
- Rupture of stand-pipe

(2)Decrease in coolabilty in the core
- Channel blockage in the fuel block
- Inner pipe failure of primary double concentric pipe
- Inner pipe failure of secondary double concentric pipe
- Rupture of secondary double pipe
- Rupture of pressurized water cooling system pipe

(3)Depressurization accidents
- Rupture of primary double concentric pipe

(4)Water ingress accidents
- Rupture of heat transfer tube in pressurized water cooler

(5)Accidents in primary coolant purification system
- Failure of primary coolant purification system

(6)Accidents in the treatment system of the radioactive waste
- Failure of the treatment system of the radioactive gas waste

(7)Accidents during the irradiation tests
- Failure of the sweep gas pipe in the irradiation test facilities

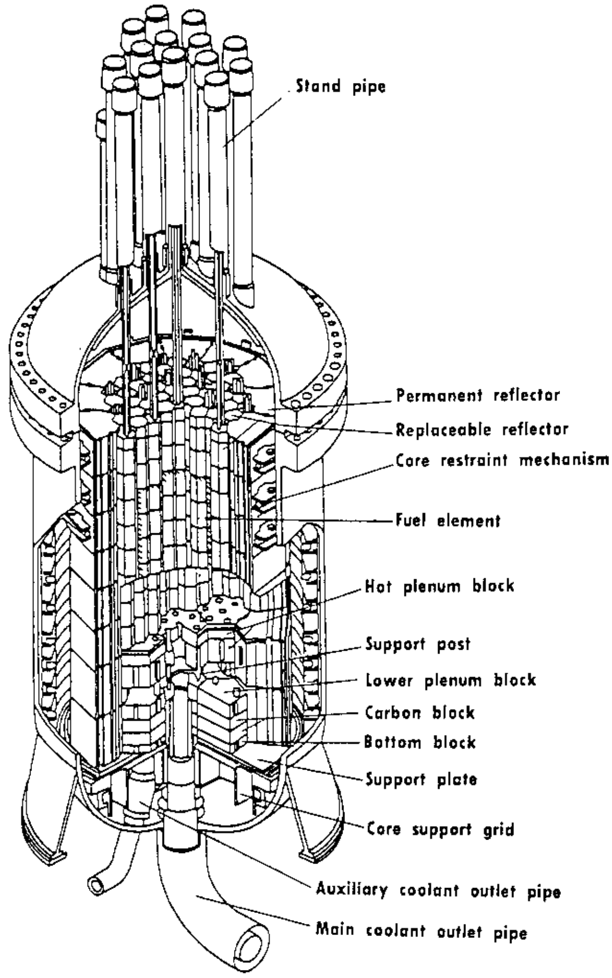

Fig.1 Bird's eye view of reactor vessel and core

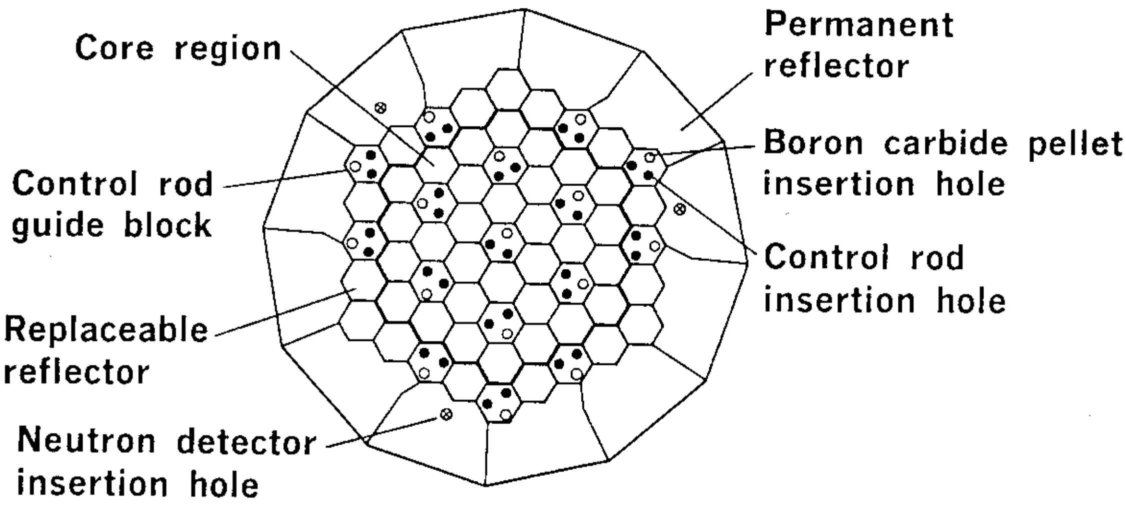

Fig.2 Cross section of HTTR core

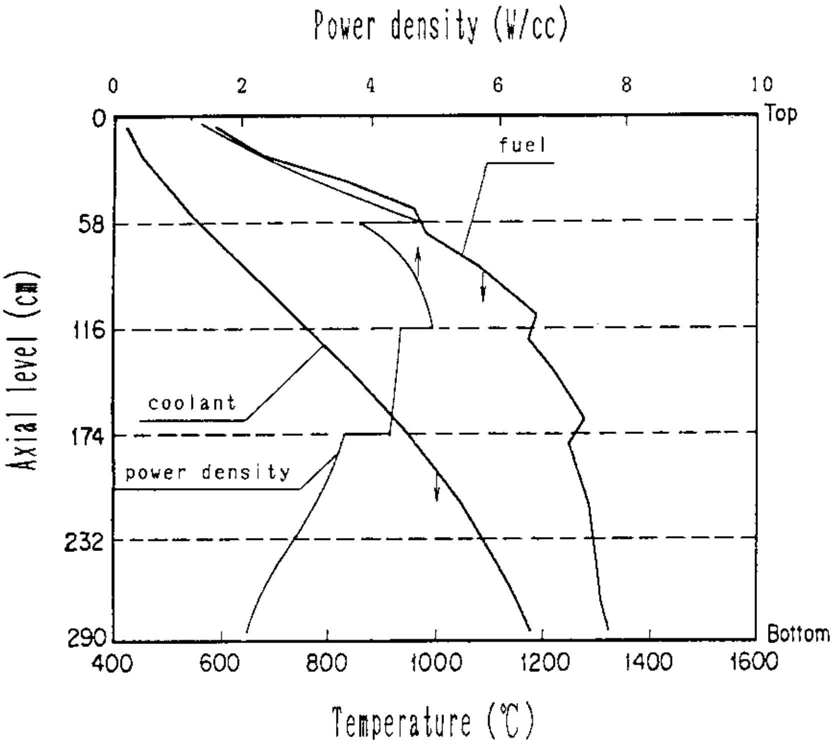

Fig.3 Axial power and temperature distributions for 950°C operation

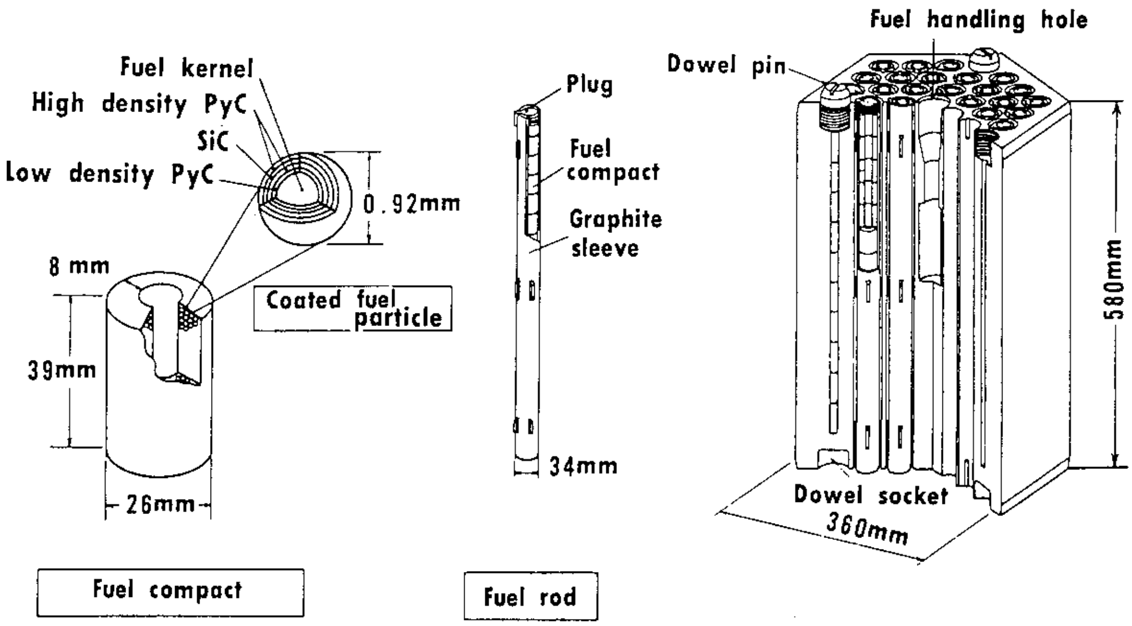

Fig.4 Block type fuel of the HTTR

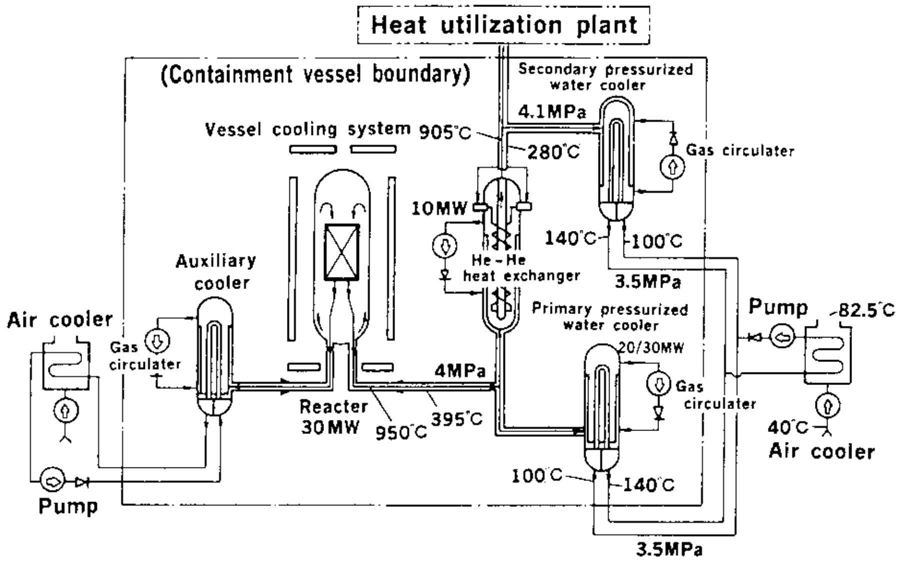

Fig.5 Flow diagram of cooling system

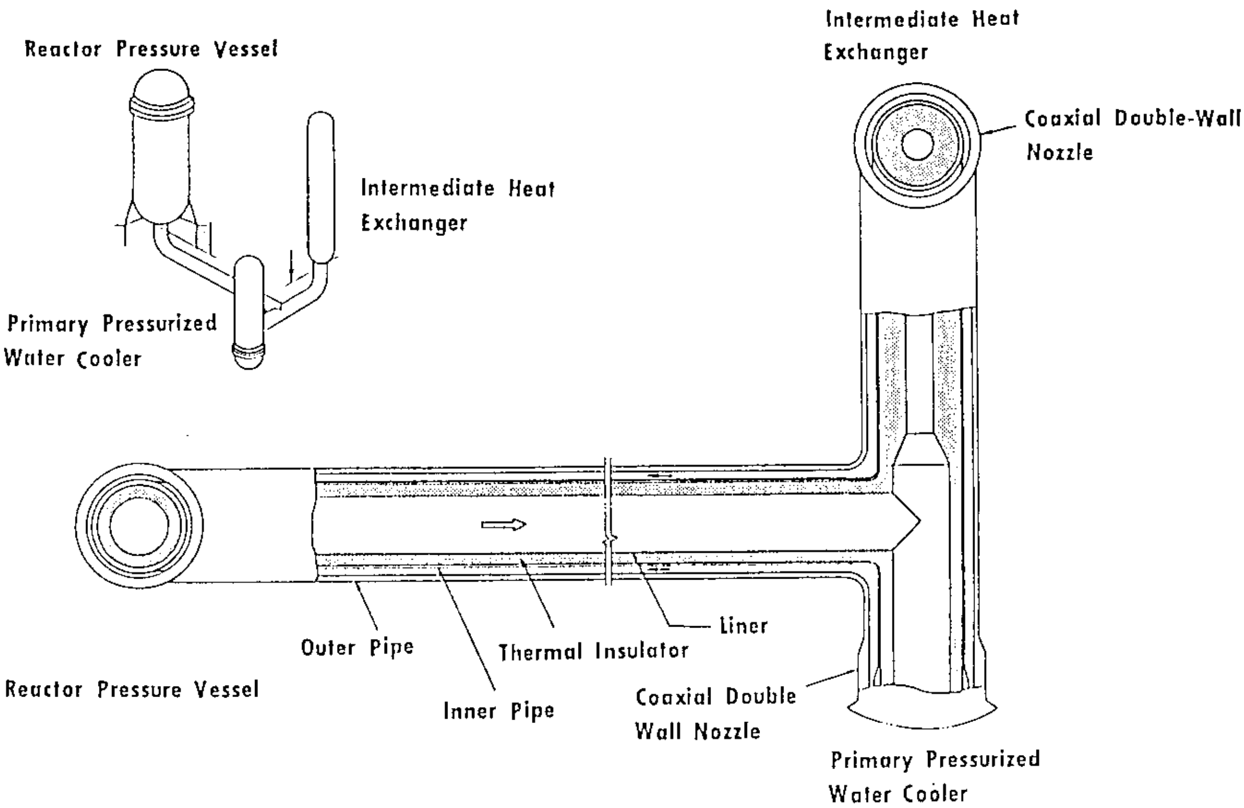

Fig.6 Co-axial double piping

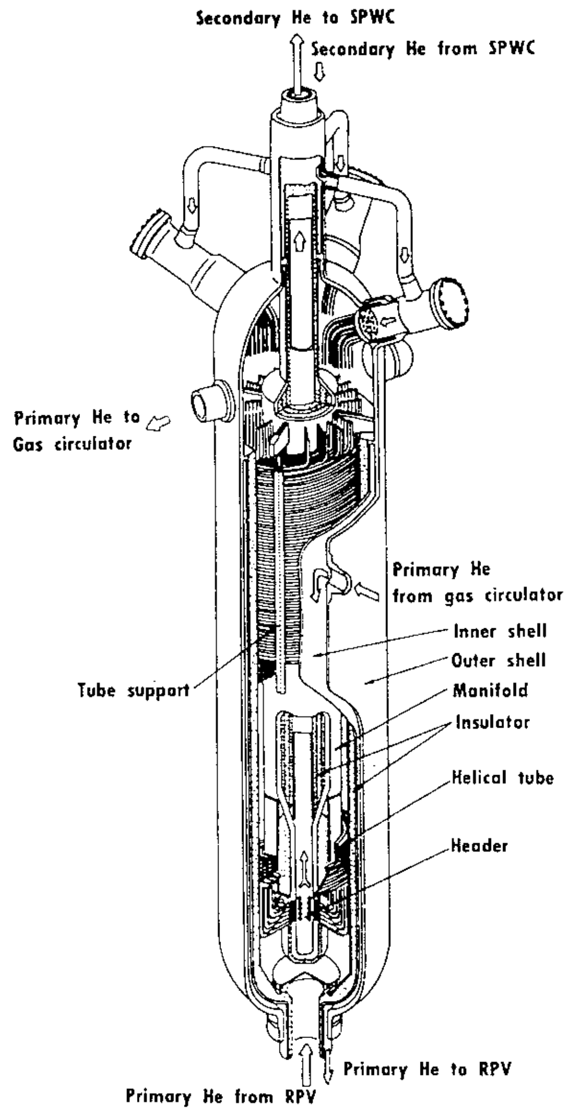

Fig.7 Intermediate heat exchanger (10MW)

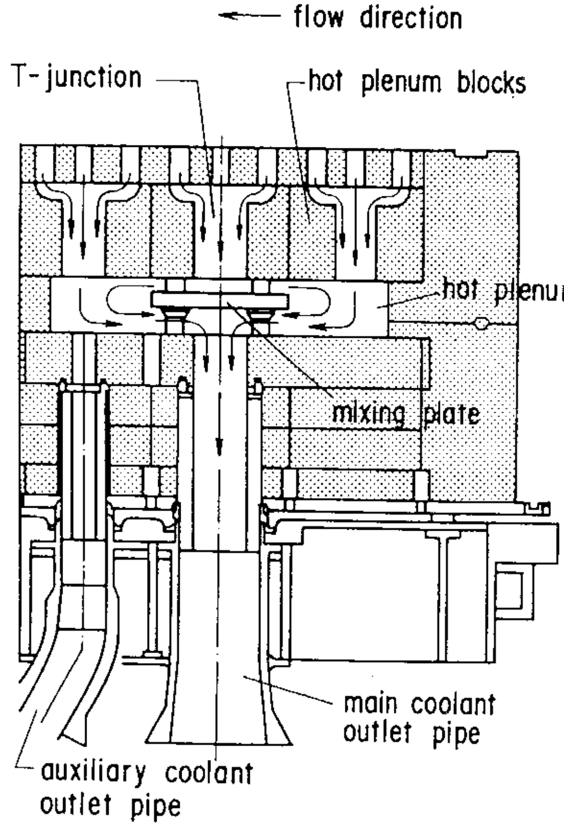

Fig.8 Details of HTTR core bottom structure

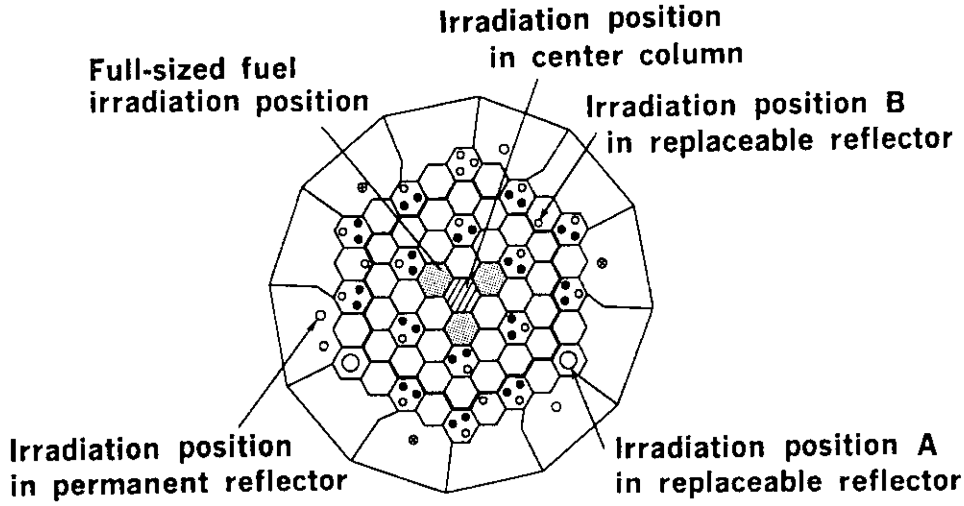

Fig.9 Cross sectional view of irradiation region

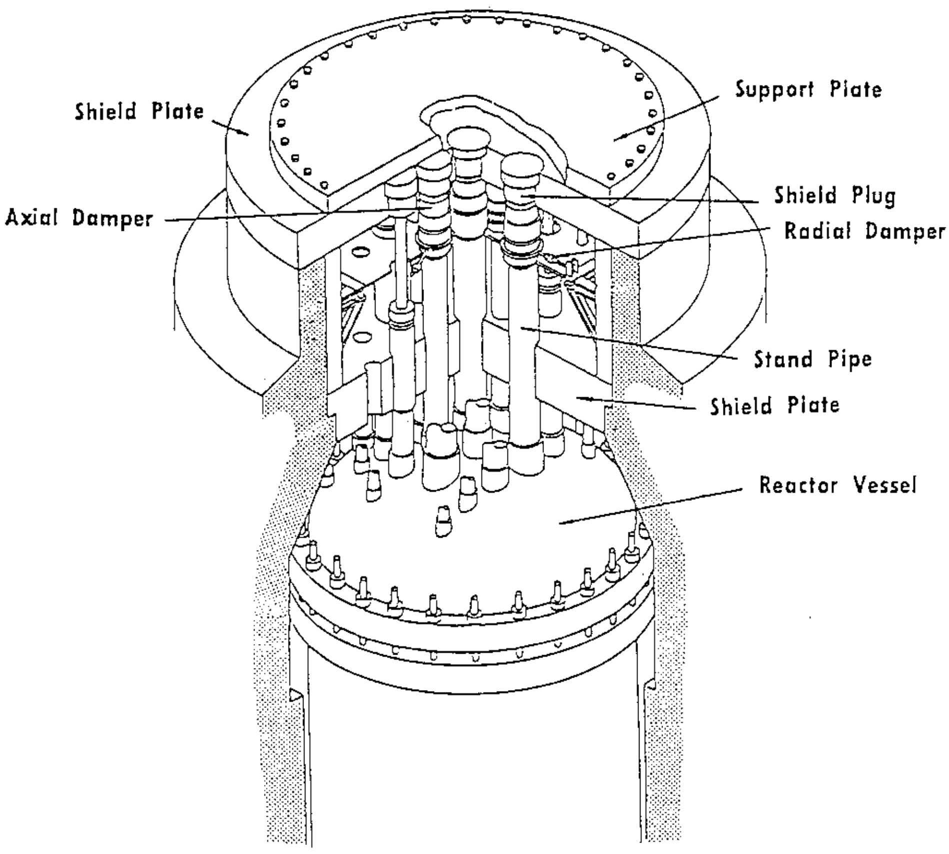

Fig.10 Stand-pipe fixing devices

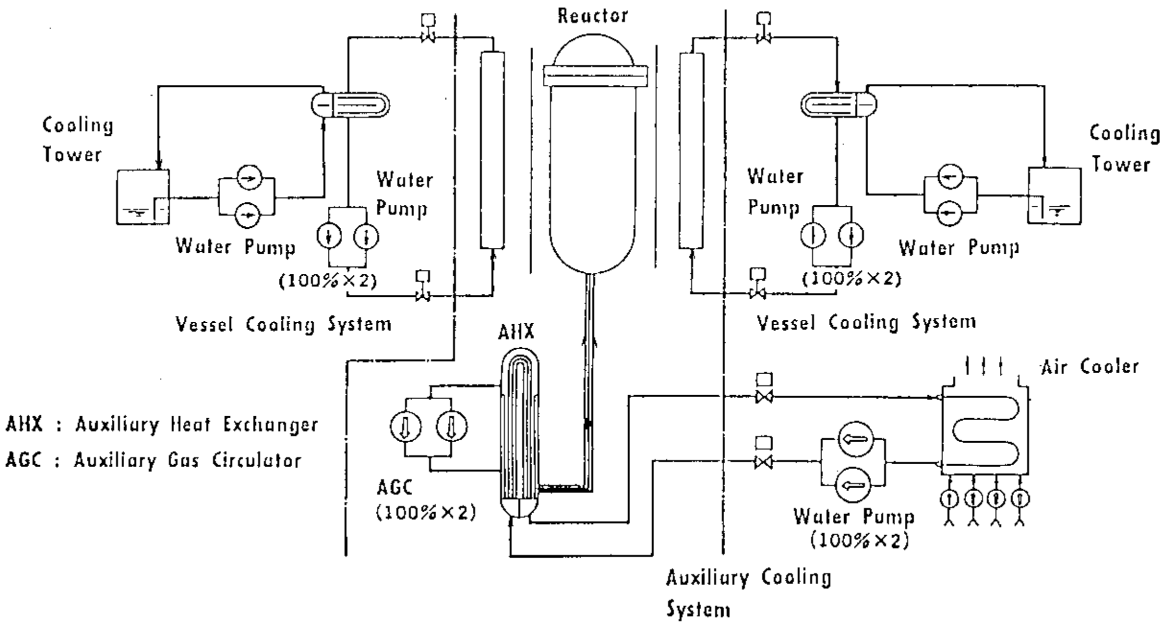

Fig.11 Residual heat removal systems

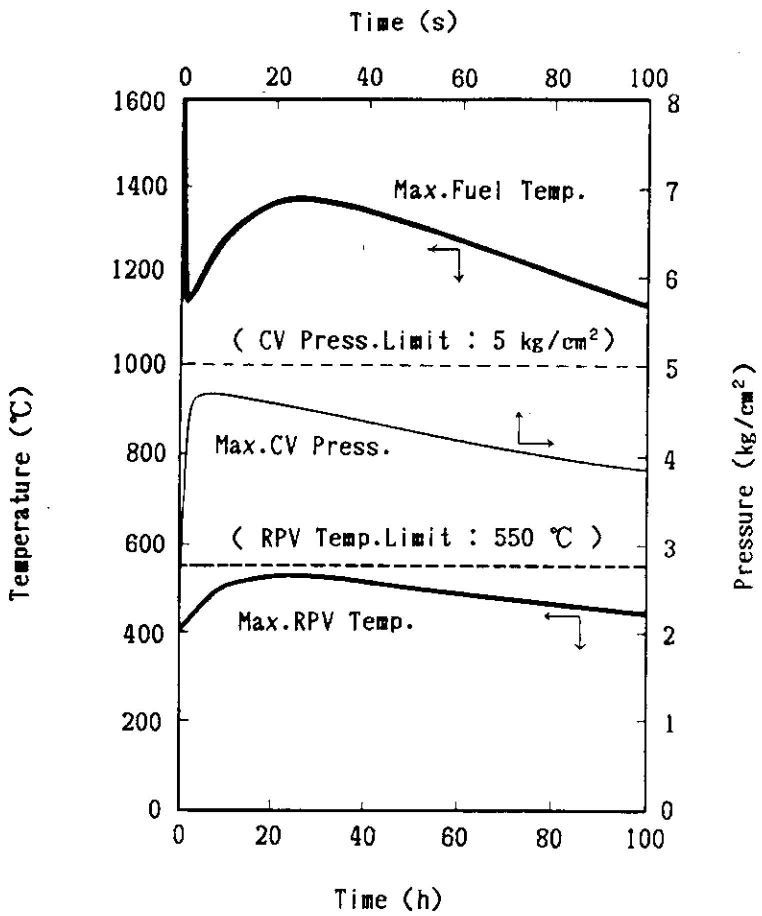

Fig.12 Temperature and pressure transient in depressurization accident

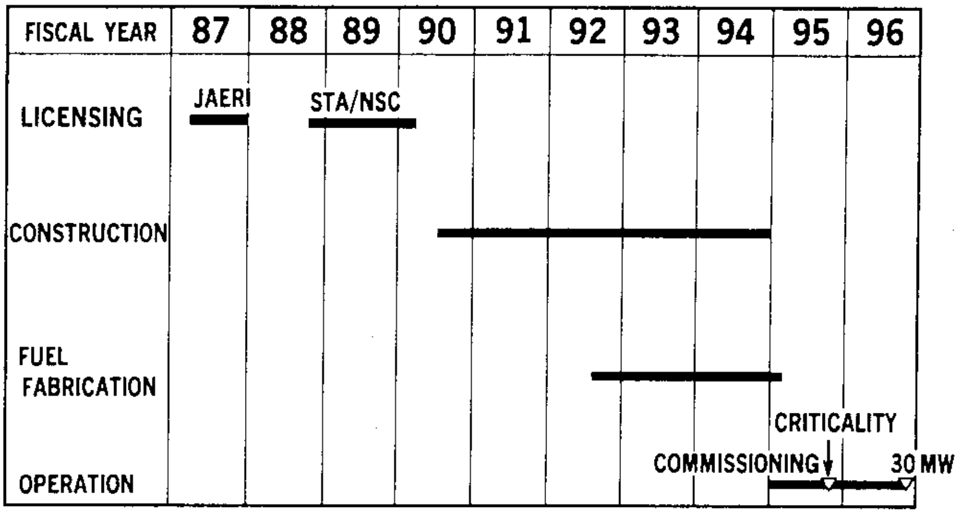

Fig.13 HTTR licensing, construction and operation schedule

2.11 MODULAR HIGH TEMPERATURE GAS-COOLED REACTOR DESIGN

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USA

Abstract

The Modular High-Temperature Gas-Cooled Reactor (MHTGR) is an advanced power plant concept which has been under development since 1984. The design utilizes basic high temperature gas-cooled reactor features of refractory coated fuel, helium coolant and a graphite moderator which are supported by design studies and experience of over 30 years. The design is being developed based on selections made in response to well defined requirements for safe, reliable and economic power. The geometric arrangement of the reactor and steam generator vessels, the core and the heat removal components has been selected to exploit the inherent characteristics associated with high-temperature materials. The design utilizes passively safe features which provide a higher margin of safety and investment protection than current generation reactors.

Design features include an arrangement of four identical 350 MW (t) modular reactor units located in underground silos covered by a single reactor building. This Nuclear Island (NI) with the reactors and associated equipment is coupled to an Energy Conversion Area (ECA) with two turbine generators producing a combined total of approximately 540 MW(e). Nuclear heat is normally removed by a steam generation system featuring a once-through design. A shutdown cooling system utilizing water and a reactor cavity cooling system utilizing air driven by natural convection provide alternate heat removal paths. The design and development program, now in the preliminary design stage, is a cooperative effort by the U.S. government, the utilities and the nuclear industry.

1. Introduction

The development of earlier HTGR plants proceeded on a trend toward very large monolithic design during the 1970s and early 1980s. In about 1984 there was a recognition by the U.S. participants within industry, the Department of Energy and the Congress that the changes in the environment for nuclear power, including the financial, electrical demand pattern and public interests, pointed toward a reevaluation of the goals for development of improved reactor designs. An evaluation by the joint industry/government participants led to a focusing of the development of the gas-cooled reactor toward a smaller modular HTGR power plant with emphasis on passive safety, reliability and competitive economics.

A design team of General Atomics, Bechtel National Inc., Combustion Engineering and Stone & Webster Engineering Co. is now focused on the development of the preliminary design that will meet these challenging demands. Base technology support is being provided by the Oak Ridge National laboratory. The program is under the sponsorship of the U.S. Department of Energy (DOE) and in cooperation with potential utility users represented by Gas-Cooled Reactor Associates (GCRA).

2. Design Approach

The plant has been designed in response to top level requirements defined by the utility/user, through GCRA, and by top level Nuclear Regulatory Commission (NRC) regulatory criteria applicable to all reactor types. The user and regulatory requirements as applied to the MHTGR have been specified as direct, quantifiable statements defining acceptable consequences or risks to the public for normal operation, transients, design basis events, and other very low probability events. The requirements for safety and investment risk have had a strong influence on the plant arrangement and the selection of components.

The resulting design, intended for deployment by the end of the century, is based on existing technology, successfully demonstrated components and reactor experience. The safety design philosophy selected for the MHTGR has been to control radionuclide releases primarily through retention at the source, within the refractory coated fuel particles, even under accident conditions. Selections including a small unit rating per module, an annular core, a large negative temperature coefficient and below grade (silo) installation with passive decay heat removal support this safety philosophy. This concept places minimal reliance upon active design features or operator action.

The safety philosophy was made possible principally by improvements in coated particle fuel technology. The retention of radionuclides within the coated fuel particles replaces reliance upon such secondary barriers as the primary coolant boundary or a containment structure.

3. Design Description

The basic modular element containing the reactor components is a steel vessel system comprised of a reactor vessel, a steam generator vessel and a connecting cross duct vessel (Figure 1). The reactor vessel is approximately the same size as that of a large boiling water reactor and contains the core, reflector, and associated supports. A shutdown cooling heat exchanger and a shutdown cooling circulator are mounted at the bottom of the reactor vessel. Top mounted penetrations house the control rod drive mechanisms and the hoppers containing boron carbide pellets for reserve shutdown. The penetrations are also used as access for refueling and inspection.

The heat transfer during power operation or normal core decay heat removal is accomplished by helium which is heated as it flows down through the core. It is collected in a plenum below the core and flows through a coaxial hot duct inside the cross duct vessel to a once-through, helical bundle steam generator.

After flowing downward over the steam generator tubes, the cool helium flows upward in an annulus between the steam generator vessel and a shroud leading to the main circulator inlet.

The main circulator is a submerged electric motor driven single stage axial compressor with active magnetic bearings. The helium is discharged from the circulator and flows through the annulus of the cross duct vessel and hot duct and then upward to the top plenum over the core (Figure 2).

In order to meet availability and maintenance requirements, a separate shutdown cooling system, including a water cooled heat exchanger and submerged electric motor driven single stage centrifugal compressor with active magnetic bearings, is provided as a backup to the primary heat transport system. The heat removal systems allow hands-on plant maintenance to begin within 24 hours after plant shutdown.

The low enriched fissile uranium (19.9%) as UCO and fertile thorium as ThO_2 provide the fission energy for the reactor. Multiple particle coatings provide the barriers for radionuclide retention. This refractory coated particle fuel is designed to maintain integrity when exposed to temperatures above those expected during licensing basis events and beyond.
The fuel particles are bonded together in fuel rods which are contained in sealed vertical holes in hexagonal cross-section graphite fuel blocks (Figure 3). These fuel blocks, provided with coolant flow channels for normal heat removal, are stacked in columns to make up an annular shaped core. Unfueled graphite blocks form the center of annulus and surround the active core to form the reflector. Key reactor core design parameters are shown in Table 1. The annular shape of the core has been selected to enhance the heat removal capabilities in the event of a loss of all forced cooling. The reactor core and the surrounding graphite neutron reflectors are supported on a steel core support structure at the lower end of the reactor vessel. A horizontal cross-section of the reactor core and vessel internals is shown in Figure 4.

TABLE 1
Reactor System Design Parameters

Core Thermal Power	350 MW
Core Power Density	5.9 w/cm^3
Annular Core Diameters:	
Outer	3.5 m
Inner	1.6 m
Core Height	7.9 m
Number of Columns in Active Core	66
Number of Fuel Elements Per Column	10
Number of Control Rods	30
Number of Reserve Shutdown Columns	12

The MHTGR utilizes a once-through fuel cycle; that is, it does not rely on recycling of spent fuel. Each module is refueled once every 20 months. The refueling is accomplished with the reactor shutdown and depressurized, utilizing a refueling machine accessing the fuel elements through the appropriate control rod penetrations in the top of the reactor vessel. The spent fuel is transported to the spent fuel storage pool for temporary storage before shipping to final storage offsite.

Each reactor module is housed in adjacent, but separate, reinforced concrete structures located below grade and under a common roof structure. The below-grade location provides significant design benefits by reducing the seismic amplifications typical of above-grade structures and by providing confinement (Figure 5).

A reactor cavity cooling system (RCCS) (Figure 6) is located in the below grade concrete structure external to the reactor vessel to remove plant residual heat. This system is totally passive and provides the alternative safety related heat sink if the forced cooling systems are inoperative. The heat is transferred by means of conduction, convection and radiation from the core to the RCCS. This system has no controls, valves, circulating fans, or other active components. The RCCS is the only safety related heat removal system utilized by the MHTGR.

Almost all components and systems of each module, which are required to meet regulatory requirements, are independent of other modules and are localized within the individual concrete structures. These include plant protection and decay heat removal systems.
Each of the four modules produces a thermal output of 350 MW(t). The plant is made up of two sets of two modules headered to feed one of two turbine generators. Thus, the reference configuration consists of a 2 x (2x1) configuration of reactors and turbines with common support systems. Thermal energy from the four reactor modules is delivered to two steam turbine generators to produce 538 MW(e) net of electric power. The turbine plant is similar to a modern fossil-fired plant except that the MHTGR plant utilizes a nonreheat steam cycle. A mechanical draft cooling tower rejects the condenser heat load to the atmosphere. A plot plan for the plant showing the nuclear island and energy conversion area is shown in Figure 7. Key plant performance parameters are summarized in Table 2. Operation is coordinated utilizing an overall plant control, data, and instrumentation system and all four reactors are operated from a single control room.

Table 2

Plant Performance Parameters

Thermal Power	1400 MW(t)
Electrical Output	588 MW(e) Gross; 538 MW(e) Net
Net Efficiency	38.4%
Steam Conditions	540°C (1000°F)/16.6 MPa (2400 psig)
Core Exit Helium Temperature	687°C (1268°F)
Cold Helium Temperature	258°C (497°F)

4. Conclusion

The MHTGR design is based on over 30 years of reactor experience with the carbon dioxide-cooled Magnox and Advanced Gas-Cooled Reactor (AGR) developed in the United Kingdom; the 15MW(e) Arbeitsgemeinshaft Versuch Reaktor (AVR) development plant and the 300MW(e) Thorium Hochtemperatur Reaktor (THTR) Demonstration Plant developed in Germany; the 40 MW(e) Peach Bottom I developed in the United States by General Atomics; and the 330 MW(e) Fort St. Vrain (FSV) demonstration plant, also a General Atomics project. The FSV, AVR, and THTR facilities have provided invaluable confirmation and demonstration of specific and generic HTGR design and operating characteristics.

As a second generation nuclear power system, the MHTGR has been designed to meet utility and regulatory requirements. The MHTGR responds to concerns of the public, the government, the utilities, and industry about nuclear safety, economic risk, and investment protection.

Based on technology developed and demonstrated in the United States, the United Kingdom and West Germany, this system makes use of the refractory-coated nuclear fuel, helium gas as an inert coolant, and graphite as a stable core structural material.

Public safety and protection of the plant investment is provided by inherent and passive features. The high-performance MHTGR provides flexibility in power output and siting, competitive energy costs, and can serve diverse energy needs both domestically and internationally.

5. Acknowledgment

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US-DOE MHTGR PROGRAM

350 MW(t)
MODULAR HTGR
PLANT

GENERAL ATOMICS

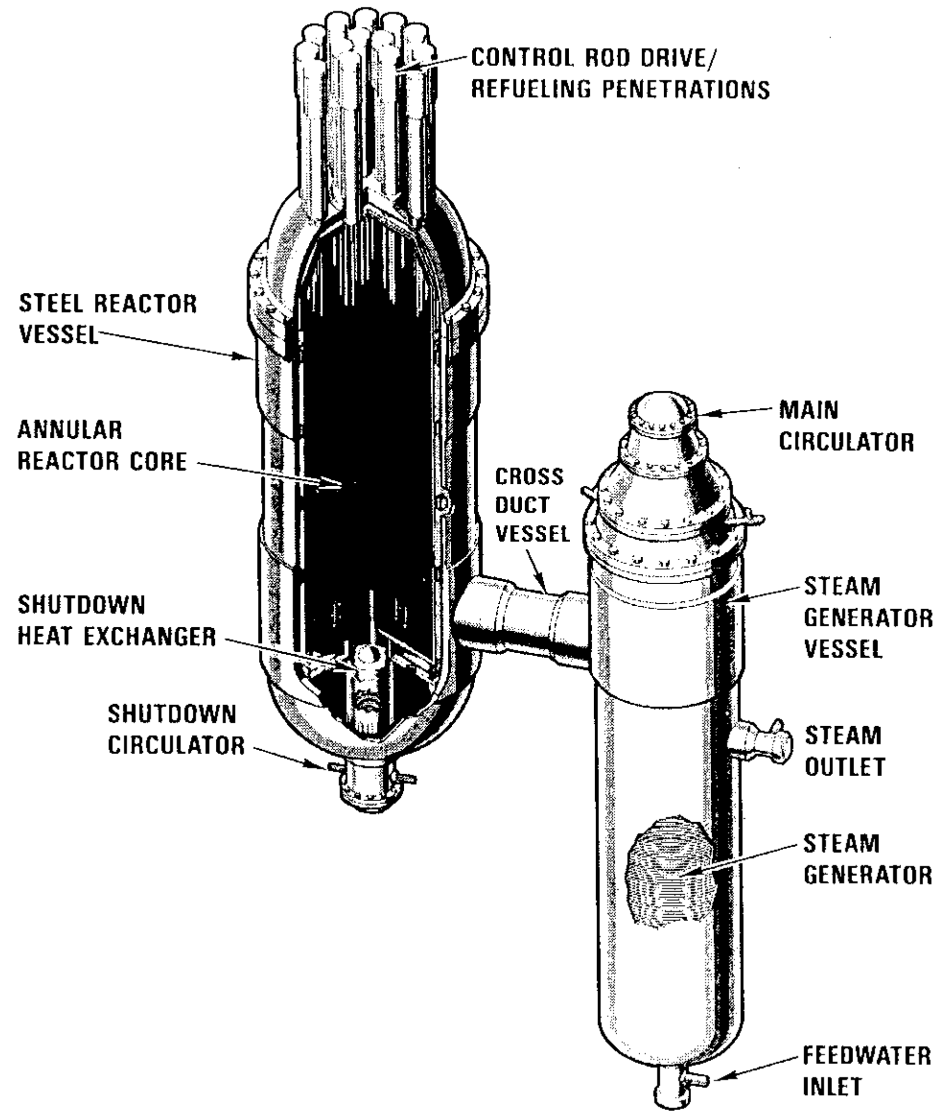

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FIGURE 1

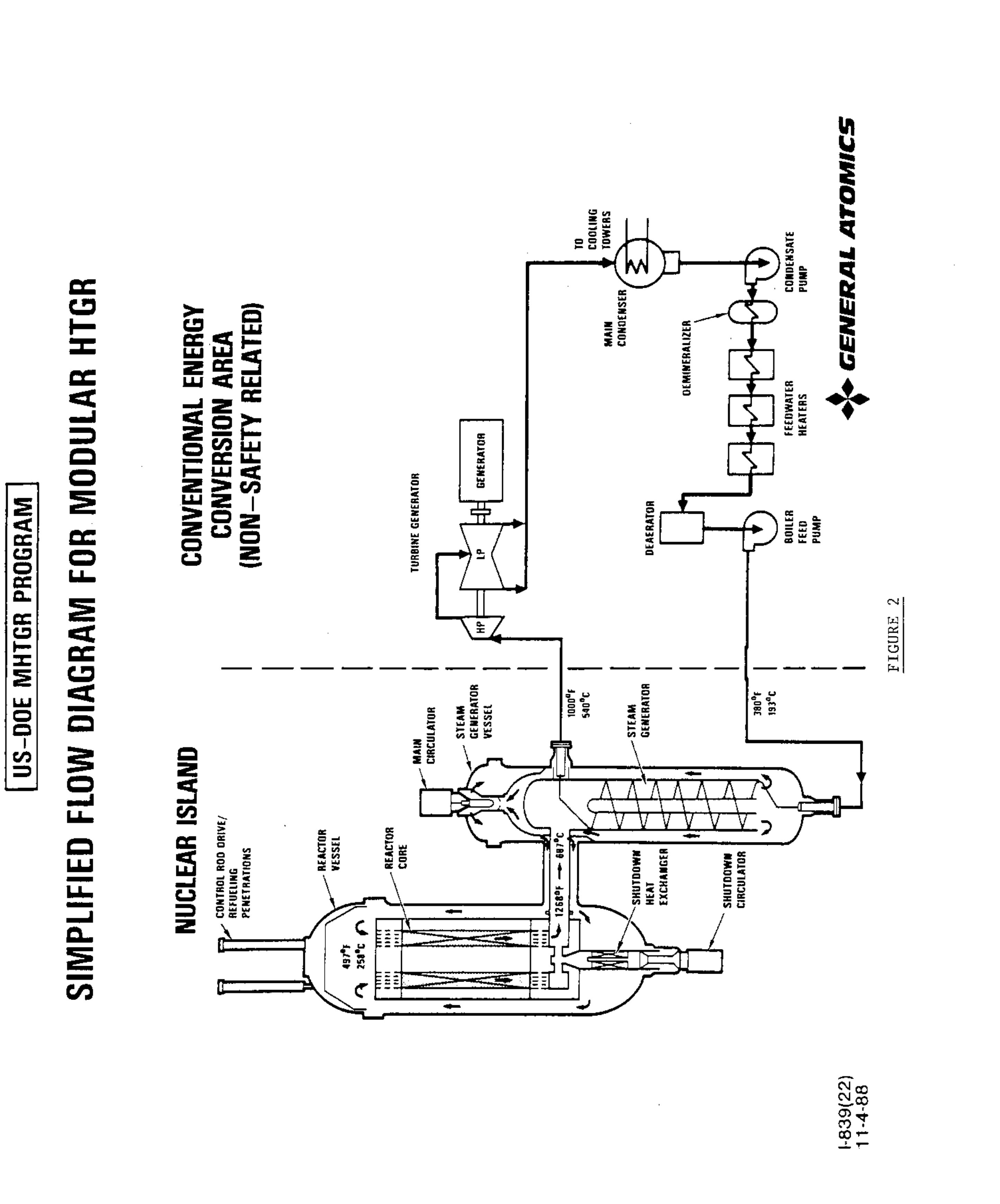
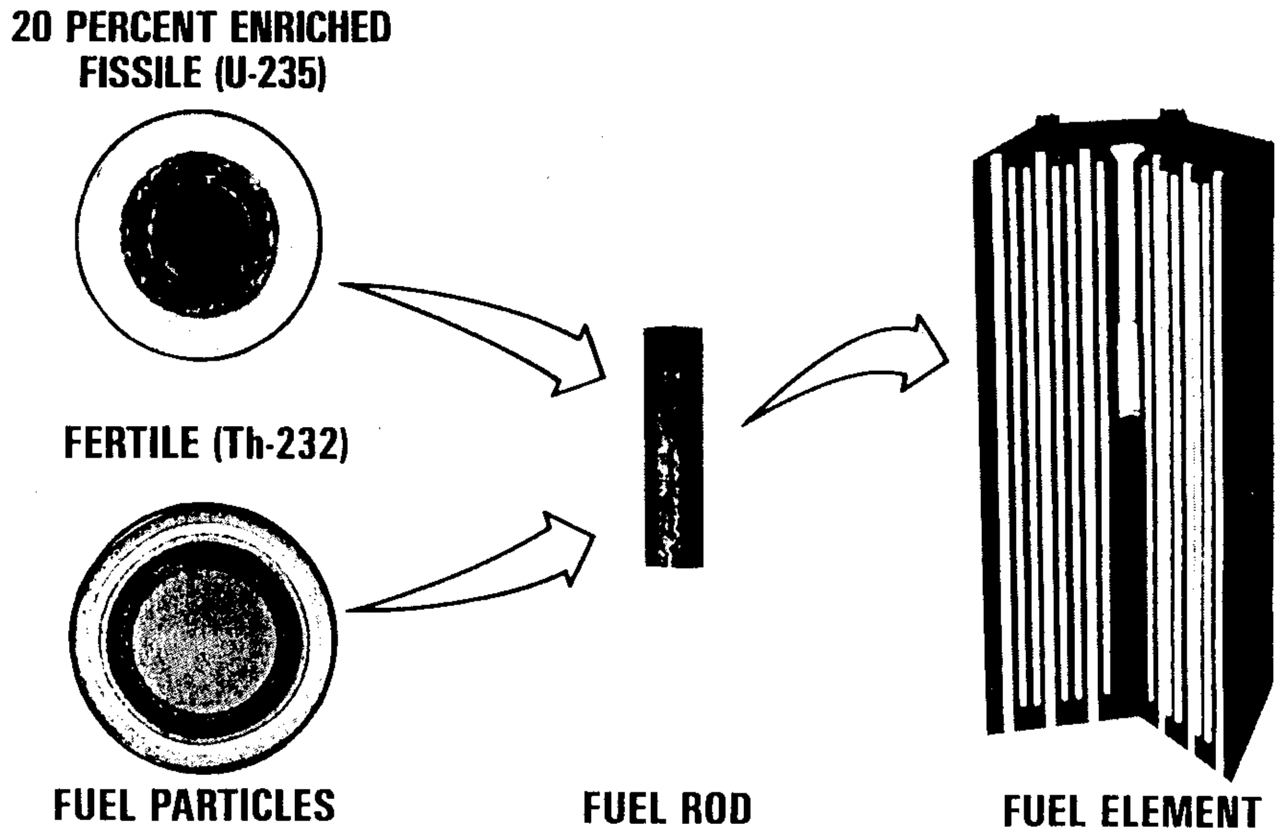

US-DOE MHTGR PROGRAM
SIMPLIFIED FLOW DIAGRAM FOR MODULAR HTGR
NUCLEAR ISLAND
CONVENTIONAL ENERGY
CONVERSION AREA
(NON-SAFETY RELATED)
CONTROL ROD DRIVE/
REFUELING
PENETRATIONS
REACTOR
VESSEL
REACTOR
CORE
497°F
258°C
1268°F
687°C
MAIN
CIRCULATOR
STEAM
GENERATOR
VESSEL
STEAM
GENERATOR
SHUTDOWN
HEAT
EXCHANGER
SHUTDOWN
CIRCULATOR
1000°F
540°C
380°F
193°C
TURBINE GENERATOR
HP
LP
GENERATOR
TO
COOLING
TOWERS
MAIN
CONDENSER
CONDENSATE
PUMP
DEMINERALIZER
FEEDWATER
HEATERS
DEAERATOR
BOILER
FEED
PUMP
GENERAL ATOMICS
I-839(22)
11-4-88

FIGURE 2

US-DOE MHTGR PROGRAM

HTGR FUEL COMPONENTS



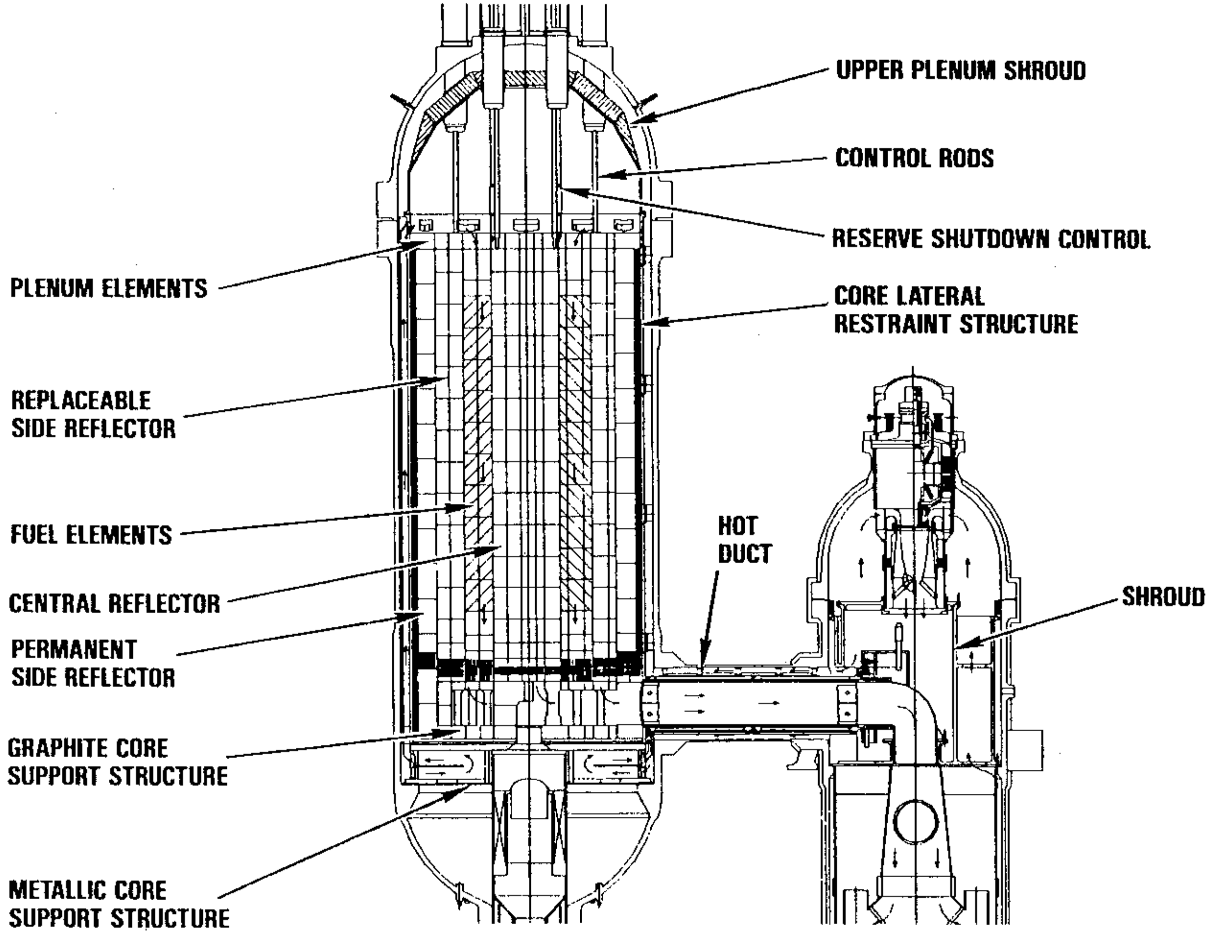
A-795(1)(a)
3-2-90

GENERAL ATOMICS

FIGURE 3

US-DOE MHTGR PROGRAM

CORE AND REACTOR INTERNAL ARRANGEMENT



J-147(17)
8-9-89

GENERAL ATOMICS

FIGURE 4

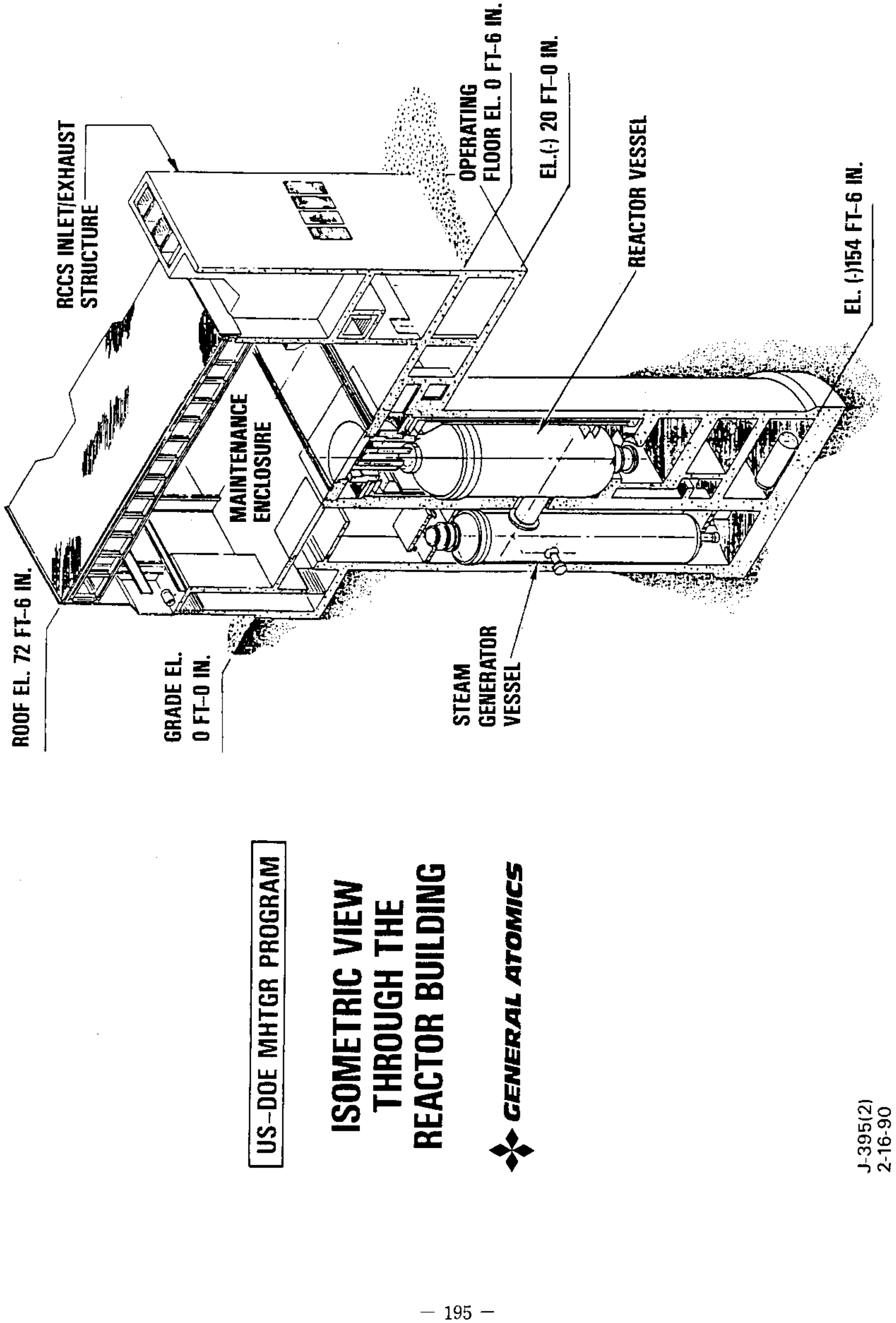

US-DOE MHTGR PROGRAM
ISOMETRIC VIEW THROUGH THE REACTOR BUILDING
GENERAL ATOMICS
J-395(2)
2-16-90
ROOF EL. 72 FT-6 IN.
GRADE EL. 0 FT-0 IN.
MAINTENANCE ENCLOSURE
RCCS INLET/EXHAUST STRUCTURE
OPERATING FLOOR EL. 0 FT-6 IN.
EL.(-) 20 FT-0 IN.
REACTOR VESSEL
STEAM GENERATOR VESSEL
EL. (-)154 FT-6 IN.

FIGURE 5

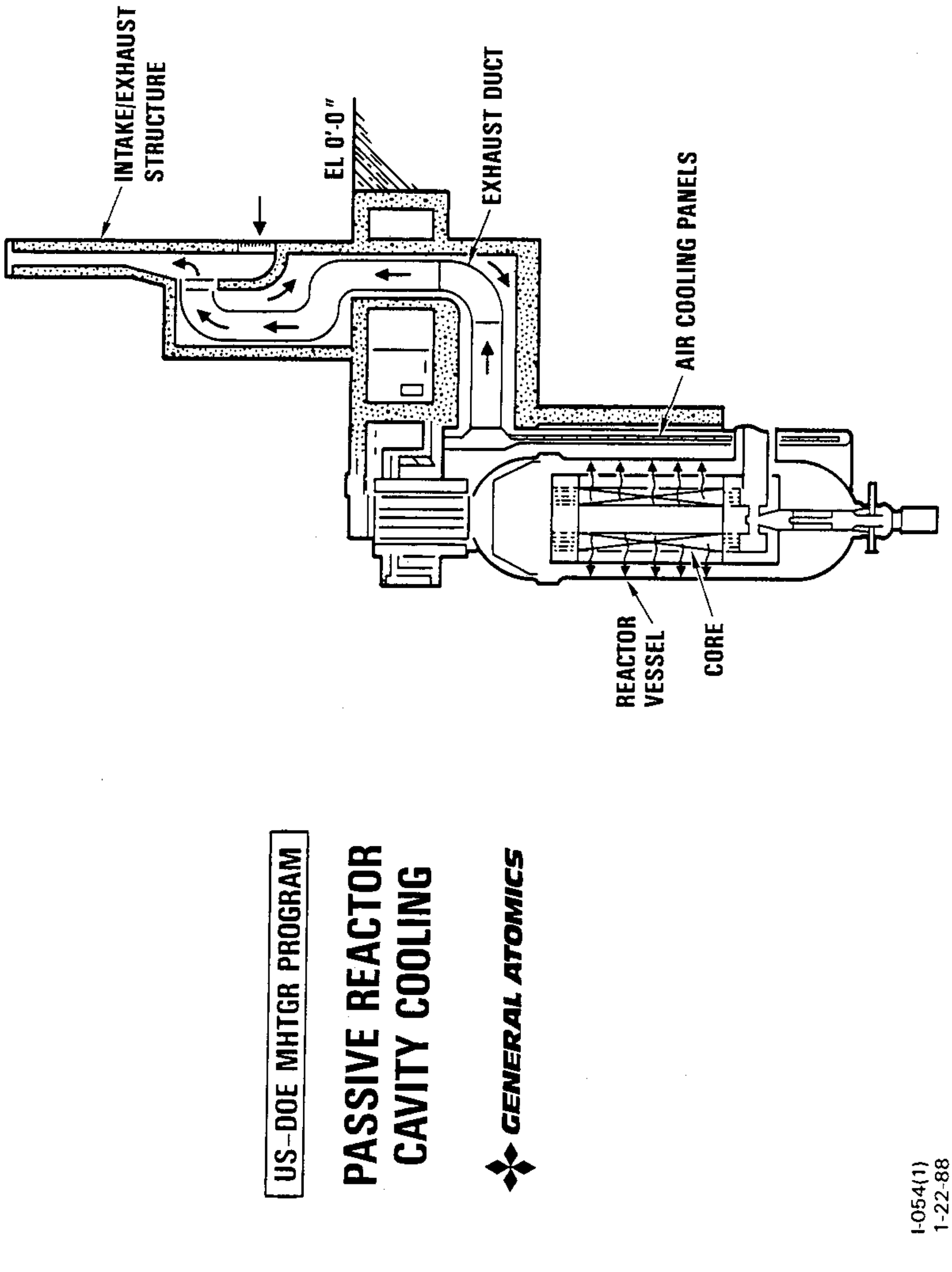
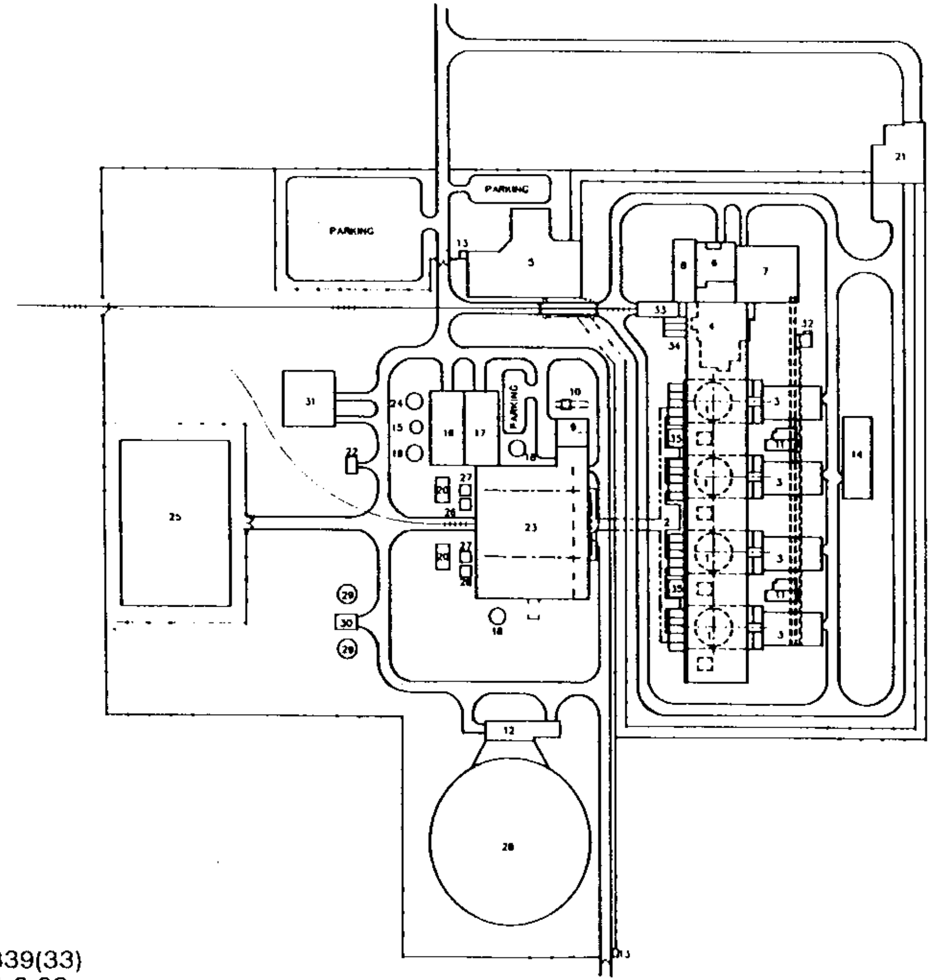

INTAKE/EXHAUST STRUCTURE
EL 0'-0"
EXHAUST DUCT
AIR COOLING PANELS
REACTOR VESSEL
CORE
US-DOE MHTGR PROGRAM
PASSIVE REACTOR CAVITY COOLING
GENERAL ATOMICS
I-054(1)
1-22-88

FIGURE 6

US-DOE MHTGR PROGRAM

NUCLEAR ISLAND SEPARATED FROM ENERGY CONVERSION AREA



LEGEND

1. REACTOR BUILDING
2. MAIN STEAM & FEEDWATER PIPING
3. REACTOR AUXILIARY BUILDING
4. REACTOR SERVICE BUILDING
5. OPERATIONS CENTER
6. PERSONNEL SERVICES BUILDING
7. RADIOACTIVE WASTE MANAGEMENT BUILDING
8. NUCLEAR ISLAND COOLING WATER BUILDING
9. STANDBY POWER BUILDING
10. FUEL OIL STORAGE TANK & PUMP HOUSE
11. HELIUM SERVICES BUILDING
12. CIRCULATING WATER PUMP HOUSE
13. GUARD HOUSE
14. HELIUM STORAGE STRUCTURE
15. CLARIFIER
16. MAKEUP WATER TREATMENT & AUXILIARY BOILER BUILDING
17. MAINTENANCE BUILDING
18. CONDENSATE SURGE TANK
19. DEMINERALIZED WATER STORAGE TANK
20. UNIT TRANSFORMER
21. NI WAREHOUSE
22. HYDROGEN STORAGE AREA
23. TURBINE BUILDING
24. FILTERED WATER STORAGE TANK
25. SWITCHYARD
26. STARTUP AUXILIARY TRANSFORMER
27. UNIT AUXILIARY TRANSFORMER
28. STATION COOLING TOWER
29. FIRE WATER STORAGE TANK
30. FIRE PUMP HOUSE
31. ECA WAREHOUSE
32. REMOTE SHUTDOWN BUILDING
33. WASHDOWN BAY
34. CHILLED WATER BUILDING
35. ELECTRICAL EQUIPMENT AREA

GENERAL ATOMICS

I-839(33)
11-8-88

FIGURE 7

2.12 DESIGN DETAILS OF HTR-MODULE AND HTR-500

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FRG

Abstract

Design Features of the HTR-Module

The modular HTR power plant is a universally applicable energy source for the co-generation of electricity, process steam or district heat.

The modular HTR concept is characterized by the fact that standardized reactor units with power ratings of 200 MJ/s (so-called modules) can be combined to form power plants with a higher power rating.

Consequently, the special safety features of small high-temperature reactors are also available at higher power ratings.

The safety features and the technical design are briefly described in the following, taking a power plant with two HTR-Modules as an example. Special attention will be given to the pressure vessel unit.

The principal safety feature of the HTR-Module is based on the fact that, even in the case of failure of all active cooling systems and complete loss of coolant, the fuel element temperatures remain within limits at which there is practically no release of radioactive fission products from the fuel elements. This guarantees that the modular HTR power plant does not present any hazard to the environment either under normal operation or in the event of accidents.

Due to its features, the modular HTR power plant is suitable for construction on any site, but particularly on sites near other industrial plants or in densely populated areas.

Design Details of the HTR-500 Power Plant

The HTR-500 has been developed as a nuclear power plant for medium-sized and large power grids, either as a base load plant or for load-following applications according to the requirements of the grid. Furthermore, the HTR-500 is well suited for the cogeneration or supply of process steam for industrial applications. The HTR-500 has a thermal power output of 1390 MJ/s, which corresponds to an electrical rating of 550 MW.

The concept selected for the HTR-500 system incorporates to a large extent the THTR-300 technology licensed and realized in accordance with the current state of science and technology.

The reactor plant concept is marked by the integrated arrangement of all primary-system components within a burst-proof pre-stressed concrete reactor vessel in single-cavity design. The integrated arrangement provides protection against impacts from internal incidents as well as radiation shielding. The pressure vessel and important systems are arranged within the confinement building which provides protection against impacts from external events.

1. Design Details of the HTR-Module

1.1 General Design Features and Philosophy

The most important design features are:

- The use of spherical fuel elements, which are capable of retaining all radiologically relevant fission products up to fuel element temperatures of approx. 1,600°C.

- The reactor core is designed and layed out in such a way that a maximum fuel element temperature of 1,600°C is not exceeded during any accident.

- Active core cooling is not necessary for decay heat removal during accidents. It is quite sufficient to discharge the decay heat by means of passive heat transportation mechanisms (such as heat conduction, radiation, natural convection) to a simple cavity cooler outside the reactor pressure vessel.

- Reactor shutdown is carried out solely by absorber elements, which on demand can drop freely into the boreholes of the reflector.

- Graphite is used in core areas with high temperatures (fuel elements, core internals). Temperature-induced failure of this material is impossible at the maximum temperature occuring which is 1,600°C.

- The noble gas helium, which from the chemical and neutron-physical viewpoints is a neutral one, is used as a coolant.

- Due to the high activity retention of the fuel elements, a pressure-tight reactor building is not necessary. The reactor building is accessible for repair work any time after accidents as a result of the low activity release.

- Reactor core and steam generator are installed in separate steel pressure vessels in such a way that there is no danger of component overheating in the event of failure of the primary-circuit cooling.

This choice of installation also increases the component accessibility for maintenance and repair.

The HTR-Module is therefore well suited for industrial applications and for the cogeneration of heat and steam, especially in cases where a base load has to be supplied continuously, such as for the chemical industry and for isolated small electrical grids.

1.2 General Layout and Reactor Plant Arrangement

The layout of the modular HTR power plant is shown in Fig. 1.

The reactor building is in the center of the plant. In addition to the two HTR modules, it contains some auxiliary systems. Components of the start-up and cool-down systems, the fast discharge system of the steam generator, and the intermediate cooling systems are located in the annex to the reactor building.

The two modular units are separated from each other by a central service area. An outer protective shell encloses the inner building structure. It meets the requirements for protecting the reactor plant from external impacts (e.g. aircraft crash).

The reactor auxiliary building directly adjoins the reactor building. It accomodates the facilities for helium purification and for treatment and storage of radioactive waste, the ventilation systems, the store for new fuel elements, the sanitary rooms and other rooms such as laboratories.

The central control building contains the control room, the switching, control and closed-loop control systems, the reactor protection system, the emergency power supply and further installations such as process computers, air-conditioning, and ventilation systems.

The machine hall basically holds the components of the water/steam circuit with the turbo sets and the components for the generation of process steam. The cooling tower (hybrid design) is positioned at a distance from the machine hall.

The nuclear system (HTR-module) generating steam basically consists of:

- The reactor pressure vessel with core, core internals, shutdown systems and facilities for charging and discharging fuel elements.

- The connecting pressure vessel with hot duct.

- The steam generator with the heating tube bundle and the primary circuit blower.

Each HTR-Module is installed in a primary cell, the concrete walls of which support the weight of the reactor pressure vessel, the connecting pressure vessel and the steam generator pressure vessel and their internals (Figs. 2 and 3).

On the inside wall of the primary cavity, surface coolers are installed with water flowing inside the tubes. These coolers discharge dissipated heat during normal operation and decay heat during reactor shutdown.

As shown in Fig. 2, the reactor and the steam generator are positioned next to one another. This layout characteristic of the HTR-Module offers substantial advantages:

- After a reactor shutdown, harmful natural circulation of hot helium through the primary circuit is prevented due to thermohydraulic decoupling of heat source and heat sink (the "hot" reactor core is higher than the "cold" steam generator). Consequently, there is no need for cooling the steam generator after shutdown. It can be shut off and left in a hot condition.

- The substantial separation of the reactor core and the steam generator by the shielding concrete walls of the reactor cell makes it possible to repair all defects in the steam generator bundle or the primary circuit blower without access problems.

Main design data of the HTR-Module are given in Table 1.

The HTR-Module uses the well-proven spherical fuel elements. The coated particles consist of a UO_2 kernel, which is coated with layers of pyrocarbon and silicon carbide. The U235-enrichment is about 7.8 %.

During normal operation, the fuel elements attain a maximum temperature of approximately 850°C.

The evaluation of numerous heat-up experiments has shown that the coatings of the particles are capable of retaining practically all radiologically relevant fission products inside the intact coated particles up to fuel element temperatures of around 1,600°C. For this reason, the HTR-Module was designed in such a way that the fuel element temperatures limit themselves to maximum values of about 1,600°C during all accidents.

The reactor core, which is located inside the reactor pressure vessel (Fig. 3), consists of approx. 360,000 spherical fuel elements in a loose pebble bed with a diameter of approx. 3 m and an average height of approx. 9.4 m. It is cooled by helium. The mean power density of the core is 3 MW/m^3, and the mean core outlet temperature is 700°C.

The reactor can be shut down by simply dropping the absorber rods into the reflector boreholes. 18 small-sphere shutdown units serve to compensate the reactivity increase due to a cold, unpoisoned core. Graphite spheres with a 10% B_4C content and a diameter of approx. 10 mm are used as shutdown elements. The spheres, which are stored in storage containers located above the top thermal shield and the side reflector, drop freely into the reflector boreholes on demand.

Fig. 3 shows the design of the reactor core:

The ceramic core structure, which essentially consists of side, bottom and top reflectors, forms the cylindrical vessel which holds the pebble bed. The ring-shaped side reflector consists of 24 individual columns made of separate graphite and carbon segments. The cold-gas plenum is located in the top reflector. The bottom reflector, which also consists of several layers, contains the hot-gas plenum.

A metallic core vessel, which is supported in the lower part of the reactor pressure vessel, acts as supporting structure for the ceramic core internals.

The cover of the core vessel is constructed as a radiation shield (the so-called top thermal shield) to provide access to the area above the core vessel for maintenance work.

This area is filled with stagnant helium during operation.

The drives of the shutdown and control systems are mounted on the thermal shield. These consist of the reflector rods and the small sphere shutdown units.

During normal operation, the pebble bed is cooled by the primary coolant flow. After flowing upwards through boreholes in the side reflector, the primary coolant is collected and deflected in the top reflector. It then flows downward through the top reflector, the pebble bed and the bottom of the core whereby it removes the heat generated in the core, before being collected in the hot-gas plenum and conveyed to the steam generator via a penetration in the side reflector.

The reactor core is located in the reactor pressure vessel which, together with the connecting pressure vessel and the steam generator pressure vessel, forms the so-called pressure vessel unit. The arrangement of this pressure vessel unit in the reactor building is shown in Fig. 2. For details of this unit see item 1.4.

The fuel element discharge tube passes centrally through the bottom of the reactor pressure vessel. Four other connecting pipes supply the fuel elements and the conveying gas for the small sphere shutdown units.

The cover, which can be removed for inspection or maintenance purposes, is bolted down with pre-stressed bolts and sealed with two metallic sealing rings.

The reactor pressure vessel is supported in the reactor cell by 3 brackets on one level with the connecting pressure vessel.

The connecting pressure vessel together with the hot-gas duct serves to guide the primary gas between the reactor pressure vessel and the steam generator pressure vessel.

The connecting pressure vessel contains the hot-gas duct comprising the metallic gas liner (through which the hot gas flows from the reactor to the steam generator), a fibre insulation and the outer metallic support tube.

The cold gas flows back from the steam generator to the reactor between the support tube and the connecting pressure vessel, thus cooling the metallic components.

The steam generator pressure vessel is positioned in the steam generator cell which is located next to the reactor pressure vessel on a slightly lower level.

It consists of the steam generator pressure vessel section and the blower pressure vessel section screwed on top of this. The steam generator is designed as a once-through, helical tube type with uphill boiling.

The primary circuit blower returns the helium to the reactor through the annular gap between the connecting pressure vessel and the hot-gas duct.

The blower consists of a single-stage radial compressor. A speed-controlled asynchronous motor is provided as drive. The motor is cooled with water. The blower with motor is installed in the blower pressure vessel section as a slide-in unit with vertical shaft and hanging impeller. A blower flap is located on the suction side of the blower.

The main auxiliary systems assigned to the nuclear steam-generating system are:

- Pressurization and cleaning of the primary circuit
- Continuous charge and discharge of fuel elements
- Removal of dissipated heat and decay heat from the primary cell
- Conditioning of the air in the rooms.

These systems are installed in the reactor building and in the reactor auxiliary building. They take advantage of the well-proven AVR and THTR designs and experience.

A novel system has been chosen for the removal of the decay heat via the cavity cooler.

The cavity cooler surrounds the reactor pressure vessel at a distance of about 1.5 m (Fig. 3). It is installed as a closed tube wall in front of the concrete walls of the reactor cavity with a clearance distance of approx. 10 cm. Water of a temperature of about 40°C flows through the cavity cooler at low pressure. Its task is to remove the dissipated heat amounting to approx. 400 kW to the connected intermediate cooling systems during normal operation, and to remove the decay heat from the reactor of up to approx. 850 kW to these systems in the event of failure of the main heat sink.

1.3 Essential Safety-Related Properties

The modular HTR power plant is designed in such a way that all accidents based on physically and technically plausible assumptions do not lead to inadmissible release of radioactivity. This is primarily due to the optimum exploitation of the favorable inherent features of small high-temperature reactors. "Inherent safety" is used in this case to describe the fact that the reactor itself reacts to certain malfunctions without the actuation of active systems or external controlling interventions in such a way that no inadmissible or even dangerous situations occur.

The technical and nuclear-physical design of the HTR-Module is such that the temperature of the fuel elements always stabilizes at a maximum of approx. 1,600°C, even in the case of assumed failure of all active shutdown and decay heat removal systems.

The selection of a low mean power density in the reactor core, the selection of a suitable geometry for the reactor core and the surrounding core internals, and the use of appropriate materials make sure that the decay heat can escape from the reactor core to the surrounding components and structure solely by means of physical processes (heat conduction, radiation, convection).

<u>Activity Release Barriers</u>

A characteristic safety feature of the HTR-Module is to be found in the fact that, under all operating and accident conditions, the radioactive materials formed during nuclear fission are enclosed in the coated particles in such a way that a significant release of activity from the coated particles can be excluded.

The safe enclosure of activity is guaranteed by the design of the fuel-particle coatings and the inherent limitation of the maximum accident temperature possible in the fuel elements of approx. 1,600°C.

The radioactive materials escaping from the few defective particles are partly retained in the fuel element matrix. The non-retained fraction is transferred to the primary coolant and distributed in the primary circuit.

The gas-borne activity in the primary circuit is reduced by radioactive decay, by extraction through the helium purification facility and by deposition on the surfaces of the primary circuit. The primary circuit therefore acts as a further barrier to prevent the release of radioactive materials.

The components of the pressure vessel unit are designed in such a way that failure can be excluded. Leakages in the connecting pipes which cannot be isolated are very improbable due to the planned quality assurance measures. In a leakage nonetheless postulated, it is in principle only the very low gas-borne activity in the primary coolant and a small fraction of the activity deposited on the primary circuit surfaces which might be released into the reactor building. For this reason, and basically because of the high retention capability of the coated particles, no tightness-related requirements are imposed on the reactor building.

Reactor Protective Actions

Accidents cause characteristic changes in the process variables which are detected by the reactor protection system which in turn initiates shutdown of the modular HTR plant if the specified limit values are exceeded.

Regardless of the accident, the same three protective actions are always performed for plant shutdown:

- Dropping of the reflector rods
- Shutdown of the primary circuit blower
- Isolation of the steam generator

The first two actions are for nuclear shutdown purposes, while isolation of the steam generator (i.e. closure of feedwater and live-steam valves) separates the reactor plant from the steam power plant.

The HTR-Module can be left standing in a hot condition until the causes and consequences of the accident have been repaired.

As there is no need to remove the decay heat by means of forced circulation within the primary circuit, no separate decay heat removal circuits are necessary.

In the case of a loss-of-pressure accident and a rupture of the steam generator heating tube, the primary circuit isolation valves are additionally closed resp. the steam generator is quickly drained.

More details on safety features and accident control will be given in the paper presented by Mr. Weisbrodt.

1.4 Description of the Pressure Vessel Unit

The design and arrangement of the pressure vessel unit, which forms the reactor coolant pressure boundary, are apparent from Figure 4. The pressure vessel unit consists of the the following components:

- Reactor pressure vessel
- Connecting pressure vessel
- Steam generator pressure vessel with blower pressure vessel section

The principal dimensions are given in Fig. 4. The reactor pressure vessel includes connections for

- Fuel discharge with failed fuel separator block
- Fuel feed with valve bank
- Small ball shutdown system with valve banks

The steam generator pressure vessel includes feedwater and main steam nozzle for connection to the water/steam cycle (secondary system).

The design of the pressure vessel unit draws on experience gained during the construction and operation of light water reactor pressure vessels by virtue of

- comparable design conditions
- comparable dimensions.

Fig. 5 compares the design and dimensions of reactor pressure vessels for

- 1300 MWe PWR plant
- 1300 MWe BWR plant
- HTR-Module plant

It is evident that the same design principles are observed for all important design features such as

- product form (only seamless forging)
- flanged joints
- nozzles
- closure head and bottom head.

This applies especially to the arrangement and design of welds in the pressure boundary and also to that of attachment welds, e.g. for vessel supports.

Against the background of light water reactor experience, it is evident that the requirements for

- proper functioning and loading
- proper selection and use of materials
- good manufacturing practice and ease of examination

which are crucial to the safety of the pressure vessel unit, are certainly met.

Design Criteria

A distinction is made in the design of the pressure vessel unit between the primary gas envelope and the areas around the main steam and feedwater nozzles (Fig. 4).

On the primary side the HTR-Module concept rules out contact of the hot gas with the pressure boundary. Furthermore, the difference in pressure between cold and hot gas protects the pressure boundary even in the event of leaks in the walls separating hot from cold gas.

The temperature range postulated in the design of the primary system is governed by functional conditions in the core region of the reactor pressure vessel (RPV). Since the annular gap between reactor pressure vessel and core barrel is filled with stagnant helium, the RPV-wall is heated or cooled solely by radiation and convection. Therefore, the RPV-wall warms up very slowly during initial plant start-up and restarts after long plant shutdowns. Therefore, the smallest margin to the actual RT_{NDT} temperature occurs during these service conditions. They are therefore the design basis for protection against brittle fracture.

As a result of the design basis conditions

- reactor scram with residual heat removal via the cavity coolers at full primary system pressure
- depressurization accident with residual heat removal via the cavity coolers

the decay heat of the reactor core and heat removal via the cavity coolers cause a temperature increase in the RPV wall. This temperature of 350°C is the maximum design temperature for the pressure vessel unit.

On the secondary side, the design criteria are governed by the water/steam cycle. The design temperature for the vessel is 350°C.

The scope of nuclear codes and standards applicable to reactor coolant-system components of light water reactors is limited to design temperatures of max. 400°C. Therefore, these codes and standards can be applied in full to the primary side of the pressure vessel unit.

Generally accepted technical rules (conventional codes and standards, ASME Code) are also available for component requirements governed by the higher secondary-side design temperature.

Materials

The base material for the pressure boundary of the pressure vessel unit is the same material used for light water reactor components, i.e.20 MnMoNi 55. Materials not covered by the codes and standards for light water reactors are used for the main steam nozzle for which design temperatures exceed 400°C. Fig. 6 shows the materials used. These materials have been qualified by many years' satisfactory performance in service in conventional power plants and by supplementary testing in connection with construction of the nuclear power plants THTR-300 and SNR-300.

Manufacture and Inspection

These statements on design and materials show that the nuclear codes and standards can be applied in full to light water reactors. It has been demonstrated that these rules can also be applied for quality requirements and verifications relating to manufacture and inspection.

For the few HTR-specific features of the pressure vessel unit, solutions have been found which are equivalent from the safety standpoint.

2. Design Details of the HTR-500 Power Plant

2.1 Technical and Commercial Aspects

The HTR-500 is an important innovation in the field of reactor technology. Due to its universal applicability in the electricity and thermal energy markets as well as due to its high flexibility regarding the site, and its safety characteristics, the HTR is predestinated for close-in siting and particularly suited for countries starting to build up their nuclear energy supply.

For reasons of slower increase of electricity demand, grid size, stand-by power reserve and scope of financing, the electricity market shows a worldwide trend towards power units between 300 and 600 MWe. The characteristics of the HTR-500 are especially suited to meet the requirements of the national and international markets for future power reactors.

The potential success of the HTR-500, from the vendor's point of view, is based on its technical and safety-related advantages, combined with its competitiveness. The risk of loss of the plant capital investment is very low. Even in the event of hypothetical accidents, evacuation is not required.

The spent HTR fuel elements are suitable for final disposal without the slightest need for conditioning.

The consistent utilization of HTR-specific characteristics as well as the optimization of components and circuits have led to the result that the electricity-generating costs of an HTR-500 are competitive with those of the large LWRs and more favorable compared to conventional power plants.

2.2 Overall Layout and Reactor Plant Arrangement

The design of the HTR-500 makes considerable use of the technology embodied in the THTR 300. Simplifications and optimization benefit in addition from recent results in the field of research, development and engineering. The fundamental features of the HTR-500 design are:

- the use of standardized components and proven materials
- the integration of all primary system components into a pre-stressed single-cavity concrete reactor pressure vessel (PCRV)
- the separation of operating and safety systems leading to a simple design.

Fig. 7 shows the reactor vessel housed within the reactor confinement building including the primary system, the shutdown facilities, parts of the decay heat removal system and other safety-relevant components. The reactor confinement building acts as a protection against external impacts. All components carrying activity are arranged inside the reactor confinement building except for the fuel element storage facility which is protected against external impact by separate measures. Fig. 8 shows a vertical section of the PCRV with its internals.

The prestressed concrete reactor vessel encloses the ceramic and metallic reactor internals, the helium circulators, the steam generators and the decay heat removal (DHR) heat exchangers. The pebble bed is completely enclosed by the graphite reflector resulting in a cylindrical core volume of about 200 m^3. The fuel element spheres are added continuously from above. Spent fuel elements are discharged from the core through three discharge pipes without interruption of power operation.

The heat produced in the core is transferred to the steam generators by the helium coolant loop. The helium is purified in the purification plant which operates in a bypass to the main helium circuit.

Under normal operation, the coolant is circulated by six circulators, each located above one of the steam generators. The helium flows downward through the pebble bed core being heated from 266°C to 723°C at a pressure of 55 bar. It is collected in the hot-gas plenum and transferred to the steam generators via the hot-gas ducts. In the steam generators, the helium flows upward cooling down to a temperature of 260°C with a pressure of 53.7 bar. From the circulators, the helium is passed to the cold-gas plenum including the outer areas of the reactor cavity. In this way, the concrete, the liner, and the liner cooling system as well as the control and shutdown rods and the metallic reactor internals are exposed to cold gas only. Passing through the gas flow guidance structures, the cold gas enters the core plenum above the pebble bed surface through slits in the top reflector.

Two auxiliary heat exchangers arranged between the six steam generators will remove the decay heat and the residual heat in case of an accident or if the main cooling system is not available. Below each of these DHR heat exchangers, a DHR circulator is arranged. The DHR heat exchangers are cooled by two separate loops of the DHR removal cooling water system. Table 2 contains the main design data of the HTR-500.

2.3 Safety Characteristics and Safety Concept

The special importance of the HTR-specific safety characteristics results from the limitation of damage in the event of accidents. Not only is the product of damage and frequency very small, but also the damage itself. Therefore, the environment is not exposed to consequences of a serious nature, even in the event of hypothetical accidents.

The main HTR-inherent safety features are:

- negative temperature coefficient of reactivity effecting self-stabilization and limitation of reactor power
- ceramic core structure and fuel elements ensuring resistance to temperatures of up to about 3,500°C, eliminating the possibility of core meltdown
- low ratio of power density to heat capacity resulting in a slow rise of fuel element temperature under accident conditions

- inert, phase-stable gaseous coolant ruling out total loss of coolant.

In addition to the inherent safety features, engineered passive safety measures are also applied in the HTR-500, such as utilization of natural convection and passive cooling water systems for residual heat removal and utilitzation of the fail-safe principle.

The plant safety system with its active, passive, and engineered safety equipment comes into action independently of human interference, thus offering a forgiving technology which even controls maloperation by the operating personnel.

The safety concept of the HTR-500 takes advantage of the experience gained from the licensing procedure of the THTR 300.

The concept is based on the classical safety systems for reactor shutdown, decay heat removal, and for activity control. Failure of the safety systems does not result in immediate inadmissible effects. Sufficient time is available for manual countermeasures after such hypothetical accidents. The multibarriers for retention and confinement of activity remain effective under accident conditions.

Reactor Shutdown Concept:

The shutdown concept consists of two shutdown systems which are different in structure and function.

The first shutdown system consists of 48 reflector rods travelling within the side reflector bore holes. It is their task to shut down of the reactor in case of accidents. During plant operation, all rods are withdrawn to their upper position. The rods drop into the side reflector under the effect of gravity. For withdrawal, each reflector rod is provided with an individual drive.

The second shutdown system consists of 72 in-core rods which travel directly into the pebble bed. The tasks of the system are long-term shutdown as well as control of activity changes during start-up and shutdown, load changes and during steady-state operation.

Decay Heat Removal Concept:

Decay heat removal following reactor shutdown takes place via the main cooling system or via the decay heat removal system. The decay heat removal system is used in case the main cooling system either is not available or not suitable for accident control. The decay heat removal system has the task to control, together with other engineered safety systems, occurring accidents according to postulated safety boundary conditions. The decay heat removal concept is characterized by the following features:

- use of the main cooling system if available
- use of natural convection in the primary system
- separate and redundant systems within the primary system which are protected against external impacts
- sufficient time for manual countermeasures after failure

Only in the hypothetical case that the decay heat removal systems cannot be restored, the liner cooling system of the pressurized concrete pressure vessel will be used to take the residual heat out.

The arrangement and configuration of the decay heat removal heat exchangers were selected to ensure sufficient natural convection in the primary circuit in the event of failure of the associated circulator. Access to the components is ensured by design, so that any manual measures required can be performed with sufficient reliability.

Activity Enclosure Concept:

The activity enclosure concept is based on the multibarrier principle (Fig. 9). The primary system with its components is arranged in a burst-safe pressurized concrete pressure vessel. This integrated design concept ensures the integrity of the primary system boundary and limitation of the postulated leakage cross sections. The maximum leakage cross section is limited by constructional measures to 33 cm.

The fuel elements contain the fuel in the form of coated particles which are embedded in the graphite matrix. The SiC coating represents an effective retention barrier for the highly mobile fission products.

The second barrier is formed by the concrete pressure vessel which is furnished with a leak-tight liner. The vessel penetrations are closed by vessel closures, the seals of which are monitored, so that any leakage can securely be detected.
Pipes containing primary gas, which penetrate the vessel, are designed to be failure-proof and supplied with shutoff valves. The pressure vessel is furnished with a redundant pressure relief system to limit pressure and prevent excessive stresses.

2.4 Main Reactor Components

Reactor Pressure Vessel:

The reactor pressure vessel is characterized by a distinct separation of its individual components (Fig. 8).

- The prestressed concrete structure carries the loads from the internal pressure and the internals and acts as a radiation shield. It is, therefore, simultaneously a biological shield.
- The liner including the penetration liners and vessel closures ensures a gas-tight containment for the coolant.

- The heat protection system consisting of the liner insulation and the liner cooling system protects the concrete and the liner from the temperatures of the coolant while minimizing the heat losses.

The prestressed concrete reactor vessel is designed as a single-cavity structure. Under the aspect of safety, this solution represents the optimum of a pressure boundary and can be well covered by calculations. This design also permits clear functional procedures and is uncomplicated with regard to its construction. This optimum design was mainly obtained by the following measures:

- an advanced and more exact know-how of the material data resulting from extensive test programs
- improved materials (e.g. more suitable concrete aggregates, use of steel of very low relaxation for the prestressing tendons)
- wire winding for horizontal prestressing instead of individual cables for circumferential prestressing resulting in substantial advantages during erection
- more comprehensive calculation procedures having a higher accuracy and, hence, reduction of uncertainties which required higher safety margins
- a more adequate design philosophy, i.e. partial prestressting instead of complete prestressing
- less penetrations due to the arrangement of the circulators above the steam generators
- prestressed concrete closures for the large penetrations are designed according to the same principles as the PCRV itself
- three safety valves for protecting the PCRV against overload which eliminate overload safety margins; the safety valves are set to three response levels between design and test pressure
- no shutdown of the reactor for in-service inspections but continuous monitoring of the structural behavior during operation.

For economic reasons, it should be tried to obtain a short construction period of the PCRV. Since the duration of the construction period is essentially determined by the installation of the mild reinforcement, the use of steel fibres may allow for a substantial reduction of the construction period. A further incentive for using steel-fibre concrete is the increase of the vessel safety. The guaranteed tensile strength and especially the toughness of this material result in an increase of the safety of the structural elements. Within a research project, a suitable steel-fibre concrete has been developed on the basis of the type of concrete used for the HTRs. The program includes the material development, testing, and calculation.

Reactor Internals:

One of the key features of the High-Temperature Reactor is the graphite structure enclosing the reactor core. This structure accomodates the forces arising from the pebble bed resulting from pressure drop or rod insertion. It acts as radiation shield, fuel element container and gas flow guidance structure.

The ceramic internals are exposed to mechanical, thermal, chemical and radiation loads. These loads result in requirements which have to be taken into consideration in design and construction of the ceramic elements.

The complete ceramic structure consists of the top reflector, the side reflector, and the bottom reflector (Fig. 10).

The side reflector is composed of an inner and an outer cylindrical wall set up from graphite blocks arranged in block columns and ring layers. All the component parts are fixed by dowels, the individual ring layers are positioned by keys. These keys, at the same time, have a sealing effect reducing the bypass flows through the vertical slits between the columns to a permissible extent.

The side reflector is supported on its outer surface by pre-stressed two-step spring packs which are attached to the thermal side shield. Azimuthal shifts of the side reflector are prevented by interlocking connection with the side shield. The side reflector can freely expand upwards.

The top reflector consists of hexagonal graphite columns which are vertically subdivided into three layers of blocks and suspended from the thermal top shield by means of a metal suspension structure. The top reflector can freely expand in downward direction.

The bottom reflector is composed of hexagonal graphite co-lumns which are vertically set up from individual sections and axially supported by columns having a circular cross sec-tion. These columns are fixed on the upper bottom layer for-ming at the same time the hot-gas plenum below the core bot-tom.

Changes compared to the THTR mainly result from increasing the THTR core to a higher power output, i.e. larger diameter, and from the different fuel cycle, i.e. the "Once Through, Then Out" cycle and its impact on neutron flux distribution.

For design optimization and verification, a series of tests has been performed with core structure models of various si-zes. The purpose of these experiments was to determine the forces upon the reflector side wall and the core bottom structure in horizontal and vertical direction due to dead load, pressure of pebble bed, and insertion forces of in-core rods. The test results have verified the design chosen.

In addition to the above experiments, various large-scale mo-dels were tested to verify the seismic safety of the whole structure. The results of these tests were compared to those of analytical methods which were developed in parallel. The system is stable under seismic loads.

Steam Generator:

The six steam generators are designed for a 40-year service life at a load factor of 0.8. The minimization of the PCRV dimensions requires a compact design, and the desired low gas pressure loss results in an arrangement of the heat exchange surfaces in cross-counter current. The shape of the assembly penetration in the vessel head requires a circular steam generator cross section. In an optimum response to the requirements stated above, such as compact design, circular cross section, and cross-counter current, the steam generator is of helical design. The structural design of the steam generator represents:

- a heat transfer tube bundle whose heat exchange surfaces are subdivided into individual tube cylinders in the form of heli-coils;
- the tubes of the individual tube cylinders are held in vertical tube support plates and pass through special sleeves provided for tube guidance and for avoiding friction and wear;
- the load of the complete tube bundle is accomodated in the cold range at the level of the feedwater inlet and steam outlet. This reduces the thermal compensation of the steam tubes and permits free thermal expansion of the overall tube bundle. Due to this arrangement, the expansion zone is located in the cold range which simplifies the design.
- Feedwater and steam are collected by passing through integrated tube sheet headers.

The thermal design of the heat exchange surfaces includes a safety margin, so as to permit plugging of defective tubes during service life without reducing the power output. Plugging can be performed from outside without opening the primary system. The materials used for the tubes and support structures are 15Mo3 in the economizer section, 10CrMo910 in the evaporator section, and Incoloy 800 in the superheater section. The main improvements compared to the THTR are:

- the simple integrated tube sheet headers replacing the complicated penetration of individual tubes through the vessel closure
- elimination of the reheaters which results in substantial economy in the steam generator bundle itself as well as in the steam-/feedwater circuit.

Helium Circulators:

In modern plant concepts for high-temperature reactors, the helium circulators with integrated electric motor drives are installed above the steam generators. The change from horizontal to vertical (as compared to THTR) involved moving from oil-lubricated bearings for supporting the circulator shaft to active magnetic bearings. Thus, the oil system is completely eliminated leading to improved safety and lower costs.

The magnetic bearing for these circulators will contribute to the desired simplification and the high availability of the HTR. By the introduction of this technology, it is possible to abandon the complicated and expensive facilities for lubricant supply of conventional circulators.

The use of active magnetic bearings in heavy components requires considerable effort in development. A special aspect of this development is safe run-down of the circulator in the event of deenergization and the resulting failure of the magnetic bearings. For this purpose, mechanical catcher bearings are provided, the function and service life of which are being tested in a special test facility (Fig. 11). In addition, a test for the prototype circulator is under construction.

3. Summary

For both reactor concepts under development at HTR-GmbH, the HTR-Module and the HTR-500, a design overview was given. Specific items for both concepts were described in more detail. The still remaining engineering tasks and development work including component testing and safety-related analyses are initiated.

With the HTR-Module and the HTR-500, advanced reactor power plants will be available which fully meet the requirements of systems of the so-called second generation. These reactor concepts may contribute to the solution of energy problems of the next century for power generation as well as for heat production for industrial applications.

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Tab. 1

HTR - 2 Module
Main Design Data, Cogeneration

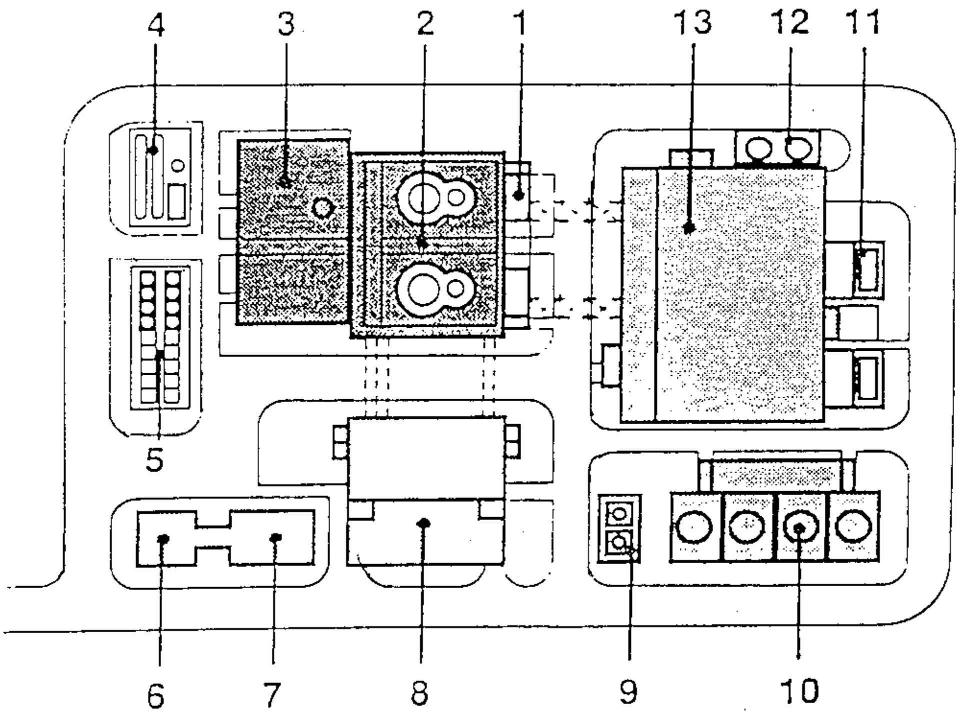
Overall plant	2 reactors
Reactor thermal power	400 MW_{th}
Generator terminal power - for max. process steam generation - for min. process steam generation	 72 MW 124 MW
Quantity of process steam max. min.	 115 kg/s 47 kg/s
Process steam pressure temperature	17 bar/272 °C
Reactor	
Number of fuel elements	360.000
Fuel element diameter	6 cm
Charging process	Multiple passage
Primary coolant	Helium
Helium flow at 100 % power	85 kg/s
Helium temperatures	250/700 °C
Mean helium pressure	60 bar
Mean power density	3.0 MW/ m^3
Steam Generator	
Live-steam pressure at SG outlet	190 bar
Live-steam temperature at SG outlet	530 °C
Live-steam mass flow	77 kg/s
Feedwater inlet temperature	170 °C

Tab. 2

HTR 500
Main Design Data

Thermal reactor power	1390	MJ/s
Net electrical output	550	MW
Efficiency	39.6	%
Average load factor	80	%
Average power density in core	6.6	MW/m^3
Primary gas pressure	55	bar
Hot- gas temperature at core outlet	723	°C
Cold- gas temperature at core inlet	266	°C
Main- steam pressure	180	bar
Main- steam temperature	525	°C
Feedwater temperature	190	°C
Condenser pressure	0.06	bar

HTR Module
Reactor Plant Arrangement



1 Reactor building annex
2 Reactor building
3 Reactor auxiliary building
4 Helium storage
5 Spent fuel element storage
6 Gate house
7 Office and staff building
8 Central control building
9 Wet- cooling- cells
10 Cooling tower
11 Transformer
12 Deionat tanks
13 Turbine building

Fig. 1

HTR - Module
Section through Reactor Building

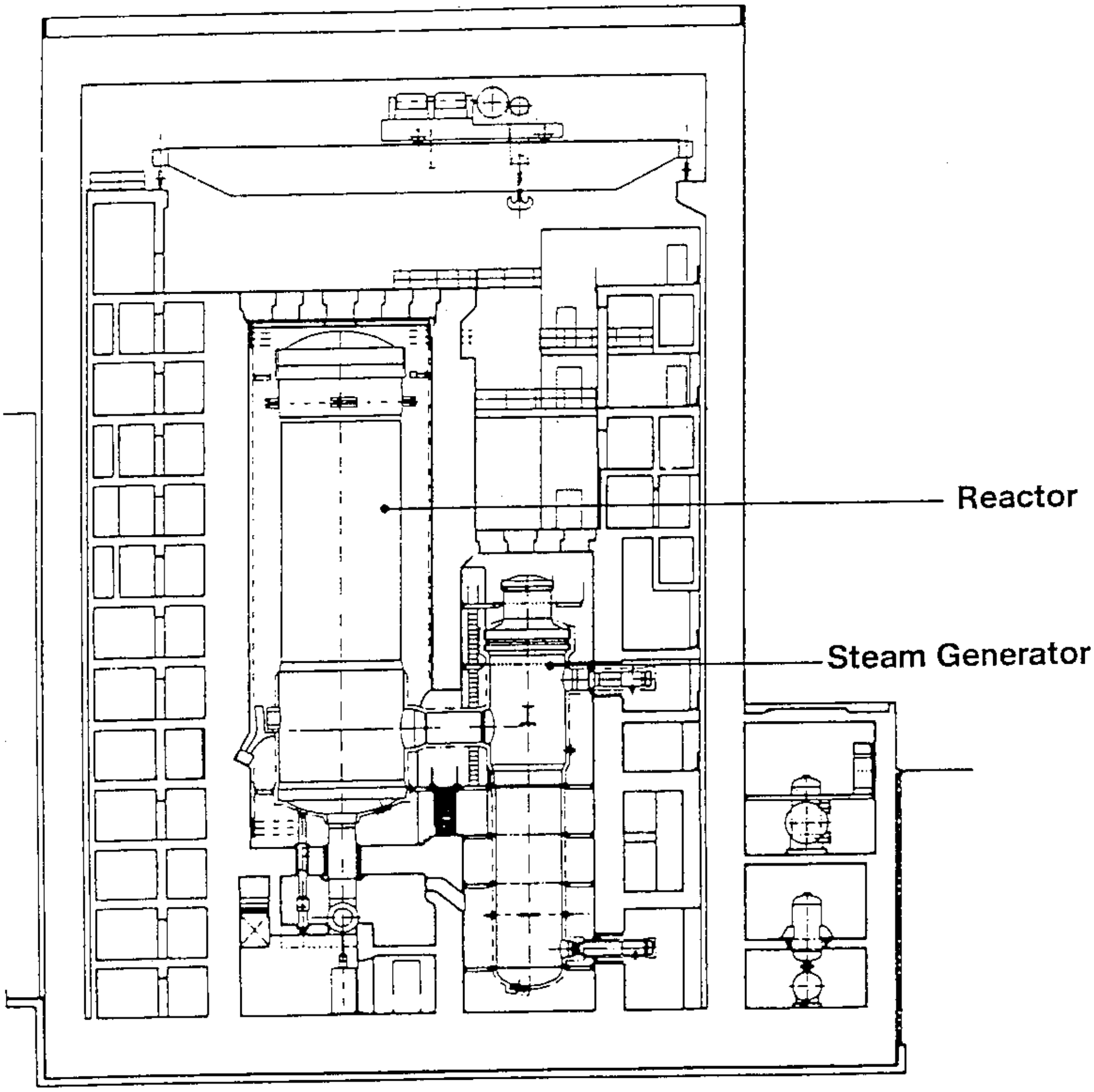


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Fig. 2

Primary Circuit of an HTR - Module

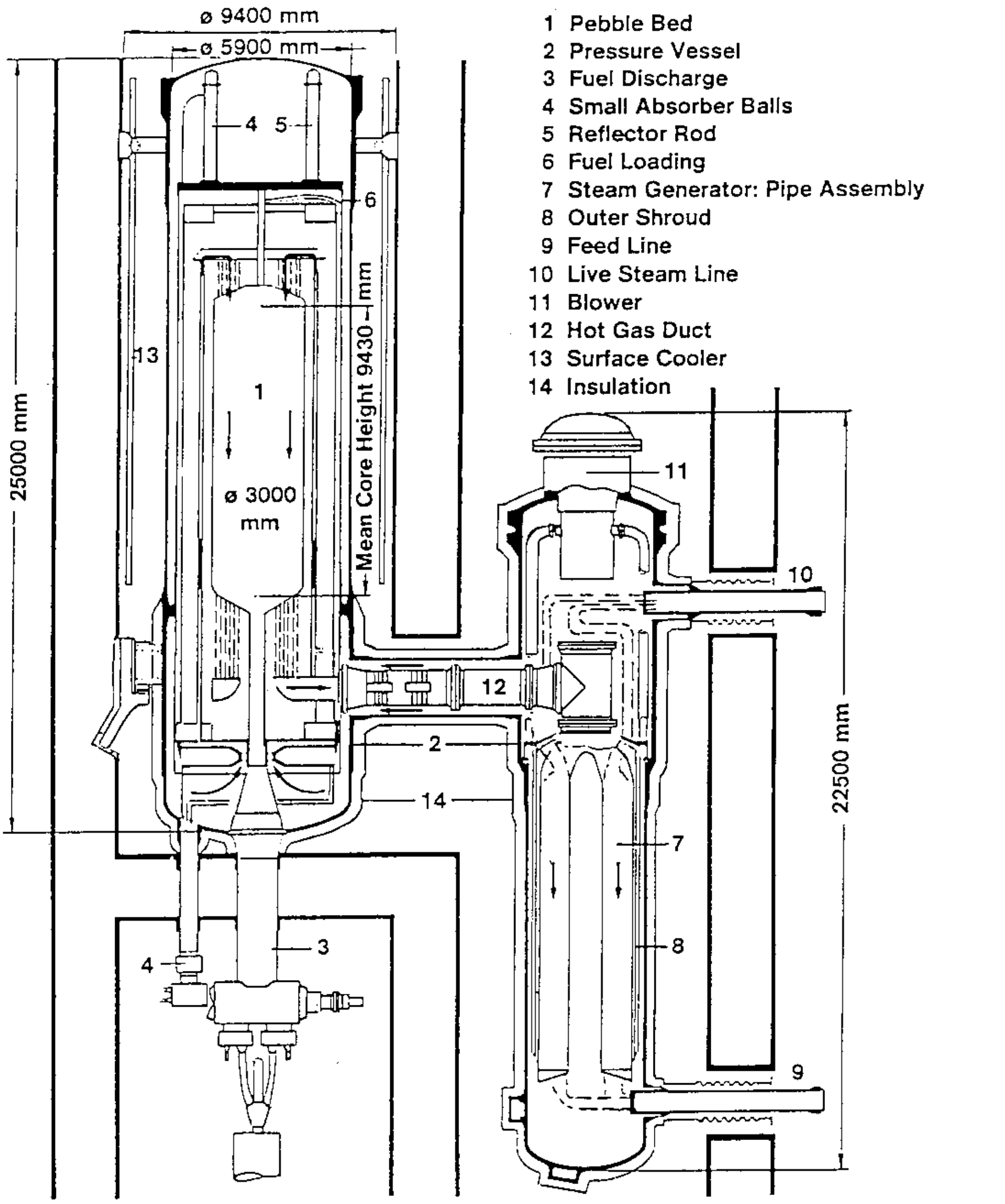

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Fig. 3

HTR Module
Pressure Vessel Unit

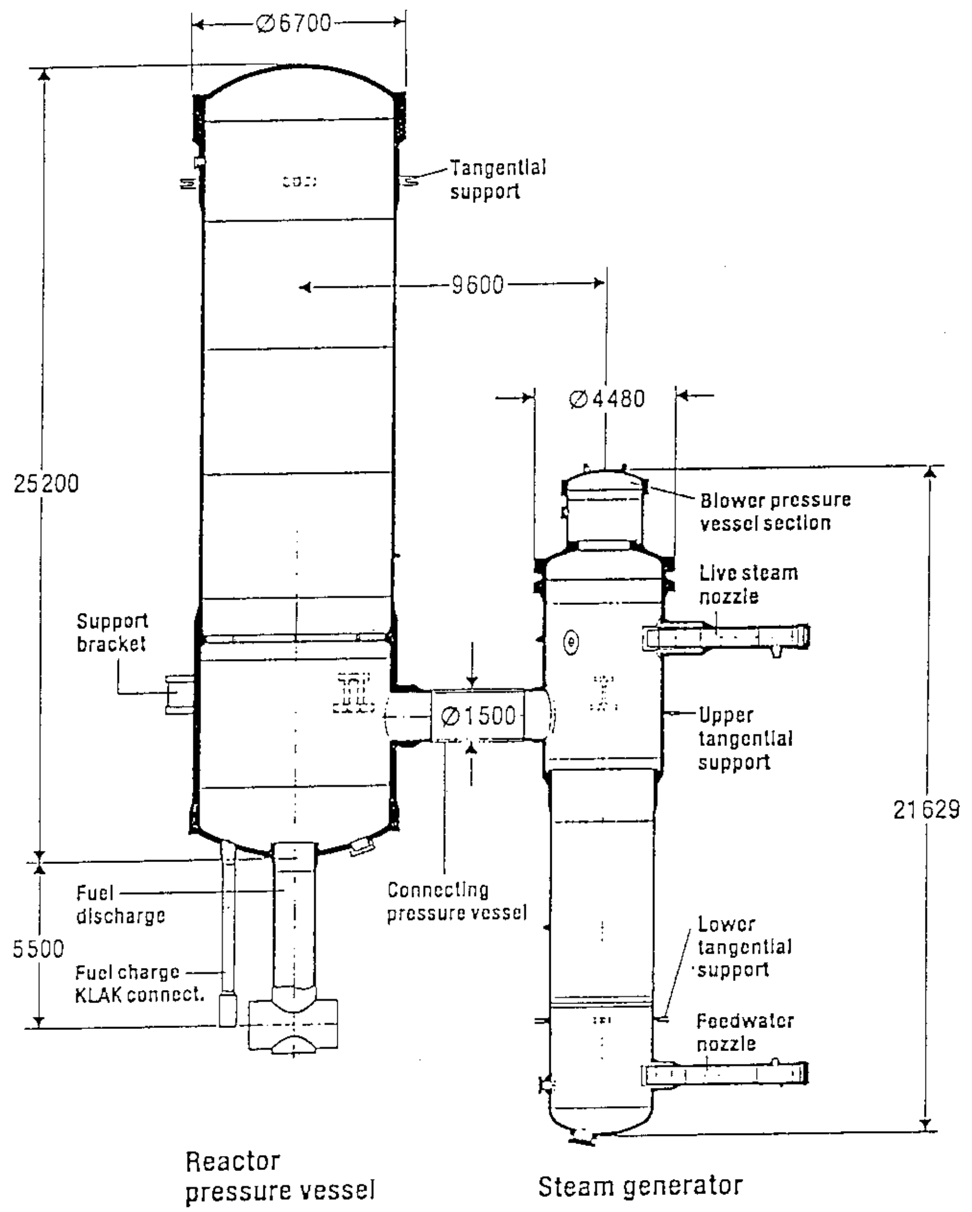
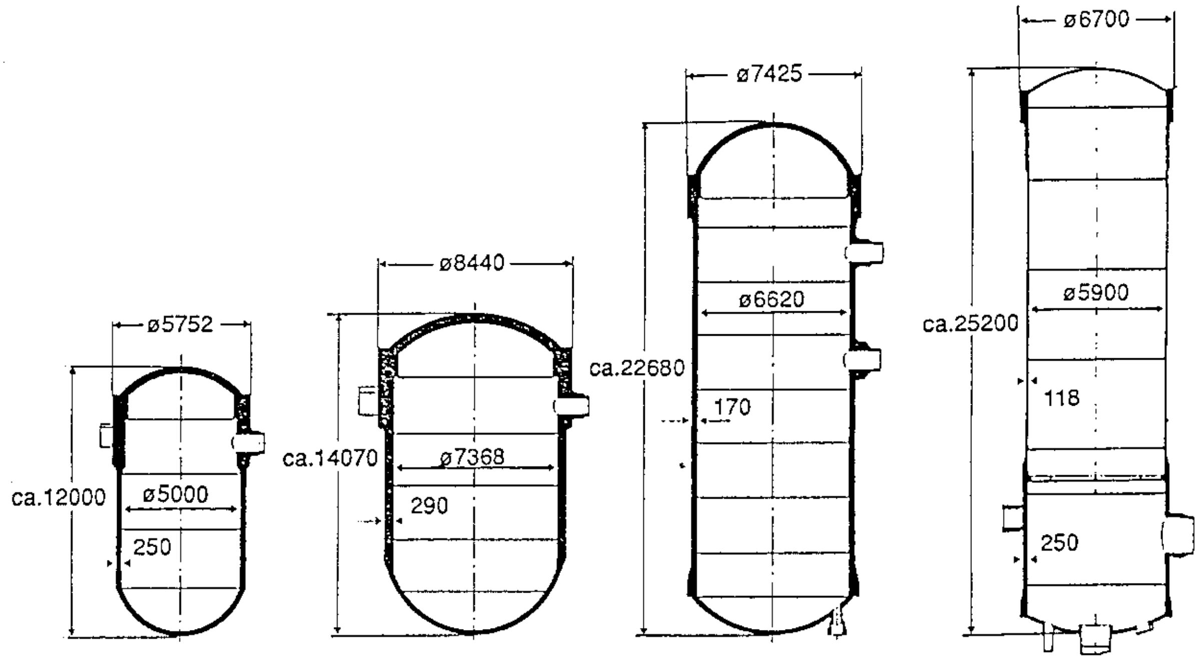


Fig. 4

HTR Module
Comparison Reactor Pressure Vessels



Pressure water reactor
PWR 1300 MW, 500 t
157 bar, 326 °C

Heavy water reactor
Atucha II, 1000 t
115 bar, 313 °C

Boiling water reactor
Grundremmingen, 780 t
71 bar, 286 °C

HTR-Module
Concept, 830 t
60 bar, 250 °C

Fig. 5

HTR Module
Steam Generator Pressure Vessel
Materials for Main Steam Nozzle

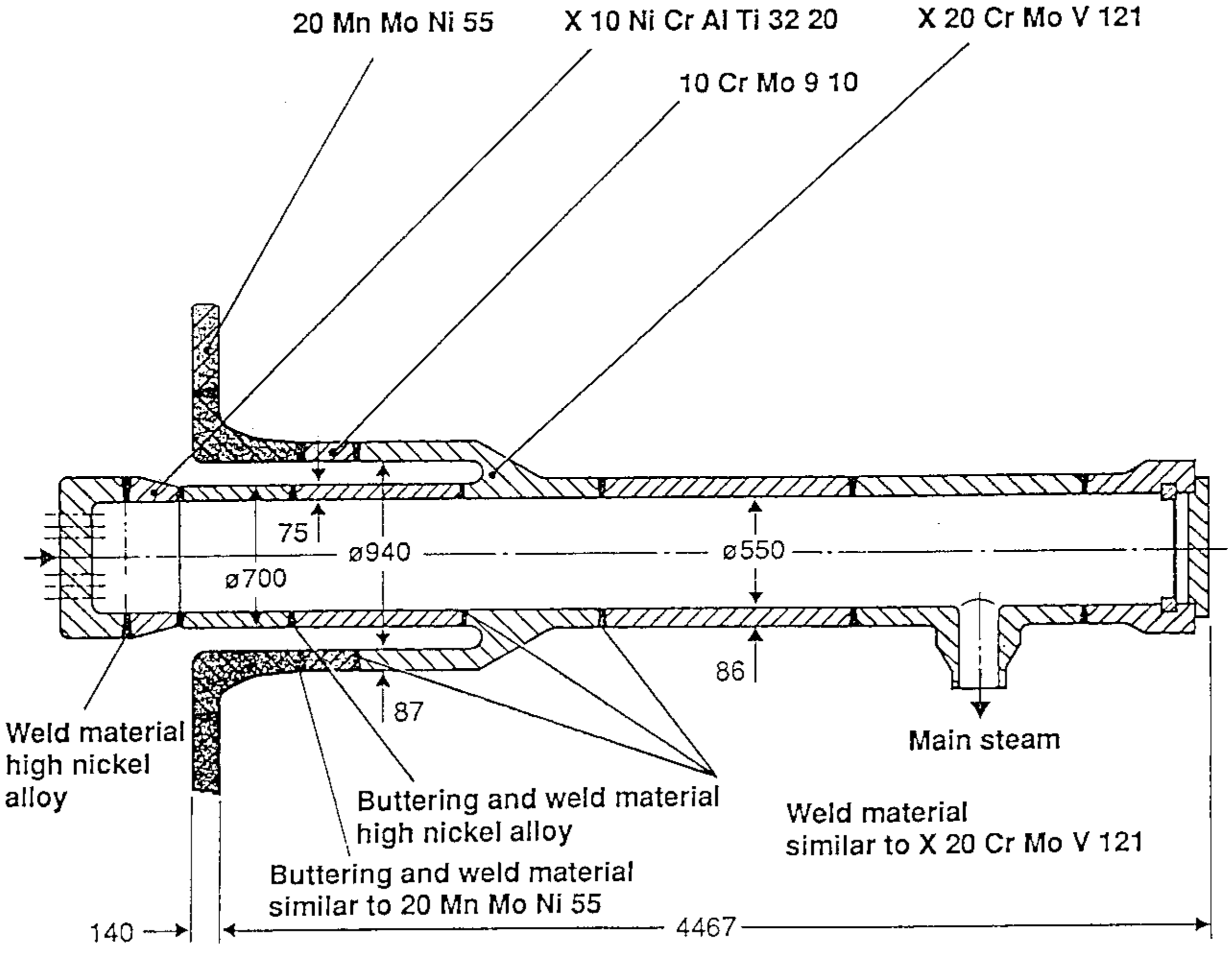


Fig. 6

HTR 500 - Nuclear Power Station
Longitudinal Section

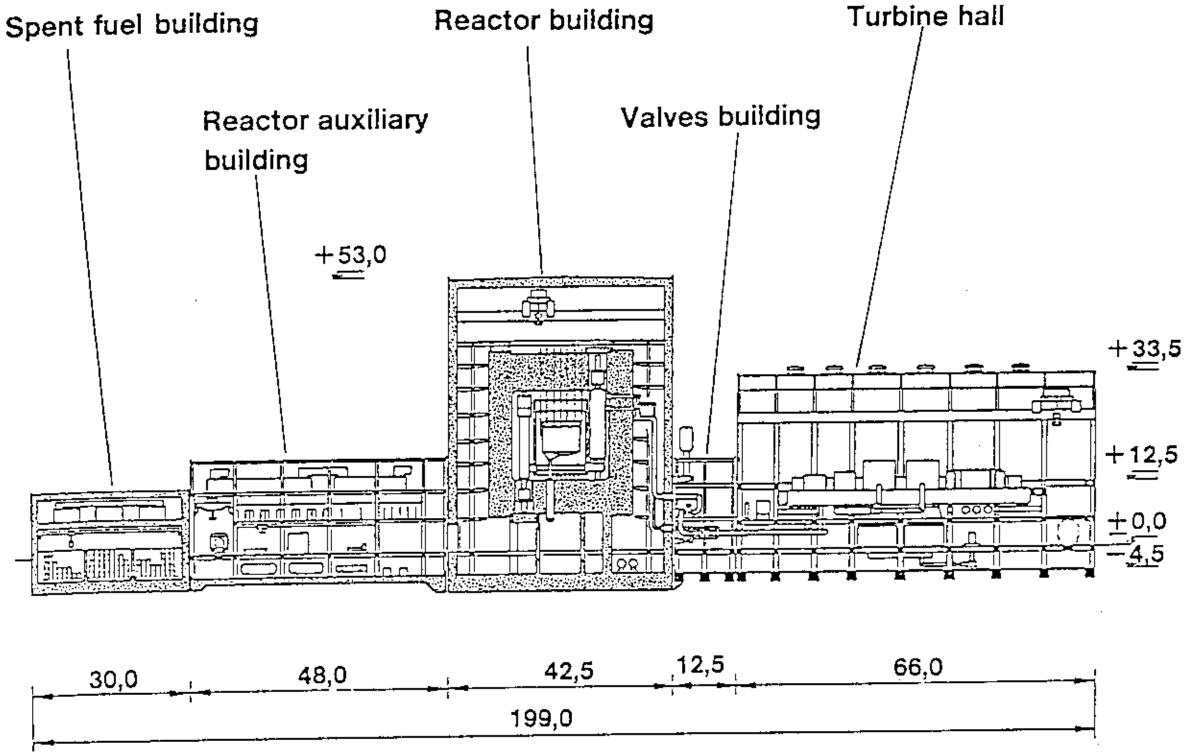


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Fig. 7

HTR 500
Reactor Pressure Vessel with Internals

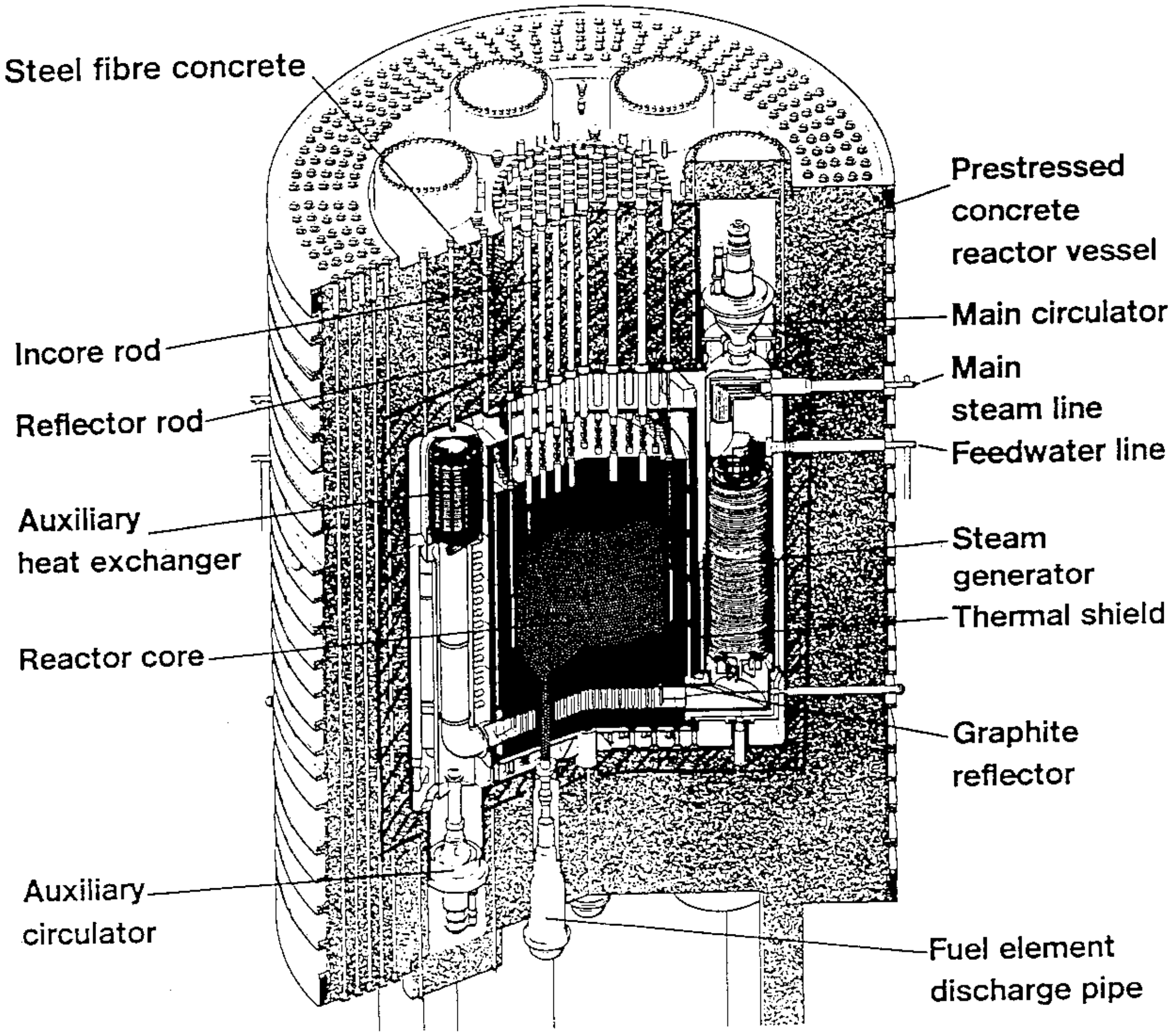

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Fig. 8

HTR 500 Multiple Barriers for Fission Product Retension

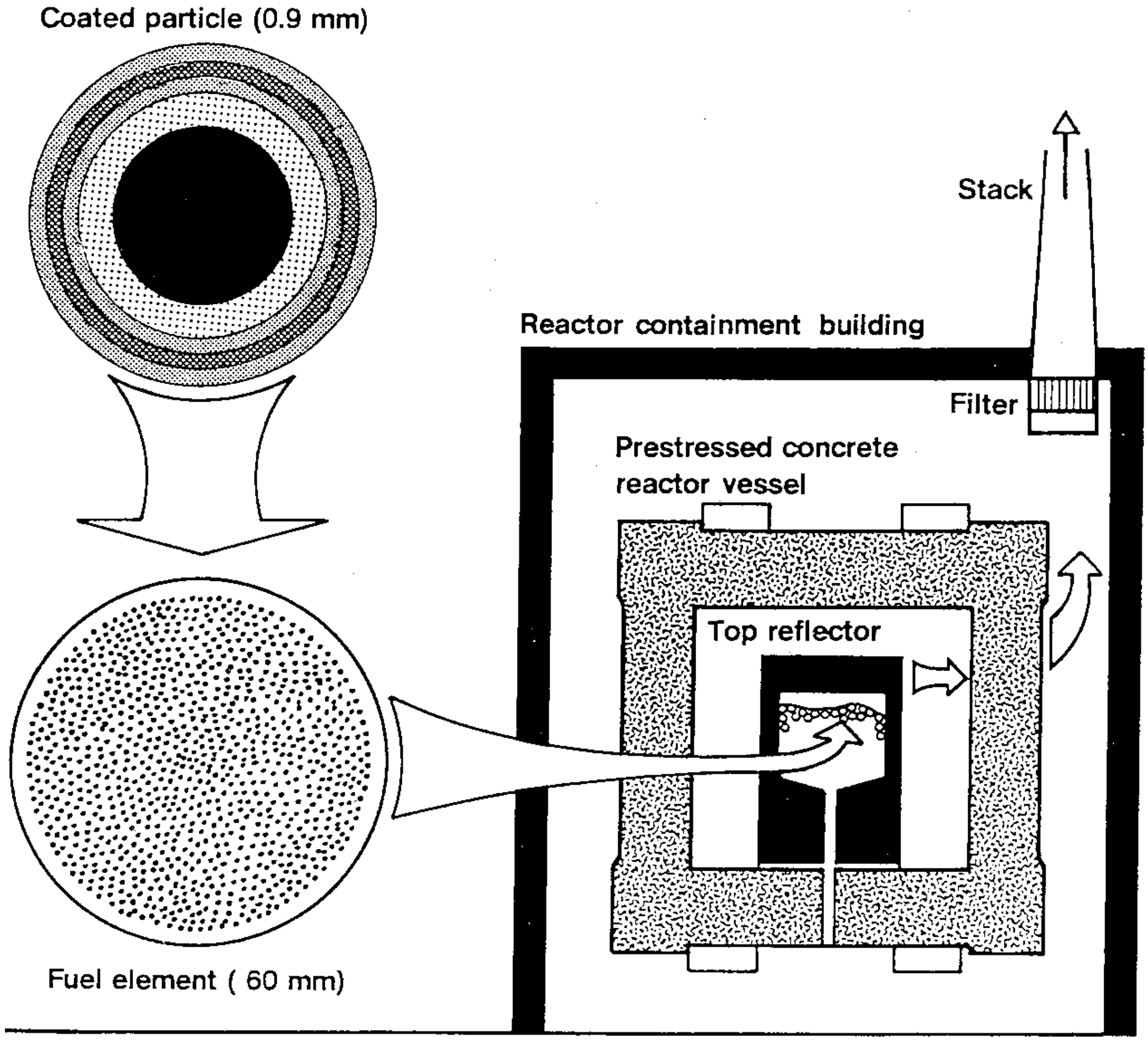
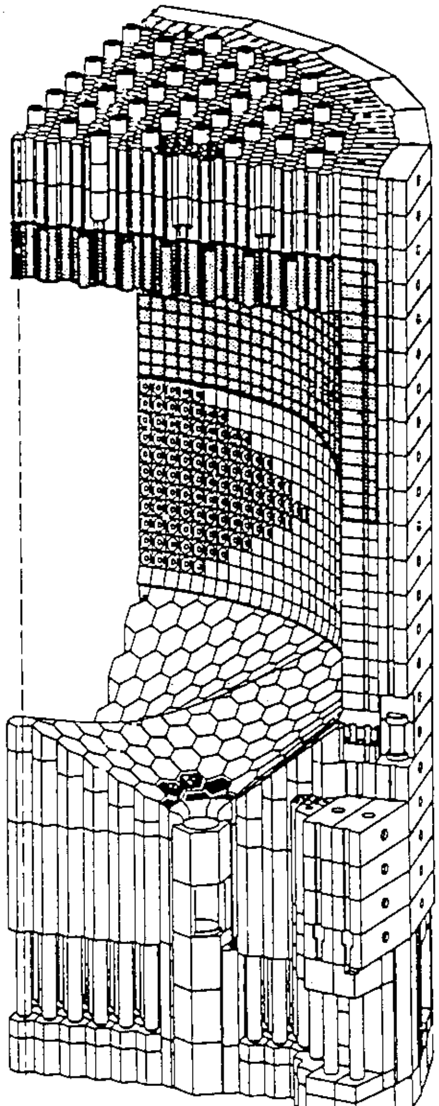


Fig. 9

Core Structure of a High-Temperature Reactor



89.14-6

Fig. 10

Retainer bearing test facility with active magnetic bearings

HTR

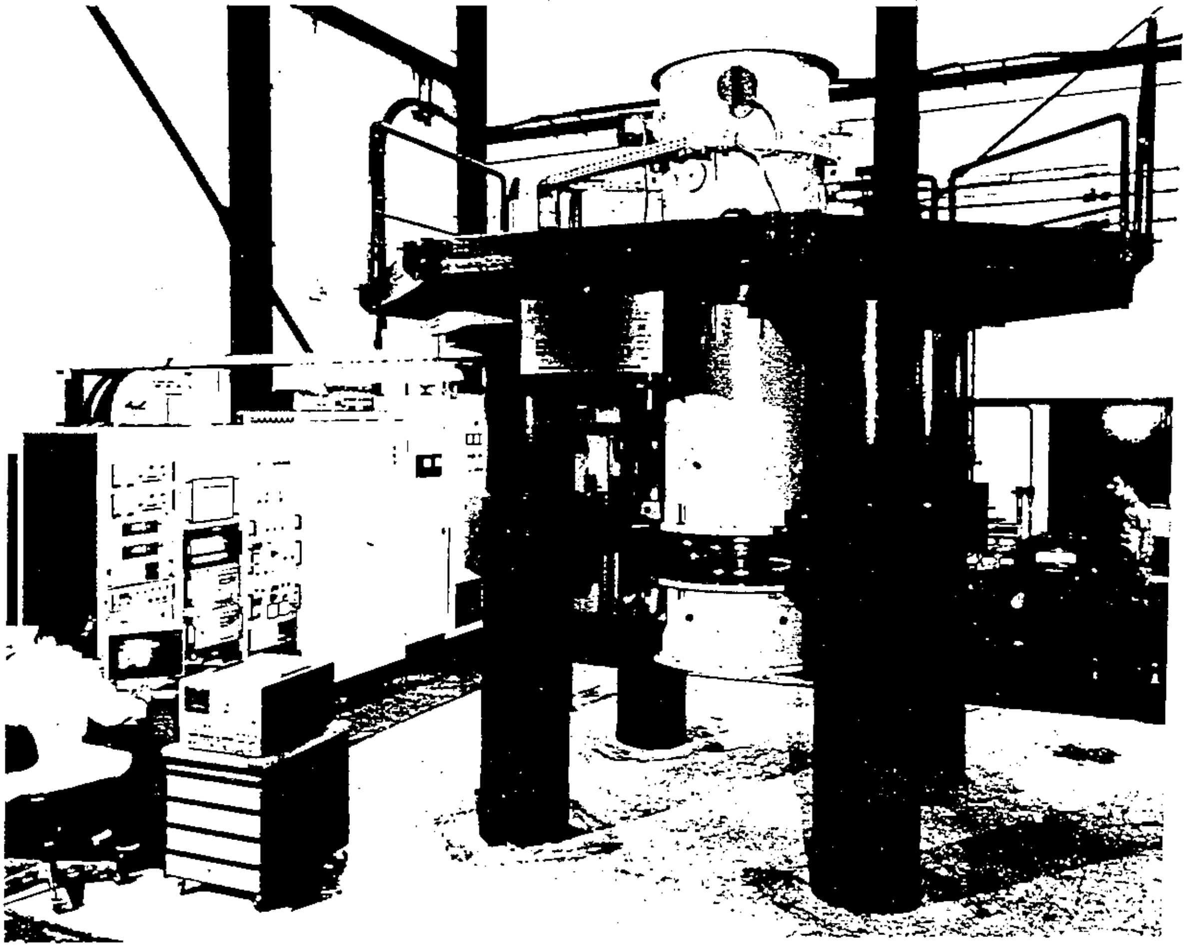


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Fig. 11

3. Licensing Safety Issues and Associated R&D of HTGR

3.1 KEY LICENSING SAFETY ISSUES AND DRAFT REVIEW RESULTS OF HIGH TEMPERATURE ENGINEERING TEST REACTOR (HTTR)

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1. Introduction

In Japan, the research and development(R&D) on the High Temperature Gas-Cooled Reactors(HTGRs) had been carried out for more than fifteen years as the multi-purpose Very High Temperature Gas-Cooled Reactor(VHTR) program for direct utilization of nuclear process heat such as nuclear steel making. However, the Japanese Atomic Energy Commission has changed the R&D program for a more basic "HTTR program" to establish and upgrade the HTGR technology basis, reflecting the change of social and energy situation and with no incentives for industries to introduce HTGRs in near future.

The HTTR is a test reactor with thermal output of 30 MW and outlet coolant temperature of 950°C, employing pin-in-block type fuel blocks, and has the capability to demonstrate nuclear process heat utilization using an intermediate heat exchanger. Since 1986 the detailed designs have been made , in which major systems and components are determined in line with the HTTR concept. The application for the construction permit of the HTTR, which is the first HTGR in Japan, was submitted by the Japan Atomic Energy Research Institute(JAERI) to the Science and Technology Agency (STA) in February 1989. After about 10 months safety review, the STA concluded the draft of safety review results on the HTTR in December 1989, taking into consideration the unique features of the HTTR and has referred it to the Nuclear Safety Commission (NSC) for double-check safety review.

This paper gives a brief presentation of the essential items on the results of safety review for the HTTR made by the STA, featuring and focusing on the major characteristics of the HTTR.

2. Major Features of HTTR

The HTTR has been so designed as to be an engineering test reactor which aims to establish and upgrade the technological basis for advanced HTGRs and to conduct various irradiation tests for innovative high temperature basic researches and various modes of operation and test for advanced HTGRs.

The HTTR plant, composed of a reactor building, a spent fuel storage building, a machinery building and so on, is to be constructed in the Oarai Research Establishment of the JAERI. The reactor building is centered in the

HTTR plant. The main reactor facilities of the HTTR such as the reactor pressure vessel, primary cooling system, reactor containment vessel and refueling machine are housed in the reactor building, as illustrated in Fig.1. The reactor pressure vessel is 13.2m high and 5.5m in diameter, and contains the core of 30 MWt , permanent and replaceable graphite reflectors, core support structure and radial restraining devices as illustrated in Fig.2. The main cooling system is composed of a primary cooling system, a secondary helium cooling system and a pressurized water cooling system. The primary cooling system has two heat exchangers, an intermediate heat exchanger and a primary pressurized water cooler, in parallel.

The major specification of the HTTR is shown in Table 1.

The reactor core is graphite-moderated, helium gas-cooled and hexagonal fuel elements are used. The active core consists of 30 fuel columns and 7 control rod columns, each column composed of 5 blocks (2.9m high) in series. The active core of 2.3m in diameter is surrounded by 15 replaceable reflector columns and 9 reflector-zone control rod columns. Some of replaceable reflector columns are used as irradiation test columns. The permanent reflector surrounds the replaceable reflector and is made up of large polygonal graphite blocks fixed by restraining devices. Each hexagonal graphite block has three dowel-pins on the top and three associated sockets at the bottom, and the blocks are fixed by setting dowel-pins into sockets.

A standard fuel element assembly, 36cm in width across the flats and 58cm in length, is made up of fuel rods and a hexagonal graphite block, as shown in Fig.3. The fuel consists of TRISO coated particles of low enriched uranium oxide whose average enrichment is about 6% and the kernel diameter is 600um. The particles are dispersed in the graphite matrix and sintered so as to form a fuel compact. These compacts are contained in a sleeve to form a fuel rod. The fuel rods of 3.4cm in diameter are contained within vertical holes of a graphite block. Helium gas flows through an annular channel between a vertical hole and a fuel rod inserted in it to remove heat produced by fission and gamma heating.

Reactivity is controlled by 16 pairs of control rods which are individually supported by the mechanisms located in standpipes connected to the hemispherical top head of the reactor vessel, and inserted into the channels in the active core and replaceable reflector regions. The reactor shutdown under the high temperature condition is made by inserting 9 pairs of control rods into the reflector region at first, then the other 7 pairs of control rods in the core region are inserted after the active core region temperature decreases. Back-up shutdown capability is provided by insertion of boron carbide/graphite pellets into the holes in the control rod blocks.

Major nuclear and thermal-hydraulic specification of the HTTR is shown in Table 2.

The reactor cooling system is composed of a main cooling system (MCS), an auxiliary cooling system (ACS) and two reactor vessel cooling systems (VCSs). The reactor cooling system is schematically shown in Fig.4. The ACS is in the stand-by condition during the normal reactor operation and is operated to remove the residual heat from the core when the reactor is scrammed. Both VCSs are operated at each 100% flow rate during the normal operation in order to cool biological shielding concretes around the reactor vessel, and they serve to cool the reactor vessel and the core under such abnormal conditions as a pipe break of the primary cooling system, when the core is no longer cooled effectively by the ACS.

3. Philosophy of Safety Review

(1) Basic policy for safety review

In the safety review of the HTTR, the review was performed by confirming the basic items of the safety design and so on , for ensuring the safety of the public, employee etc., under the condition of not only the normal operation but also accidents and then, for judging that the HTTR facility conforms to relevant standards of the licensing established by the Law for Regulation of Nuclear Source Material,Nuclear Fuel Material and Reactors.

The following items were selected as the basic policy, taking into account the characteristics of the HTTR.

1) Reactor facility shall be so designed as to withstand against anticipated natural phenomena including earthquake, meteorology, hydrology, etc. and anticipated artificial phenomena such as fire, missile, etc. in the site without loss of required safety function.
2) Reactor facility shall be designed for dose equivalent of the public caused by the release of radioactive materials during normal operation to be kept lower than the value of the off-site dose equivalent established and limited by the law and as low as reasonably achievable.
3) Protection and control of radiation shall be so designed that reactor operators may have no exposure over the value of dose equivalent by the law during normal operation.
4) Reactor facility shall be so designed that prevention and detection of occurrence of anomaly and mitigation of anomaly can be performed during normal operation.
5) Reactor facility shall be designed for integrity of fuel and reactor coolant pressure boundary not to be damaged against the trouble of equipment, miss-operation, etc. during the operation.

6) Reactor facility shall be so designed that influence of accidents including the rupture of reactor coolant pressure boundary, reactivity initiated accident, etc. may not spread and release of radioactive materials may be well mitigated.
7) Site location shall be selected so properly that a distance between the facility and the public may be kept enough so as to ensure the safety of the public even under the major and hypothetical accidents.

(2) Method of safety review

1) The review was performed based on the application documents for the construction permit of the HTTR submitted by the JAERI for the safety review.
2) As for the evaluation of site location,the natural condition such as geology, ground of site and social environment were reviewed by the submitted documents and field survey as occasion demands.
3) The review was performed taking into consideration that the HTTR has many characteristics such as adoptions of coated fuel particle as fuel, graphite as moderator and core internal structure, helium gas as coolant, outlet coolant temperature of about 950°C, etc. in comparison with the LWRs.

A basic method of security for safety should not be different from that of the LWRs and therefore, the review was performed, based on such existing guidelines established by the NSC as "Examination Guide of Reactor Siting and Guidelines for Interpretation in their Application" and "Examination Guide of Safety Design for Light Water Nuclear Power Facilities" and relevant standards established by the law, taking into account the characteristics of the HTTR and the differences of the HTTR from the LWRs.

(3) Key issues for safety review

Concerning the unique features of the safety design of the HTTR, the exposure evaluation during normal operation and safety assessment which are different from those of the LWRs, the review was performed to confirm that the design values have appropriate margins based on the experimental data, and especially, the following key issues were reviewed with emphasis.

1) Fuel; Integrity of the fuel should have been confirmed by irradiation tests, etc. in which the conditions of fuel temperature, fuel burn up, etc. under the normal operation condition and anticipated transients were appropriately considered.
2) Designs for graphite structure and high temperature structure; The design guideline for graphite structure for the HTTR shall have been established properly based on the information obtained from the result of

R&D, because the method of the strength design for graphite structure have not been defined by the law, the standard, etc.. The design guideline shall have also been established for high temperature structure properly because the reactor coolant pressure boundary is exposed to the conditions at high temperature and high pressure.

3) Design and safety assessment codes; As for design and safety assessment, verified computer codes shall be used. The uncertainty shall be considered in the evaluation as the appropriate safety margins.
4) Aseismic; The graphite blocks, etc. composing the core shall be designed based on the aseismic design, considering that these show a non-linear vibration behavior under earthquake. It should be confirmed that the supporting ground for reactor building has no possibility of liquefaction.
5) Selection of evaluated events; The following items shall be considered for selection of events to be evaluated in the HTTR.
   a. Anomaly of irradiation specimen and experimental equipments under irradiation tests.
   b. Graphite oxidation due to air and water ingress into the reactor core.
6) Source terms and release paths; Both of the source terms and release paths shall be properly considered in the dose evaluation during the abnormal occurrences as well as normal operation condition.
7) Irradiation tests and various modes of operation and test; Each test and operation shall be so designed that anomaly of the specimen under irradiation tests and abnormal occurrences during the operation and test may not cause loss of integrity of the HTTR.

4. Review Results for Key Safety Issues

(1) Fuel

The fuel consists of coated particles of low enriched uranium dioxide. The coated fuel particles are bonded together into fuel compacts which are contained in a graphite sleeve to form a fuel rod. The fuel rods are inserted into vertical holes of a graphite block, making up a fuel assembly, so-called a pin-in-block type fuel assembly. The fuel assembly is illustrated in Fig.3, and the major specification for fuel assembly is listed in Table 3.

The main safety requirements for the fuel are as follows.

1) The coated fuel particles shall not fail significantly under normal operation condition.
2) The maximum fuel temperature shall not exceed 1600°C in any abnormal operational transient.

3) The fuel element shall be designed not so as to lose the structural integrity during the full service period.

Fuel performance tests have been made in an in-pile gas loop (OGL-1) and in-pile capsules as well as out-of-pile tests under the simulated conditions of abnormal operational transients in the JAERI. It was confirmed from the test results that the fuel would be capable of maintaining its functions over the life time of the maximum burn up, 33,000 MWd/t, which meets the requirement 1) described above.

Regarding the requirement 2), the review confirmed that the maximum allowable fuel temperature of 1600°C in any abnormal operational transient has been determined based on the various core-heatup simulation test results in order to avoid significant failures of the coated fuel particles. The failure behavior of fuel under abnormal conditions is exemplified in Fig.5, together with the data of GAT Inc., USA.

The review also confirmed that the graphite sleeve and block would be designed so as to meet the graphite structural design code which has been developed in the JAERI and that the structural integrity of the fuel assembly could be maintained even against postulated abnormal conditions, satisfying the requirement 3).

(2) Designs for graphite structural and high temperature structure

The graphite materials and heat resistant metallic materials are employed for the reactor internal structures and the reactor pressure boundary components of the HTTR, respectively. Since these materials are not specified to be used in reactors by the Japanese criteria, standards, guidelines, etc., the graphite structural design code(1) and the high temperature structural design code(2) have been developed for the safety design of the HTTR on the basis of test data. Tables 4(a) and (b) show the features of these codes, respectively. The review confirmed that the material properties of such less ductility and larger scatter in the strength data are well reflected in the graphite structural code and that anticipated failure modes in the high temperature range are well taken into account in the high temperature structural code. These codes were, therefore, concluded to be reasonable to apply in the HTTR design.

(3) Verification of design and safety assessment codes

The nuclear characteristics of the HTTR were calculated by a nuclear design code system which consists of the DELIGHT(3), TWOTRAN(4) and CITATION(5) codes. The DELIGHT and TWOTRAN codes are used to produce the group constants. The CITATION code is a core analysis code. The thermal hydraulic characteristics were calculated by the use of a flow network analysis code

FLOWNET[6] and a fuel temperature analysis code TEMDIM[7]. The calculation flow of nuclear and thermal hydraulic designs is illustrated in Table 5.

The review confirmed that the nuclear design code system was verified through the comparison between the experimental results obtained by the VHTRC and SHE facilities and analytical results. The analytical and experimental results were agreed well as illustrated in Fig.6. The verification of the FLOWNET and TEMDIM codes was performed by the comparison between experimental results obtained by the HENDEL facility and analytical results. The review also confirmed that the verification results have satisfied the analytical accuracy and conservativeness required for the FLOWNET and TEMDIM codes.

Plant dynamics were analyzed by the use of the ASURA code[8] which simulates the characteristics of the HTTR plant control system. The verification of the code was performed by cross check with the core dynamics code BLOOST-J2[9] which is described later. The review confirmed that the agreement between the two codes was good.

Abnormal events anticipated in the HTTR were analyzed by the safety assessment codes which are shown in Table 6 with typical events analyzed by the corresponding codes. Since the basic design of the HTTR is different from that a typical LWR, the codes relevant to an LWR can not be applicable to the safety analysis of the HTTR. The main differences of safety assessment codes for the HTTR from those for an LWR are as follows. Transients of heat transfer and hydraulics with helium and graphite are treated, and the oxidation of graphite structure and the generation of inflammable gas by air and water ingress are treated.

The characteristics of each code are as follows. The BLOOST-J2 code is a code to analyze core dynamics. The THYDE code was originally developed to treat the transient of heat transfer and hydraulics for a LWR and the THYDE-HTGR code[10] is a modified code to analyze the transients of the whole plant of the HTTR, the TAC-NC code[11] is to analyze heat transfer in the reactor core, the RATSAM6 code[12] to analyze the mass and energy injection into the containment vessel during a depressurization accident, the COMPARE-MOD1 code[13] to analyze the temperature and pressure distributions in the containment vessel during a depressurization accident, i.e. pipe rupture of the primary cooling system, the GRACE code[14] to analyze the oxidation of graphite due to the air ingressed into the core during a depressurization accident, the OXIDE-3F code[15] to analyze the oxidation of graphite structure due to the water ingressed when a pressurized water cooler heat transfer tube is ruptured and the FLOWNET/TRUMP code[16] is to analyze the temperature distribution in a fuel assembly in case of channel blockage. Exposure estimations are performed by the use of the PLAIN and HTCORE codes which are not shown in Table 6.

It was confirmed that all of the codes have been verified by the comparison with experimental results or the cross check with other codes already verified. An example of the verifications is shown in Fig.7 for the BLOOST-J2 and THYDE-HTGR codes. The results of both codes were compared with the control rod withdrawal experiment with 50% full power of Fort Saint Vrain Reactor. The agreement is good as seen in the figure.

(4) Aseismic Design

For aseismic design, the following were mainly reviewed with respect to the supporting foundation and reactor core design.

1) Seismicity and geology at the site, and stability of supporting foundation, and
2) Structural integrity of the core and core support components, and scrammability of the control rods

Seismicity at the site has been assessed by investigating the historical earthquakes, active faults in land and sea, seismotectonic structures and micro-earthquake observations. It was confirmed to be reasonable that the maximum design earthquake S_1(:the maximum earthquake during lifetime of the plant in the area expected from historical earthquakes) with the maximum acceleration of 180 cm/sec^2 and the extreme design earthquake S_2(:the potentially maximum earthquake in the area determined from the seismotectonic considerations) with 350 cm/sec^2 were defined, respectively, based on the assessment.

Geology of the site has also been examined by means of boring, seismic reflection survey, etc..

Stability of supporting foundation of the HTTR, which is very dense fine sand, was confirmed to be enough in strength, settlement, possibility of slip and liquefaction, based on the test results and investigation results.

Since the reactor core is designed to pile up the graphite blocks on the core support structures connected with each other by the key/keyway system, structural integrity of the graphite structures and scrammability of the control rods are major concerns in seismic events. Various aseismic tests were performed to verify the analytical codes and demonstrate the aseismic assurance, as shown in Fig.8(a). The following were confirmed from the test results:

1) The aseismic integrity of the core and core support structures has been guaranteed by the demonstration tests and the analyses.
2) Seismic effect on the delay of scram time was not significant, as can be seen in Fig.8(b).

(5) Selection of postulated events for HTTR

Abnormal events to be postulated in the HTTR have been, first, selected based on the investigation on the key factors which affect on the items of safety evaluation criteria identified for the HTTR; (i) fuel temperature, (ii) core damage, (iii) temperature of reactor coolant pressure boundary, (iv) coolant pressure in the reactor coolant pressure boundary, (v) pressure of containment within its pressure boundary and (vi) risk of radiation exposure for the public.

Tables 7(a) and (b) show the selected abnormal events of the HTTR classified into anticipated operational occurrences and accidents, respectively.

Abnormal events concerning the reactor core and the reactor cooling system include the factors originating from the failure or malfunction of the irradiation specimen or experimental facilities during irradiation tests in addition to the factors originating from the failure or malfunction of the reactor or reactor cooling system itself. Events resulting in the air ingress and the water ingress, which are the principal design basis events in the HTGR, are also taken into account as abnormal events.

Abnormal events concerning the risk of radiation exposure are arranged from the standpoint of the differences in the release paths of radioactive materials into the environment.

Principal initiating events which cause each of the abnormal events have been identified based on the malfunction of the system or the equipments such as the primary cooling system, the pressurized water cooling system, the auxiliary cooling system, equipment containing radioactive materials, experimental facilities and so on. Initiating events for anticipated operational transients have been classified into similar event groups such as the abnormal change in the reactivity or power distribution in the core, the abnormal change in the heat generation or the heat removal in the core, the abnormal change in the pressure or inventory of the primary coolant, the transient caused by the irradiation specimen or an experimental facility, the transient during various modes of test operation in accordance with "Evaluation Guide for Safety Assessment of Light Water Nuclear Power Plant".

Initiating events for accidents have also been classified into similar event groups such as the decrease in the coolability for the core, the depressurization of the primary cooling system, water ingress into the core, the release of the radioactive materials, failure of irradiation specimen and experimental facilities in accordance with "Evaluation Guide for Safety Assessment of Light Water Nuclear Power Plant".

The severest events with respect to the safety evaluation criteria within each similar event groups are selected and have been analyzed as the postulated typical events for each group.

By the review, the selection of postulated events was judged to be reasonable and acceptable for the safety assessment of the HTTR.

(6) Siting-source-terms and release paths

The site evaluation event proposed for the HTTR is a double-ended rupture of co-axial double pipe of the primary cooling system. The fission products of the siting-source-terms are released to the containment vessel through the rupture of the co-axial double pipe along with the primary helium gas. The siting-source-terms for the HTTR are evaluated considering following radionuclide inventories and the review confirmed that the siting-source-terms are properly evaluated in the HTTR.

1) Radionuclide inventory contained in the core.
2) Radionuclide inventory contained in the coolant of the primary cooling system, which is released from initially defected fuel(1% of total loaded fuel). In the calculation, the minimum plate out fraction of iodine is conservatively used based on the test results.
3) Radionuclide inventory except for noble gases plated out on the surface of the primary cooling system, which is estimated conservatively expected for noble gases as 100% of plate out fraction.
4) Radionuclide inventory contained in the specimen of the fuel failure test which is planned as the irradiation test in the HTTR core.

The source terms have been developed in accordance with "Examination Guide of Reactor Siting and Guidelines for Interpretation in their Application".

The exposure dose to the public is evaluated from the amount of radionuclides which are released to the atmosphere and which are contained in the containment vessel for direct and skyshine exposure.

The amount of radionuclides which are released to the atmosphere is calculated, considering the release paths describe below:

1) Release path from the containment vessel pressurized during the accident to the service area surrounding the containment vessel, according to the elevation of the pressure.
2) Direct ground release path from the service area, according to the elevation of the pressure in the service area.
3) Stack release path from the service area through the emergency air cleaning system.

Figure 9 illustrates the release paths described above.

The review confirmed that these paths have been properly evaluated reflecting the characteristics of the HTTR.

(7) Safety ensurance for various tests planned in HTTR

Various kinds of irradiation tests and various modes of test operation including the demonstration of inherent safety features for advanced HTGRs are planned to be carried out in the HTTR. The test operation includes demonstration tests in which the primary coolant flow rate is decreased and a pair of control rods is withdrawn under the power operation condition.

The review investigated if tests could be carried out without the loss of required safety function of the HTTR even when an abnormal event occurred either in the HTTR facility or in additional equipments associated with tests, and it was confirmed by the review.

In the case of the test operation in which the primary coolant flow rate is decreased, some signal values set for the reactor scram, for example the signal for decrease of the primary flow rate, are designed to be changed automatically with a test operation mode switch. It was confirmed that the test operation could be achieved safe enough from the normal operation condition.

Every abnormal transient and accident postulated during the test operation modes is evaluated to confirm the reactor safety. Figure 10 shows analytical results of reactor transients when the operating circulator accidentally stops during the test in which two out of three circulators are stopped. It was confirmed that the maximum fuel temperature during the abnormal occurrence would not reach the maximum fuel temperature of the normal operation.

The review confirmed that during the irradiation tests and the test operation the safety ensurance was proper and the reactor was remained safe in any postulated event.

5. Concluding Remarks

The STA assessed the safety of the HTTR, fundamentally reflecting the safety requirements and evaluation criteria established for the LWR power plants and taking into account the major key features of HTGRs.

Referring the safety requirements for the LWR and based on various verification tests and many R&D results, the safety design on nuclear reactor core, fuel, graphite components, pressure boundary, irradiation tests and various modes of test operation in the HTTR could be confirmed with enough safety margins. As for the aseismic designs of foundation of reactor building and graphite core components, the seismic stability could be assured with related testings.

Taking into consideration the criteria of safety evaluation of the LWR, the review confirmed that assessed events have been properly selected for anticipated transients and accidents in the HTTR.

The draft review results has ,thus, concluded that the HTTR design is appropriate with respect to the safety.

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Table 1 Major specification of the 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

Table 2 Major nuclear and thermal-hydraulic specification of the HTTR

Thermal power	30 MW
Core diameter	2.3 m
Core height	2.9 m
Average power density	2.5 W/cm^3
Fuel loading	off-load, 1 batch
Excess reactivity	15% Δk
Uranium enrichment	3～10 wt%
average	about 6wt%
Fuel burn up (average)	22 GWd/t
Reactivity coefficient	
Fuel temperature coefficient	$-(1.5 \text{ to } 4.6)\times 10^{-5}\ \Delta k/k/^\circ C$
Moderator temperature coefficient	$(-17.1 \text{ to } 0.99)\times 10^{-5}\ \Delta k/k/^\circ C$
Power coefficient	$-(2.4 \text{ to } 4.0)\times 10^{-3}\ \Delta k/k/MW$
Prompt neutron lifetime	0.67～0.70 ms
Effective delayed neutron fraction	0.0047～0.0065
Total coolant flow	10.2kg/s (950°C Operation)
Inlet coolant temperature	395 °C
Outlet coolant temperature	950 °C (max.)
Power peaking factor	
Radial	1.1
Axial	1.7
Effective core coolant flow rate	88 %
Max. fuel temperature	1492 °C

Table 3 Major Specfication of HTTR fuel.

Kernel	
Material	: UO_2
Diameter	: 600 μm
Enrichment	: 3 - 10 % (av. 6%)
Density	: 95 %TD (Theoritical Density of UO^2, 10.96 g/cm^3)
Coatings	
Buffer(Low-density pyrocarbon)	: 60 μm
IPyC(Inner Pyrocarbon;high density)	: 30 μm
SiC (Silicon Carbide)	: 25 μm
OPyC (Outer Pyrocarbon;high density)	: 45 μm
Coated fuel particle	
Diameter	: 920 μm
Fuel compact	
Outer diam.	: 26 mm
Inner diam.	: 10 mm
Length	: 39 mm
Packing fraction of coated fuel particles	: 30 vol%
Fuel rod and graphite block	
Rod length	: 580 mm
Rod outer diam.	: 34 mm
Number of rods per graphite block	: 31 or 33

Table 4(a) Major features of the graphite structural design code

Item	Content
Category of structures	Core components and Core support components
Failure theory	Maximum principal stress theory
Analysis	Visco-elastic stress analysis to core components
Stress limits	• Secondary stress limits are specified in the same manner as primary stress limits. • Safety factor are specified to be equal to and/or severer in the graphite components than the metallic core suppurt strucures. • Static and cyclic fatigue limits are specified to all plant conditions.
Specified minimum ultimate strength	Survival probability : 99% Confidence level : 95%

Table 4(b) Feature of the high temperature structural design code

Materials	Feature of Code
2 1/4Cr-1Mo Steel	Design rules are the same as those of high-temperature design code for FBR "Monju" (FBR Code).
Austenitic Stainless Steel (SUS 321 and SUS 316)	
Ni-base corrosion resistant and heat resistant superalloy (Hastelloy XR)	Referred to the FBR Code, design rules are established on the basis of material properties and component test data.

Significant Failure Modes	Feature of Design rules
Creep Rupture	Limiting the primary stress intensities
Creep-Fatigue Failure	Limiting the accumulated creep-fatigue damage developed by primary+secondary+ peek stress
Loss of Function by Excessive Deformation	Limiting the Strain
Creep Buckling	Limiting the loads and strains

Table 5 Calculation Flow of Nuclear and Thermal-Hydraulic Design

(Code name and objectives)

CODE NAME	Objectives
DELIGHT, TWOTRAN	· Group constant
CITATION	· Reactivity · Control rod worth · Power distribution · ℓ , β
FLOWNET	· Coolant flow rate distribution · Coolant temperature
TEMDIM	· Fuel temperature

Table 6 Typical Events and Computer Code Used in the Safety Analysis

COMPUTER CODES	TYPICAL EVENTS
BLOOST-J2	<ANTICIPATED OPERATIONAL TRANSIENTS> · Abnormal Control Rod Withdrawal from Subcritical Condition · Abnormal Control Rod Withdrawal at Power Operation · Transients During the Irradiation Tests Reactivity Addition by the Move of Irradiation Specimen
BLOOST-J2, TAC-NC, GRACE	<ACCIDENTS> · Rupture of Stand-Pipe
THYDE-HTGR	<ANTICIPATED OPERATIONAL TRANSIENTS> · Decrease in Primary Coolant Flow Rate Coast-Down of IHX-Side Primary Helium Circulator Primary Helium Circulator Exhaust Valve Opened by Error in the Primary Helium Storage and Supply System · Increase in Primary Coolant Flow Rate Revolution Increase of IHX-Side Primary Helium Circulator Revolution Increase of PWC-Side Primary Helium Circulator Feed Valve Opened by Error in the Primary Helium Storage and Supply System · Decrease in Heat Removal in the Secondary Cooling System Bypass Flow Control Valve Opened by Error in the Air Cooling System Exhaust Valve Opened by Error in the Secondary Helium Storage and Supply System · Increase in Heat Removal in the Secondary Cooling System · Loss of Off-Site Electric Power · Transients During Safety Demonstration Test <ACCIDENTS> · Rupture of Secondary Double Concentric Pipe · Inner Pipe Failure of Secondary Double Concentric Pipe · Rupture of PWC System Pipe
THYDE-HTGR, TAC-NC	<ACCIDENTS> · Inner Pipe Failure of Primary Double Concentric Pipe
THYDE-HTGR, TAC-NC, RATSAM6, COMPARE-MOD1 GRACE	<ACCIDENTS> · Rupture of Primary Double Pipe
THYDE-HTGR, OXIDE-3F	<ACCIDENTS> · Rupture of Pressurized Water Cooler Heat Transfer Tube
TAC-NC	<ANTICIPATED OPERATIONAL TRANSIENTS> · Transients During the Irradiation Tests Insulation Degradiation of Irradiation Capsule
FLOWNET/TRUMP	<ACCIDENTS> · Rupture of Stand-Pipe

Table 7(a) Selection of Abnormal Events for Anticipated Operational Occurrences Based on Factors Essential for Safety Evaluation Criteria

Factors Essential for Safety Evaluation Criteria	Sequence of Abnormal Events			
Increase in fuel temperature		Increase of power	Reactivity addition	Malfunction of the reactivity control system
				Malfunction of the experimental facility
				Increase in the primary coolant flow
				Increase in the heat removal by the secondary cooling system
		Decrease in heat removal in the core	Decrease in the core coolant flow	Decrease in the primary coolant flow
			Increase in the core inlet temperature	Decrease in the heat removal by the pressurized water cooling system
				Decrease in the heat removal by the secondary helium cooling system
			Heat release from the experimental facility	Insulation degradation of the irradiation capsule
Increase in the temperature of the reactor coolant pressure boundary	Increase in the primary coolant temperature	Increase of power	Reactivity addition	Malfunction of the reactivity control system
				Malfunction of the experimental facility
				Increase in the primary coolant flow
				Increase in the heat removal by the secondary cooling system
		Decrease in heat removal in the core	Decrease in the core coolant flow	Decrease in the primary coolant flow
			Increase in the core inlet temperature	Decrease in the heat removal by the pressurized water cooling system
				Decrease in the heat removal by the secondary helium cooling system
Increase in the pressure of the reactor coolant pressure boundary	Increase in the primary coolant temperature	Increase of power	Reactivity addition	Malfunction of the reactivity control system
				Malfunction of the experimental facility
				Increase in the primary coolant flow
				Increase in the heat removal by the secondary cooling system
		Decrease in heat removal in the core	Decrease in the core coolant flow	Decrease in the primary coolant flow
			Increase in the core inlet temperature	Decrease in the heat removal by the pressurized water cooling system
				Decrease in the heat removal by the secondary helium cooling system
				Increase in the primary coolant inventory

Table 7(b) Selection of Abnormal Events for Accidents Based on Factors Essential for Safety Evaluation Criteria

Factors Essential for Safety Evaluation Criteria	Sequence of Abnormal Events			
Core damage	Increase in fuel temperature	Increase of power	Reactivity addition	Malfunction of reactivity control system
		Decrease in heat removal from the core	Local decrease in the core coolant flow	Decrease of coolant flow in the fuel channel
				Decrease of the core coolant flow
				Loss of primary coolant
			Increase in the core inlet temperature	Decrease in the secondary helium flow
				Loss of secondary helium coolant
				Decrease in heat removal by the pressurized water cooling system
			Heat release from the experimental facility	Decrease in coolant flow of the fuel channel for the fuel failure test
	Loss of core support function of support post		Graphite corrosion by the air and water	Air ingress
	Drop of fuel element			Water ingress
Increase in the temperature of the reactor coolant pressure boundary	Increase in the primary coolant temperature	Increase of power	Reactivity addition	Malfunction of reactivity control system
		Decrease in heat removal from the core		Decrease of the core coolant flow
				Loss of primary coolant
			Increase in the core inlet temperature	Decrease in the secondary helium flow
				Loss of secondary helium coolant
				Decrease in heat removal by the pressurized water cooling system
Increase in the pressure of the reactor coolant pressure boundary	Increase in the primary coolant temperature	Increase of the power	Reactivity addition	Malfunction of reactivity control system
		Decrease in heat removal from the core		Decrease of the core coolant flow
				Loss of primary coolant
			Increase in the core inlet temperature	Decrease in the secondary helium flow
				Loss of secondary helium coolant
				Decrease in heat removal by the pressurized water cooling system
				Pressurized water or secondary helium ingress into the primary circuit
Increase in the pressure difference of reactor coolant pressure boundary(IHX tube)				Loss of secondary helium coolant
Increase in the pressure of containment pressure boundary				Loss of primary coolant
				Loss of secondary helium coolant
				Loss of pressurized water
Risk of radiation exposure				Leak of radioactivity into the containment
				Leak of radioactivity into the service area
				Leak of radioactivity outside the service area
				Leak of radioactivity into the glove box

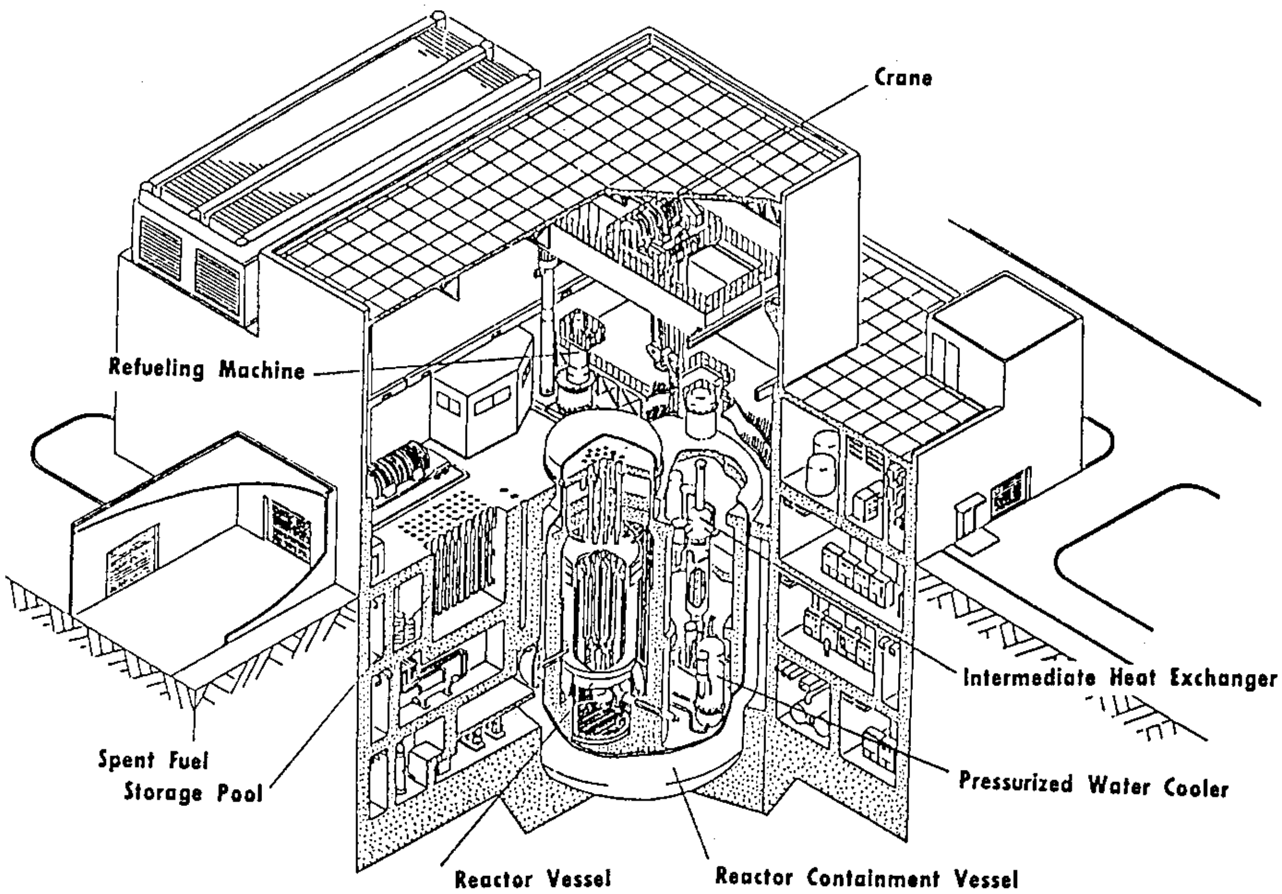


Fig. 1 HTTR reactor building

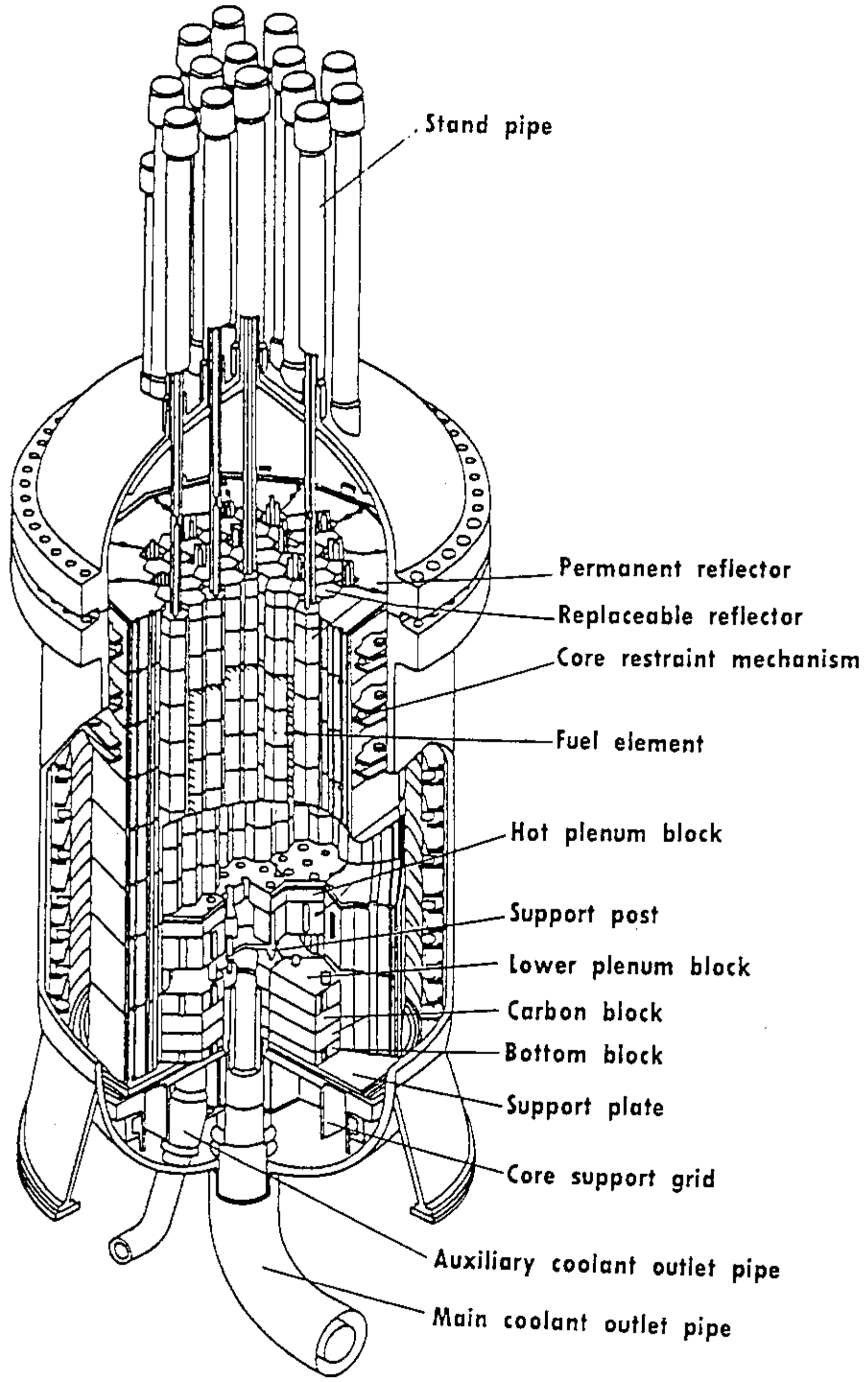


Fig. 2 Bird's eye view of the reactor vessel and core

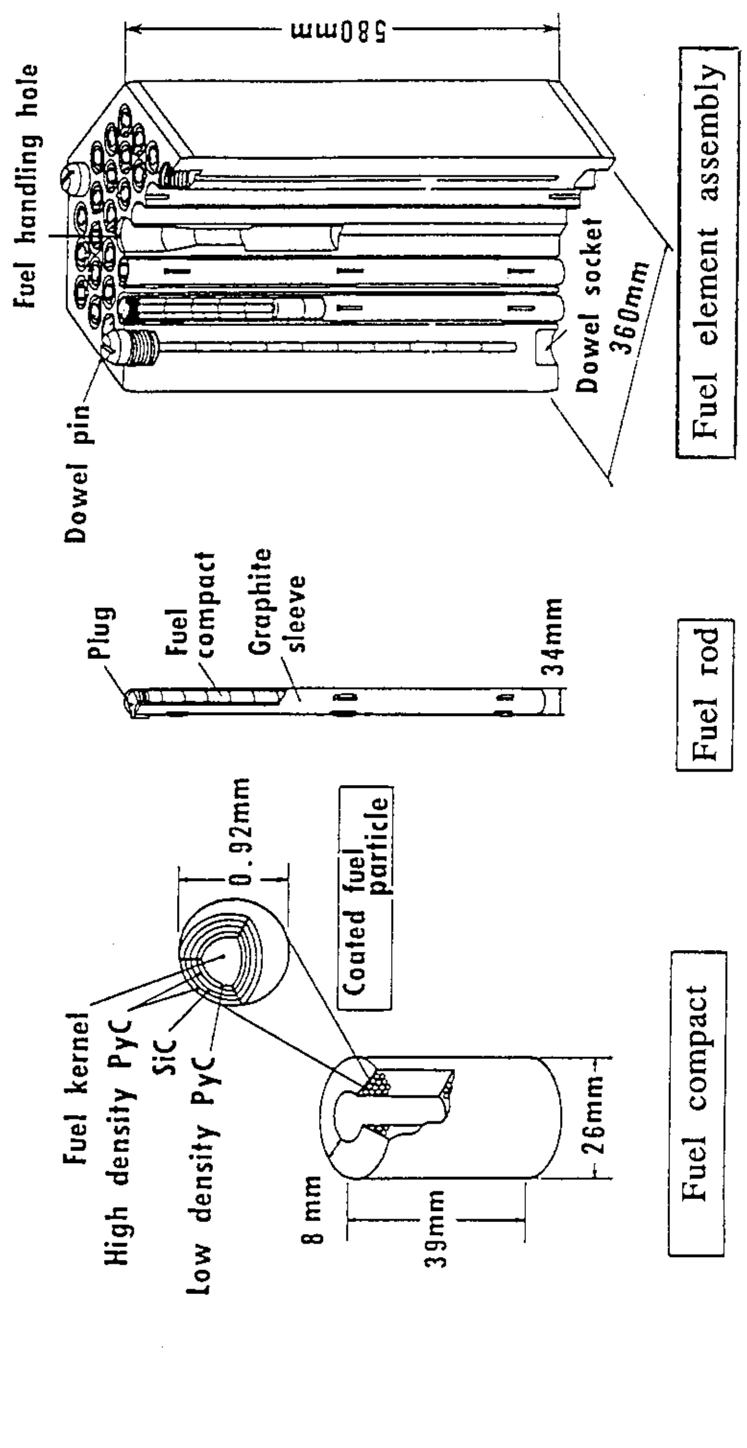


Fig.3 Structure of HTTR fuel assembly.

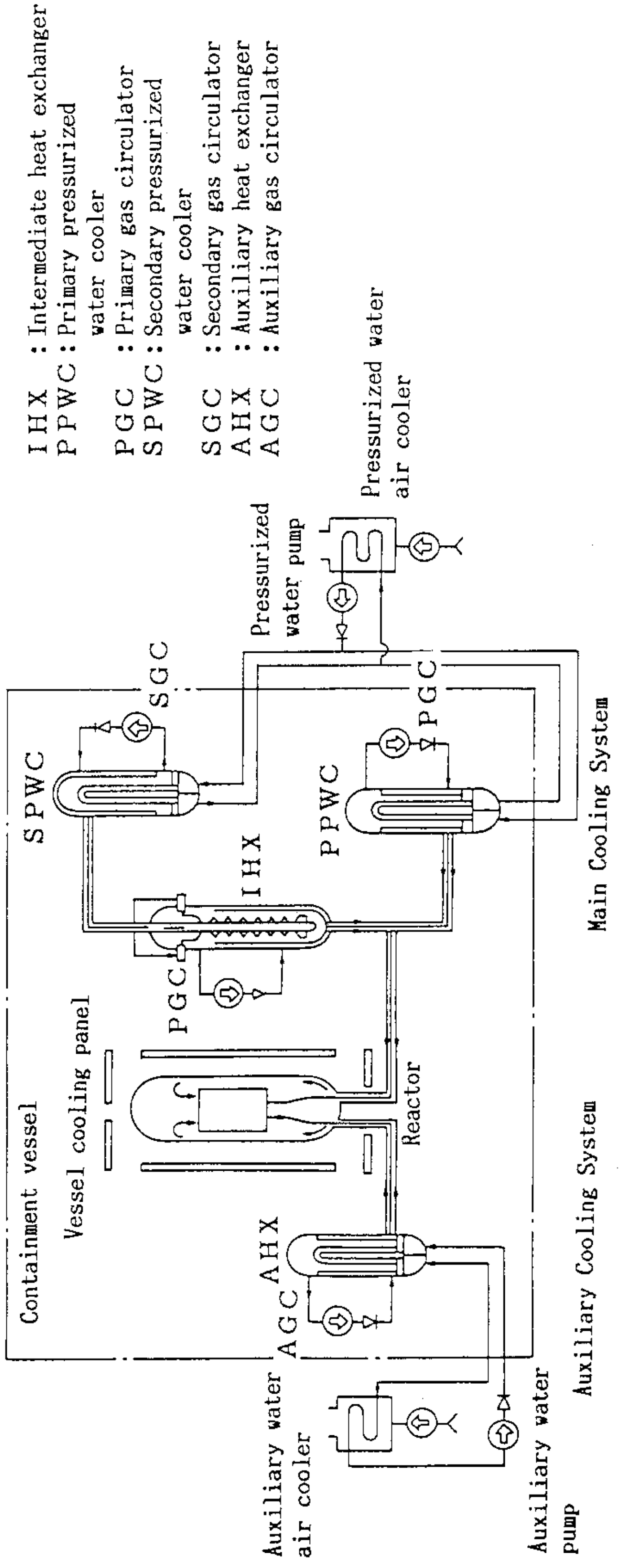


Fig. 4 Cooling system

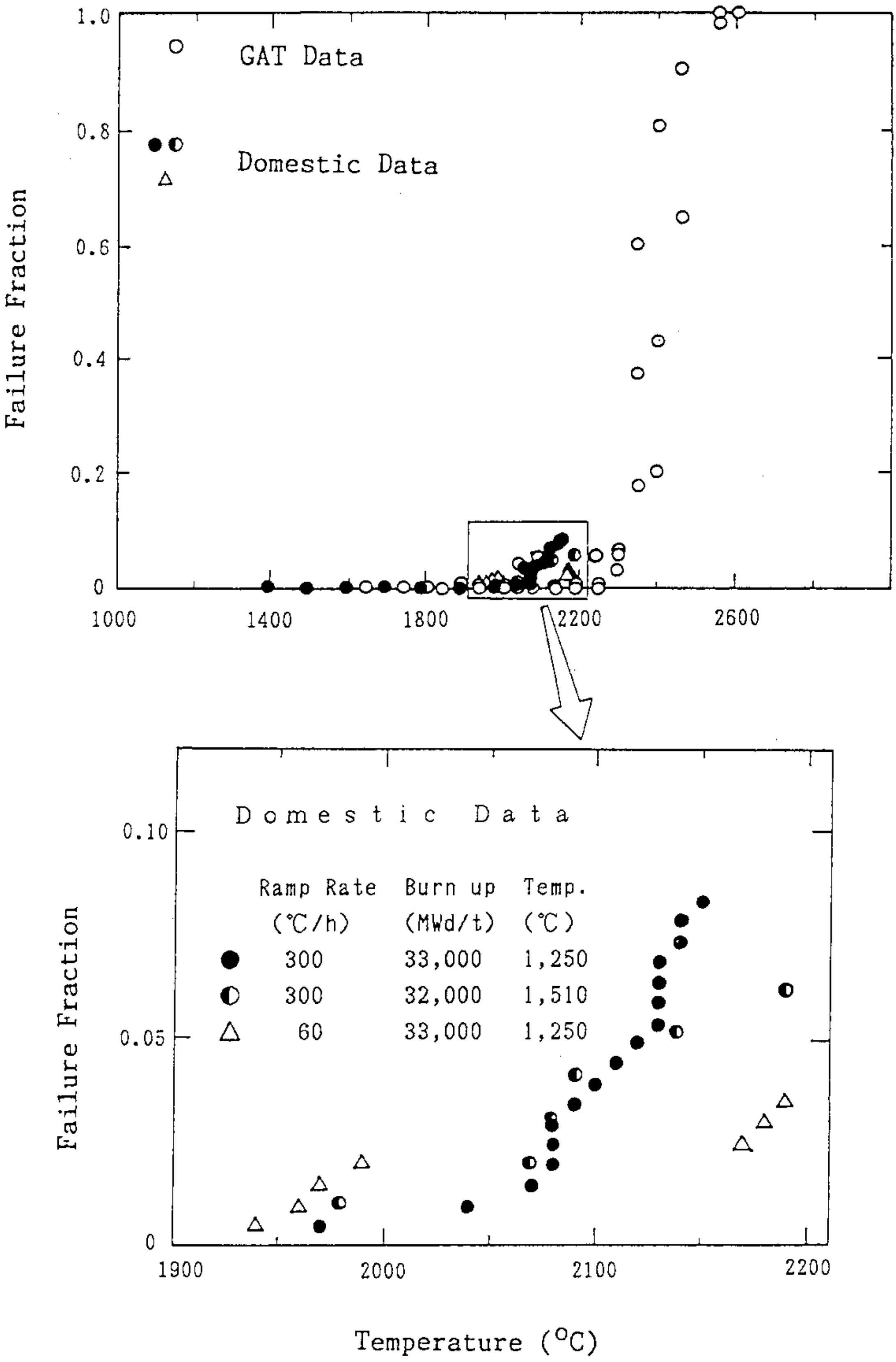


Fig.5 Fuel Failure Behavior under Abnormal Conditions.

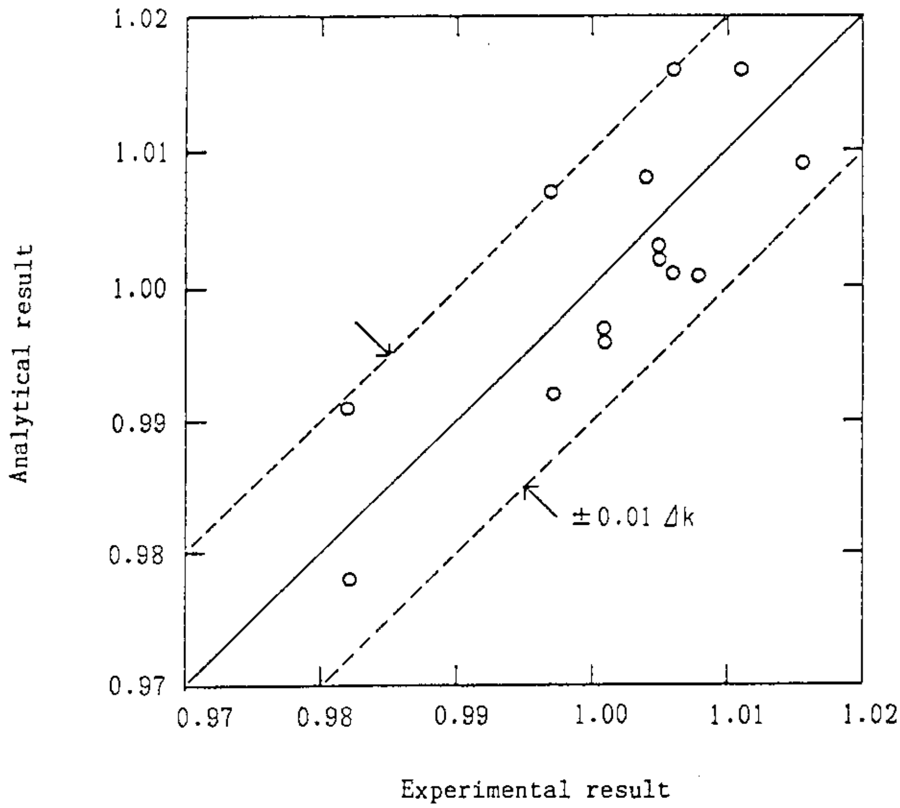


Fig.6 Comparison of the multiplication factors obtained by VHTRC experiment with the analytical results

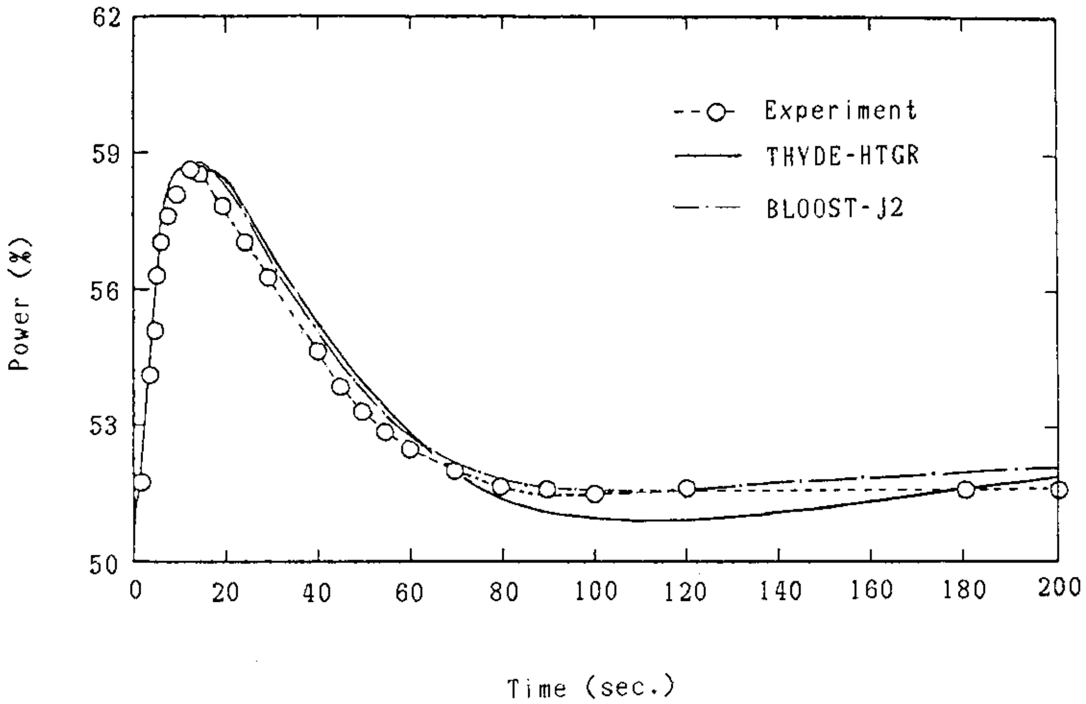


Fig.7 Comparison of core dynamic results between control rod withdrawal experiment of FORT Saint Vrain Reactor with 50% power and the analysis of BLOOST-J2 and THYDE-HTGR.

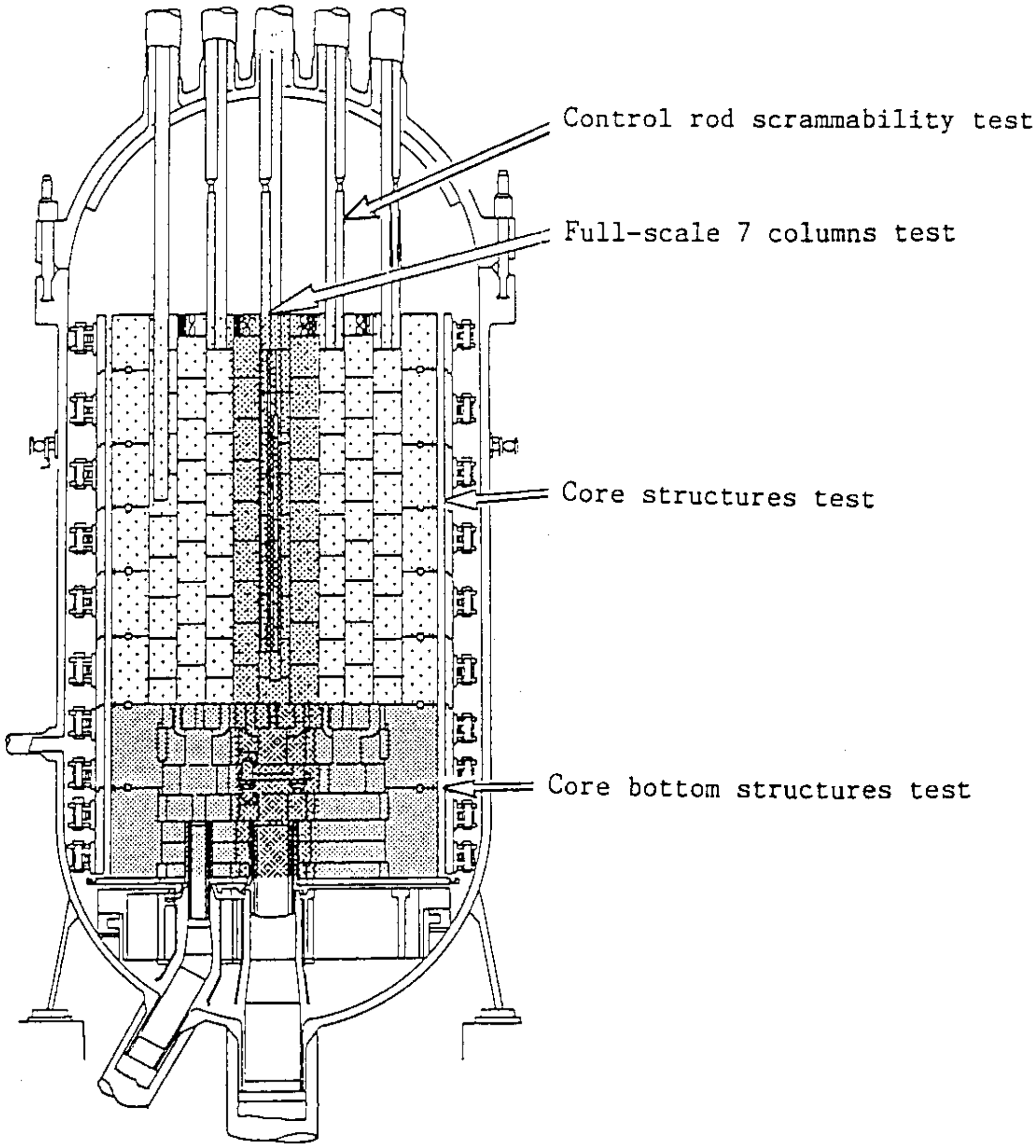


Fig.8(a) Aseismic test items

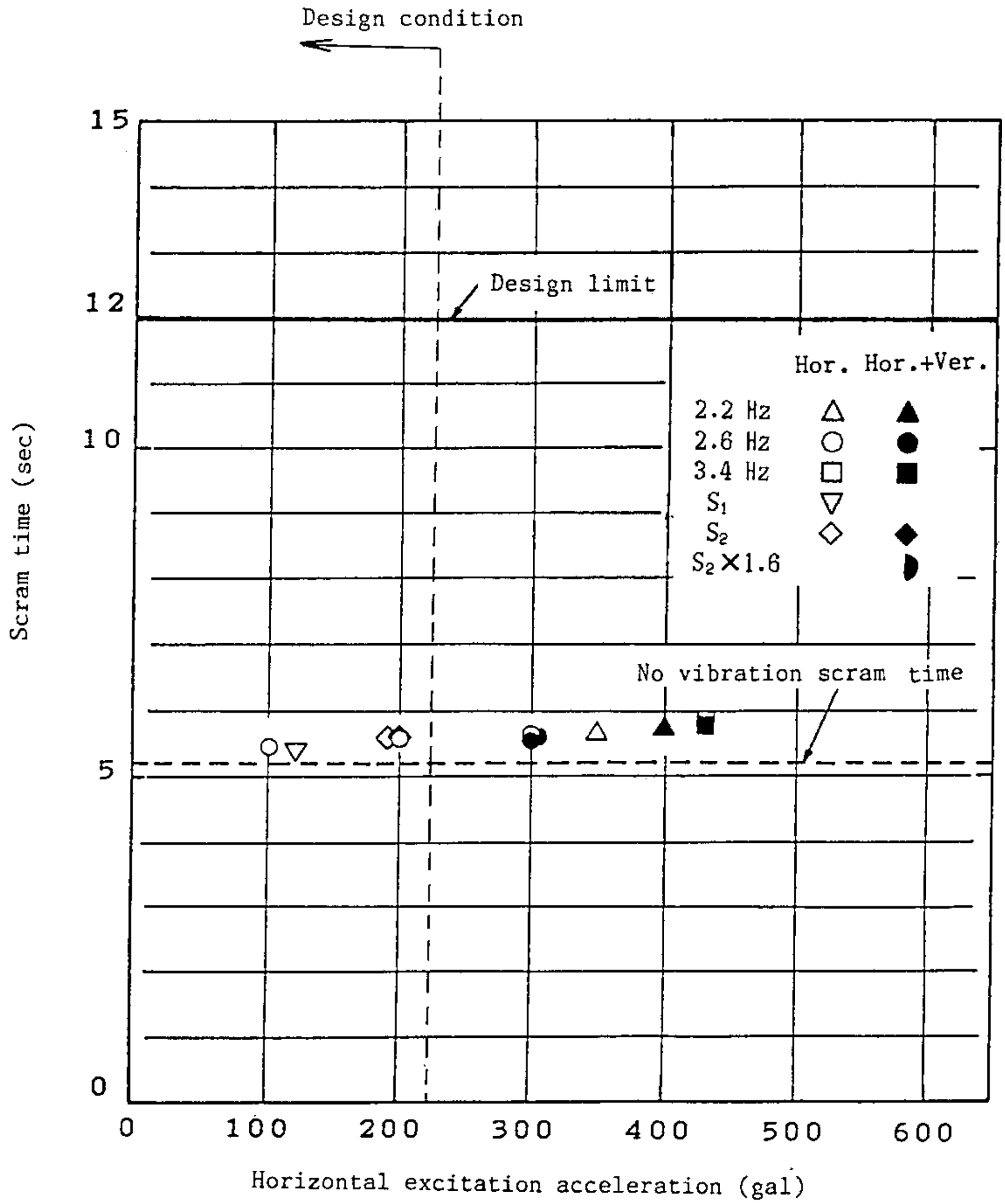
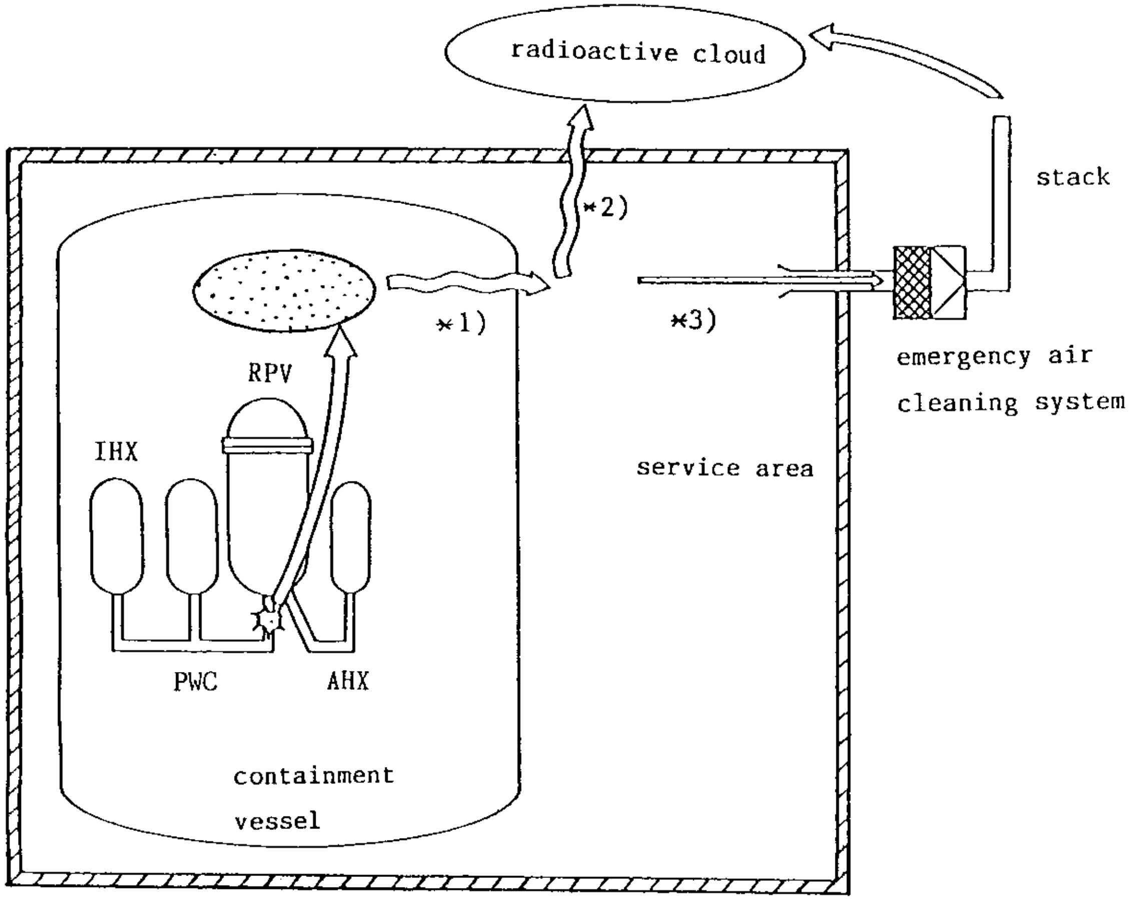


Fig.8(b) Control rod scrammability test results



*1) Release from the containment vessel

*2) Direct ground release from the
service area

*3) Stack release from the service area

⇦: fission product release path
⇜: leak

Fig. 9 Fission Product Release Path

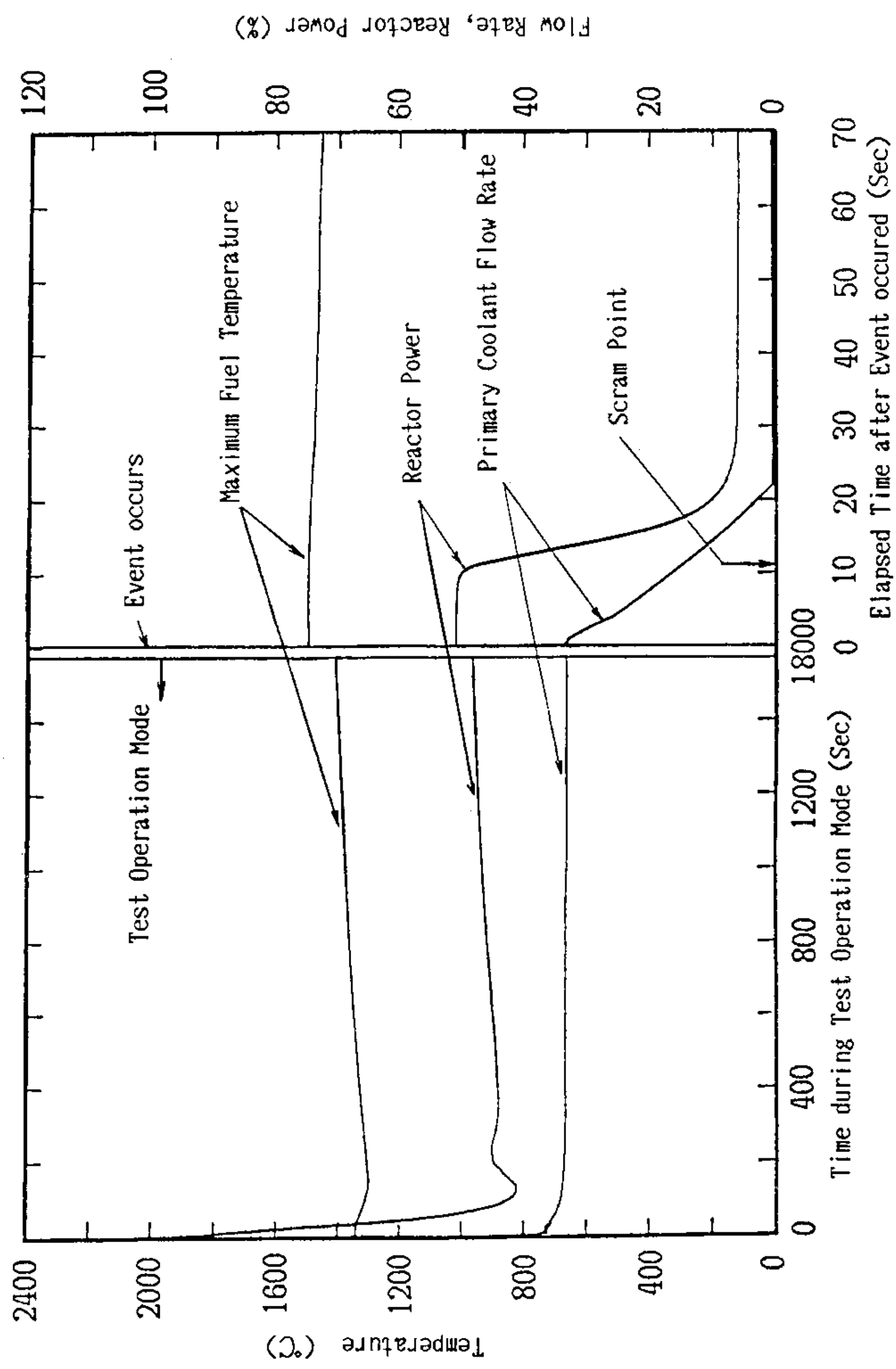


Fig.10 Analytical results of reactor transients when working circulator accidentally stops during test in which two out of three circulators are stopped.

3.2 LICENSING OVERVIEW OF THE MODULAR HIGH TEMPERATURE GAS-COOLED REACTOR (MHTGR)

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1. ABSTRACT

The MHTGR is an advanced reactor concept being developed under a cooperative program involving the U.S. Government, the nuclear industry, and the utilities. The design utilizes the basic HTGR features of ceramic fuel, helium coolant, and a graphite moderator. However, the specific size and configuration are selected to utilize the inherent characteristics of these materials to develop passive safety features that provide a significantly higher margin of safety than current generation reactors. Results to date indicate that the design will meet the U.S. Environmental Protection Agency's Protective Action Guidelines for ionizing radiation at the site boundary, hence precluding the need for sheltering or evacuation of the public during any licensing basis event. This safe behavior is not dependent upon operator action and is insensitive to operator error.

This report discusses the MHTGR Licensing Plan. A discussion of the NRC preapplication phase review of the MHTGR is presented including a summary of the safety response to events challenging the functions relied on to retain radionuclides within the coated fuel particles. The regulatory interaction process related to the application for Preliminary and Final Design Approvals and a standard design certification is also presented.

2. INTRODUCTION

In 1985 the U.S. DOE MHTGR Program submitted a Licensing Plan (Reference 1) to the Nuclear Regulatory Commission that proposed licensing activities prior to submittal of an application. In 1986 the NRC issued the Advanced Reactor Policy (Reference 2) which encouraged the earliest possible interaction between the advanced reactor developers and the NRC and its staff. Both groups sought preapplication interactions to provide this early communication in the design and licensing process. This process, wherein the regulator worked with the designer to develop regulatory criteria and the designer worked with the regulator to set the conceptual design configuration, is in contrast to the previous procedure in which a complete preliminary design was presented to the regulator for approval. This paper describes the process for preapplication review by NRC, the results to date of the NRC review, and the process for obtaining Preliminary and Final Design Approvals and a standard design certification.

3. BACKGROUND

While the MHTGR utilizes basic features and inherent characteristics common to all HTGRs, the fundamental difference of the modular HTGR's design is that its size and configuration have been specifically selected to passively remove core heat thereby retaining radionuclides within the ceramic coated fuel particles. This is best illustrated in Figure 1 which shows chronologically the core sizes and geometries of the U.S. HTGRs and the corresponding maximum accident core temperature under conditions of loss of helium pressure and flow. Up until the development of the MHTGR, all previous designs sustained damage to a fraction of the silicon carbide (SiC) coated fuel. Only with the slender, annular core of 350 MW thermal rating does the geometry and size assure that sufficient heat could be removed passively by conduction and radiation to maintain fuel integrity. Results to date indicate that the design meets the U.S. Environmental Protection Agency's Protective Action Guidelines (Reference 3) for ionizing radiation at the site boundary, hence precluding the need for sheltering or evacuation of the public during any licensing basis event. This safe behavior is not dependent on operator action and is insensitive to operator error.

Since the MHTGR's passive approach to radionuclide control is fundamentally different from existing reactors in the U.S., elements of the Licensing Plan cover the development of regulatory criteria. While the manner in which the criteria are derived is generic, the actual criteria are specific to the MHTGR.

4. LICENSING PLAN

The MHTGR Licensing Plan identifies the licensing-related activities, administrative process, organizational responsibilities, and schedule necessary to support U.S. Nuclear Regulatory Commission (NRC) reviews, approvals, and certification of the Standard MHTGR. The objective of the Plan is to assure that necessary licensing activities are identified, planned, and executed sufficiently for the NRC to issue a Preliminary and a Final Design Approval and a standard design certification for the Standard MHTGR.

For planning purposes, all licensing activities within the scope of the Plan are scheduled within one of two periods, namely the Preapplication and Application periods. Preapplication period licensing activities started in 1985, and have as an objective the issuance of an NRC Safety Evaluation Report on the conceptual design of the MHTGR. Application period activities started in 1989 and have as an objective the issuance by the NRC of Preliminary and Final Design Approvals and a standard design certification for the Standard MHTGR design.

Figure 2 displays the overall licensing logic for the Licensing Plan and identifies interfaces to the DOE-MHTGR Program schedule. The top two bars in the figure display the major elements of this Plan. The third bar displays relevant program milestones.

4.1 Preapplication Activity

Extensive interactions have taken place during the past five years with the NRC, its staff, and the Advisory Committee on Reactor Safeguards (ACRS). During the Preapplication period, licensing submittals were made to the NRC for their review, including the Licensing Plan (Ref. 1), Top-Level Regulatory Criteria (Ref. 4), a Regulatory Technology Development Plan (Ref. 5), a 2-Volume Probabilistic Risk Assessment (Ref. 6), a 5-Volume PSID (Ref. 7), and an Emergency Planning Bases Report (Ref. 8). The NRC's Advanced Reactor Group has performed a technical review of the Preapplication submittal with input from their contractors and consultants, including Oak Ridge and Brookhaven National Laboratories.

Briefings have been held to familiarize NRC staff, the ACRS and various of its subcommittees with the MHTGR design and technology. From time to time the Commissioners have been collectively and individually briefed on the status of the interactions.

The formal review of the Preapplication submittals by NRC and the ACRS has been completed. The NRC has drafted the Safety Evaluation Report (SER) on the MHTGR design. The draft SER provides initial guidance to the designers for the preliminary design. The draft SER reviews the design with an emphasis on the equipment relied on to meet regulatory criteria. Independent contractor evaluations of the response of the MHTGR to off-normal events are included.

The draft SER concludes that the MHTGR has the potential to have a high level of safety, is expected to exceed the safety level of current LWRs, and has the potential to meet or exceed the NRC Safety Goals. Particular aspects of the MHTGR design and licensing approach have been noted by NRC staff in the draft SER to require policy guidance from the NRC Commissioners. These aspects include the absence of traditional pressure-retaining containment structure, the use of a mechanistic source term to determine site suitability, accident selection, and the absence of the need for an offsite plan for evacuation or sheltering of the general public.

In order to provide additional information to the NRC on the first of these aspects, i.e., the traditional containment structure, a containment study (Ref. 9) was submitted to the NRC.

This study included an assessment of the impact of various containment system alternatives on plant safety and cost, as well as, on the MHTGR approach to decay heat removal. The containment study showed that the containment system and the defense in depth provided by the commercial MHTGR design mitigate consequences of accidents sufficiently to meet all the safety requirements with a large margin as shown in Figure 3. The study confirmed that the commercial MHTGR possesses sufficient and assured decay heat removal capability. The study also concluded that the containment systems for the commercial MHTGR and the MHTGR New Production Reactor will meet safety requirements, are appropriate for their missions, and are not contradictory design selections.

The issuance of the final SER is anticipated soon, and will influence the continued development of the MHTGR design, as well as the private initiative to market the MHTGR. The preapplication interactions with the NRC have been very constructive and valuable to the MHTGR program. The feedback has enabled the preliminary design and Application activity to get underway with a better understanding of the regulatory requirements for advanced reactors and of the specific areas of interest to the regulator.

4.2 Application Activity

The objective of the Application activity is to prepare and submit to NRC formal licensing documents describing the Standard MHTGR and its safety features and support NRC review of the documents sufficiently for NRC to issue a Preliminary Design Approach (PDA), a Final Design Approval (FDA) and a standard design certification. The temporal scope of the Application activity extends from 1989 through the receipt of Preliminary and Final Design Approvals and standard design certification. The Application activity will take place in two parts: 1) Activities prior to submittal of an application for a PDA, and 2) subsequent activities.

4.2.1 Application Activity Prior to Application for a PDA

Activities prior to application for a PDA have as their principal objective, resolution of issues documented in the Preapplication activity stage Safety Evaluation Report (SER). The resolution of these issues will be accomplished by interactions between the DOE-MHTGR Program and NRC in two areas. These interactions will keep the NRC abreast of developments in the DOE-MHTGR Program, enable the early resolution of Preapplication activity stage SER issues, and increase the effectiveness and efficiency of NRC review of the MHTGR application for a PDA.

The first area of interaction is directed to resolution of Preapplication activity stage SER issues related to the licensing approach and regulatory criteria. The licensing approach and regulatory criteria interactions will be based on an application pre-submittal of licensing related sections of PSSAR Chapters 1 and 3. This submittal is to be reviewed by NRC and, after interactions to resolve comments, NRC will be asked to issue a Safety Evaluation Report or a Licensing Review Basis document on the licensing approach and regulatory criteria. The topics to be addressed in the application pre- submittal include:

- Assessment of NRC Policies, Regulations and Regulatory Guidance
- Standardization Philosophy, including scope and level of detail of design, approach to Design Approval/Certification, and relevance of an operating plant.
- Criteria for Public Consequence and Risk
- Principal Design Criteria
- Approach to Offsite Emergency Planning
- Licensing Bases, including licensing basis events, siting source term, classification of SSCs, defense in depth and safety evaluation of design relative to the adequacy of controlling release of radionuclides.

The second area of interaction between the DOE-MHTGR Program and NRC prior to an application for a PDA is directed to the early resolution of Preapplication activity stage SER issues relative to the MHTGR design, codes and models, and regulatory technology development. For these interactions, a series of meetings will be held with NRC to discuss SER issues and DOE-MHTGR Program resolution of these issues. The objective of these meetings is to reach mutual agreement on resolution of SER issues. These agreements will be documented formally in the PSSAR to be submitted in support of a PDA application.

4.2.2 <u>Application Activity Subsequent to Application for a PDA</u>

The procedural and administrative approach of the application phase will be guided by existing NRC policies and regulations. Specifically, the review procedures for standard designs found in 10CFR52, Appendix 0 will be used to obtain a Preliminary and a Final Design Approval prior to filing for a standard design certification. This stepwise approach is in accordance with the three primary objectives of the Advanced Reactor Policy statement (Ref. 2). These objectives are: 1) to encourage early interactions, 2) to provide interested parties with the Commission's views on desired characteristics of advanced reactor designs and 3) permit the Commission to issue timely comment on advanced reactor designs. For design certification, it is planned to use the requirements and procedures of 10CFR52, Subpart B, Standard Design Certifications. On these bases the procedural steps discussed below will be followed:

1. <u>Submit PSSAR</u> - A Preliminary Standard Safety Analysis Report (PSSAR) will be prepared. The PSSAR will incorporate resolution of PSID stage SER issues, as discussed above. The PSSAR, along with a Regulatory Technology Development Plan and a Probabilistic Risk Assessment based on preliminary design will form the basis of an application for a Preliminary Design Approval under 10CFR52, Appendix 0.

2. Review by NRC - Subsequent to docketing by the NRC, meetings will be held with the NRC to discuss further details and/or clarification of the design and its assessment. Based on these meetings NRC will issue comments. Responses to these comments will be submitted to the NRC as amendments to the PSSAR.

3. Issue Safety Evaluation Report - The NRC's review and technical acceptance of the design will be documented in its Safety Evaluation Report (SER). Upon resolution of major open issues, the final SER, which will form the technical base for the NRC staff's approval of the design, will be issued.

4. Review by ACRS - The PDA application will be referred to the ACRS for a review and issuance of a letter report to the Commission on the acceptability of the Standard MHTGR.

5. Issue Preliminary Design Approval - Initial NRC staff approval of a standard MHTGR design will be granted in the form of a Preliminary Design Approval.

6. Submit FSSAR - Following the review of the Standard MHTGR preliminary design, a similar process will occur when near-final design information becomes available. Prior to application for a Final Design Approval, DOE-MHTGR Program interactions with NRC will seek early resolution of PDA stage SER issues. A Final Standard Safety Analysis Report (FSSAR) will be prepared similar to the PSSAR, but containing more detailed final design and analysis information, and PDA stage SER issues resolution. The FSSAR will support an application for a Final Design Approval under 10CFR52, Appendix 0.

7. Issue Final Design Approval
   An NRC review process similar to that associated with the PSSAR (steps 2-4 above) will be conducted and conclude with the NRC staff's issuance of a Final Design Approval.

8. Issue Standard Design Certification
   The standard design certification process provided by 10CFR52, Subpart B, will be pursued, including the opportunity for informal hearings before an Atomic Safety and Licensing Board, leading to NRC issuance of a standard design certification for the Standard MHTGR.

The PSSAR and FSSAR will define the scope of design for which design approvals and certification are being sought, and provide a description and evaluation of the design including the characteristics of a generic plant site. The PSSAR and FSSAR will also provide a representative design for that portion of the MHTGR which is not intended to be part of the approved/certified scope of design and describe and evaluate the interfaces between the two.

Compliance with applicable regulatory criteria will be demonstrated in the PSSAR and FSSAR. In support of the preparation of the PSSAR and FSSAR, selected safety assessments will be conducted and documented to support compliance of the MHTGR design with applicable regulatory criteria. Additional safety assessments will be prepared as necessary to supplement the PSSAR and FSSAR, for example, a PRA will be performed and submitted to the NRC at the time of the PSSAR and FSSAR submittal and the Regulatory Technology Development Plan will be updated to reflect its current status.

The application for a standard design certification will include the experience and results of the Standard MHTGR licensing review through FDA and will also reflect experience gained from start-up and operation of an MHTGR plant.

5. REFERENCES

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3. "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," Environmental Protection Agency Report EPA-520/1-75-001, September 1975 (revised June 1980).

4. "Top-Level Regulatory Criteria for the Standard HTGR," DOE Report DOE-HTGR-85002, Rev. 2, October 1986.

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This report does not include Fig. 1~3.

3.3 LICENSING SAFETY ISSUES AND RESULTS OF HTR-MODULE SAFETY CONCEPT REVIEW BY INDEPENDENT EXPERTS

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ABSTRACT

Humanity is facing great challenges. On the one hand, it is imperative to ward off the impeding menace of an environmental catastrophe, on the other it is necessary to overcome poverty in the third world and to supply the increasing world population, particularly in third countries, with a sufficient amount of energy while at the same time preserving the natural environment.

One of the most convincing solutions to the future energy supply is nuclear energy. However, due to the accidents in Three-Mile Island and Tchernobyl, public acceptance of nuclear energy has suffered a severe blow.

The nuclear industry has taken note of the public concern and started acting accordingly by striving to develop a new generation of nuclear reactors featuring a high degree of passive safety properties.

As early as 1979, the development of the modular HTR was launched in the Federal Republic of Germany, taking exacting standards and layout criteria for a basis. The supplying industry soon realized that her own judgement was not sufficient and that the evaluation of the newly developed modular concept by neutral authorities and experts was absolutely indispensable in order to ensure the basis of this development.

Thus, the German Board of Experts on Environmental Questions critically reviewed the new modular concept as early as 1981 in a special expertise submitted to the Federal Government and gave a basically positive statement in this matter. Further evaluations of the modular concept were conducted in 1981/82 by a subcommittee of the Reactor Committee, in 1982 by the Rhineland Technical Control Board, and in 1984/85 by a counselling group established by the Federal Minister of the Interior.

Encouraged by the positive evaluations, a safety report was developed in the course of 1985 and 1986. In 1987, it was submitted and

a review of the safety concept by independent experts was initiated.

1. Target of the HTR-Module Safety Concept Review

The target of the development of all advanced reactors is to prevent the release of appreciable amounts of radioactivity into the environment and thus to prevent the evacuation of the surrounding population respt. the contamination of large areas of the surroundings. This is achieved by retaining the fission products within the fuel element, the primary circuit, the containment resp. in the reactor building. Compared to the concepts of other advanced reactors - e.g. the advanced light-water reactor or the sodium-cooled breeder - the HTR, while pursuing the same targets, is characterized by a number of special properties and features. For example: whereas the probability of a core-meltdown of a light-water reactor has been reduced to a minimum due to passive and/or improved fast-starting active measures, and whereas the effects of a core-meltdown are here controlled by purposive, but active measures, the HTR-module can rely upon the properties and features of a graphite reactor based solely on natural physics.

These are:

- low power density in combination with a high heat capacity
- high heat resistance of the fuel particles and the graphite
- maximum fuel temperature achieved remains far below the melting point and also below the temperature at which damage to the fuel particles sets in
- the activity in the primary circuit during operation is so low that in case of a relief from pressure due to pipe fraction a direct unfiltrated emission via the stack is possible.
- the retainability of the fuel particles is such that a pressure-proof containment is unnecessary even in the incident of a core heat-up; filtrated emission via stack is possible.
- even in superhypothecial incidents, decay heat can be removed from the primary system of the HTR-module to the environment without active measures in the primary system and this based only on thermal conduction, convection and thermal radiation.
- long period of time available for initiating countermeasures in the case of incidents
- low dose rates for personnel during operation, maintenance and repairs

- simplified auxiliaries and emergency power supply
- conventional water/steam cycle.

Most of the above mentioned properties of an HTR have alreradу been confirmed in the licensing procedures of the AVR and THTR-300 reactors as well as in the preconceptual evaluation of the HTR-500 and the HTR-module reactor concepts. Moreover, they have been verified/ validated in particular by the operation of the AVR and the THTR-300 and the goal-oriented test operation of the AVR.

The objective of the safety concept evaluation of the HTR-module by independent experts was to critically review the individual safety properties as well as the integral concept of the HTR-module as a whole. This was to confirm that the HTR-module meets the requirements of an advanced future reactor concept under normal and abnormal operating conditions as well as in superhypothecial incidents, and that therefore unrestricted licensability prevails. In addition, the review was to create a solid and legal basis for the continuation of the detailed design.

2. HTR-Module Plant and Safety Analysis Report

2.1 Plant Description

Figure 1 shows a longitudinal view of the HTR-Module with steam generator.

Figure 2 shows a longitudinal view of the Reactor building and the arrangement of the HTR-Module and Steam generator.

Figure 3 shows a cross section of the reactor building with HTR-Module, Steam Generator, and Primary Cell.

Figure 4 shows the site plan for the arrangement of the overall HTR-Module Power Plant.

The HTR Module power plant underlying the safety concept review is a power plant for the cogeneration of electricity and process steam (Fig. 5). The process steam can be used for a wide variety of applications in the chemical or oil industries.

The place of the fossil-fired heat sources of conventional power plants is taken in the reference HTR Module plant by two nuclear steam supply systems (modular units). Each modular unit comprises one high-temperature reactor, one steam generator and one primary gas blower.

The heat generated by nuclear fission in the high-temperature reactor is transported from the reactor via the coaxial duct to the steam generator by the primary coolant helium (Fig. 1), which is circulated by the primary gas blower. In the steam generator, the heat is transferred to the water/steam system which is designed and operated as a purely non-nuclear system.

Auxiliary and supporting systems connected to the primary system are provided for operation of the reactor; furthermore, safety of the reactor is ensured by systems which fulfil the task of keeping loadings on components and structures within acceptable limits under accident conditions and which minimize the impact of accidents on the operating personnel and the environment.

The reactor auxiliary systems are installed in the reactor building and the reactor auxiliary building (Figs. 2 and 3).

The cooling requirements in the power plants are met by closed water systems which remove the rejected heat to service water systems.

Operation of the power plant is controlled and monitored from the central control room. An emergency control room - with independent separate and deverse accesses - is provided in the lower part of the reactor building.

Normal operation is largely automated by means of open and closed-loop controls which correct minor deviations from required setpoints. In the event of major deviations, automatic operational limiting controls restore the plant to normal conditions. If trip limits set in the reactor protection system are reached, the necessary safety-related countermeasures are automatically initiated.

2.2 Safety Analysis Report

For light water reactors the contents of Safety Analysis Reports are governed by a Bulletin of the German Federal Ministry for the Interior (BMI) with the title "Contents Checklist and Format for a Standard Safety Analysis Report for Pressurized Water Reactor and Boiling Water Reactor Nuclear Power Plants". For the HTR module this "Contents Checklist" was modified to reflect

- the site-independent Safety Concept Review
- the application of a preliminary HTR-ruling and
- the new HTR-technology.

Figure 6 shows the contents of the HTR-Module Safety Analysis Report.

Section 1 of the Safety Analysis Report contains postulated site conditions to the extent necessary for reviewing the concept and for arriving at an expressive preliminary overall judgement. The postulates made for this purpose are representative of the largest number of potential sites in FRG.

Section 2 contains a list of general design features of the HTR Module power plant on which the application for a preliminary ruling on the concept is based (Fig. 7). Sections 3 to 9 of the Safety Analysis Report contain descriptions of the power plant and operation thereof in accordance with the requirements stated in the above mentioned "Contents Checklist" by the BMI, with modifications specific to the HTR Module. For purposes such as providing a basis for a preliminary overall assessment, these sections give more details on the concept described in Section 2 for postulated, exemplary conditions of service.

Since the HTR Module has been designed as a universally usable energy source, detailing of the concept for actual conditions of service would in essence only have a bearing on:

- overall layout and non-nuclear structures
- steam power conversion system
- electrical systems (grid connection).

On the contrary, actual conditions of service and site conditions would hardly affect construction of the reactor systems such as

- reactor core
- nuclear steam supply system
- reactor auxiliary systems

and, by definition, have no influence on the general design features described in Section 2.

Section 10, titled "Guidelines and Technical Rules", gives a survey of the applications or adaptation of the Bulletins by the BMI and of the KTA Safety Standards to the HTR Module.

3. Rules of Review

The independent experts for the HTR Module Safety Concept review were

- the "Technischer Uberwachungsverein" TUV (Technical Control Board), Hannover (main contractor) and Cologne (subcontractor)
- the "Gesellschaft fur Reaktorsicherheit" GRS (Reactor Safety Association), Cologne/Munich
- the "Civil Engineering Institute" (IBMB), University of Brunswick
- the "Institute of Reactor Safety" (ISF) of KFA, Julich

- The "Reaktorsicherheitskomitee" RSK (Reactor Safety Committee), Bonn (an advising committee to the minister of environmental protection BMU).

  The main results of the review are indicatd subsequently:

3.1 Fire Protection Review

In addition to the fire protection concept described in Section 2.6 of the Safety Analysis Report, a wide range of in part very detailed additional documents were submitted to the experts, in this case the Civil Engineering Institute (IBMB) of the University of Brunswick.

The expertise completed in February 1989 approved the fire protection concept. The few additional requirements and recommendations can be fulfilled easily and have no effect on plant technology and safety.

3.2 Plant Security Concept Review

The plant safety concept - the entirety of provisions for protection of the plant and operation thereof against sabotage and other external interferences - was reviewed by the Reactor Safety Association (GRS).

The expertise completed in March 1989 approves the concept and concludes that the inherent features of the HTR Module warrant less stringent security precautions than those to be taken for LWRs.

3.3 Safety Concept Veview by TÜV's

The main document for the safety review by the TÜV's was the Safety Analysis Report, above all Section 2 which describes the main features of the HTR Module relevant to the concept. The findings of the independent expert are listed subsequently following the same breakdown as Section 2.

Section 2.2: Characteristic Safety Features

- The engineering configuration and nuclear design of the HTR Module is such that, even in the event of postulated failure of all active shutdown and decay heat removal systems, the fuel temperature stabilizes at 1620°C. No appreciable release of radioactivity from the fuel assemblies occurs below this temperature.
- Active decay heat removal systems which limit the loadings on components and structures surrounding the core can fail for several hours without the allowable limits being exceeded.

Assessment in expertise: approved

Section 2.3: Technical Design Features

- Fuel element
  - Coatings (TRISO)
    Enrichment (8.5%)
    1620°C max. temperature due to SiC layer
  - Particle failure curve (manufacturing defects = 6×10^{-5}, irradiation-induced = 2×10^{-4}; accident-induced = 5×10^{-4}

Assessment in expertise: approved

- Reactor core
  - By virtue of selected core design, fuel temperature stay below 1620°C under all accident conditions, even on loss of active decay heat removal
  - Due to uranium content of 7 g per fuel element, the reactivity excursion on water inleakage is less than on inadvertant withdrawal of all reflector rods
  - Design for unrestricted load cycling between 50 and 100%

Assessment in expertise: approved; restriction on part-load operation below 50% during the first core start-up phase (because no analyses submitted for this case): limitation of absorber ball level in storage vessels

- Shutdown systems
  - Hot shutdown by absorbers in reflector holes
  - Cold shutdown by 6 rods and 18 absorber ball units
  - Location of rod drive mechanisms inside reactor pressure vessel
  - Location of all absorber ball unit components needed for shutdown inside reactor pressure vessel

Assessment in expertise: design and configuration approved. Reactivity balances for equilibrium core approved but because those for first core start-up phase up to several months show relatively small margins, reactor power might be below 200MW in the first few weeks.

- Pressure Vessel Unit
  - Consists of reactor pressure vessel, gas duct pressure vessel and steam generator pressure vessel inclusive of valve banks on reactor pressure vessel and nozzles of steam generator pressure vessel
  - Offset configuration, thus limiting natural circulation in the primary system
  - leak before break, assured safety for entire pressure vessel unit

Assessment in expertise: approved after discussion of dissimilar-metal weld and change of material for main steam nozzle. New requirement: preservice pressure test to include nozzles at steam generator pressure vessel.

• Primary and secondary system isolation
  - Primary system by two valves in each line of which only one operated by reactor protection system (fail-safe)
  - Secondary system by two valves in each line (fail-safe) both actuated by reactor protection system whenever reactor is shut down. Consequently, rest of secondary system outside reactor building has no functions relevant to safety.
  - Primary System overpressurization protection: two safety values; secondary system: one safety valve backed up by steam generator relief system

Assessment in expertise: approved

• Confinement envelope
  - Consisting of reactor building and other features (secured sub-atmospheric pressure system, building pressure relief system, heating/venting/air conditioning systems isolation)
  - Normal operation: no filtering
  - At maintenance work: filtering by exhaust air filtering system (aerosols)
  - During major depressurization accident (non-isolable DN 65 line): unfiltered venting through two dampers to vent stack
  - Other depressurization accidents: possibility of filtering by sub-atmospheric pressure system (iodine filter)
  - environmental impact of all accidents far below limits prescribed in the Radiological Protection Ordinance even without active measures taken or filtering as low as reasonably possible; consequently no containment necessary

Assessment in expertise: approved. New requirement: higher grade exhaust air filtering system.

• Decay heat removal
  - Provided by secondary system, cavity coolers, helium purification system
  - On loss of active cooling, decay heat removed from core to cavity coolers solely by thermal conduction, natural convection and radia-

tion

- Secured component cooling system, two trains
- With cavity coolers intact and loss of core cooling, core can heat up for lengthy period of time (15 h) without exceeding design limits for reactor pressure vessel and concrete of reactor cavity being violated
- External supply can be connected to cavity coolers in the event of severe accident conditions

Assessment in expertise: approved (see emergency power supply below)

- Emergency Power Supply
  - Two trains served by two diesel generator sets, started by operational sequencing controls or manually
  - DC buses (e.g. reactor protection system) battery-buffered for two hours
  - Reactor system can sustain loss of power for at least fifteen hours (loss of auxiliary power supply, failure of diesel generator sets) without design limits being violated

Assessment in expertise: approved. Restriction: quality assurance for diesels must be so stringent that the diesel generators can certainly be started within the fifteen-hour period.

- Reactor protection system
  - Few process variables
  - Three protective actions always actuated on shutdown (reflector rod drop, blower trip, steam generator isolation); additionally steam generator pressure relief on tube failure and Primary System isolation on reactor pressure vessel depressurization
  - All actions failsafe
  - Station blackout longer than two hours can be sustained since all protective actions are initiated, plant is transferred to safe condition, reactor protection system has no further tasks to fulfil

Assessment in expertise: approved. Source-range neutron flux instrumentation to be of reactor protection grade.

- Remote shutdown room
  - Located in reactor building (designed for aircraft crash, blast wave); several separate diverse accesses
  - Power supply by diesels in switchgear building
  - On station blackout, single train battery power supply for fifteen

hours, possibility of connecting up to external power supply after that

- Monitoring functions only, except for manual absorber ball shutdown system initiation

Assessment in expertise: approved.

Section 2.4: Nuclear Classification and Quality Requirements

- Definition of classification criteria and establishment of classes for
  - pressure-retaining and activity-carrying systems
  - Heating/Venting/Air conditioning system
  - Hoists and cranes
  - structural steelwork
- Assignment of systems to defined classes
- Identification of quality requirements for the different classes

Assessment in expertise: assignment criteria correctly selected; assignment of systems as correct as possible at the present status. Final assessment of assignment of systems and identification of quality requirements can only be performed during construction licensing procedure.

Section 2.5: Summary of Design Basis Incidents

- Listing of representative accidents in analogy to "Accident Guidelines for Pressurized Water Reactors"

Assessment in expertise: approved. Listing of all design basis incidents is correct.

Section 2.6: Postulates and measures for in-plant events

- Break postulates

  Primary System: one DN 65 connecting line (2D)

  Secondary System: main steam or feedwater line (2D)

  Steam Generator Tubes: one tube (2D)

- Main steam line and steam generator tube rupture not postulated

Assessment in expertise: approved. Requirement: in-service inspection of steam generator tubes

Section 2.7: Postulates and measures for external events

- Building design for earthquake:
  - Reactor building
  - Reactor building annex
  - Switchgear building

  - Reactor auxiliary building;
    only sealed concrete pit and
    main load-bearing structures
- Building design for aircraft crash, blast wave:
  - Reactor building
- System design for earthquake, aircraft crash, blast wave:
  - pressure vessel unit
  - steam generator tubes
  - reactor coolant piping up to and including isolation valves
  - secondary system inside reactor building
  - remote shutdown room
  - components of reactor protection system inside reactor building
  - shutdown systems inside reactor pressure vessel
  - cavity coolers
- System design for earthquake:
  - secured closed cooling system
  - secured service water system
  - reactor protection system
  - emergency power systems

Assessment in expertise: approved.

3.4 Overall Safety Concept Review by Reactor Safety Committee (RSC)

Based on the expertises by the TÜV Hannover and Cologne, by the Civil Engineering Institute, by the Reactor Safety Association (GRS) as well as on additional analyses by the Reactor Safety Committee and by the Institute of Reactor Safety (Research Center Jülich) concerning residual risks, an overall statement on the Safety Concept and on the licensability will be made by the end of February, 1990.

The order for this Overall Safety Concept Review was placed by the Minister of Environmental Protection (Bundesminister fur Umwelt, BMU) in close cooperation with the Minister for Research and Technology (Minister fur Forschung und Technologie, BMFT) to the Reactor Safety Committee in spring 1989.

Based on this order, the safety concept of the HTR-Module power plant was first debated in the subcommittees of the Reactor Safety Committee in 1989. In February 1990[1], the Reactor Safety Committee is

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1) On the occasion of the JAERI HTGR Symposium in March, 1990, the exact wording of this statement can probably be presented.

expected to give a recommendation concerning the following items:

- Nuclear Steam Supply System
  - Primary System
  - Fuel Element
  - Shutdown System and Reactor Control
  - Reactor Protection System
  - Electrical Power Supply
  - Decay Heat Removal System
  - Containment
  - Emergency Control
- Design and Quality Assurance of the pressurized metallic components of the primary system and the secondary circuit
  - Design and Quality Assurance
  - Support of the Reactor Pressure Vessel and the Steam Generator
  - Secondary Circuit
- Control of design basis incidents and events beyond design basis limitations
  - Design basis incidents
  - Events beyond design basis limitations, residual risks
- External Impacts
- Staff's Exposure to Radiation

4. Summary, Next Steps

Even though the final statement on the Safety Concept is not available yet, previous discussions with experts and members of the Reactor Safety Committee allow for the deduction that the Safety Concept of the HTR-Module will presumably be awarded an extraordinarily positive evaluation.

When the detailed expertise on the Safety Concept will have been submitted, it first needs to be analysed. The necessary conclusions concerning the lay-out are yet to be drawn as well as possibly required experimental proof of the data of the analyses and the components/ systems is yet to be shown.

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1) SCHULTEN, R., "High-Temperature gas-cooled reactors", Technical Committee Meeting, IAEA, Julich (Oct. 1986)

2) REUTLER, H., LOHNERT, G.H., "Advantages of going modular in HTRs", Nuclear Engineering and Design, Vol. 78, No. 2 (April 1984)

3) HUBEL, H.J., LOHNERT, G.H., "The safety-related properties of future HTR-Module power plants", Atomkernenergie-Kerntechnik, Vol. 47, No. 3 (Nov. 1985)

4) REUTLER, H., "Konzept des modularen HTR in Zweibehälterbauweise", Atomwirtschaft (Aug./Sept. 1985)

5) RAGOSS, D., HEIT, W., NABILEK, H., SCHENK, W., "Absicherung der brennelementspezifischen Quellterme für die radiologische Auslegung des HTR-Moduls", Jahrestagung Kerntechnik, Travemünde (Mai 1988)

6) BRINKMANN, G., WILL, M., "Concept Licensing Procesure for HTR-Module Nuclear Power Plant", (to be published in "Nuclear Engineering and Design" in 1990)

7) WEISBRODT, I., STEINWARZ, W., KLEIN, W., "Status of HTR-Module plant design", Technical Committee Meeting, IAEA, Jülich (1986)

8) WEISBRODT, I., "Inherent safety design features of the HTR-Module", International Conference on Nuclear Power Performance and Safety, IAEA, Vienna (Sept./Oct. 1987)

9) WEISBRODT, I., "Engineering and Licensing Progress of the HTR-Module", Technical Committee Meeting, GCRA 10th International HTGR Conference, San Diego (Sept. 1988)

10) WEISBRODT, I., "Status of Development of the HTR-Module, IAEA, San Diego (Sept. 1988)

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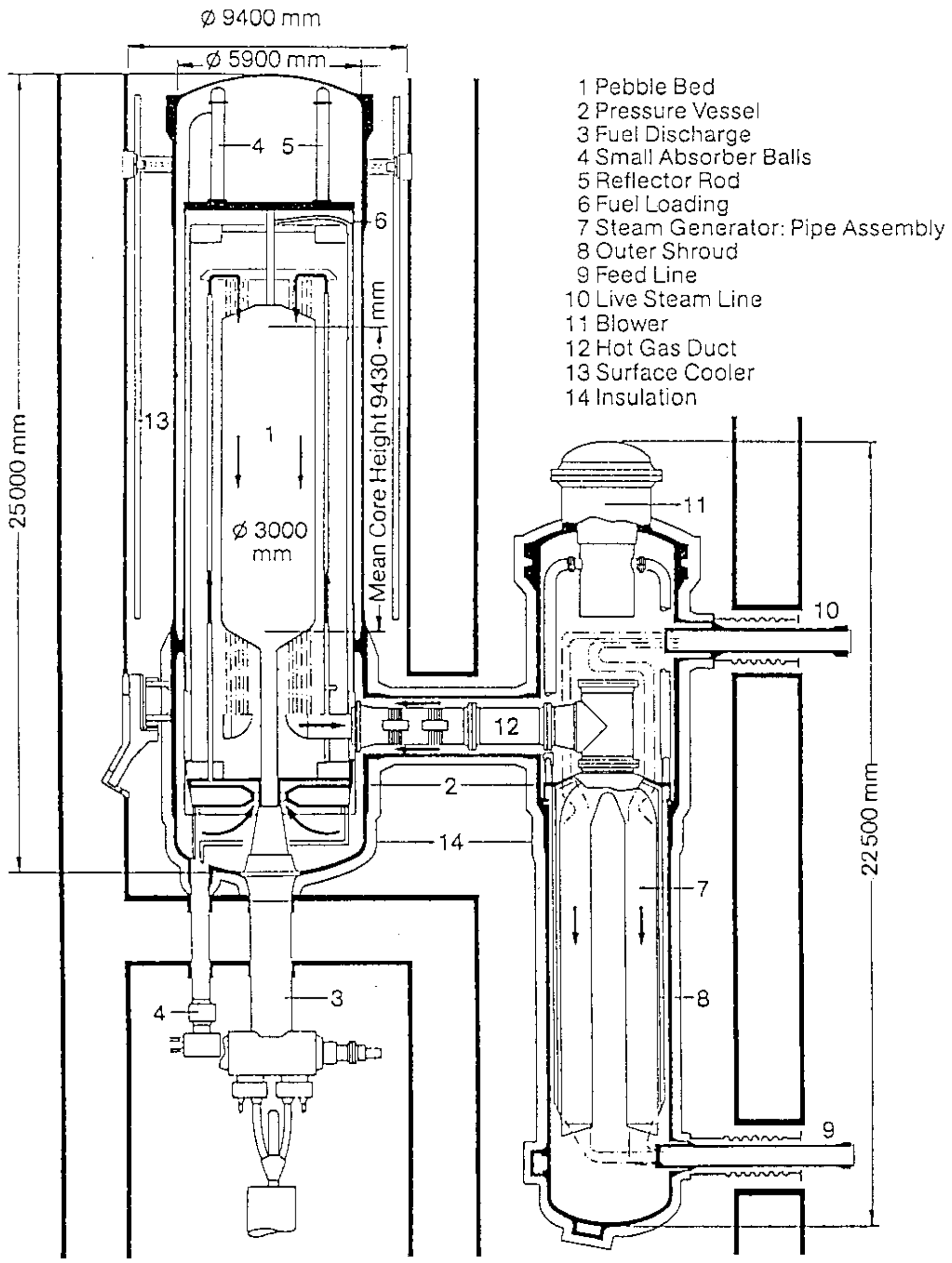

Fig. 1 **Primary Circuit of an HTR-Module**

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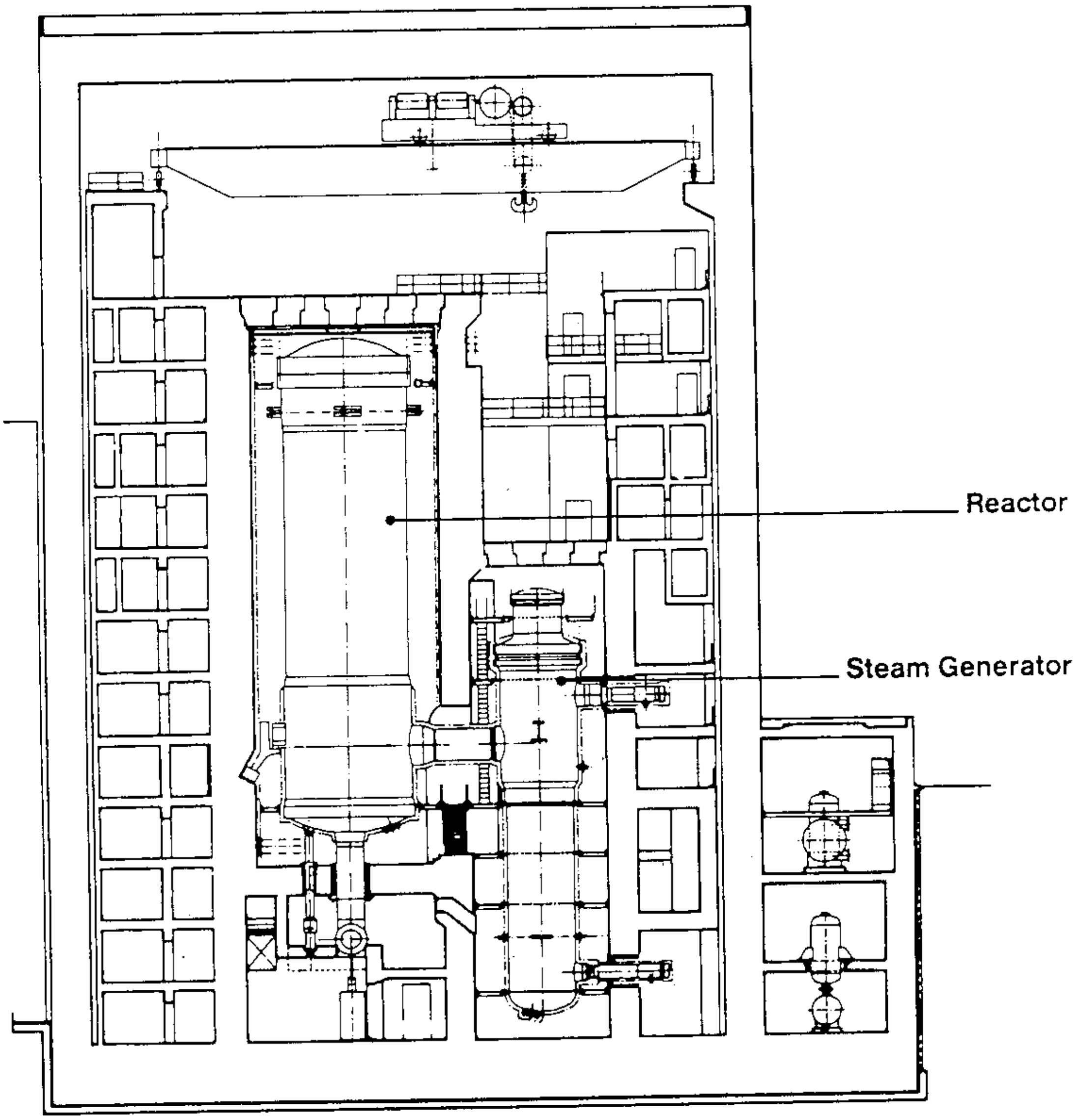

Fig. 2 Section through the Reactor Building

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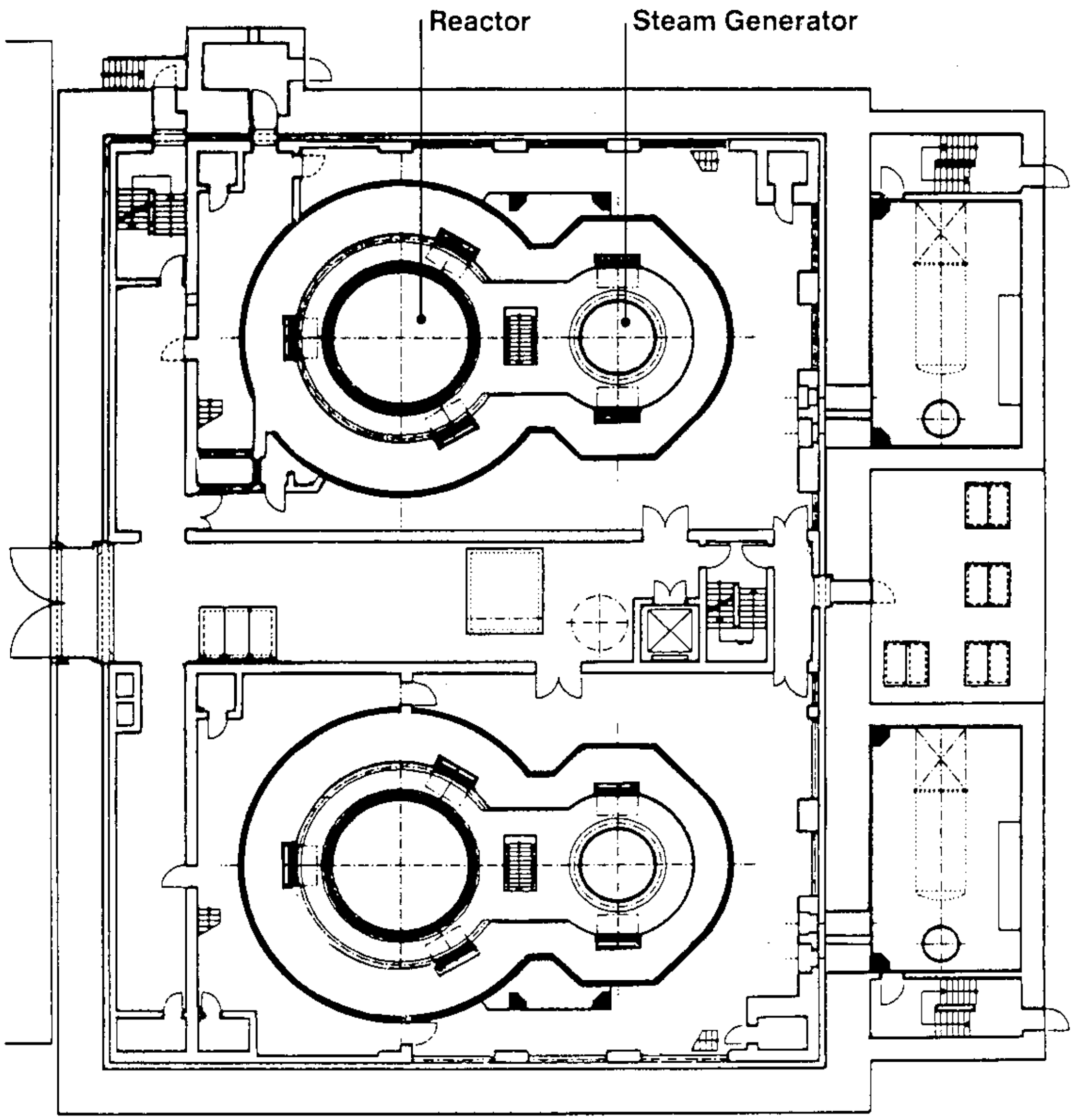

Fig. 3 **Cross Section through Reactor Building**

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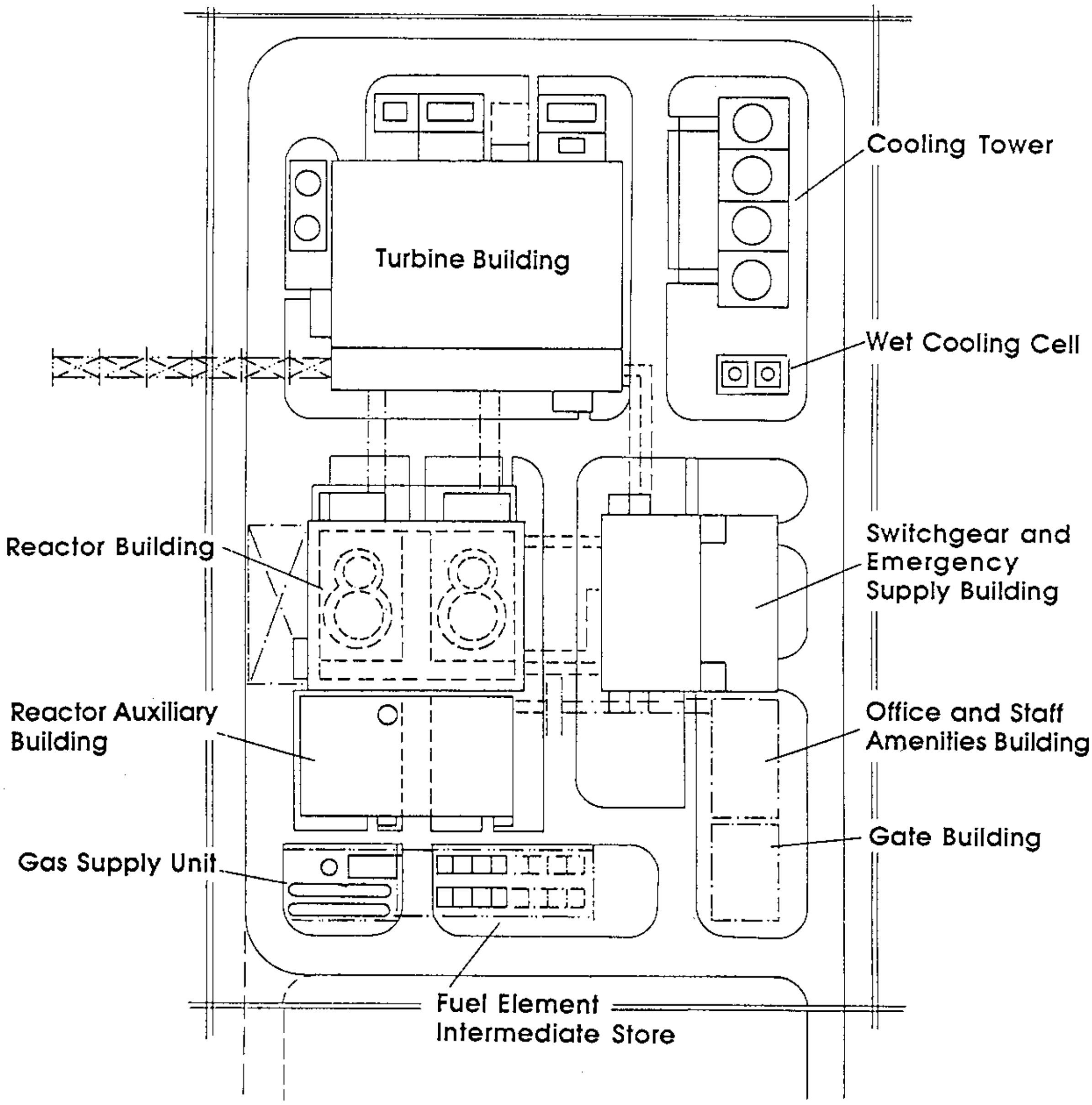

Fig. 4 **HTR-2 Module -Plant: Site Plan**

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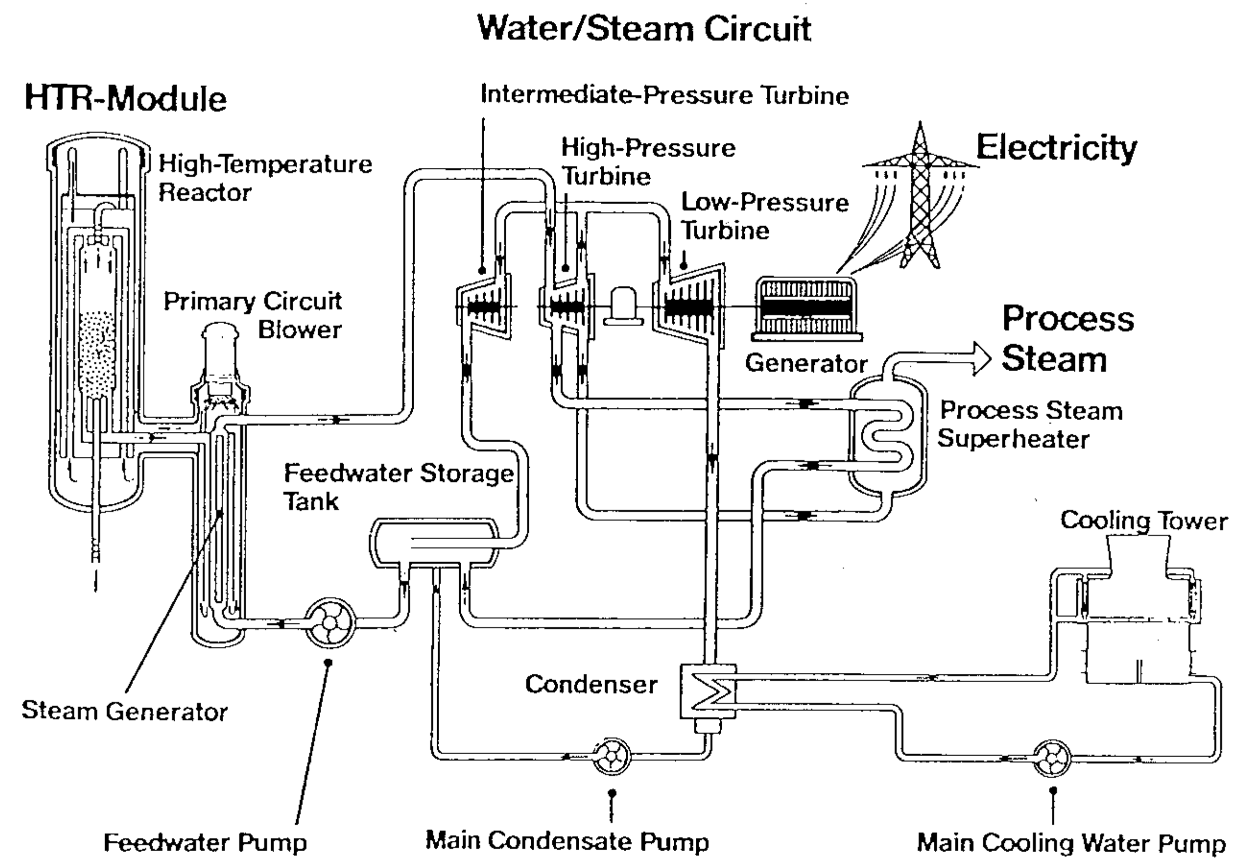

Fig. 5 Schematic Diagram of an HTR-Module Power Plant

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- I Introduction
- II Table of Contents
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- 3 Power Plant
- 4 Radioactive Materials and Radiological Protection
- 5 Power Plant Operation
- 6 Accident Analysis
- 7 Quality Assurance
- 8 Decommissioning
- 9 Waste Management Provisions
- 10 Guidelines and Technical Rules

Fig. 6 Content of Safety Analysis Report

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2 General Design Features of the HTR Module Power Plant
2.1 Introductory Remarks
2.2 Characteristic Safety Features
- Barriers against Release of Radioactivity
- Inherent Safety

2.3 Technical Design Features
- Reactor (Fuel Elements, Reactor Core, Control and Shutdown Systems)
- Nuclear Steam Supply System (Pressure Vessel Unit, Primary System and Steam Generator Isolation)
- Confinement Envelope
- Residual Heat Removal
- Emergency Power Supply
- Reactor Protection System
- Remote Shutdown Station

2.4 Nuclear Classification and Quality Requirements
2.5 Summary of Design Basis Events
2.6 Postulates and Measures for In-Plant Events
2.7 Postulates and Measures for External Events

Fig. 7 **Content of Section 2, Safety Analysis Report**

3.4 RESEARCH AND DEVELOPMENT RELATED TO LICENSING OF HTTR

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ABSTRACT

The research and development on HTGR in Japan Atomic Energy Research Institute (JAERI) were initiated by the program of the experimental very high temperature reactor (VHTR) in 1969. According to the experimental VHTR program, the various kinds of research and development works were started at that time for the experimental VHTR construction.

The items of research and development were design study of the experimental VHTR, material research, developments on helium engineering and construction of large test rig. Through the design study of the experimental VHTR, the survey of the technology to be developed and the basic data for VHTR design and construction, was performed to complete the research and development plan.

In 1987, the construction of the HTTR was decided instead of the experimental VHTR by the revision of the Long-Term Program for Development and Utilization of Nuclear Energy. According to this new program, JAERI plan of research and development was reviewed and supported the conclusion that no large modification of the existing plan was necessary.

JAERI was carrying out various kinds of research and development for the HTTR licensing and construction. These works were conducted in the Oarai in-pile Gas Loop, the Helium Engineering Demonstration Loop, Very High Temperature Reactor Critical Assembly and the others for long time.

This report covers research and development works performed for the HTTR licensing and construction.

1. Introduction

The research and development on HTGR in Japan Atomic Energy Research Institute (JAERI) were initiated by the program of the experimental very high temperature reactor (VHTR) in 1969. According to the experimental VHTR program, the various kinds of research and development works were started at that time for the experimental VHTR construction.

The items of research and development were design study of the experimental VHTR, material research, developments on helium engineering and construction of large test rig. Through the design study of the experimental VHTR, the survey of the technology to be developed and the basic data for VHTR design and construction, was performed to complete the research and development plan. Concerning material research, the coated particle fuel, graphite and high temperature alloy were tested. To accumulate the design data of helium heat transfer, the small He-loop was constructed, and for irradiation experiment of the materials, the in-pile loop, OGL-1, was installed in the Japan Material Testing Reactor.

The approach to complete the detailed program of research and development were as follows :

(1) to evaluate the data required for the design process of the VHTR,

(2) to survey and review existing technology to confirm what is currently available, and

(3) to define additional data needed and specific technology development required.

In 1987, the construction of the HTTR was decided instead of the experimental VHTR by the revision of the Long-Term Program for Development and Utilization of Nuclear Energy.

According to this new Program, JAERI plan of research and development was reviewed and supported the conclusion that no large modification of the existing plan was necessary. It was generally understood that the results of the research and development works for the VHTR are applicable for the HTTR program because the major specification and design philosophy are very similar.

Therefore, the programmed previously works were continued and the some experiments based on the HTTR design were conducted additionally.

This paper covers research and development works performed for the HTTR licensing and construction.

2. Progresses of the Research and Development on the HTTR

2.1 Reactor Physics and Nuclear Design

In order to verify the accuracy of calculations related to the neutronic design of the HTTR, various experiments have been conducted using a critical assembly VHTRC (Very High Temperature Reactor Critical Assembly). The VHTRC cores were assembled to study the basic neutronic characteristics of the HTTR core. The experiments are being conducted for the VHTRC-4 core which is loaded with the 2, 4 and 6 % enriched fuel in the axially-zoning pattern. The major items investigated are critical mass, temperature coefficient of reactivity, neutron flux distribution, reactivity worth of the HTTR burnable poison rod and reactivity worth of the HTTR control rod.

For an example of the experiments in VHTRC, the measured reactivity change caused by core temperature change is shown in Fig. 1. The calculated values were obtained using the SRAC code with the ENDF/B-IV nuclear data. The double-heterogeneity of coated particles in fuel compacts and fuel rods in a graphite block is taken into account in the cell calculation by the collision probability method. The core calculations were performed using the three-dimensional diffusion theory.

Such nuclear characteristics as power distribution, control rod worth, shut down margin and temperature coefficients are calculated by the nuclear design code system for HTTR, which consists of the computer codes DELIGHT, TWOTRAN-2 and CITATION-1000VP. DELIGHT is an one-dimensional lattice burn up code which is used to obtain multi-group neutron spectrum of a fuel cell and to produce the few-group constants for core analysis. TWOTRAN-2 is a transport code which is used to produce the average group constants of the graphite block and the inserted control rods for core analysis.

CITATION-1000VP is a core analysis code. The verification of the nuclear design code system was made by the comparison between the experimental data of VHTRC as shown in Fig. 2.

2.2 Fuel and Graphite Materials

2.2.1 Fuel

Performance of the HTTR fuel under normal reactor conditions has been tested in both the OGL-1 as shown in Table 1 and the capsule irradiation experiments. In the OGL-1 experiments, irradiation of the 11th fuel element, which aimed at testing the performance of the fuel fabricated by the scale-up facilities, was finished, and irradiation of the 12th element, which aimed at a long-term irradiation, was subsequently started. Fractional release (R/B) of Kr-88 from the 11th fuel element remained in the order of 10 to the -6th power.

Concerning the safety-related research, post-irradiation heating experiments on the irradiated coated particles and a fuel-burning experiment by air-ingress were carried out. Figure 3 shows typical results of the fission product release under isothermal heating of about 100 coated particles, which were irradiated up to 3.6 % FIMA at the maximum temperature of 1510 ℃. The cesium and silver releases by heating after 50 hours at 1800 ℃ were fairly large compared with the releases at 1600 ℃, which was considered that, thermal decomposition of SiC layer or SiC layer/Pd interaction occurred.

2.2.2 Graphite Material

Research and development have been carried out to clarify corrosion behavior and physical properties of the candidate carbon and graphite materials.

Mechanical properties of the candidate graphite and carbon materials have been investigated to accumulate the data required for licensing and safety evaluation of the graphite and carbon components of the HTTR. The research work has been carried out on the high-temperature Young's modulus, impact strength, fracture mechanics properties, low cycle fatigue life, irradiation creep properties and so on.

The multi-axial fracture behavior of the isotropic fine-grained IG-110 graphite has been examined using two types of newly manufactured bi-axial fracture testing machines. One can apply a combined axial load and internal pressure and the other can apply a combined axial load and torsion to the cylindrical specimen. Figure 4 shows the fractures of IG-110 graphite specimens under tension-compression stresses in air at room temperature. It is apparent that the bi-axial fracture behavior of IG-110 graphite is exactly followed a modified maximum strain energy theory. Furthermore, it is pointed out that the design fracture curves (solid line) derived from a modified Coulomb-Mohr theory have an enough margin of safety in the states of tension and compression stresses.

2.3 Metallic Material

2.3.1 2 1/4 Cr - 1 Mo Steel

In the HTTR, 2 1/4Cr-1Mo steel is used for the pressure boundary components (pressure vessel, pipes etc.) and reactor core support structures. The pressure vessel should retain sufficient reliability against any kind of failure. The material needs to have good characteristics against high temperature, and time-dependent material properties have to be taken into consideration in the design because the design temperature is 440 ℃. At the same time, the material has to retain sufficient fracture toughness throughout the whole operation period of the HTTR.

The fracture toughness requirement will follow the ASME design criteria which are used for the light water reactor pressure vessels. From recent experiments, the crack arrest fracture toughness (K_{Ia}) was obtained as shown in Fig. 5. The data have shown higher toughness values compared with the K_{IR} curve which is used in the ASME Code Sec. III. This fact provides a basis for assessing a structural integrity by the present ASME method using the K_{IR} vs. ($T-T_{NDT}$) curve, where T is a metal temperature and T_{NDT} is a nil-ductility transition temperature.

2.3.2 Hastelloy XR

The tests for creep, fatigue, fracture toughness, corrosion and other critical items have been undertaken as some parts of the comprehensive qualification tests for the accumulation of design data of Hastelloy XR, which is a modified version of Hastelloy X and is used for the IHX of HTTR. Corrosion tests in the JAERI-type B helium environment (20 Pa H_2, 10 Pa CO, 0.1 Pa CO_2 and 0.5 Pa CH_4) have been performed for 30 thousands hours. Creep tests are also being continued in the JAERI-type B helium for more than 30 thousands hours as seen in Fig. 6.

The service temperature of the class 1 components of HTTR is about 900 ℃, and Hastelloy XR is used as the structural materials. Since the service conditions exceed the high temperature limit of well-established structural design codes, the design guidelines for class 1 components of HTTR is required. Table 2 shows some feature of the high-temperature design code.

2.3.3 Cladding Material of neutron absorber rods (Alloy 800H)

Alloy 800H is used for the cladding material of the neutron absorber rods of the HTTR. To examine the effects of the chemical environment and thermal aging, creep tests on unirradiated Alloy 800H have been carried out. The post-irradiation creep tests have been carried out. The results of the first series of the post-irradiation creep tests, showed that the creep rupture lives of irradiated Alloy 800H were 10 to 30 % of those of unirradiated one.

2.4 High-temperature Components

2.4.1 Helium Engineering Demonstration Loop(HENDEL)

The HENDEL has been utilized to perform demonstration tests of large scale high-temperature components of the HTTR from March 1982. The HENDEL consists of the Mother (M), Adapter (A) and Test (T) sections. As seen in Fig. 7 the Mother and Adapter (M+A) section circulates helium gas at flow rate of 4 kg/s, pressure of 4 MPa and at the maximum temperature of 1000 ℃. The M+A section has been operated for more than 16,000 hours since 1982. The test section consists of the Fuel Stack (T_1) Test Section and the In-core Structure (T_2) Test Section.

The purpose of the T_1 test section is to confirm the characteristics on heat transfer in a fuel channel, flow rate distribution in a fuel stack and mechanical integrity of a control rod driving mechanism under the same operating conditions as in the HTTR. The T_1 test section consists of a single channel (T_{1-S}) and a multi-channel (T_{1-M}) test rigs. A heat transfer test was performed with the T_{1-S}. A fuel rod used in the test had the same configuration and size as the HTTR fuel rod. Figure 8 shows a relation between Nusselt number (Nu) and Reynolds number (Re). This fuel rod has higher heat transfer performance than that (Nu_S) for a concentric smooth annulus.

The verification of the temperature analysis code TEMDIM which was used in the HTTR fuel design, was made by the comparison between the experimental results obtained by the T_1 test section and analytical results.

The purpose of the T_2 test section shown in Fig. 9 is to investigate high-temperature characteristics on such core support structures, as the reflector block, the plenum block, the outlet hot gas duct, the core support post and the insulation layer in the HTTR.

Figure 10 shows a relation between leakage flow rate and pressure difference between inside and outside regions of reflector blocks. The data presently obtained after 6000 hours operation were agreed very well with the data obtained in the functional test at the installation of T_2.

The coolant flow rate distribution in the HTTR core is calculated by the flow network analysis code FLOWNET. The verification of the FLOWNET code was made through the comparison between the analytical result and experimental results of the thermal-hydraulic experiments with the T_2 test section and an one-column test facility.

2.4.2 High Temperature Component

The hot gas duct shown in Fig. 11 is an annular type duct which is one of the important components developed for the HTTR and consists of the liner tube, the pressure tube, the guide tube, the insulation, the seal plate and the stud. In this duct, the hot helium gas of 950 ℃ flows inside the liner tube and the cold helium gas of 400 ℃ flows in the counter direction of hot gas flow between the pressure tube and the guide tube. Figure 12 shows a relation between effective thermal conductivity (λ) and mean temperature (T) of the insulation settled between the liner tube and the pressure tube.

2.5 Reactor Instrumentation and Control

For HTTR control, high-temperature fission counter-chambers(HTFC), gamma-compensated ionization chambers(CIC) and high-sensitive gamma-uncompensated ionization chambers (HSUIC) have been developed. The HTFC withstood accelerated irradiation tests at 600 ℃, long-term in-reactor operation tests at 600 ℃ for 1000 days and over-heat tests at 800 ℃ for about 500 hours in simulating an accidental condition. The CIC were also tested in the JRR-4 reactor at temperatures of 400 ℃, and temperatures of 500 ℃ and 600 ℃ for more than three years. The sensitivity of HSUIC measured was about 7.3×10^{-13} A/nv in a graphite pile with a neutron source and the test is being continued in the JRR-4. As for a nuclear instrumentation system, a Wide Range Monitoring System which can measure neutron flux ranging about 10 figures was fabricated and tested in the JRR-4 and in the AVR.

Materials for high-temperature use have been investigated and developed for a noise thermometer, and the trial fabrication has been made. In addition, the high-temperature stability of Ni-Cr alloy thermocouples and their compatibility with high-temperature graphite in the helium gas atmosphere have been examined to be used in the range from 850 ℃ to 1100 ℃ (long-term use) and at 1200 ℃ (short-term use).

Both classical and modern control methods have been studied for assuring high degree dynamic stability and accuracy required for the controlled variable against various kinds of disturbances. Emphasis is placed on the development of advanced control and diagnostic methods for enhancing operational safety of the HTTR.

A computer-based integrated supervisory control and instrumentation system of the HTTR is being designed to promote automation of plant control and thereby to facilitate its operation. It consists of distributed microcomputer-based control systems for individual subsystems of the HTTR such as reactor power control, inlet and outlet helium gas temperature control and of a supervisory control system which coordinates their functions as an integrated control system including start-up and shutdown controls.

2.6 Experiments Related to Reactor Safety

2.6.1 Fission Product Plate-Out

Fission product (FP) plate-out distribution along the main pipe of OGL-1 primary circuit has been measured for the verification of a plate-out analysis code for the HTTR design.

Distributions of Cs-137 along the distance from the reactor core are shown in Fig. 13, which indicates that the plate-out activity by Cs-137 atoms increases with the increase in irradiation cycles.

Quantity of released Cs-137 was also estimated using the diffusion tubes irradiated with the 10th OGL-1 fuel specimen.

2.6.2 Gaseous FP Monitoring

Figure 14 shows a schematic diagram of the experimental monitoring system in the primary circuit of the OGL-1. The system consists of a moving-wire precipitator and a Ge gamma-ray spectrometer. The precipitator monitors total activity of gaseous FPs, and the Ge gamma-ray spectrometer analyzes contributions of each nuclide to the total activity. Performance tests have been carried out in the flow of OGL-1 primary coolant. The sensitivity of the monitoring system for gaseous FPs was 35 cps/Ci. The monitoring system has more than enough sensitivity to monitor the allowable limit of FPs in the HTTR coolant.

This system was adopted as the HTTR fuel failure detection system.

2.6.3 Heat Transfer and Fluid Dynamics

(1) Heat transfer of natural convection in the top cover of the pressure vessel

In loss of forced cooling due to an inner tube rupture of the primary cooling system, heat transfer experiments of natural convection in a hemisphere were examined to estimate a top cover temperature of the pressure vessel. Heat transfer correlations were obtained from an experiment carried out in the wide range of Prandtl numbers from $1x10^{6}$ to $5x10^{10}$.

(2) Momentum and mass transfer of oxygen gas flow in the reactor core

Flow distribution and heat transfer of oxygen gas entering the core were studied to verify the graphite corrosion rate at high temperature during air ingress accident.

(3) Hydraulics on a gas mixture flow with graphite/oxygen reaction

In order to estimate a flow rate during air ingress into the core, natural circulation of the mixture flow through a reverse U-shaped circular tube has been studied experimentally and analytically. Concentrations of oxygen, carbon monoxide and carbon dioxide were measured at various points in the flow passage to examine the behavior of the gas mixture flow.

(4) Development and application of visualization techniques

Visualization of complex thermo-hydraulic phenomena makes understanding easy and effective. Visualization techniques consist of pearl pigment method (Pearl pigment of a mica-platelet type is used as a tracer), liquid crystal suspension method (Thermo-sensitive liquid crystal is used as suspension particles) and liquid crystal thermo-camera method (A system consisting on thermo-sensitive liquid crystal film on a heated surface, video camera, image processor).

3. Concluding Remarks

Many results of the research and development have been highlited in this paper, and nevertheless the safety review of the HTTR by the Nuclear Safety Commission is now going on, JAERI considers that the research and development works for the HTTR licensing are almost completed.

The reasons of the acceleration of the R and D works include (1) the VHTR program aiming the outlet temperature of 1000 ℃, (2) suitable and well-managed plan of the R and D, (3) accumulation of know-how on high-temperature components by the operation and maintenance of the large He-Loop, (4) cooperation and assistance of the industries, and (5) the international collaboration.

JAERI now resolved that the new research plan for upgrading the HTGR technology must be progressed with the JAERI's high potentiality of innovative development, according to the revised Long-Term Program for Development and Utilization of Nuclear Energy issued by the Japan Atomic Energy Commission.

References

" Present status of HTGR Research & Development " January 1988, 1989 and 1990, Japan Atomic Energy Research Institute.

Table 1 Fuel Irradiation test in OGL-1

No.	Enrichment (%)	Max. fluence (n/cm^2)	Max. burnup (MWD/t)	CP failure fraction (post)	(before)
1	12.0	1.2×10^{20}	5800	2.7×10^{-5}	3.7×10^{-6}
2	12.0	2.0×10^{20}	8500	1.4×10^{-5}	1.0×10^{-4}
3	12.0	8.9×10^{19}	4800	7.1×10^{-4}	8.5×10^{-4}
4	19.9	2.3×10^{20}	18000	7.3×10^{-5}	1.1×10^{-4}
5	19.9	3.8×10^{20}	30000	3.7×10^{-3}	1.9×10^{-3}
6	19.8	4.1×10^{19}	3900	9.6×10^{-5}	5.0×10^{-5}
7	19.7	1.6×10^{20}	12000	7.1×10^{-5}	2.7×10^{-5}
8	19.7	1.2×10^{20}	9100	1.1×10^{-4}	5.9×10^{-6}
9	19.1	2.8×10^{20}	24000	1.4×10^{-3}	8.7×10^{-4}
10	19.5	-	26000	4.9×10^{-4}	2.5×10^{-4}

Fluence E > 0.18MeV

Table 2 Features of the High Temperature Structural Design Code

Materials	Feature of Code
2 1/4Cr-1Mo Steel	Design rules are the same as those of high-temperature design code for FBR "Monju" (FBR Code).
Austenitic Stainless Steel (SUS 321 and SUS 316)	
Ni-base corrosion resistant and heat resistant superalloy (Hastelloy XR)	Referred to the FBR Code, design rules are established on the basis of material properties and component test data.

Significant Failure Modes	Feature of Design Rules
Creep Rupture	Limiting the primary stress intensities
Creep-Fatigue Failure	Limiting the accumulated creep-fatigue damage developed by primary+secondary+ peek stress
Loss of Function by Excessive Deformation	Limiting the strain
Creep Buckling	Limiting the loads and strains

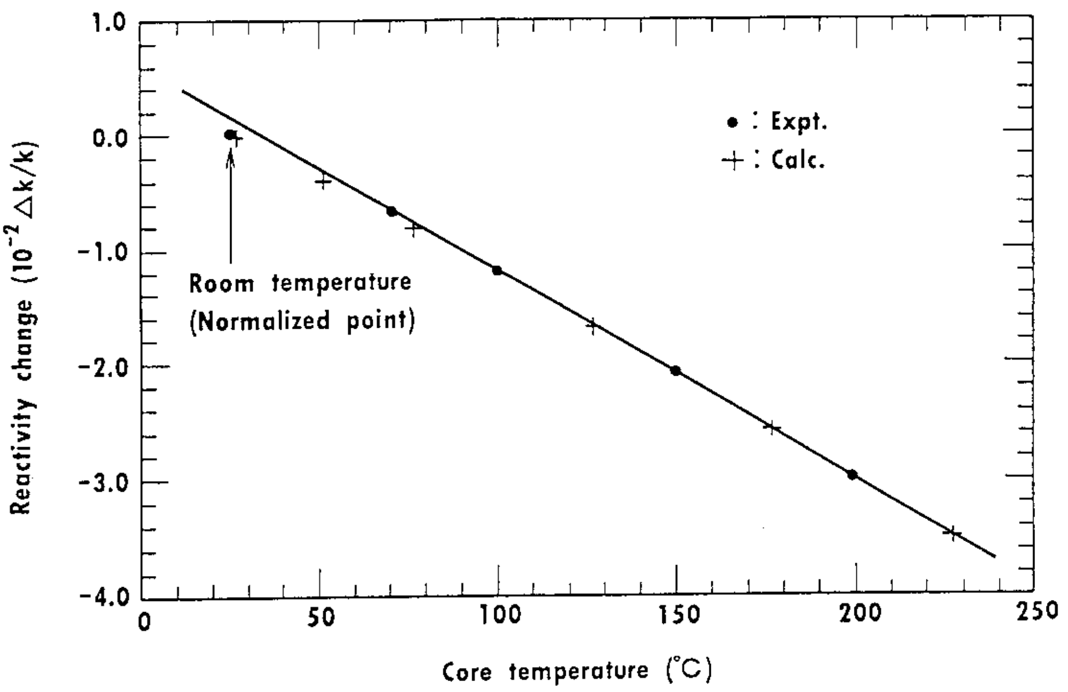

Fig. 1 Reactivity change v.s. core temperature

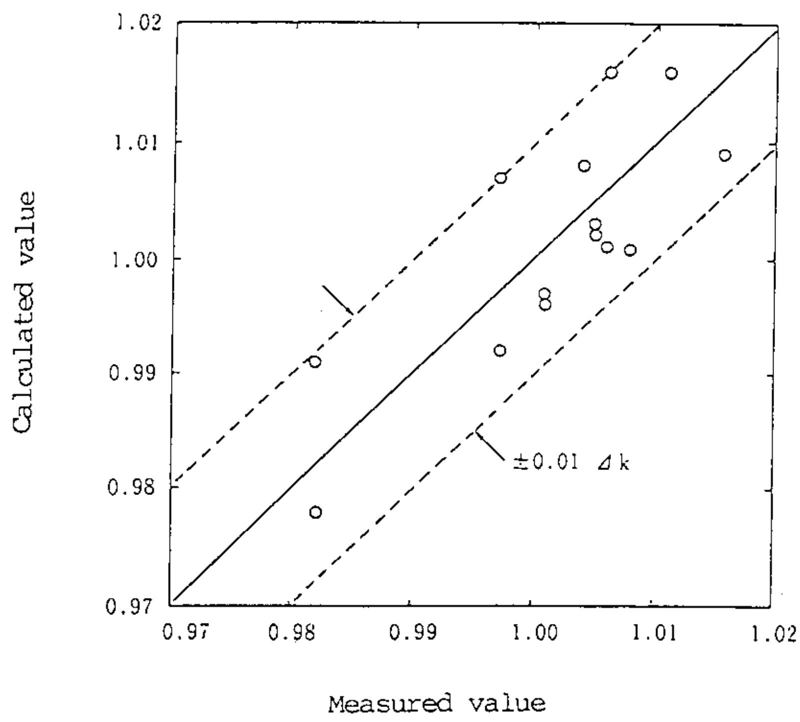

Fig. 2 Comparison between the measured and calculated values of effective multiplication factor

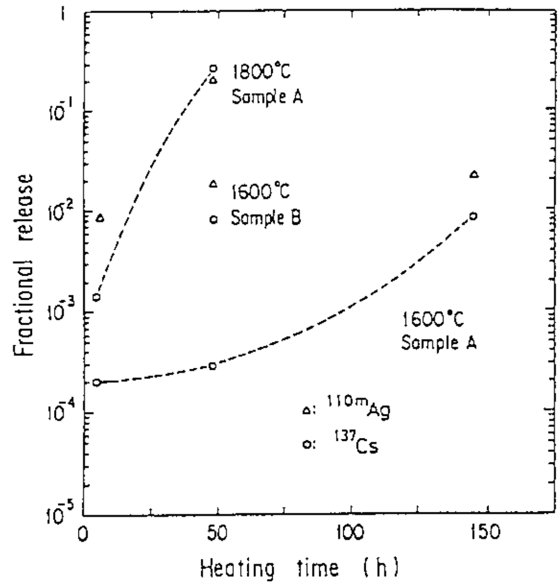

Fig. 3 Cesium and silver release from irradiated particles

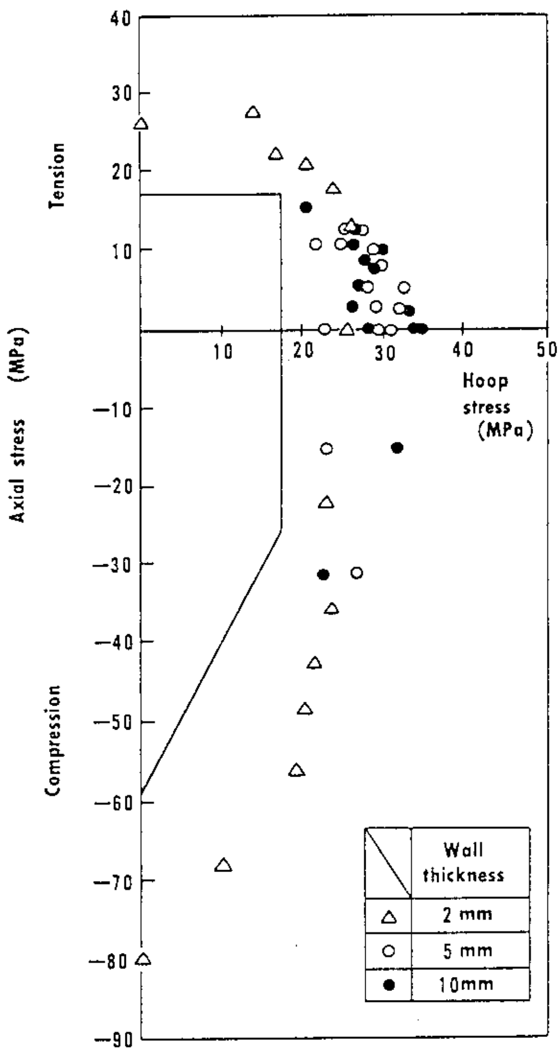

Fig. 4 Results of biaxial stress fracture test of fine-grained isotropic graphite IG-110

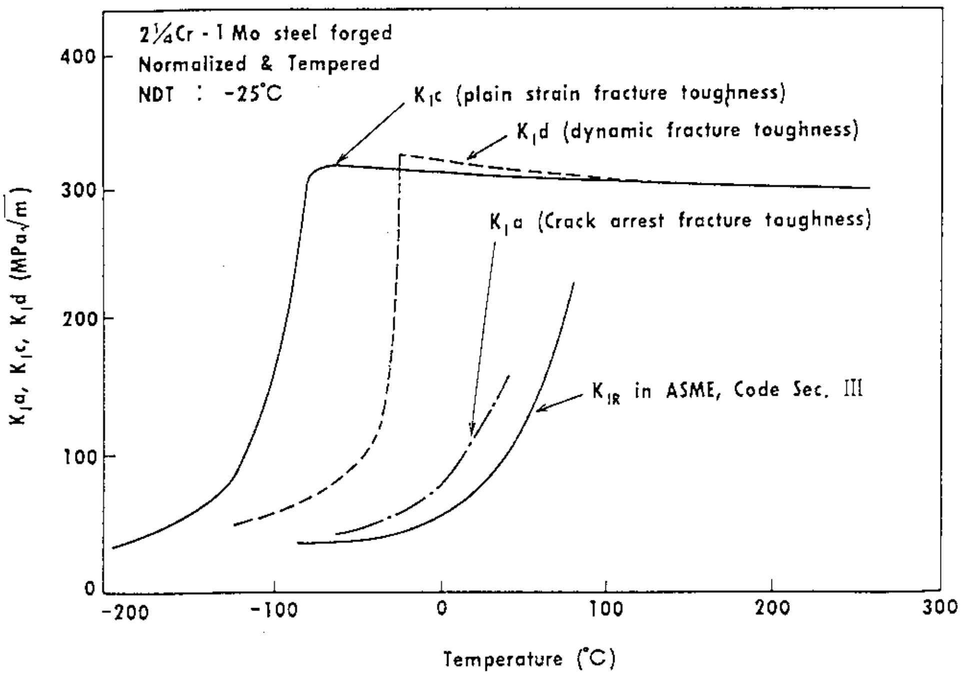

Fig. 5 Fracture toughness v.s. test temperature

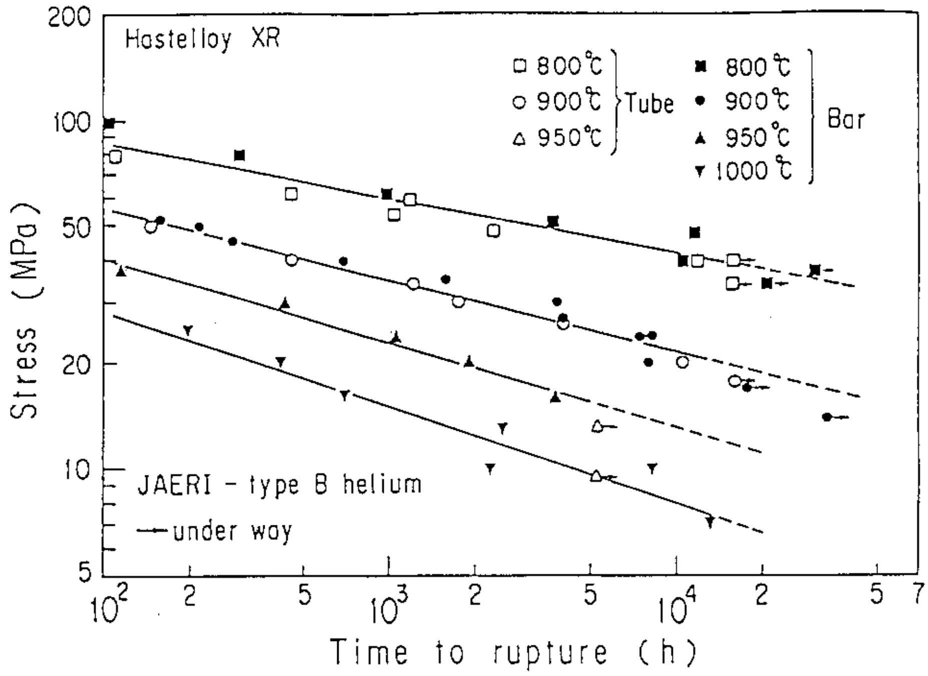

Fig. 6 Long-term creep rupture data for Hastelloy XR in JAERI-type B helium

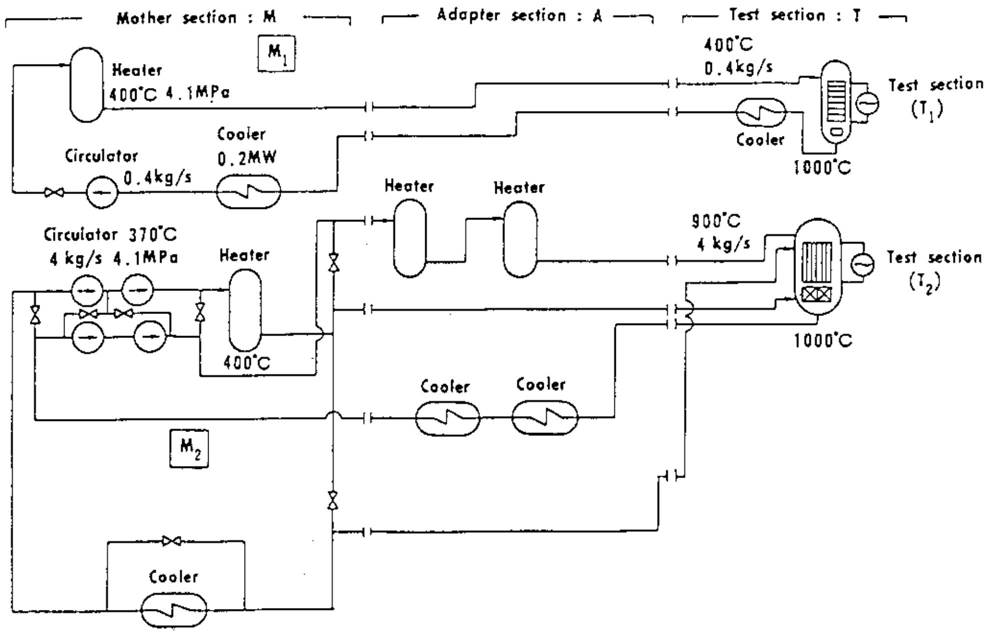

Fig. 7 Schematic flow diagram of HENDEL

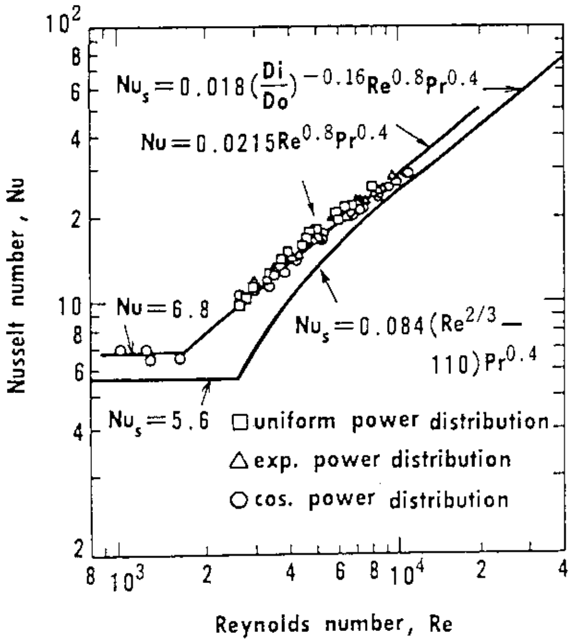

Fig. 8 Relation between Nusselt and Reynolds numbers

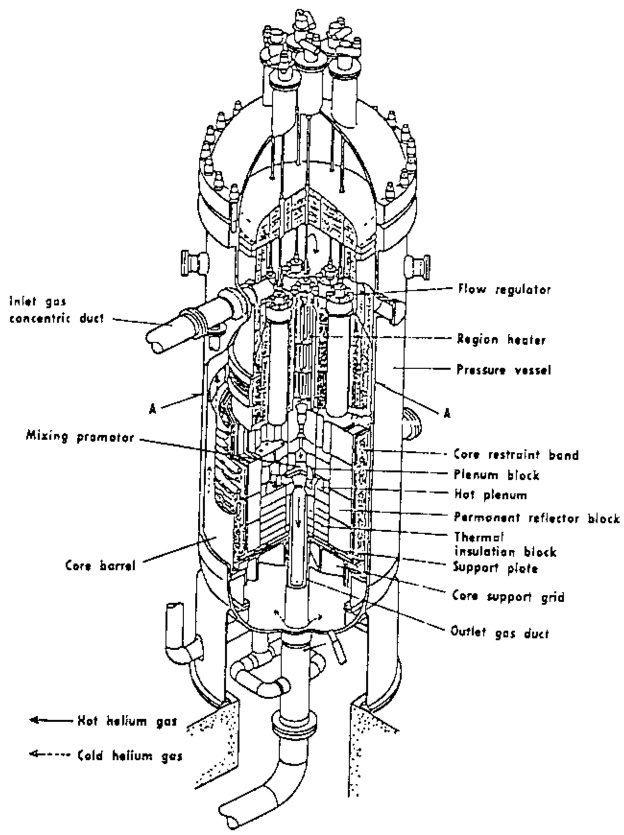

Fig. 9 In-core structure test section T-2

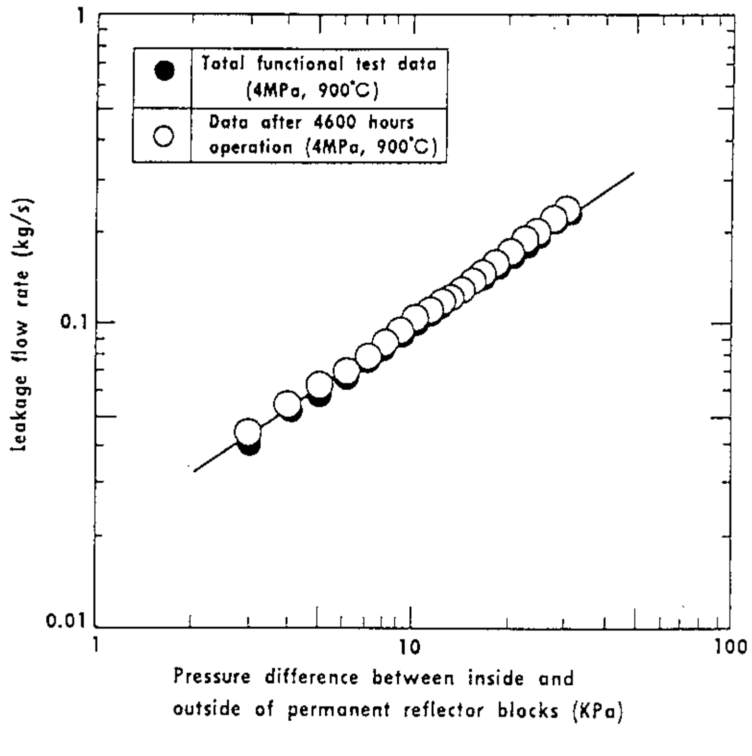

Fig. 10 Relation between leak flow rate and pressure difference

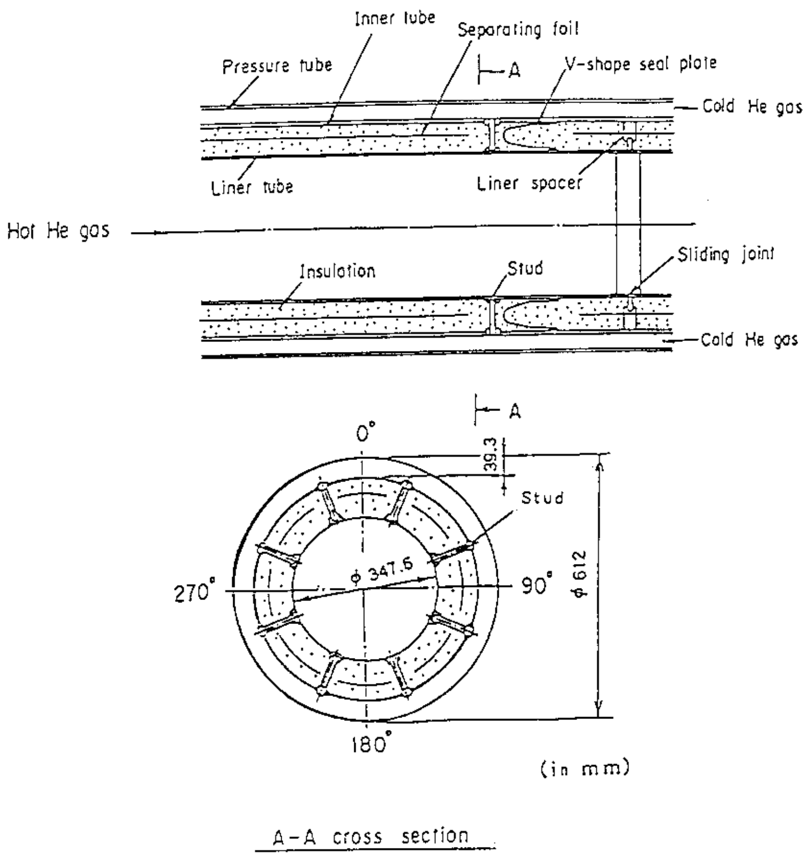

Fig. 11 Hot gas duct

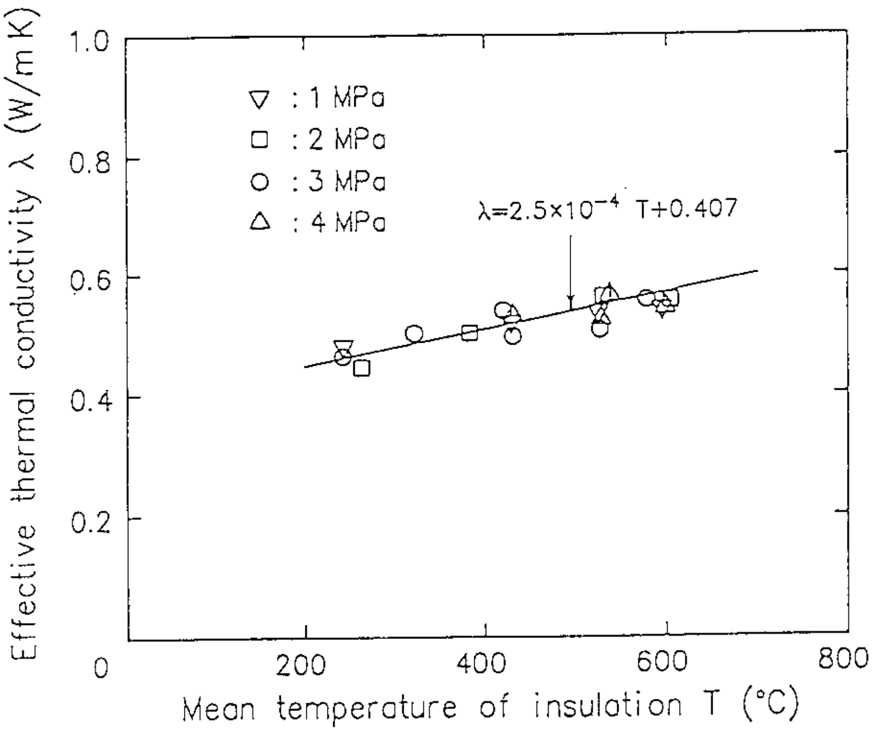

Fig. 12 Relation between effective thermal conductivity and mean temperature of insulation

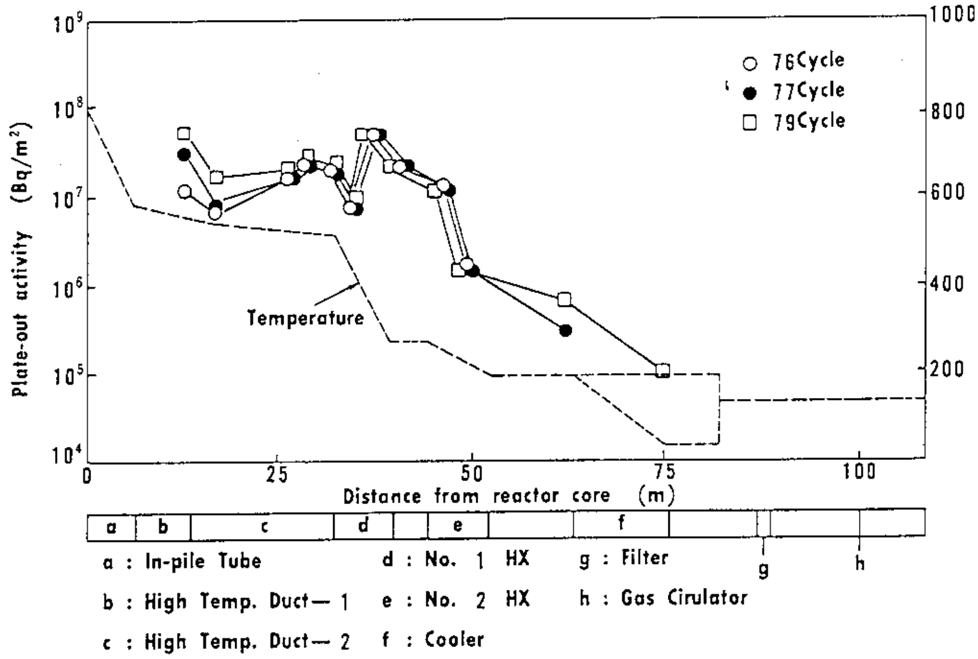

Fig. 13 Plate-out activity of Cs-137 in OGL-1

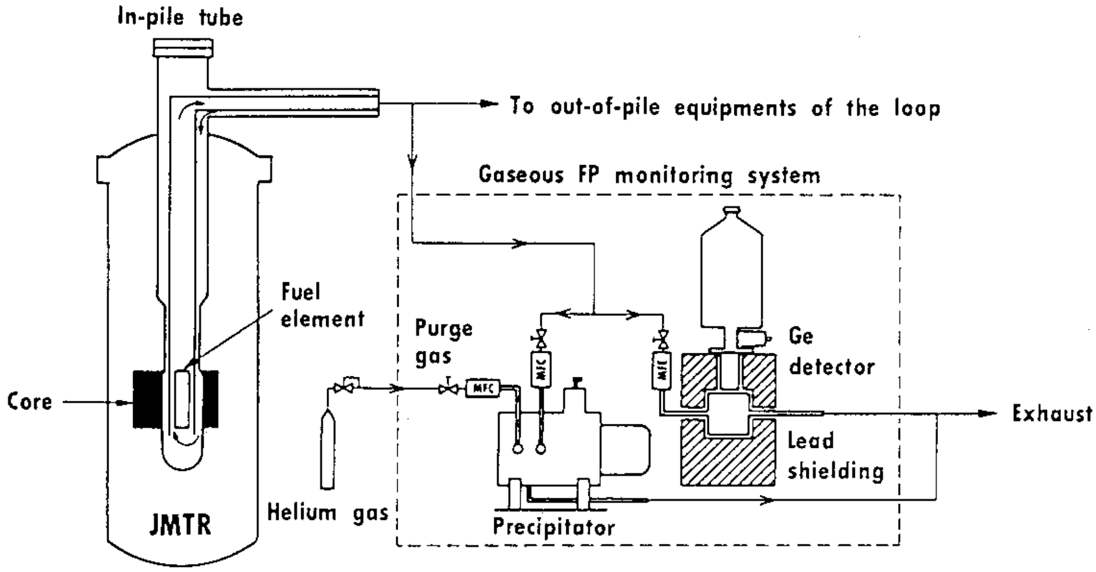

Fig. 14 Schematic diagram of gaseous FP monitoring system

3.5 RESEARCH AND DEVELOPMENT ASSOCIATED WITH LICENSING OF MHTGR

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ABSTRACT

RESEARCH AND DEVELOPMENT ASSOCIATED WITH LICENSING OF MHTGR

The Modular High Temperature Gas-Cooled Reactor (MHTGR) currently under development by the U.S. Department of Energy (US-DOE) for commercial applications has top-level goals of producing safe, economical power for the U.S. utility industry. The utility industry has been represented in formulating design and licensing requirements through both a "Utility User Requirements Document" and by participating in the DOE system engineering process known as the "Integrated Approach." The result of this collaboration has been to set stringent goals for both the safety and operational reliability of the MHTGR. To achieve these goals, the designer must have access to a more comprehensive data base of properties in several fields of technology than is currently available. A technology development program has been planned to provide this data to the designer in time to support both his design activities and the submittal of formal licensing application documents. The US-DOE has chosen the Oak Ridge National Laboratory (ORNL) to take the lead in planning and executing these technology programs. When completed these will augment the designer's current data base and provide the necessary depth to meet the stringent goals which have been set for the MHTGR. It is worth noting that the goals of safety and operational reliability are complementary, and the data required from the technology development program will be similar. Therefore, the program to support the licensing of the MHTGR is not separate from that required for design, but is a subset of that which meets all the requirements that result from implementing the US-DOE's integrated approach. This paper will provide an overview of the technology development activities now in progress or planned by ORNL to meet these requirements. Work will be described in the fields of fuel development, fission product release and transport, graphite technology, metallurgical technology, reactor shielding, safety research,and the validation of design methods in thermal/hydraulics.

1. INTRODUCTION

The MHTGR Technology Development Program was formulated through a disciplined review of technical needs to support both design and licensing. This process was conducted as part of the Systems Engineering Method implemented for planning the entire MHTGR effort by the U.S. Department of Energy, known as the "Integrated Approach."

Top-level user and regulatory requirements were formulated, and a detailed functional analysis was done to link top-level requirements with specific design selections. The conceptual design was completed using existing data provided by:

- 25 years of research and development on HTGR fuels and materials.
- Experience from AVR, Dragon, Peach Bottom 1, FSVR, THTR.
- Design study results from several large HTGR (LHTGR) designs.

Assumptions were made to account for differences in the MHTGR and LHTGR design features or where data were missing, then a technology program was formulated to validate (or modify) the assumptions prior to final design. The resulting program is viewed as critical to the licensing process, but it also is designed to satisfy the entire set of comprehensive requirements that are generated by the integrated approach.

2. ADVANCED REACTOR LICENSING PROCESS

2.1 NRC POLICY ON PROCEDURAL APPROACH

The U.S. Nuclear Regulatory Commission (NRC) issued a statement of policy in 1986 regarding the regulation of advanced nuclear power plants.[1] Primary procedural objectives of the policy follow.

2.1.1 Early Interactions

This policy encourages the earliest possible interactions of the applicant (and supporting organizations) with the NRC.

2.1.2 Desired Characteristics

It also states the NRC's intent to provide all interested parties with their views concerning the desired characteristics of advanced reactor designs.

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1. U.S. Nuclear Regulatory Commission, *Policy for the Regulation of Advanced Nuclear Power Plants*, 51FR24643, July 8, 1986.

2.1.3 Timely Comment

Finally, it expresses the NRC's intent to issue timely comment on the implications for such designs for safety and the regulatory process.

2.2 NRC POLICY ON TECHNICAL APPROACH

2.2.1 NRC Minimum Requirement

The NRC's minimum requirement is for advanced reactors to provide at least the same degree of protection of the public and the environment that is required for current LWRs.

2.2.2 Enhanced Margins

The policy also states that NRC expects that advanced reactor designs will provide enhanced margins of safety and/or utilize simplified, inherent, passive, or other innovative means to accomplish their safety functions.

2.2.3 Novel Regulatory Approaches

In addition, it notes that "advanced reactor designers are encouraged as part of their design submittals to propose specific review criteria or novel regulatory approaches which NRC might apply to their designs."

3. MHTGR LICENSING SPECIFICS

The approach to licensing the MHTGR is responsive to NRC's advanced reactor policy statement. In particular, the US-DOE has taken a strong position in introducing an innovative approach to integrating the development of comprehensive requirements for the MHTGR to include the regulatory requirements, and is thus pursuing a novel licensing approach as is encouraged by the NRC policy.

The goal is for issuance of a design certification for the standard MHTGR design based upon standard safety analysis reports (SSAR) and other submittals, including a regulatory technology development plan (RTDP).

In accordance with the objective of achieving the earliest possible interactions with the NRC, a number of preapplication activities are in progress.

These include the submittal and review of a preliminary safety information document (PSID) and the issue of the NRC response to that submittal, a safety evaluation report (SER) issued as a draft (2/89) and soon to be issued in final form.

Currently the program is completing preapplication activities and preparing for submittal of a preliminary SSAR (PSSAR) which would start the formal application phase.

3.1 MHTGR PREAPPLICATION ACTIVITIES

Submittal of the PSID and subsequent interactions with NRC established the following.

(1) An innovative approach to the licensing process is to be employed, using a systems engineering approach to generate requirements for this project by functional analysis.

(2) The DOE implementation of this approach is called "The Integrated Approach." It emphasizes a top-down approach to developing Goals, Requirements, and Design Selections. It will be employed in parallel with the existing NRC requirements, and its products will be utilized for selecting regulatory criteria, etc., which are specific to the MHTGR (Fig. 1).

From the technology-development point of view, a significant result of following this logic was the selection of a plant which does not require a conventional containment structure (Fig. 2).

Instead, it was shown during the development of the requirements by functional analysis that because of the unique characteristics of the MHTGR, the goals of safe, economical power could best be realized by taking advantage of the high temperature capabilities of the ceramic fuel and the favorable thermal response of the reactor system.

Thus, the coated fuel particle becomes the fission product containment. The technology development program must show that this containment fulfills all the requirements in both normal operation and accident conditions. It requires the development and proof of performance of a higher quality fuel than that used in the U.S. previously in the FSV HTGR.

NRC response to this design selection in the draft SER indicates that acceptance will be strongly dependent on the success of the technology development programs, especially those dealing with the development of high-quality fuel and the proof of its ability to limit fission product release. (Figs. 3, 4, 5).

Interactions with NRC are continuing with reviews of progress in specific technology tasks, including the fuel development and fission product transport areas.

4. INTEGRATED APPROACH AND TECHNOLOGY PLANNING

4.1 IN THE DESIGN PROCESS

The following four goals guide the development of the MHTGR under the Integrated Approach (Fig. 3):

(1) MAINTAIN PLANT OPERATION

(2) MAINTAIN PLANT PROTECTION

(3) MAINTAIN CONTROL OF RADIOACTIVE NUCLIDE RELEASE

(4) MAINTAIN EMERGENCY PREPAREDNESS

The process of developing the design and the approach to safety for the MHTGR begins with the quantification of top-level criteria with which each of these four goals is to be achieved (Fig. 6).

Next, an integrated systems engineering approach is systematically applied to develop the Functions, Requirements, and Specific Design Selections necessary to achieve the plant design.

4.2 TECHNOLOGY PLANNING STAGE

When a design selection has been made, the designer is asked to determine if he has all the data on hand necessary to execute the design and to completely support the expected licensing activities (Fig. 7).

When the answer is "no," the designer formally specifies a DESIGN DATA NEED (DDN) and adds it to the TECHNOLOGY DEVELOPMENT PLAN (TDP) which covers that field of technology. TDP documents are maintained as the formal record of the planning which governs the scope of all technology development activities, as well as funding/scheduling priorities.

5. TECHNOLOGY DEVELOPMENT PLANS

For the MHTGR Program, Technology Development Plans are the vehicle by which the design organizations systematically express their needs for support by the technology organization while employing the logic of the Integrated Approach. TDPs are a bridging document between the design and technology portions of the program in which the requirements of the design organizations are documented, as well as the responses of the technology organization.

The Oak Ridge National Laboratory (ORNL) is the responsible organization for planning and executing the technology development activities.

The following TDPs are currently available:

(1) Fuel and Fission Product Technology Development Plan

(2) Graphite Materials Technology Development Plan

(3) Metals Technology Development Plan

(4) Physics Verification/Validation Plan

Each of the TDPs is a comprehensive statement of all the technology development needs and the responding tasks in that technical field. Since the Integrated Approach generates the complete set of requirements to meet all four of the program goals, the range of data to be obtained under each plan must be adequate to assure that both Safety and Investment Protection Criteria will be met.

Generally, the Investment Protection Criteria (Reliability Goal) are more stringent.

5.1 REGULATORY TECHNOLOGY DEVELOPMENT PLAN (RTDP)

A Regulatory Technology Development Plan document has been prepared by assembling the individual TDPs in one document and identifying the specific items which are of particular interest to the NRC. The scope of the total tasks specified is not altered by the RTDP.

6. SCOPE/CONTENT OF TECHNOLOGY PROGRAM

6.1 INTERNATIONAL COOPERATION

The US-DOE MHTGR technology development program features extensive international cooperation.

6.1.1 Fuel and Fission Product Transport

(1) High-quality fuel production feasibility first demonstrated by Germany.

(2) Fuel development irradiation testing at ORNL includes German and Japanese fuel.

(3) Fission product transport tests at CEA in France for the U.S. MHTGR Program are managed by ORNL.

6.1.2 Graphite Materials Development

The graphite materials development program involves cooperation with Japan. This includes testing in both countries with exchange of data.

6.1.3 Metals Development Programs

These have involved past cooperation with Germany and with renewed interest in advanced versions of MHTGR. More cooperation with Japan is anticipated.

6.1.4 Physics Methods Validation

Cooperation is anticipated with an IAEA program at the Paul Scherrer Institute in Switzerland and with JAERI in Japan.

6.2 FUEL PERFORMANCE TECHNOLOGY DEVELOPMENT PROGRAM SPECIFICS

6.2.1 Significance to Licensing

Fuel produced commercially in the U.S. for the Fort St. Vrain Reactor must be upgraded in quality to support the current licensing position that MHTGR does not require containment or public evacuation. Behavior of this improved-quality fuel must be demonstrated/proven (Fig 8).

Required quality was previously shown to be feasible by German data, and was reproduced by U.S. pilot process. Currently the U.S. program is concentrating on production-scale process development, refinement of performance models, and validation for licensing with statistically large samples in tests.

The coated particle is the primary fission product containment. There are two major sources of release from this containment:

(1) heavy metal surface contamination, and

(2) failure of the coating under irradiation and/or elevated temperature conditions during accidents.

Heavy metal contamination is the source of almost all of the gaseous fission product release (>90%). The requirements can be met, however, with a fractional limit of $<10^{-5}$. The primary source of failed coatings is particle manufacturing defects. There are several kinds of defects contributing to the defective population, with missing buffers predominating. To meet the overall limits on fission product release and provide an adequate margin, the requirement for missing buffer coating type defects is $\leq 5\times10^{-5}$ fraction.

Characterization of fuel performance for licensing requires irradiation exposure tests of large numbers of particles for high (95%) statistical confidence, followed by postirradiation accident condition testing to 1800°C. This testing simulates (exceeds) core conduction cooldown accident temperature conditions.

6.2.2 Status

These tests will establish failure rates and fractional releases under normal and accident conditions. Capsules containing 10^5 particles with either U.S. (HRB-21) or JAERI (HRB-22) high-quality fuel are ready for irradiation in the ORNL High Flux Isotope Reactor (HFIR). A high-temperature exposure facility is being prepared for postirradiation accident condition tests.

6.3 FISSION PRODUCT TRANSPORT TECHNOLOGY DEVELOPMENT SPECIFICS

6.3.1 Significance to Licensing

Even given that the improved-quality fuel will retain most of the fission products within the particles, the behavior of the released fraction during accidents must be established.

Large-scale plateout, liftoff, and washoff experiments are being conducted to provide needed data and to validate computational models. These data will cover metallic fission products on both graphite and metal surfaces.

The liftoff/washoff experiments simulate accident (depressurization and moisture ingress) conditions.

Computational models are being updated and validated.

6.3.2 Status

Two loops are in final stages of preparation for the above test programs. Both are managed by ORNL for the US-DOE:

(1) The COMEDIE Loop at CEA (CENG), Grenoble, France.

(2) The DABLE Loop at The Massachusetts Institute of Technology (MIT), Cambridge, Massachusetts

6.4 GRAPHITE MATERIALS TECHNOLOGY DEVELOPMENT SPECIFICS

6.4.1 Significance to Licensing

Integrity of the fuel elements and the core support structures is significant to licensing.

Designers have identified extensive data needed to complete the design of these components and to guarantee meeting safety and reliability requirements developed by the integrated approach method. Because the requirements for reliability in particular are more stringent than current practice, and since graphite as a structural material has not previously been thoroughly characterized, the designer's data needs are extensive. (The cooperative program with JAERI has been most effective in this area).

6.5 AREAS OF TEST AND DATA GATHERING

(1) Statistical variability of properties - mechanical, physical

(2) Multiaxial stress - ASME design code support, develop facility

(3) Fatigue behavior - lifetime, verify Minor's rule

(4) Oxidation - kinetics, effect of impurities, effect on other properties

(5) Irradiation effects - dimensional changes, property changes, creep

(6) NDE - development of improved methods

(7) Composites development - replace metal components for improved performance

6.5.1 Status

This is a relatively mature program with diverse multiple activities currently in progress at labs within ORNL and JAERI. HFIR shutdown has delayed irradiation effects testing at ORNL but is expected to resume shortly.

6.6 METALS MATERIAL TECHNOLOGY DEVELOPMENT SPECIFICS

6.6.1 Significance to Licensing

The integrity of the reactor vessel, heat transport system, and internals needs to be confirmed under normal and accident conditions.

The designer has developed a list of data needed to support his design analysis. Although the MHTGR uses a vessel derived from PWR practice, the irradiation and temperature (normal operation and accident) environments are different and require new MHTGR specific data.

6.6.2 Areas of Test and Data Gathering

Existing pressure vessel steels are being ASME Code qualified to permit their use over the range of environmental conditions expected for MHTGR. (SA533B, SA508 Steels)

Irradiation tests are ongoing to assure integrity of the metals for the reactor internals under appropriate MHTGR conditions. (Alloy 800H).

The effects of MHTGR thermal history and primary coolant environment on the properties of all metallic materials in the Reactor Internals and Heat Transport System are being studied extensively.

The steam/water corrosion behavior of MHTGR Steam Generator materials is being confirmed with emphasis on stress corrosion cracking in the bimetallic weld area under alternating wet/dry conditions. FSV experience is being factored into these studies.

6.6.3 Status

Multiple activities are currently in progress at labs within ORNL. Irradiation testing at the University of Buffalo is temporarily suspended due to leaks at that reactor. We are interested in expanding the international cooperation in this area.

6.7 SAFETY RESEARCH TECHNOLOGY DEVELOPMENT SPECIFICS

6.7.1 Significance to Licensing

The desirable safety characteristics of the MHTGR during accidents depend on physics/thermal hydraulic relationships which passively limit the maximum temperatures to levels which can be accommodated by the fuel without increasing fission product release. These programs relate to the demonstration of these characteristics on a large systems scale and to validation of the computational tools used in safety analysis.

This program has also benefitted from international cooperation with Germany, Switzerland, Japan and the IAEA.

(1) The AVR Test Program - US-DOE participated in the AVR demonstration which conducted tests demonstrating inherent safety features of the AVR, which are adapted to and optimized into the MHTGR design.

(2) Validation of Computational Codes - Participation in the AVR program also provided integral data for validation of physics and thermal fluid-flow methods used for the safety and performance analyses of MHTGR. These data are being employed in an overall study of the approach to validation of the codes employed in MHTGR design and licensing.

6.7.2 Status

AVR tests are now complete, ORNL has proposed to the US-DOE program that we participate in additional international cooperative activities in the form of the IAEA's PROTEUS program for further validation of physics computational methods.

7. SUMMARY OF STATUS

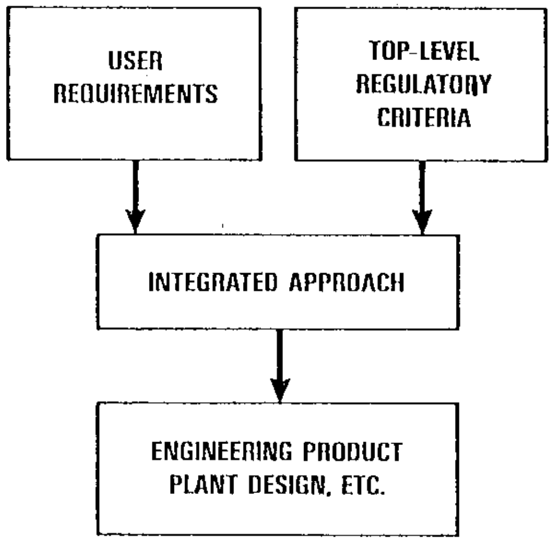
The technology programs supporting the design of the MHTGR are conducted within an integrated set of criteria/requirements which are sufficiently comprehensive that they also cover the requirements to satisfy licensing.

Given adequate funding rates, the program supports the schedule of licensing events for the standard MHTGR Design Certification.

The provisions of the NRC Advanced Reactor Policy Statement are being fully utilized in the technology area to expedite the flow of information to the NRC and to implement an innovative approach to both design and licensing known as the Integrated Approach.

US-DOE MHTGR PROGRAM

MHTGR IS RESPONSIVE TO USER & REGULATORY REQUIREMENTS



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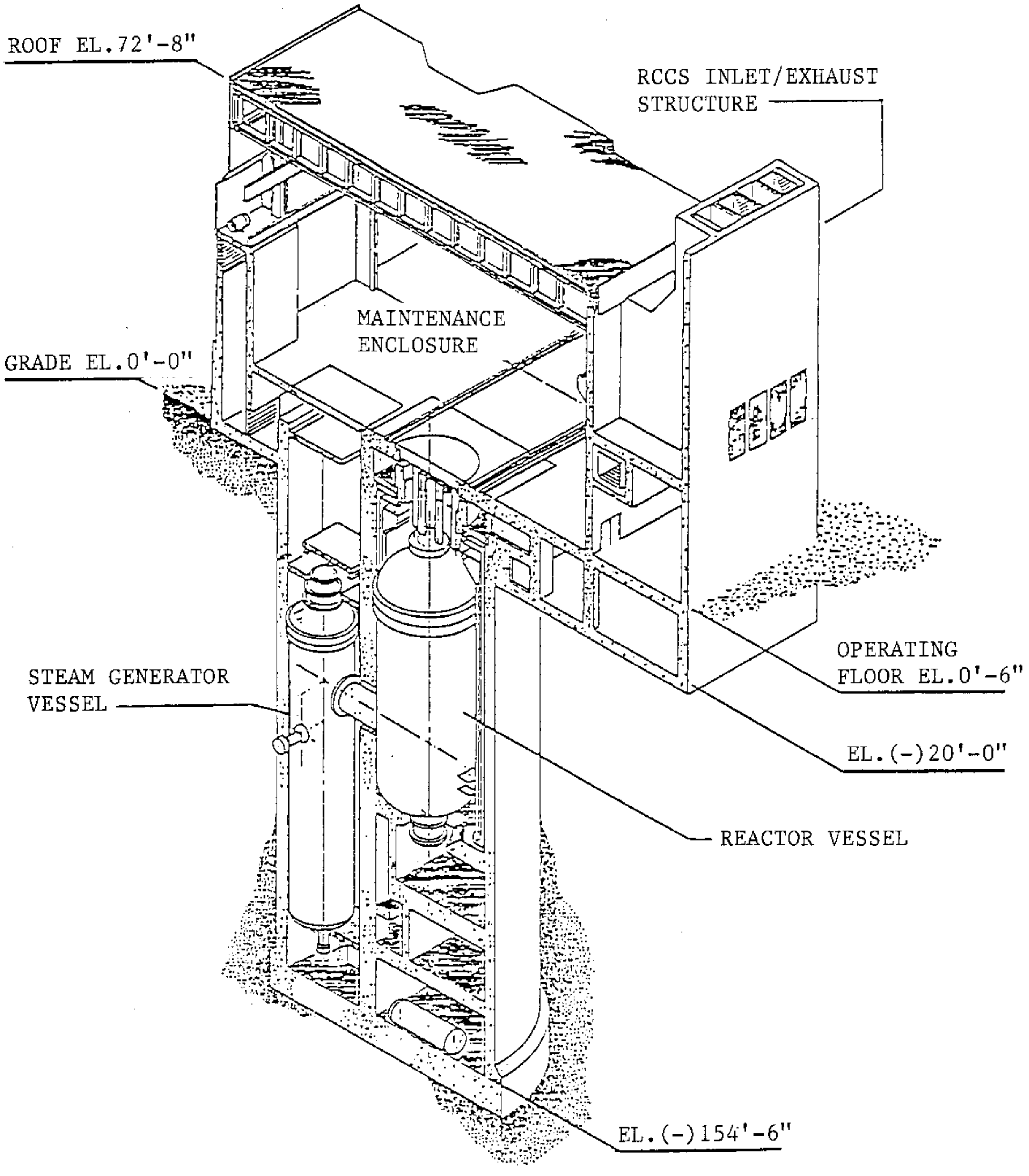
GENERAL ATOMICS

Figure 1



NUCLEAR ISLAND ELEVATION

GENERAL ATOMICS

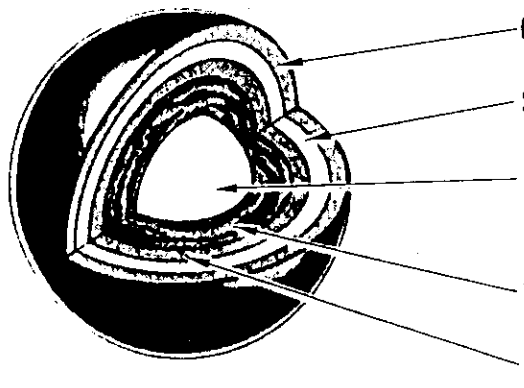
Figure 2

GENERAL ATOMICS

CERAMIC COATED FUEL PARTICLES RETAIN FISSION PRODUCTS AT HIGH TEMPERATURES

RETAIN RADIONUCLIDES IN FUEL PARTICLES

CERAMIC TRISO-COATED FUEL PARTICLE



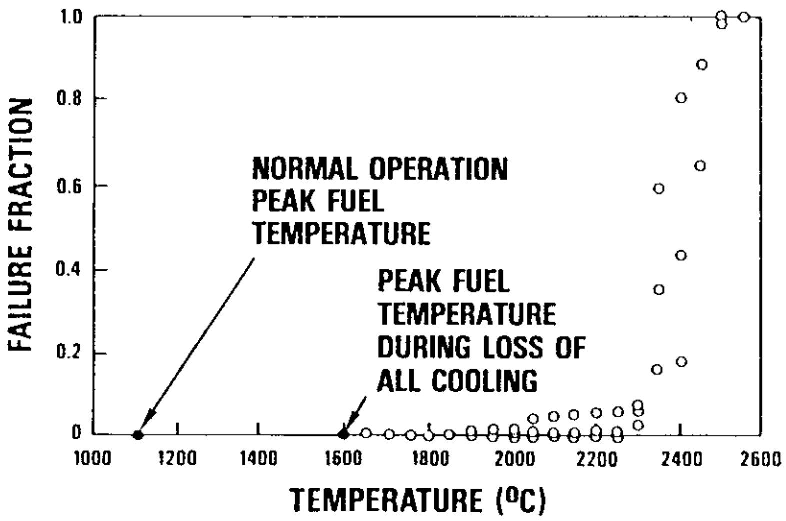


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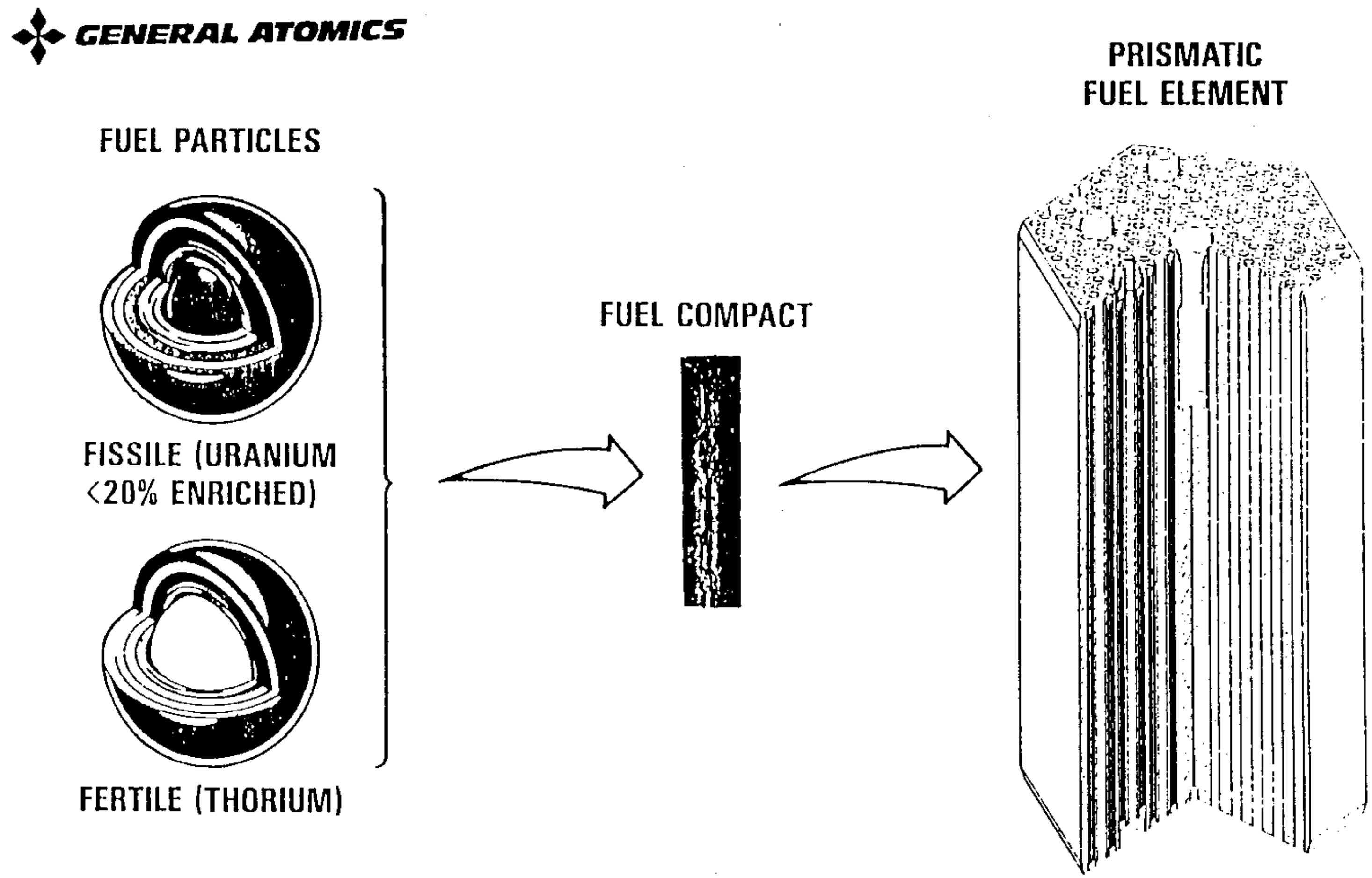
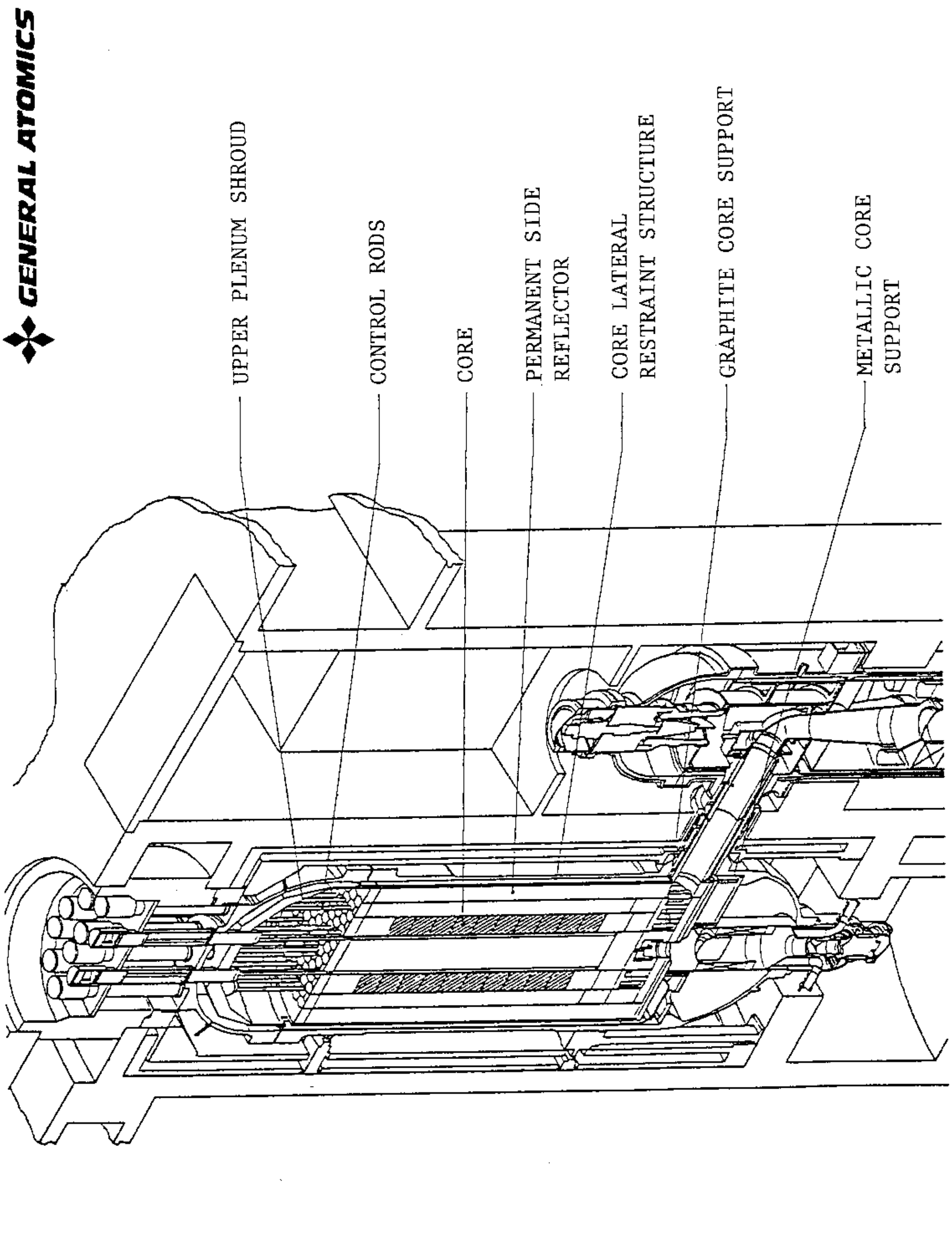
Figure 3



FUEL ELEMENT COMPONENTS

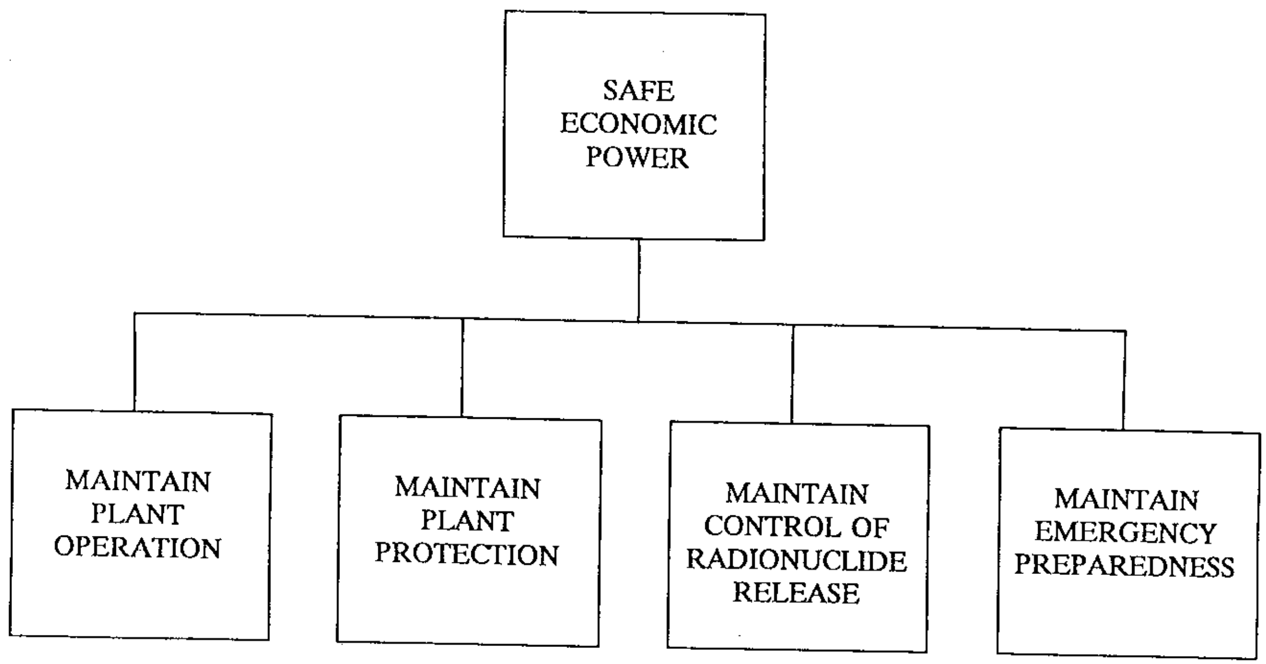
Figure 4



REACTOR CORE AND INTERNALS ARRANGEMENT

Figure 5

ORNL-DWG89-6373



Top-Level Functional Structure

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Figure 6

ORNL-DWG 88-8737

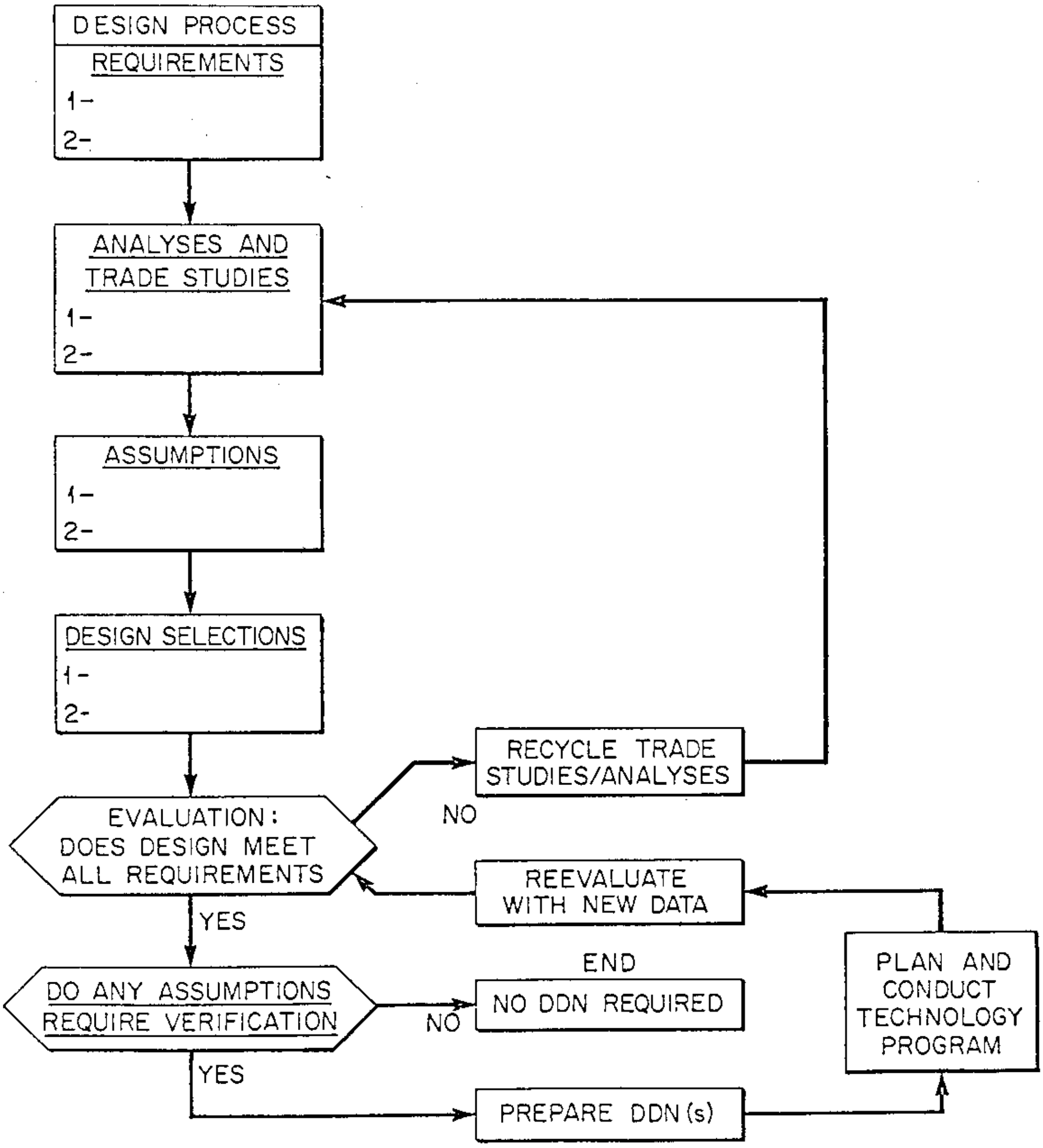

Figure 7

	FSVR	LHTGR	MHTGR Design Limit	MHTGR Expected Values
Fuel quality/performance requirements				
Beginning of life (BOL)				
Contamination (exposed U/total U)	1E−4	1E−4	2E−5	1E−5
SiC defect fraction	1E−3	1E−3	1E−4	5E−5
Fraction with missing buffers			1E−4	5E−5
^{85}Kr R/B	3E−5			
End of life (EOL)				
Failure rate (%)	0.8	0.1	2E−4	5E−5
Failed SiC (fraction)				
^{85}Kr R/B	3E−4			
Design basis accident				
Incremental failure fraction			6E−4	1.5E−4

ORNL/VG18

Fig. 8 FUEL QUALITY/PERFORMANCE EXPECTION

3.6 RESEARCH AND DEVELOPMENT REQUIREMENTS BEFORE AND BEYOND LICENSING THE HTR-MODULE

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1. Status of HTR-Development in FRG

Since nearly 30 years research and development work for high temperature reactors has been carried out in the Federal Republic of Germany. This work was always oriented towards the ideas and plans of plant design and application of the high temperature reactor: These were at first AVR, THTR and large steam cycle plants. In the seventies, when the plant designers took into account the direct cycle and the process heat application, the research program followed these advanced techniques. Today, the program supports the modern HTR design with steam cycle, the HTR-Module and the HTR-500. Despite of these changing aims of reactor design and application the R+D-program went on continuously and every task was performed up to a clear result. This has led to a high level of maturity of HTR development and to a broad knowledge in the field of HTR physics and techniques:

- Basic physical phenomena are well understood.
- The limits of materials and components are explored.
- The necessary data and methodological tools for plant design and for licensing procedures are available.

This program is controlled and managed by the "Entwicklungsgemeinschaft HTR" and is carried out by the Research Center Jülich, KFA, and the Reactor Companies, Hochtemperatur-Reaktorbau and Interatom /1/. Important contributions have been provided by universities and other research institutes in addition and the program is embedded in an international R+D cooperation all over the world.

For the success of this program it was significant, that a permanent support was given by the government, namely by the Federal Republic of Germany and additionally by the state government of Northrhein-Westfalia. This allowed for long term planning and execution of large part programs like e.g. the fuel element development or very principal investigations on fission product retention and graphite reactions with water and air.

Looking at the special question of R+D for licensing the HTR-Module, one should have in mind that in Germany 25 years ago the AVR-reactor has been licensed. At that time, there were no nuclear rules or official safety criteria to define the reliability of the safety systems, the necessary redundancies or more general the means for nuclear safety. Therefore, the constructors had much room for a safe design according to their own ideas. Without knowing the words "passive" and "inherent" in context with nuclear plants they used as far as possible the special safety properties of HTR's. With high appreciation of physical-technical relationships they succeeded to construct a very safe plant as it has been proven by 21 years of successful operation. According to younger expert opinions it fulfils basically the todays safety requirements.

Since rather exactly 10 years we have now the concept of the German modular HTR. The designers claim, that it is based on all the experience gained from the R+D-program and especially from the AVR. So, something should be wrong if now we should find, that a big program is necessary before licensing this reactor.

During these ten years of HTR-Module development the concept has been subjected to several safety reviews the first time in 1983 by an expert committee constituted by the Federal Minister of the Interior (BMI) and the last time recently by the Reactor Safety Commission (RSK) which advises the Federal Minister of the Environment Nature Conservation and Reactor Safety (BMU). The RSK could base its assessment on a draft safety document of the vendor, a comprehensive safety evaluation by the Technischer Überwachungsverein e.V. Hannover, which had been involved as technical expert in the interrupted site-independent licensing procedure of the state Lower Saxonia, and a small-effort probabilistic safety analysis carried out by KFA/ISF in 1983/84 /2/ and updated prior to the consel of the RSK. All these safety reviews and evaluations came to a very positive conclusion. There was no doubt that the safety concept is suitable to meet the safety related licensing requirements in the FRG. This includes explicitly those items of the safety concept which differ from current solutions for LWR, in particular the enclosure system for radioactive substances and the passive after-heat removal system. For instance RSK stated that it had no objections against the so-called confinement concept since it is in principal suitable to match the regulations of the German Radiation Protection Ordinance for normal operation and for design basis accidents. RSK also dealt with beyond design basis accidents and attested favourable safety features to the HTR-Module in this accident regime.

2. Main R+D Field related to Safety and Licensing

With these statements we are shure, that indeed only a small R+D effort really is necessary before licensing the HTR-module. This chapter will show the status we have reached and the points where work still has to be done. However, we should realize, that the dominating influence on our total R+D program results not from the licensing question. Like in any other technical area the development is going on steadily: novel materials and techniques, higher general safety requirements and the advanced reactor designs inspire and require additional investigations. The results hopefully lead to simpler components and plant design, to more economical solutions and to safety improvements and proofs mainly in the hypothetical area.

2.1 The Fuel Element and its Fission Product Retention

The design of the HTR-Module is based to a high degree on the fuel quality. In manufacturing, this fuel quality has been achieved in the eighties and has since been further improved. This improvement consisted of steps to reduce TRISO particle defects during pressing and in minimizing uranium contamination in the fuel element. Another sphere important aspect was the development of a sensitive quality control method; the burn-leach technique provides the necessary data to establish a design limit of less than 6×10^{-5} free uranium which has been identified as arising from particle defects.

For normal-reactor conditions, irradiation testing has been performed in material test reactors and in operating HTRs. Parameters such as heavy metal burn-up, operating temperature, and fast neutron fluence are varied to assess fuel performance. Continuous monitoring of released fission gases during irradiation tests gives a direct indication of the integrity of fuel coatings. In the German program, relevant irradiation tests with 212,000 particles were performed without a single coated particle failure. Statistically, this result corresponds to a 95 % confidence level that the coating failure fraction is less than 2×10^{-5}. As final step for the licensing procedures proof tests in the HTR in Petten are under preparation. The manufacture of proof test fuel has been completed under near-production scale conditions. With 1.4×10^{-5} defective SiC fraction, this production fuel has one of the highest quality standards ever achieved with HTR fuel elements /3/.

Quality assurance has also been completed. The irradiation rig for a test under Module-spezific conditions (Fig. 1) is under preparation. The test not only will simulate the multipass scheme typical of the reactor, but it will also cover all anticipated transient conditions. In the planned postirradiation work, several spheres will be subjected to 1600° C heating tests.

Fuel testing under off-normal conditions has provided fuel performance information as a function of fuel temperature, up to 2500°C. In small, modular HTRs, temperatures are limited to below 1600° C. Here, the fuel does not suffer irreversible changes and continues to retain all safety-relevant fission products. Experiments are performed with higher temperatures, longer heating times and with fuel from highly accelerated tests to establish the performance margins under accident conditions. A long-term program is also improving the statistical significance of the data base and narrowing the uncertainty limits.

Beyond the licensing question, this program will explore, if somewhat higher temperatures can be taken into account during accidents. If one realizes that only a very small fraction of the fuel reaches the highest temperature levels and that this occurs only during a short time, one might perhaps think of higher power rating. This can be supported by reducing the uncertainties in the temperature predictions as will be shown later.

In the frame of a basic research program for HTR investigations with coated fuel elements are carried out. Such coatings, e.g. consisting of SiC, shall provide a corrosion barrier in the case of severe water and air ingress accidents . Thus, the fuel element might cover even very extreme hypothetical accidents and reduce or possibly even eliminate the safety function of passive plant components. Coatings of still unirradiated fuel elements have given first promising results. Nevertheless it has to be realized that a high effort with certain developing risks has to be undertaken before the design of the plant and the accident analysis can be based on these novel fuel elements.

2.2 Fission Product Behaviour

As regards the accidental release of fission products from the HTR-Module to the environment the contamination of the fuel sphere graphite by uranium and of the surfaces of the primary circuit by fission products released during normal operation from coated particles with defective coating plays an important role. The graphite contamination determines the release of iodine from the core in the case of a depressurization accident. The vendor assumes that this contamination stems from the natural contamination of the graphite and not from the particles with defective coating and that it consequently consists of natural instead of enriched uranium. This must be proven and guaranteed by quality assurance since otherwise the maintenance of the planning guide values of the German Radiation Protection Ordinance for design basic accidents cannot be guaranteed without active measures. In this context the retention of iodine on colder graphite components is also important and needs further investigation.

The surface contamination by fission products particularly of the steam generator exhibits the largest contribution to the source term in the case of a water ingress accident (Table 1). The reaction of the deposited fission prodcuts with steam and water called steam-off and wash-off, may result in a partial entrainment of the fission products into the gas phase. A release into the environment would happen if the pressure in the primary circuit exceeds the set-pressure of the safety valve or if the dump line of the steam generator fails to close after demand /4/. There are discrepancies in the results of steam-off and wash-off experiments which require additional experimental work. This must include the investigation of the depositon of relevant nuclides on steam generator materials as well as of the stripping-off by the exposure to water and steam, both under realistic boundary conditions. The latter means for instance that the wash-off and steam-off experiments must be carried out under pressure to achieve the necessary temperatures. Samples from AVR are to be included in this experimental work. It must be complemented by the investigation of the size of the wettable surface of the steam generator. A computer model is being developed for this investigation.

Another aspect related to water ingress accidents is the oxidation of UO_2-kernals of defective coated particles by the reaction with steam. Theoretical investigations have shown that the release of iodine by fuel oxidation cannot be neglected (Table 1). In order to clarify the significance of the oxidation the heating tests of burned-up spheres will by expanded to tests in a steam atmosphere.

The picture would not be complete if one forgets to mention the graphite dust contaminated by fission products. The significance of the dust is probably reduced in the HTR-Module since the charging system as the main producer of dust will be equipped with a dust trap. But this does not mean that the problem has been solved fundamentally. Further experimental and theoretical work are necessary to clarify the formation, contamination, deposition and remobilization of dust in a pebble bed HTR. The AVR post-decommissioning program places emphasis on this subject.

2.3 Steel-Pressure Vessel

Based on the good experiences with Light Water Reactors, the pressure vessels for the HTR-module will consist of the optimized steel 20 MnMoNi 55. The materials data, the manufacturing, the design methods and the operational procedure will be in accordance with the German nuclear codes and standards, KTA rule 3201.1 to 3201.4 and the RSK-guide lines. Compared to the pressurized water reactor the HTR has lower pressure and thinner walls, which suggests a design with low stresses. There is no additional anticorrosive stainless steel cover.

Since the standards were developed for LWR's specifically, some moderations and supplementary stipulations for materials and design calculations are unavoidable:

- The pressure test concept and inservice inspection has to be adapted due to the HTR-specific pebble bed core, graphite structures and the fiber insulation. This will require no extra experimental work.

- Regarding the lower wall temperatures, we evaluated the effect of neutron irradiation on possible embrittlement of the core-pressure vessel: The nominal operation temperature of the section exposed to the highest neutron fluence (0_S ($E > 1$ MeV) $= 9 \times 10^{21}$ m^{-2}) is expected to be 195° C.

Neutron irradiation embrittlement of the material 20 MnMoNi 55, exposed to HTR irradiation environment at that temperature differs from the well known and evaluated shift of the nil-ductility-transition (NDT)-temperature during neutron irradiation at the LWR-specific operational temperature level of 300° C. There are some indication that the transition temperature for impact ductility is slightly increased, the level of fracture toughness somewhat more reduced at HTR-conditions. The flux level seems to have little influence on the embrittlement: embrittlement values abtained with a flux of 6×10^{15} n/m^2s were same as with 3×10^{17} n/m^2s at 150° C /5/.

To examine the irradiation effect under HTR Module specific conditions in more detail, a small irradiation program is being prepared. This program could be carried out within 3 years and would fit with a possible licensing procedure.

2.4 Other Safety Relevant Components

These are primarily the components which ensure the shut-down of the reactor and the protection of the pressure boundary from too high pressure.
The six reflector rods, which serve for the control of the reactor power and the hot shut-down largely correspond with the design of the THTR-reflector rods. They can be considered as proven components. Therefore, there should be no need for additional research work. But it must be proven yet - and this is also valid for the second shut-down system - that the components particularly the actuators do not become to hot under passive heat removal conditions. This is more or less a question of design. As regards the second shut-down system it is to be expected that the functionality and reliability must be proven at least for the first plant. The RSK-Subcommitee HTR acknowledged that the release mechanism of the system by the closed-circle current principle is very reliable.

The pressure boundary of the primary circuit is protected from overpressurization by two trains of safety/block-valves. There are no extraordinary requirements to be met. Therefore there is no need for specific research work.

2.5 Passive Heat Removal

The most interesting safety feature of the German HTR-Module is the abondonment lack of an active afterheat removal system. Heat transfer to the environment by physical phenomena is sufficient to keep the maximum temperature of the fuel below any damaging limit. It was very important that this concept was accepted in the site-independent licensing procedure of the state lower saxonia and just now by the reactor safety commission as mentioned before.

The computer codes for calculating the heat transport and the temperatures, especially the optimized two dimensional code THERMIX, take into account all mechanisms like natural convection, heat conduction and heat radiation. THERMIX includes the latest available data for the effective heat conductivity and is verified in laboratory experiments like LUNA /6/.

Experiments have been performed for normal accident as well as for conditions. In case of depressurization the heat is mainly transported by heat conduction and thermal radiation which is described by an effective thermal conductivity of the pebble-bed core. The effective thermal conductivity in packed beds of spheres has been measured for the high temperature range up to 2000 K in a special test facility. The influencing physical properties like annealing effects of the matrix graphite, the surface emissivity, and the pebble-bed porosity are updated for modern HTR fuel elements.

Solid cornerstones for the safety proofs of passive heat removal are the LOCA-experiments with the AVR plant. In these tests, after plant shutdown and depressurization, the reactor was operated at 4 MW to establish a steady initial state whose temperature profile was approximately identical to that during full power operation. The actual experiment was initiated by stopping the blowers. The afterheat meanwhile decayed was simulated by fission power as if the accident were to occur during full power operation.

Thermohydraulic behaviour typical of an HTR was especially demonstrated by the two experiments conducted for a period of more than 100 hours in October 1988. The measuring points next to the core centre in the graphite noses at core mid-height showed a maximum temperature increase of 300 K established 13 hours after experiment initiation. A computer simulation of the temperature redistribution during the tests is shown in figure 2.

The experimental results with open and closed main circuit valves are almost congruent, which shows that natural convection through cold gas recycling does not contribute towards heat removal in a depressurized reactor. Prior to the experiments in October 1988, monitor spheres had been added for temperature measurement in the core. At the time of the experiment, they were in the region of the highest temperatures. These spheres meanwhile are discharged and show maximum temperatures between 1070 and 1085°C. First results of corresponding calculations indicate somewhat higher temperatures and a slower decrease; i.e. the codes give conservative figures.

This leads to the question, whether beyond the needs for a licensing procedure additional specific experiments could be helpful: In a large scale, preferable 1:1, facility the effective heat conductivity could be measured under realistic conditions and geometries to reduce possible safety margins regarding the maximum fuel temperature. As a result, the power rating of the HTR-Module, and thereby the economy, could be increased. At the moment, consideration are going on in Germany, whether such demonstration experiments are worthwhile and where they could be done.

2.6 Air and Water Ingress

Air ingress is of special concern since it is the only mechanism in the HTR-Module which may have the potential for a release of large quantities of fission products from the core /7/. A presupposition for this mechanism is the access of unlimited quantities of air to the hot core. This requires large openings in the pressure boundary of the primary circuit and in the primary cell; events, which are extremely unlikely and thus fall into the beyond design accident regime. Nevertheless it is necessary to consider such events in order to prove either that they do not result in catastrophic consequences or that these can be avoided by accident management measures. Research work is being carried out in both directions. A computer code (REACT/THERMIX) exists which was successfully applied to chimney draught scenarios in the past. But it needs improvements particularly in the modelling of the transport phenomena under natural convection and of the corrosion at the periphery of the pebble bed. This work must be supplemented by validation experiments and by corrosion experiments for advanced graphite materials. An additional experiment is in preparation to investigate the resistance of the SiC-coating of burned-up particles under the condition of unlimited air ingress. Tests of unirratiated particles revealed that the SiC-coating remains intact.

Water ingress which may be caused by a leak in the steam generator has two additional aspects apart from remobilization of deposited fission products (see section 2.2), and these are the increase of reactivity and the reaction of the steam with graphite forming burnable gases. The reactivity effect is kept so small in the HTR-Module by choosing a low heavy metal content that it can be balanced by the automatic control system. But there is the need for an experimental verification of the effect for low enriched fuel. This experiment will also take into account the effect of an uneven distribution of the water content in the core. A code (TINTE) is being developed for the coupled treatment of neutron physics, thermodynamics and graphite corrosion. The latter is determined amongst others by the inhibition of the reaction by the reaction products particularly hydrogen. Neglecting this inhibition can mean that ignitable concentrations of burnable gases can occur in the reactor hall in the safety valve of the primary circuit opens. Experimental data are available for a rough assessment of the inhibition effect (Fig. 3). This is considered to be sufficient for the time being; new experiments will only be carried out if the formation of equitable mixtures become a great issue.

Large quantities of water mostly in liquid form can enter the primary circuit, if the steam generator is not dumped errornously in the case of a tube leak.

Assessments /2/ revealed that such an event would not result in fundamentally new accident scenarios, but there is the need for a computer code which is capable to describe water transport phenomena in the primary circuit under natural convection.

2.7 External Events

The degree of protection of the HTR-Module against external events is primarily a question of the safety concept and of the design. According to the safety concept only the reactor building and its interior will be designed against rare external events such as pressure wayes from gas cloud explosions and air craft impacts. The committee "civil engineering" of the RSK had no doughts that the requirements concerning the design against external impacts can be matched. Additional research work is only necessary related to beyond-design events such as earth quakes exceeding the safe shut-down earth quake (SSE). The latter is for instance part of investigations in the framework of a probabilisticly oriented safety analysis and is restricted to theoretical work. The aim of this work is to clarify the contribution of beyond-design earthquake intensities to the risk where by the frequency of the earthquake intensity and the probability of the failure of important components are taken into account.

3. Final Disposal of Spent HTR Fuel

Similar to other countries, the operation of a new nuclear plant in the FRG is approved by the competent authority only, if a clear, acceptable treatment route for the radioactive waste, especially for the spent fuel can be shown. Corresponding to the time period to be regarded, different levels of assuring this route have to be passed:

The storage at the reactor site is part of the licensing procedure for the plant itself.
The intermediate storage from the reactor needs, of course, an independent licensing procedure. Under todays circumstances, it is impossible to obtain an operation approval for a nuclear plant without producing a site and license for a storage facility sufficient for the planned operation time of the plant.

For the final disposal, a concept must be shown which is technically proven by throrough investigations i.e. presents the state of the art. At least one site should be available which fulfills the necessary conditions.

Direct final disposal, as mentioned before, without reprocessing (fig. 4) is taken as the reference case for spent HTR fuel in Germany /8/. Indeed, especially for the low enriched fuel, this line offers many advantages:

(Table 2): HTR LEU fuel allows high burn-up figures beyond 80.000 MWd/t HM. In fact, fuel element burnup in the AVR mass testing is peaking at around 150.000 MWd/t HM. Both, 80 - 90% of the initial U 235 and the bred, fissionable plutonium are burned insitu. Generally, in terms of electricity generation, uranium consumption in HTR's without fuel reprocessing is roughly the same as in LWR's with fuel reprocessing. In addition, one has to realise that the isotopic composition of plutonium in spent HTR fuel represents a low neutron value. I.e from the economic viewpoint, the direct final disposal is very promising.

Of even higher importance are the technical aspects for the spent fuel treatment concept. The HTR fuel element, in that respect, bears many positive features. Most of the safety related properties, quoted in plant accident safety analyses, are likewise effective in intermediate storage and for final disposal of the spent fuel.

- The fuel particle coating, providing an effective long-term barrier against fission product transport, reduces the effort for additional engineered barriers

- The low power density allows for passive air cooling systems in intermediate storage facilities at an early stage after discharge. Furthermore, the disposal techniques developed for medium active waste forms with their relatively low heat generation can be applied for spent HTR fuel.

- The homogeneous graphite matrix of the fuel element minimizes any spent fuel conditioning effort.

- The corrosion resistance of both fuel element matrix and fuel particle coatings against repository related salt brines helps to simplify the final disposal packaging concept.

- The small dimensions of the spherical HTR fuel element allow for small and easy-to-handle equipment in intermediate and final storage.

Two basic technical solutions can be offered for the safe intermediate storage of the spent HTR pebble bed fuel: canister storage behind concrete shielding or canister/direct storage in shielded containers licensed for transportation too.

Both solutions are being applied for spent fuel from the AVR and from the THTR.

For the final disposal, it is decided to dispose of nuclear waste in deep geologic formations as the best possible option of thorough separation from the biosphere. For heat generating radioactive waste forms, salt rock was chosen as host medium because of its unique visco-plastic properties, which, by lithostatic pressure plus temperature as the driving forces, will ensure the natural barrier of the host rock to build up within a few years time after emplacement of the waste packagings.

Intermediate storage is practiced already and needs no essential additional R+D. The R+D work with respect to final disposal of spent HTR fuel covers the following main areas:

- Characterization of the radiological, mechanical and chemical properties of the spent fuel elements.

- Investigation of the mechanical and chemical behaviour of the HTR fuel under realistic repository conditions including the attack of salt brines.

- Development of a safe and economical packaging concept, which takes into account the fuel element properties and answers the long-term immobilisation of relevant nuclides.

- Development of a suitable emplacement technique in the salt rock from a conceptual design up to manufacture and in situ testing of prototype components.

In all areas, the necessary qualifying tests are under way with hot cell experiments and with field tests. All results so far confirm the chosen concept.

As an example, figure 5 shows a demonstration test with spent AVR fuel elements in unlined, vertical boreholes off the 800 m gallery in the ASSE salt mine. This in situ disposal test with a planned duration of 5 years will finish with retrieval and examination of the packages. The test is combined with a measurement program, which will render additional information on in situ package and borehole behaviour. A precursory simulation test has started 1989 for e.g. verification of moisture enhanced rates of convergence at relevant temperature.

It should be pointed out that other technical solutions for disposal than embedding in salt rock seem possible and could easily be developed for countries with a different infrastructure lacking suitable underground salt strata.

4. Conclusions

Due to a broad R+D program since more than 25 years including the successfull operation of AVR and THTR in Germany, the HTR system can be regarded as a mature technology. Since about ten years, the German HTR-Module concept is under investigation. Several risk and safety assessments have given evidence of the unique safety features of this system. Nearly all proofs for the licensing procedure could be given today; R+D tasks to facilitate the procedure have been identified and are under way. It is very clear, that licensing and construction of a HTR-Module will not fail due to missing R+D results.

Most of the ongoing German R+D program is dedicated to questions beyond the direct licensing procedure: Direct final disposal, control of hypothetical accidents, and improvements in economy are the main goals.

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Table 1 Sources and source terms (expected values) of the HTR-Module for water ingress accidents resulting in an opening of the primary circuit either by the failure to close of the steam generator dump line or by the pressure built-up in the primary circuit due to a complete failure of the gas purification plant.

		Radiologically Relevant Nuclides [GBq]		
		I-131	Sr-90	Cs-137
	Contamination Steam Generator	44	13.4	860
	Stripping-off by Water		2.2%	2.3%
	Stripping-off by Steam	100%		0.4%
Source Nr.	In Gas from:			
1	Stripping-off from Steam Generator	44	0.3	23
2	Oxidation of def. Particles	14		
3	Heating-up of Core due to Depressurization	560		
Release Path into Environment and Conditions of Primary Circuit	Steam Generator Leak -Dump Line- Roof of Building Complete Depressurization Gas-Discharge 100%	83	0.3	23
	Relief Valve- Reactor Hall- Stack No Depressurization Gas-Discharge 10%	5.8	0.03	2.3

Table 2 Utilization of low enriched uranium in HTR.

Reactor Concept	200 MW MODUL	1390 MW HTR-500
Burn up (MWd/t)	80 000	105 000
In situ utilization		
fissile U (%)	87	81
fissile Pu (%)	86	80

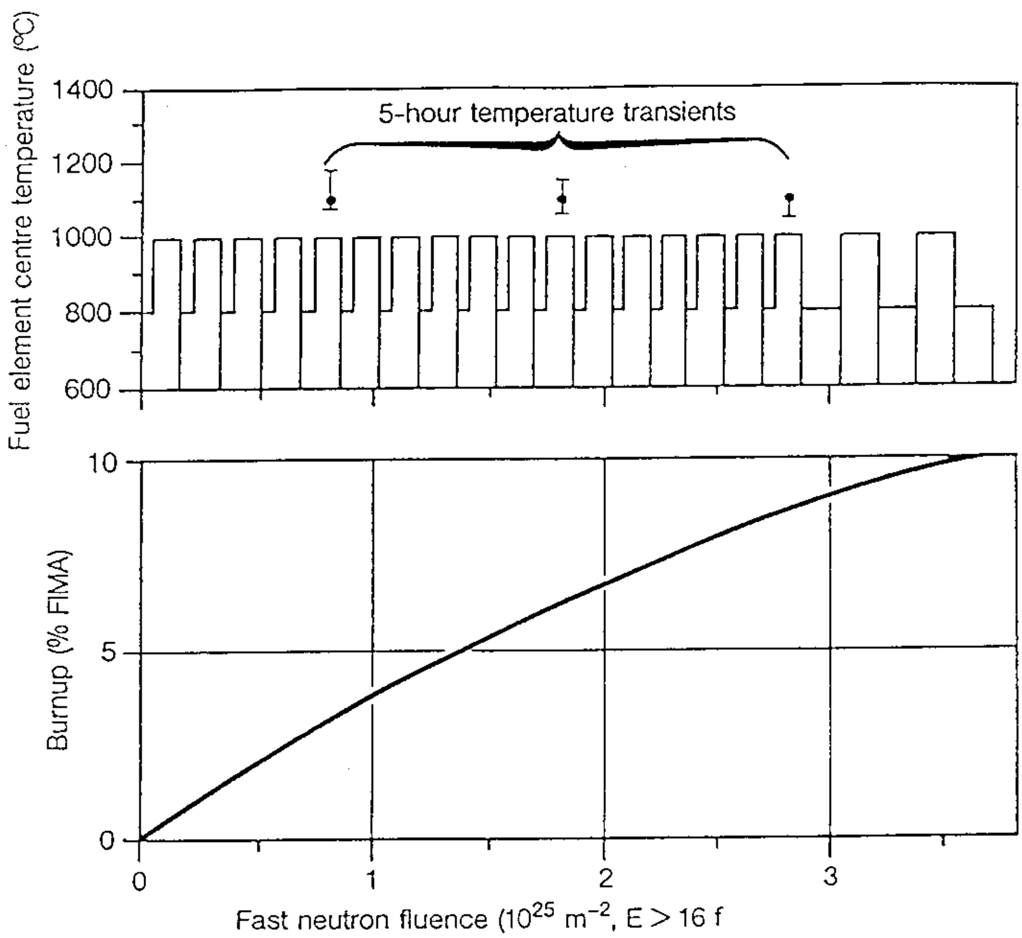


Fig. 1 Irradiation conditions in planned MODUL proof test HFR-K6 cover the requirements both by the 200 MWth steam-generating and the 170 MWth process-heat MODUL.

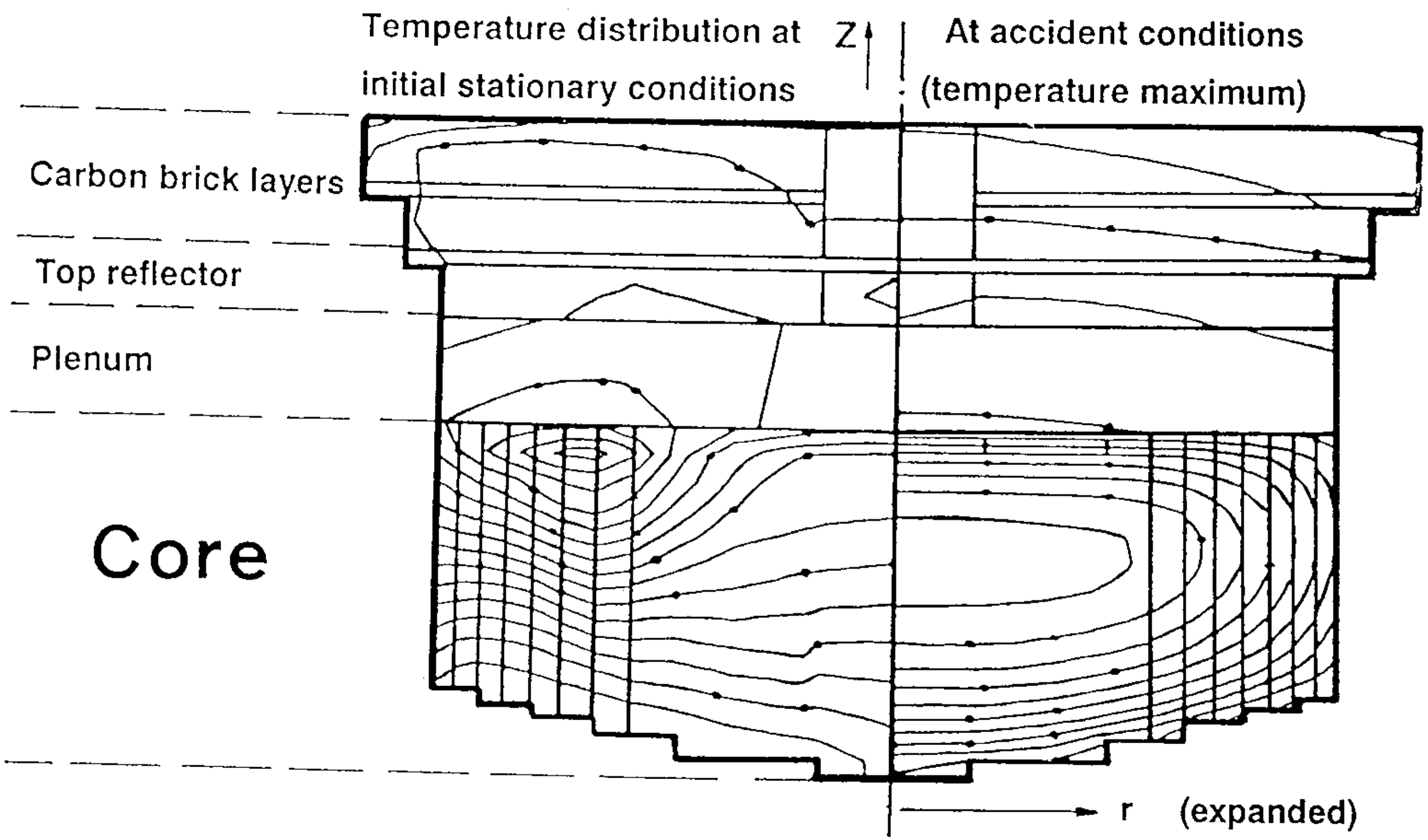


Fig. 2 Loss-of-coolant computer simulation. Isotherms calculated for the steady state of normal operation (left) and for the accidental condition (right).

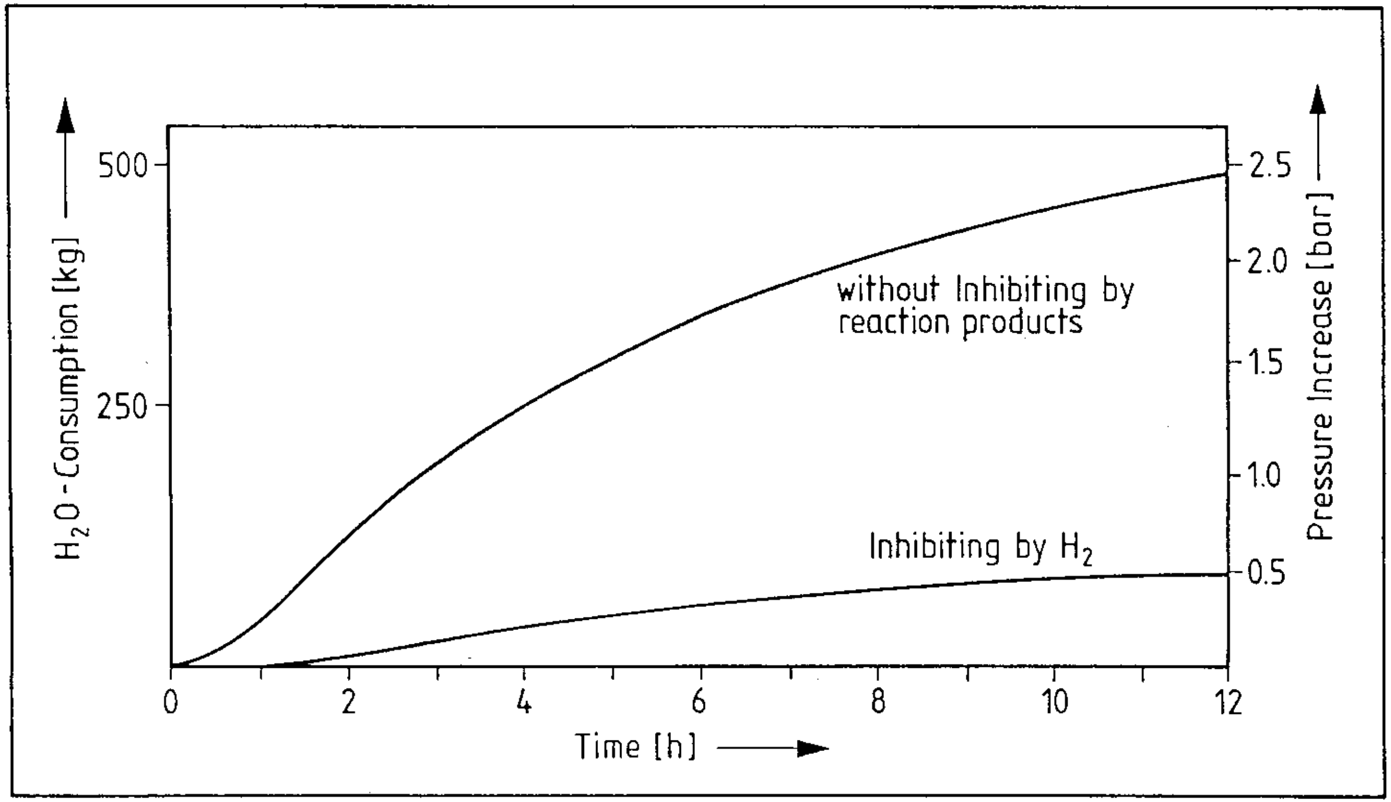


Fig. 3 Time dependent H_2O-consumption and pressure increase after the ingress of 600 kg steam into the HTR-MODUL with the blower valve open.

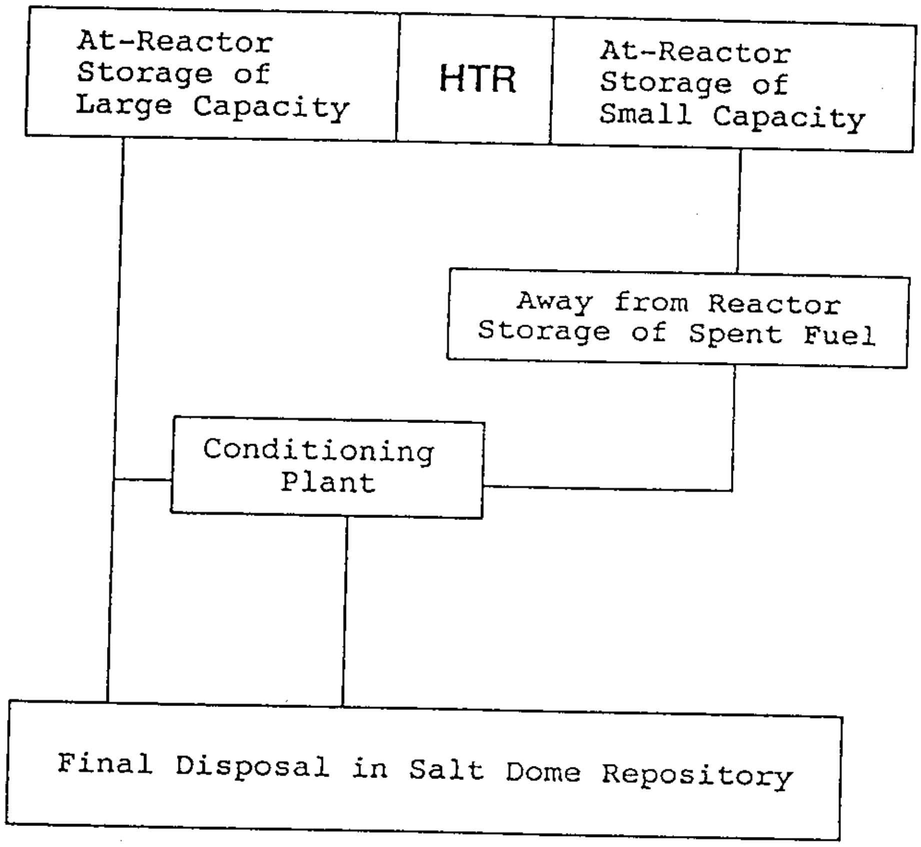
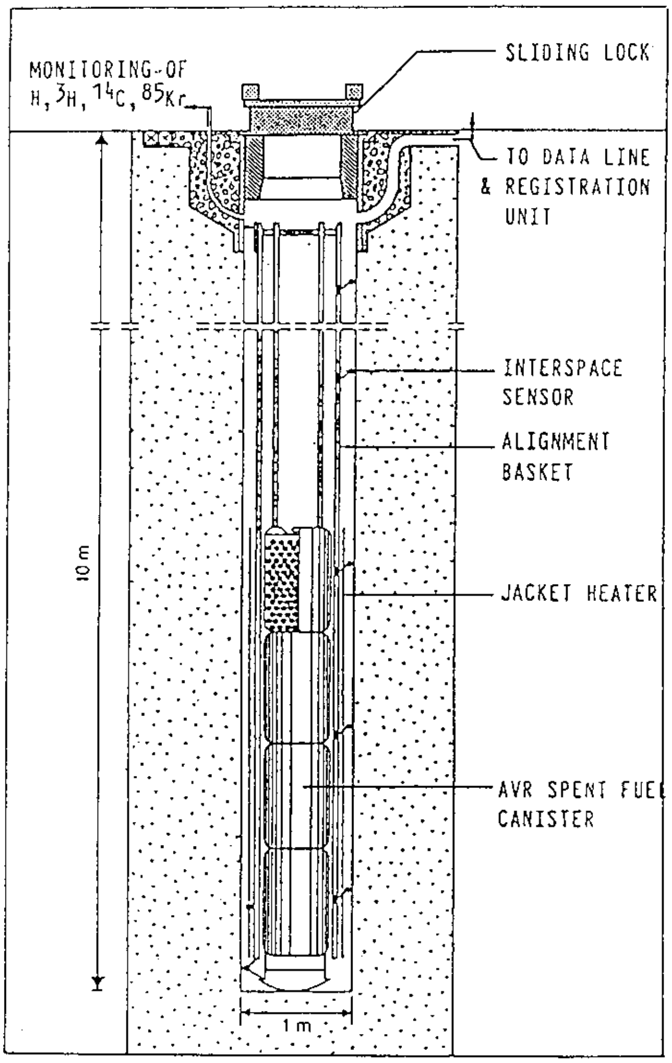
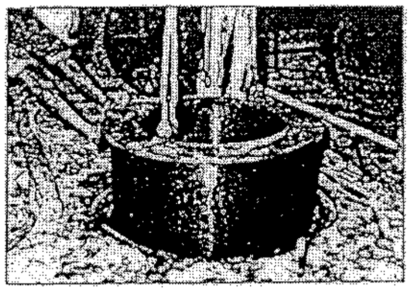


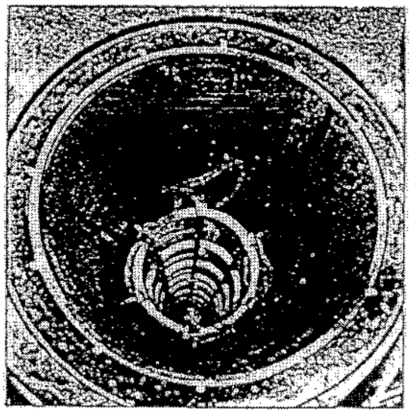
Fig. 4 Concepts for HTR spent fuel treatment



a) VERTICAL SECTION OF LOADED BOREHOLE



b) LOWERING THE SHIELDING COLLAR



c) VIEW ONTO INSTRUMENTED ALIGNMENT BASKET IN SITU

Fig. 5 Retrievable disposal test with spent AVR fuel elements in the ASSE salt mine.

CLOSING ADDRESS

Kazuo SATO
Executive Director
JAERI

Views were exchanged and discussed on the technologies relating to the HTGRs in this Symposium. Before summarizing the outcome of this Symposium, let me first confess very honestly that at the very early stage of preparation of this Symposium there had been a kind of pessimism as to how many people would have been interested, how many papers could have been expected to this Symposium and so on. It was a surprise, and I would say it was an encouraging surprise for us, that so much interest and enthusiasm were shown to this Symposium.

We have, by now, 240 participants beyond the capacity of this room, and many people joined the Symposium in the next room, looking at the TV screens. We have 27 participants from abroad from 9 countries and one international agency. One thing we deeply missed was that we could not have Soviet colleagues in due time by some reasons but we still note their contribution to the Symposium by submitting abstracts of their would-be presentations.

In the 1st session, we received three keynote addresses from Professor Mukaibo, Dr. Kupitz and Professor Schulten from respective point of view that delineated the multi-dimensional profile of the current status and future prospect of the HTGRs.

The following 2nd session was a really long one. We have 12 presentations covering the policies, planning, designs as well as experimental works. One remarkable and unique feature of this Symposium was that we had a precious chance to hear the views from developing countries such as Bangladesh, Indonesia and China. These were very good inputs for us all.

It appears to me that there exists much interest and expectation for HTGRs and recent progress in technologes has been quite remarkable and significant. However, HTGRs are not yet making a breakthrough for a large scale commercial deployment. Of course, such a breakthrough is not solely dependent on technologies. There are many other factors such as economics, social problems, public acceptance and so on. Nevertheless, technology is one of the vital components for the future of the HTGRs, and therefore, we should further make efforts to take advantages of the HTGRs such as inherent safety, high heat efficiency or very wide application area.

In the 3rd session, we received 6 presentations concerning licensing and relating R & D. In many countries the licensing, as well as relating R & D, is following the practices for light water reactors. While this situation is to some extent understandable, HTGRs have different features and different advantages from light water reactors. Therefore, there must be new views and new practices in the licensing process for HTGRs, while of course maintaining a consistent philosophy on nuclear safety. There are quite many research activities going on relating to the licensing. However, the need of R & D is not only for the licensing. It is more fundamental to solve very many issues and licensing issues are a part of thess to which we are now confronting.

In the last session, we had a panel discussion with distinguished panelists under an excellent chairmanship of Mr. Murata, who, as he himself said, was really the inventor of JAERI's HTGR project. Broad aspects were discussed and since our memory is still fresh, I would not repeat the excellent resume of Mr. Murata.

Through the whole program of this Symposium, we have felt from all of you high interests and very warm cooperation. Being so much encouraged, we are planning to have Symposia of this kind with adequate intervals as Mr. Murata suggested. We have currently a plan to have the second Symposium hopefully in late 1992 or early 1993. We are very much looking forward to seeing again you all, as well as those who could not come this time, at the 2nd Symposium.

Finally but not least, on behalf of the host organization, I would like to express our hearty thanks to all of you and related people, and look forward to seeing a very rapid and fruitfull development of High Temperature Gas-Cooled Reactors.

Thank you very much again and the Symposium is now closed.

APPENDIX Panel Exhibition with Brief Explanation on the Results of Research and Development for HTTR

CONSTRUCTION OF THE HIGH TEMPERATURE ENGINEERING TEST REACTOR

HTTR Development Division
Department of HTTR Project, JAERI

The HTGR development program has a long history in Japan. The JAERI started the HTTR development program in 1969 so as mainly to construct an experimental reactor for direct heat application, along with the Japanese industries. The energy situation, however, has changed remarkably in Japan during the last 20 years, then, in June, 1987, the Japan Atomic Energy Commission issued the revision of the Long-Term Program for Development and Utilization of Nuclear Energy, recommending that Japan should proceed with the development of more advanced new technologies for the future, in parallel with the existing nuclear systems. It also emphasizes that HTGR is one of the most promising reactors with high efficiency and inherent safety, therefore it should be explored for the broad use of nuclear energy, not only for power generation. Then, the construction of the HTTR was decided instead of an experimental HTGR. The HTTR aims at establishing and upgrading the technology basis necessary for an HTGR, serving at the same time as a potential tool for new and innovative basic researches.

The JAERI carried out the detailed design of the HTTR based on R&D results accumulated so far in Japan along with the environmental survey at the Oarai Research Establishment of JAERI as the HTTR site. The site and the facility arrangements of the HTTR are shown here. The HTTR reactor building is also shown.

Major objectives of the HTTR are (i) to establish basic technologies for advanced HTGRs in future and (ii) to utilize the HTTR as an irradiation test reactor in order to conduct researches in innovative high-temperature technologies.

The HTTR consists of a core of 30MWt, a main cooling circuit, an auxiliary cooling circuit and related systems. The reactor pressure vessel is 13.2 m high and 5.5 m in diameter and contains the core, graphite reflectors, core support structure and radial restraining devices. Major specification of the HTTR is shown in the table.

Schedule of the HTTR construction is shown in this figure. The first reactor criticality is expected to attain in 1995 and the operation in Phase I will last several years in order to conduct normal operation under the outlet coolant temperature of 850°C and high temperature test operation under the outlet coolant temperature of 950°C. During the operation of Phase I, a heat utilization system is also planned to be connected to the HTTR. Candidates of the heat utilization are the hydrogen production by IS process, the hydrogen production by high-temperature electrolysis of water and the stream reforming. The detail of the heat utilization system will be determined in a few years based on the evaluation program which is in progress in JAERI. A high performance fuel assembly will be used in Phase II.

(shown in Page 387)

STRUCTURES OF THE HTTR

HTTR Development Division
Department of HTTR Project, JAERI

Cooling system

The reactor cooling system is composed of a main cooling system (MCS), an auxiliary cooling system (ACS) and two reactor vessel cooling systems (VCSs). The reactor cooling system is schematically shown in the figure. The MCS is separated into two lines outside the reactor vessel. The heated helium gas is cooled in a He/He intermediate heat exchanger (IHX) in one line or cooled directly in a pressurized water cooler (PWC) in the other line. The heat is finally removed by an air cooler in both lines, while another PWC provided after the IHX in the first line. When the first line with heat transfer capacity of 10 MW is operated, the second line, which has heat transfer capacity of 30 MW, is operated at 20 MW. Coaxial double pipes are used for transferring hot helium gas.

The ACS consists of an auxiliary heat exchanger, auxiliary gas circulators and an air cooler. The heat transfer capacity of the ACS is about 3.5 MW and these components are similar to those of the MCS.

Each VC has heat capacity of 100% to cool reactor vessel and core in emergency.

Core

The reactor core is graphite-moderated, helium gas-cooled and hexagonal fuel elements are used. The active core consists of 30 fuel columns and 7 control rod columns, each column composed of 5 blocks (2.9 m high). Reactivity is controlled by control rods which are individually supported by the mechanisms located in stand-pipes connected to the hemispherical top head of the reactor vessel, and inserted into the channels in the active core and replaceable reflector regions. The reactor core is cooled by helium gas of 395°C at the reactor inlet temperature which flows downward through the core. The maximum fuel temperature in the normal operational condition is approximately 1500°C under the high temperature test operation with the reactor outlet coolant temperature of 950°C.

Fuel Elements

A fuel element assembly, 36 cm width across the flats and 58 cm in length, is made up of fuel rods and a hexagonal graphite block. The fuel consists of TRISO coated particles of low enriched uranium oxide whose average enrichment is about 6% and the sintered to form a fuel compact. These compacts are contained in a sleeve to form a fuel rod. The fuel rods of 3.4 cm in diameter are contained within vertical holes of a graphite block. Helium gas flows through a gap between a vertical hole and a fuel rod to remove heat produced by fission and gamma heating.

IHX

The intermediate heat exchanger is a vertical helically-coiled counter flow type heat exchanger in which primary coolant flows on the shell side and secondary coolant on the tube side as shown in the figure. the material of the shell is 2 1/4 Cr-1 Mo steel and Hastelloy XR is used for the boundary between the primary and secondary circuit. The heat transfer tubes are designed so as to withstand the differential pressures of 0.3 MPa between the primary and secondary helium in normal operational condition whereas it is designed to stand for a short duration of a pressure of 4 MPa in case of depressurization accident of secondary helium cooling system.

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RESEARCH AND DEVELOPMENT OF HTTR FUEL

Fuel Irradiation and Analysis Laboratory
Department of Fuels and Materials Research, JAERI

The HTTR employs pin-in-block type fuel elements (Fig. 1). Helium cools outer surface of the fuel pin which consists of graphite sleeve and fuel compacts. The Triso-coated UO microspheres are dispersed in the graphite matrix of the fuel compact. Volumetric loading of the coated fuel particles in the compact is 30%. Maximum fuel temperature is 1350°C nominal. The maximum burnup and fast neutron fluence in 660-days fuel life will be 3.6%FIMA and 1.3×10^{25}n/m^2 (E>29fJ), respectively.

The fuel is fabricated by a private company, Nuclear Fuel Industries, Ltd. (NFI) except the graphite components (Fig. 2). The sleeves and blocks which are made of fine-grained isotropic graphite, IG-110, are supplied by Toyo Tanso. Fuel qualities have been improved by the cooperation of JAERI and NFI. Heavy metal contamination, which is due to either fuel particles with through-coating failure or graphite matrix contamination, is kept lower than 1.5×10^{-4}, as-fabricated defective particle fraction is much lower than the design limit, 1.5×10^{-3}. The first-core fuel fabrication is scheduled in 1993-1996 by NFI with capacity of 400 kgU/year.

The HTTR fuel performance is studied in JAERI. Short-lived fission gas release has been studied in Japan Materials Testing Reactor in gas-swept capsules and the OGL-1 gas loop. In the OGL-1, the life-sized fuel pins can be irradiated. A data base on the short-lived noble gases and iodine behavior has been formed and applied to the in-situ diagnosis of fuel integrity. The recent tests confirmed the good performance of the fuel near the maximum fuel temperature and upto the maximum burnup of the HTTR (Fig. 3). The irradiated particles have been examined in the hot laboratories to generate the data on SiC corrosion, thermal and mechanical response of the fuel compact, etc. The SiC corrosion by fission-product palladium has been studied to propose a model where the rate-controlling step is assumed to be the release of Pd from the fuel kernel into the coating. In the light of experimental evidences, neither the diffusion through pyrolytic carbon layers nor the reaction at the SiC surface is regarded as rate-determining.

Behavior under abnormal conditions has been studied mainly by out-of-pile heating tests of the irradiated fuel particles. The particles are sampled from the irradiated fuel compacts by electrolytic decomposition of the latter. It has been confirmed that the significant failure occurs only above 1900°C (Fig. 4), and no failure was observed in the prolonged heating at 1600°C. Metal fission products are retained by the coating up to 1700°C (Fig. 5). Above 2100°C, however, the coating becomes pervious to the metal fission products such as Cs, because the SiC dissociates rather rapidly and open pores penetrate the grain boundaries. A mechanistic model of coating failure, in which both the internal pressure buildup and the SiC thermal degradation are taken into account, has been developed. The model successfully predicts the loss of coating integrity to the metal fission products and the ultimate coating failure. The latter occurs when radial displacement of the outer pyrolytic carbon layer by thermal creep reaches about 10%, following the mechanical failure of the SiC. It was found from the measurements of ^{85}Kr release and the x-ray radiographs that the pyrolytic carbon retains gaseous fission products for a short time after the SiC mechanical failure.

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MECHANICAL CHARACTERIZATION STUDIES ON GRAPHITES AND FERRITIC STEELS FOR HTTR

Materials Strength Laboratory
Department of High Temperature Engineering, JAERI

INTRODUCTION

The HTTR reactor design necessitates new advanced materials such as nuclear grade graphites and ferritic steels to be utilized for core, core iternals and pressure boundary components. Materials Strength Laboratory (MSL) has been involved in establishing mechanical property data bases in order to select, qualify and specify these materials.

1. Mechanical Design Features of the HTTR Reactor Components

The HTTR reactor components possess features in structural design aspects. There are two types of graphite components. One is a replaceable component like a prismatic fuel element block. The other is a permanent component like a hot plenum block. Particular attentions have to be placed on whether the effect of neutron irradiation is significant or not and whether the sizes of original blocks are important or not.

Metallic components such as pressure vessel, primary circuit pipes and core internal components are also subjected to neutron irradiation at ~400°C. Special care must be paid on materials degradation during the long-term service life. These metallic materials are 2.25Cr-1Mo steel (JIS SFVA F22B, JIS SCMV4) and 1Cr-0.5Mo-0.3V steel (JIS SNB16). Although the materials have been used widely for the reactor vessels in petrochemical industries, several specific materials characterizations have been needed in the licensing for nuclear applicaton. Fracture toughness values such as dynamic fracture toughness K_{Id} and crack arrest fracture toughness K_{Ia}, neutron irradiation embrittlement behavior and material properties in helium environment are necessary items to be investigated.

2. Mechanical Characterization of Graphite Materials

1) Three kinds of graphites and carbon were selected for the individual graphite components: grade IG-110 for core components, grade PGX for support components and grade ASR-ORB carbon for insulation blocks.

2) Extensive mechanical tests have been performed to characterize the selected materials with respect to deformation and fracture behavior. A representative fatigue data analyzed statistically on IG-110 graphite is shown in the left Figure. Fatigue data could be characterized by plotting a maximum applied stress normalized by mean tensile strength as a function of number of cycles to failure (N_f).

3) Effects of fast neutron irradiation and oxidation on mechanical properties including creep coefficients have been determined by a series of irradiation tests including the cooperative tests with KFA Jülich. Irradiation creep coefficients of IG-110 graphite are shown in the right Figure together with those of other nuclear graphites.

4) Test results have been utilized for establishing the graphite structural design codes and the design material properties of the individual graphite materials.

3. Mechanical Characterization of Ferritic Steels

The flow chart in the bottom Figure shows the scope of work which MSL has done for clarifying the mechanical properties of the ferritic steels under service conditions. Degradation of material strength have been investigated through thermal and stress aging tests up to 5×10^4 h. Fatigue tests were performed focusing on the effects of creep and helium environment on fatigue life. Neutron irradiation tests were conducted in order to evaluate the toughness degradation. The results of these tests have been considered in alloy selection. Main results are as follows:

1) Through thermal and stress aging tests, alloy compositions with low Si content and with a reduced P level of less than 100 ppm are specified for the pressure vessel materials.
2) Fatigue properties have been systematically evaluated including oxidation, helium and creep effects.
3) The degree of embrittlement by neutron irradiation have been evaluated. The ductile-to-brittle transition temperature (DBTT) shift would be about 10°C and the upper shelf energy will decrease by about 20% of the initial value after the design lifetime.

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HIGH-TEMPERATURE ALLOYS FOR HTGR APPLICATIONS

Material Performance and Testing Laboratory
Department of Fuels and Materials Research, JAERI

1. Introduction

Studies on high-temperature metallic materials for high-temperature gas-cooled reactor (HTGR) applications have been made in Japan Atomic Energy Research Institute (JAERI) for more than 15 years. The items of research activities carried out are classified into three different categories, i.e., those for near-term target structural alloys, for the long-term target heavy dust structural alloys and cladding materials of neutron absorber rods as summarized in the table (top).

2. Near-Term Target Alloy (Hastelloy XR)

Comprehensive qualification tests such as creep, fatigue, corrosion and other fracture-relevant properties on Hastelloy XR have been carried out in order to accumulate the test data for structural design and safety evaluation.

The figure (middle left) shows the status of on-going creep tests with a complete record of deformation characteristics in JAERI-type B helium environment (20 Pa H_2, 10 Pa CO, 0.1 Pa H_2O, 0.2 Pa CO_2 and 0.5 Pa CH_4), some of which are exceeding 3×10^4 hours. A test data base has already been prepared for the performance in air up to 2.4×10^4 hours. One of the important observations is that no statistically significant difference has been recognized between the helium and air environments in the creep rupture performance. Improved oxidation resistance of the alloy is believed to have contributed to the results, and such a feature is a special advantage in prediction of the long-term performance in the service environment.

As for fatigue tests the effects of strain rate, hold time and aging on fatigue properties have been studied in JAERI-type B helium environment. The figure (middle right) shows the fatigue life as functions of test temperature and total strain range. It can be seen from the figure that the fatigue lives of the aged specimens are shorter than those of the solution annealed ones at test temperatures ranging from RT to 700°C (except one datum point at 500°C with a total strain range of 0.4%). The tendencey becomes more pronounced under higher strain range conditions. At 800 and 900°C, no significant difference in the fatigue strength was resolved between the solution annealed and the aged specimens. Reductions in the fatigue life due to the aging treatment observed through the tests conducted at and below 700°C are considered to be closely related to the loss of ductility, because generally speaking the low-cycle fatigue strength is dependent on the material's ductility.

As for corrosion tests it was confirmed that Hastelloy XR still revealed the stable corrosion characteristics in maintaining protective oxide integrity without severe spallation or intergranular attack even after exposure to the helium environment at 900°C for 3×10^4 hours under severe thermal cycling.

For further improvement and optimization of the specification of Hastelloy XR, a high creep strength version with the additions of B by 40 to 60 ppm has been established, i.e., Hastelloy XR-II. The chemical composition of Hastelloy XR-II, however, falls still within the standard specification of Hastelloy XR.

The filler metal for Hastelloy XR has been also developed successfully with special emphasis placed on the manufacturing process.

3. Long-Term Target Alloys (Ni-Cr-W Superalloys)

R&D on Ni-Cr-W superalloys as a component material at service temperatures around 1000°C have been promoted by JAERI with the cooperation of the authorities from a number of industrial and academic organizations. The basic composition, Ni-18.5%Cr-20.5 to 21.5%W, of the experimental alloys was specified in the first step.

As the next step, effects of the further alloying elements, i.e., C, Nb, Fe, Mn, Si, Ti, B and Y, on creep-rupture strength, tensile property, hot workability and corrosion resistance have been investigated. Based on the experimental results, the final composition has been proposed.

After the heat treatment conditions are fixed with the performance tests, qualification tests as well as piping and welding tests will be conducted on an industrial scale heat.

The figure (bottom left) shows the comparison of newly developed Ni-Cr-W superalloy (predicted values for the above-mentioned industrial scale heat) with Hastelloy XR and EARNS alloy.

4. Cladding Material of Neutron Absorber Rods (Alloy 800H)

In the HTTR Alloy 800H is used for the cladding material of the neutron absorber rods. For the generation of engineering data, post-irradiation creep tests have been carried out. Creep tests on un-irradiated Alloy 800H have been also carried out to examine the effects of the chemical environment and thermal aging. The figure (bottom right) shows the results of the first series of the post-irradiation creep tests as well as the creep data for unirradiated Alloy 800H. The creep rupture lives of irradiated Alloy 800H were 10 to 30% of those of unirradiated one, and there was no significant difference in the creep strength among three environments.

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REACTOR INSTRUMENTATION

Reactor Instrumentation Laboratory
Department of Reactor Engineering, JAERI

1. Nuclear Instrumentation

In the field of nuclear instrumentation of HTGR, a Wide-Range Monitoring System (WRMS), High-Temperature Gamma Compensated Ionization Chambers (HTCICs), High-Sensitive Gamma Uncompensated Ionization Chambers (HSUICs) and Wide-Range In-Core Fission Counter-Chambers (WIFCs) have been developed under the HTTR development program of JAERI.

The WRMS includes High-Temperature Fission Counter-Chambers (HTFCs) and has a monitoring range of about 10 decades of reactor power. The HTFCs, the neutron sensors of the WRMS, can be operated in high temperature (e.g. 800°C) and high gamma-ray environment (e.g. 1.1×10^6 R/h). The overall operating performance tests of the WRMS, including accelerated irradiation tests and a long-term in-reactor operating test of 3 years at 600°C, were carried out using reactors of JMTR, JRR-4 by JAERI, HFR (EURATOM) and AVR (Germany) under KFA-JAERI cooperation. One of the test results is shown in the panel.

The HTCICs were also developed and tested for more than 3 years in the JRR-4 at 400°C, 500°C and 600°C.

The HSUIC was developed for power range monitoring and the protection system of the HTTR. As the neutron flux density outside the pressure vessel of the HTTR is very low and estimated at 10^{11}n/m^2.s (about 1/500 of that of PWR), a special high-sensitivity-performance is required for the HSUIC. A ^{3}He gas is used as an ionization gas of the HSUIC to get a high neutron sensitivity. The sensitivity is 5×10^{-16}A/(n/m^2 .s), i.e. 5×10^{-12}A/nv, and the HSUIC has 3 sensitive sections to make it possible to monitor an axial flux distribution.

The WIFC with an outer diameter of 10mm was developed for measurement of in-core neutron flux distribution in HTGR. It has a measuring range over 8 decades at 800°C and was used for a KFA-JAERI experiment for measuring neutron flux in a steam generator region above the AVR core.

2. High Temperature Measuring Instrumentation

A new type of thermocouples, so-called Nicrosil/Nisil thermocouple (N-type TC), seemed to be the most suitable sensors for in-core gas-temperature measurement in the HTTR. The actual experience on the use of N-type TCs is, however, very little so that a long-term out-pile test of them having different types of sheaths was started at a high-temperature He-gas environment. The N-type TCs under the test have sheaths with four different materials; i.e., inconel 600, incoloy 800, hastelloy X and Nicrosil. The insulator is MgO in common. The test is being performed at the temperature of 1200°C. The results of 1200°C test are shown in the panel. N-type TC having Nicrosil sheath seems to have the least amount of emf drifts, so far forth. The test will be continued for the total test hours of more than 10000.

Noise thermometer (NT) has been developed and joint experiment was carried out in cooperation between KFA and JAERI. Irradiation tests of an NT system developed by KFA were carried out in reactor JMTR in JAERI (1984-1985). JAERI's NT system will be tested using high temperature testing furnaces in KFA (1990).

3. Fuel Failure Detection System

In an HTGR, development of a Fuel Failure Detection (FFD) system is important to detect a fuel failure within a small scale for the purpose of the reactor's safety operations and core management. The behavior of noble-gas fission products (FPs) released in the primary coolant gas is very complicated because the FPs are released into the coolant helium gas even during normal operation as background FPs. Moreover, the amount of the FPs changes considerably depending on various factors such as fuel temperature, reactor power, burn-up rate of fuel and et cetera.

We have developed an FFD system using a wire-precipitator and performed fundamental experiments using FGS (gas sweep capsule) and OGL-1 in the JMTR for the evaluation of its performance. The FFD system for the HTTR was developed on basis on selective detection of short-life noble-gas FPs and a state function for estimating the effect of back-ground due to long-life noble-gas FPs in the primary circuit. By comparison with the measured value of the precipitator and background estimated by the state function, it is determined whether fuel is failed or not.

As a result of the experiments, the fuel failure rate of 0.04% can be detected by this FFD system. Moreover, the dependency of precipitator counting rates on the fuel temperature and the reactor power was obtained by analyzing measured data. By integrating these results, a state function for the fuel failure detection is determined.

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STRUCTURAL INTEGRITY OF CORE & HIGH TEMPERATURE COMPONENTS

Thermal Structure Laboratory
Department of High Temperature Engineering, JAERI

1. The Coolant Flow In Core

A crossflow through the interface gap of fuel elements and a leakage flow at seal mechanisms between hot plenum blocks have a significant impact on the core thermal hydraulic design. The crossflow rate and the pressure drop were measured for parallel- and wedge-shaped gap configurations over the range of expected widths, and also analyzed using a finite element method. A very good agreement was found between calculations and experiments, and the empirical crossflow equation was devised to analyze the core flow distribution.

A new seal mechanism, which consists of graphite seal elements with a triangular cross section placed on the V-shaped seats of hot plenum blocks, was proposed to reduce the leakage flow effectively. The pressure loss coefficient factor of this seal mechanism is very stable and predictable under the various conditions of hot plenum blocks.

2. The Core Structure

A safety assessment of the support post structure required an evaluation of fracture strength of the posts and a vibrational characteristics in earthquakes. Stress states of the support post under the reactor core weight were analyzed by means of a photoelastic experiment and compared with the results of the finite element method. Fracture criterion of the support post was proposed experimentally.

Vibrational characteristics of the support post structure were evaluated by the forced vibrational tests and analyzed by the Lagrange's vibrational equation. Seismic behaviour in the core model was simulated by the modified SONATINA-2V code which could take into consideration of the vibrational characteristics of the post structure. Then the impacts force to the graphite components such as key-keyways, dowel pins and plenum blocks were estimated to assure the strength of fracture.

3. The High Temperature Structural Design

An assessment of the high temperature structural design is required especially for long term life prediction in Hastelloy XR at elevated temperature regime, following the construction and the operation of 1.5 MWt intermediate heat exchanger by the ERANS. Lots of efforts have been paid to furnish a creep constitutive equation, rupture time and creep ductility under the conditions of multiaxial stress states. As long as creep-fatigue life prediction was concerned, the design work was done under the criterion of linear damage summation rule based on the time to rupture on constant-loaded creep tests, which has a margin for life comparative with the results based on the constant-stressed creep. It is expected to formulate the more satisfactory methods to predict the creep-fatigue life in service, reflecting fracture mechanisms of the material at high temperatures, not only for base metal but also for welded one.

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THERMOHYDRAULIC STUDY ON SAFETY

Heat Transfer Laboratory
Department of High Temperature Engineering, JAERI

Thermohydraulics of a fuel rod cooling channel and a control rod cooling channel under the conditions of the normal operation and transient events have been studied in a high temperature gas cooled reactor. Thermohydraulics and mass transfer on reactor safety in the accidents have also been studied. Recent researches in relation to HTTR (High Temperature gas cooled Test Reactor) are summarized in Table of the panel.

In the following, thermohydraulic study is described on safety during air ingress accident. The accident starts by guillotine rupture of the primary double concentric pipe at the outlet of the reactor vessel. Then the helium gas coolant with a pressure of 4 MPa blows up into the reactor containment vessel (RCV). After this depressurization down to a pressure of about 0.5 MPa, the gas mixture of air in the RCV and the helium gas coolant enters the reactor core from the breach of the primary coolant pipe because of buoyancy force resulting from temperature difference between the hot channels of the reactor core and the cold inlet passage, as shown in Fig. 1 of the panel.

The corrosion of hot graphite materials by oxygen in the gas mixture during the air ingress might lead to the consequences of (1) reduction in mechanical strength of graphite, (2) release of fission products, and (3) production of inflammable carbon monoxide. Accordingly, in the air ingress accident it is necessary to examine the flow rate of the gas mixture entering the reactor core and the corrosion rate of graphite especially at the temperature higher than about 900°C for evaluation of the amounts of the corroded graphite and the generated carbon monoxide in the safety analysis of the HTTR design.

Figure 2 in the panel shows the experimental and analytical results of the flow rate in addition to the result of the safety analysis in the HTTR design. The analytical result obtained by a numerical calculation code agrees well with the experimental one. The time delay from the pipe rupture until the onset of the massive air ingress is caused by the fact that the buoyancy force in the reactor vessel is not sufficiently large to initiate the natural circulation of the gas mixture through the reactor vessel. As the time elapses, the buoyancy force increases gradually because helium gas in the reactor vessel exchanges grandually with the gas mixture oxide. Finally the natural circulation of the gas mixture takes place and the massive air ingress into the reactor vessel begins. The flow rate used in the safety analysis is found to be conservatively overestimated in comparison with that predicted by the code.

It is known that mass transfer of oxygen in a gas mixture to graphite surface is the dominant process to determine the corrosion rate of the graphite at the high temperature. Since the mass transfer data so far obtained are insufficient in the graphite corrosion at the high temperature, mass transfer correlation, from which the corrosion rate is calculated, is obtained from heat transfer correlation on the assumption of analogy between heat and mass transfer in the safety analysis of the HTTR design. Experimental data for Sherwood number, which is a dimensionless number of mass transfer coefficient, are lower than the correlation used in the safety analysis as shown in Fig. 3 of the panel. Therefore it is confirmed that the conservative corrosion rate is employed in the safety analysis of the HTTR design.

Safety evaluation on HTTR verified that all the safety criteria for the air ingress accident are satisfied by the evaluation based on the conservative analytical conditions.

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HELIUM ENGINEERING DEMONSTRATION LOOP (HENDEL)

HENDEL Development Laboratory / HENDEL Operation Division
Department of High Temperature Engineering, JAERI

1. Purpose

The HENDEL is a test facility to perform large scale demonstration tests of high temperature components for the HTTR. The tests are performed by using full scale model of the components and test data obtained have been used for the safety review of the HTTR.

2. Major Specifications

The HENDEL building is about 30m 90m of length and 35m of height, and located at sourthern part of the Tokai Research Establishment of JAERI. The HENDEL consists of Mother(M), Adapter(A) and Test sections(T) as shown in figure of schematic flow diagram. The Mother and Adapter (M+A) section is able to circulate helium gas at the maximum flow rate of 4 kg/s, the maximum pressure of 4MPa and the maximum temperature of 1000°C, as shown in the first Table of major specifications.

3. Development of Loop Technology and Its Component

The Mother and Adapter (M+A) section was constructed in March 1982, and it has been operated for about 14000 hours. Through the operation of HENDEL, technologies necessary for helium circulation at high temperature, high pressure and large flow rate are developed. Performance of helium purification system is investigated. Above all, the development of supporting technology of a rotating shaft for helium circulator is the most important matter. Characteristics of gas bearing type circulator were made clear, and research and development of magnetic type circulator and reciprocating high speed compressor have been performed. The bottom right picture shows a magnetic type reciprocating compressor which is used for advanced purification system. The bottom left figure shows a relation between rotational speed and amplitude of a rotor for gas bearing and magnetic bearing, respectively. It is found that magnetic bearing has excellent characteristics and is more stable than gas bearing.

4. Reactor Component Tests

The high-temperature components to be tested in HENDEL are (1) fuel stack and (2) in-core structure.

(1) Fuel stack test section (T_1 test section)

It consists of a single channel (T_{1-S}) and a multi-channel (T_{1-M}) test rigs. Thermal and hydraulic test of simulated fuel rod, as shown in the top center figure, and control rod and verification test of HTTR control rod drive mechanism (CRDM) were performed with T_{1-S}. Thermal and hydraulic tests of the fuel stack including crossflow tests have been carried out with the T_{1-M}. The top left figure shows a relation between Nusselt number and Reynolds number for simulated fuel rod of the HTTR. There are no significant difference among heat transfer coefficients (Nusselt numbers) obtained with three types of power distribution set in simulated rods. Moreover, these data is about 15% higher than those in smooth annuli, and there is no transitional flow region between laminar region and turbulent region. The reason is mainly because simulated fuel rod has three spacer ribs in the azimuthal direction as shown in the figure, which may give diturbance to gas flow.

(2) In-core structure test section (T_2 test section)

The T_2 test section, as shown in the bottom right picture, is to simulate the core support structure of the HTTR, such as the plenum block, the core support post, the insulation layer, the graphite reflector, and the outlet gas duct. The bottom left figure shows a relation between leakage flow rate and pressure difference between inside and outside of permanent reflector blocks. Present data obtained after 6500 hours' operation agrees very well with the data in the functional test performed just after the installation of the T_2 test section.

No detectable change in the leakage flow rate has occurred since the T_2 test section was constructed. This indicates that there is no enlargement of the gaps induced by irregular dimensional changes of blocks or by degradation of the core restraint band mechanism for the permanent reflector blocks.

The bottom center figure shows temperature distribution in the core bottom structure. The temperature of the metal support plate was below the design value of 500°C due to the cooling by the cold helium gas and the carbon insulation blocks.

(shown in Pages 396 and 397)

REACTOR PHYSICS RESEARCH FOR HTTR

Thermal Reactor Physics Laboratory
Department of Reactor Engineering, JAERI

1. Experimental Program at Very High Temperature Reactor Critical Assembly

Experimental research on reactor physics of HTGRs (High Temperature Gas-cooled Reactors) started in 1971 through critical experiments at the SHE (Semi-Homogeneous Experiment) of which fuel element was a homogeneous mixture of 20% enriched UO_2 and graphite. As the design of the HTTR was going into details, mock-up experiments were required. A new critical assembly VHTRC (Very High Temperature Reactor Critical assembly) was constructed by modifying the SHE to obtain experimental verification for the neutronic design accuracy of the HTTR.

The VHTRC reached critical on May 13, 1985. The mock-up core experiments for the basic design verification of the HTTR were carried out in simple core geometry. Since 1988, a more detailed mock-up core has been investigated for the detailed design verification. The experimental items are divided roughly into three categories: reactivity balance (critical mass and reactivity worth of control rods and burnable poison rods), power distribution and temperature effect on core neutronic characteristics. Kinetic parameters β_{eff}/Λ and fuel lattice parameters are also to be measured.

2. Main Features of VHTRC

The VHTRC is a low-enriched uranium fueled and graphite moderated critical assembly. The assembly has a shape hexagonal prism (2.4m across the flats and 2.4m long) and it can be horizontally split into two equal half assemblies of which one is fixed and the other is movable. The assembly is entirely covered with thermal insulator and can be heated up to 200°C by using 40 electric heaters (30kW in total) to study the temperature effect; a single fuel rod can be heated up to 800°C using a simple heating device which is inserted into the central column.

The core of VHTRC consists of pin-in-block type fuel. A fuel rod is a stack of fuel compacts packed in a graphite sheath. The fuel compact is a hollow cylinder of coated fuel particles dispersed in graphite matrix. The uranium content is 21 grams per fuel compact. Three kinds of fuel compacts are prepared; they are classified by the uranium enrichment of coated particle fuel, i.e. 2%(BISO type), 4%(BISO type) and 6%(TRISO type). A great variety of core configuration come from the availability of these three kinds of fuel compacts as well as the loading pattern of fuel rods in a fuel block.

3. Initial Critical Experiment at VHTRC-1 Core

The initial critical core VHTRC-1, which is reflected both radially and axially, is loaded with 4% enriched fuel in the lattice pattern of twelve fuel rods per a fuel block where $C/^{235}U$ ratio is about 8600. After initial critical approach at room temperature, the whole assembly was heated up to 200°C and the 2 and 4% enriched fuel rods were added to the core to bring it critical again. The measurements for the following items were carried out.

(1) Critical mass at room temperature and 200°C.

(2) Neutron flux distributions at room temperature and 200°C.

(3) Reactivity worth of an HTTR mock-up burnable poison rod in the central column.

(4) Reactivity worth of HTTR mock-up control rods in the central column and the radial reflector column.

(5) Temperature coefficient of reactivity in the range from 25 to 200°C.

The experiments were analyzed with the SRAC code using the ENDF/B-IV nuclear data and good agreements between the calculations and the experiments were obtained. The results indicate that the present calculation method and nuclear data are in good accuracy for the fundamental neutronic design of the HTTR.

Four different cores having radial and axial reflectors were assembled so far to study the basic neutronic characteristics of the HTTR core. At present the experiments at the forth core VHTRC-4, which is loaded with 2, 4 and 6% enriched fuel in axially zoning pattern, are almost being completed. As the next step, experiments will be continued in a more detailed mock-up core into which many burnable poison rods are inserted.

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NUCLEAR HYDROGEN PRODUCTION

Department of Research (Branch Office at Tokai)
Takasaki Radiation Chemistry Research Establishment, JAERI

Since the awareness of decreasing fossil fuel resources in the 70's, an enourmous amounts of R&D's on the alternative energy systems have been carried out worldwide. The requirement of the alternatives is further emphasized by the appearance of environmental problems in the 80's. A concept of hydrogen production with nuclear heat has been recognized as one of the promising systems because of its potential cleanness and high thermal efficiency.

Methods so far proposed to realize the nuclear hydrogen production can be classified as follows, according to the source material of hydrogen and the degree of electricity utilization; 1) steam reforming of methane, 2) thermochemical process, 3) hybrid process and 4) high temperature steam electrolysis. The last three methods are for water splitting hydrogen production. In a thermochemical process, several endo-thermic and exo-thermic chemical reactions constitute a system, where net inputs are heat and water and net outputs are hydrogen and oxygen. In a hybrid process, an electrolysis reaction is combined with thermochemical reactions. For these two methods, many combinations of chemical reactions have been studied. The high temperature steam electrolysis is carried out with solid electrolyte, e.g., yttria stabilized zirconia at temperatures of ca. 1000°C, and aims at a reduction of electricity required compared with the conventional water electrolysis.

Thermochemical processes utilizing iodine and sulfur oxides as materials circulating in the process are well-studied ones (iodine-sulfur family). At JAERI, our group has been studying one version of the iodine-sulfur family composed of the following chemical reactions;

$$SO_2 + I_2 + H_2O = 2HI + H_2SO_4 \qquad (1)$$

$$2HI = H_2 + I_2 \qquad (2)$$

$$H_2SO_4 = H_2O + SO_2 + 0.5O_2 \qquad (3)$$

Reaction (1), the so-called Bunsen reaction, is carried out in an aqueous solution in a presence of excess iodine. The reaction is known to reach equilibrium very rapidly within a second. Hydriodic acid and sulfuric acid are produced separately in a two liquid phases, i.e., a heavy solution composed of HI, I_2 and H_2O (HI_x phase), and a light solution of sulfuric acid. Iodine in the HI_x is separated from hydriodic adic by distillation. Then, the hydriodic acid is vaporized and fed to a column packed with Pt/Active Carbon (Pt/AC) kept at 200°C for Reaction (2). Reaction (2) is carried out by batch operation, because the prodcut iodine is adsorbed on Pt/AC and, after iodine saturation, a desorption of iodine is necessary to regenerate the decomposition activity of Pt/AC. Reaction (3) is carried out at 800°C in a gas phase using Pt/Al_2O_3 or Fe_2O_3/Al_2O_3 catalyst. This reaction requires the highest temperature in the process, and is therefore assumed to be coupled with the high temperature heat source. At present, the feasibility of the process is studied with a bench scale cycle demonstration plant made of quartz and glass. Also, a flowsheeting of the process is going on to optimize it.

The equilibrium conversion of Reaction (2) is low (15% at 200°C) irrespective of pressure, and its dependency on temperature is small. The method using Pt/AC is one way to overcome the disadvantage of chemical equilibrium. With the method a one-path conversion of 60-70% could be attained owing to the adsorption of iodine on Pt/AC. In the studies on the iodine-sulfur family, many other ideas have been so far proposed for Reaction (2), most of which also try to shift the equilibrium to the favorable side of hydrogen production by removing the product iodine from the reaction zone. Researchers at General Atomic Co. proposed to carry out the decomposition at a composition of liquid-liquid phase equilibrium, where pure liquid HI is in equilibrium with $HI/I_2/H_2O$ mixture, using a solid catalyst (Pt/TiO_2) or a homogeneous catalyst (PdI_2). Knoche et al. at RWTH Aachen has proposed a reaction distillation of HI_x. At the top of distillation column, the equilibrium in the vapor phase is expected to shift largely to the hydrogen production owing to the absorption of iodine in an iodine-free azeotropic hydriodic acid. These efforts are encouraged by international co-operations, e.g., in the framework of IEA workshop on the hydrogen production.

Concerning the construction materials for the commercial plant, screening tests have been performed for the main reactors and candidates for the next step of material tests have been selected.

(shown in Page 399)

Panel Photographs

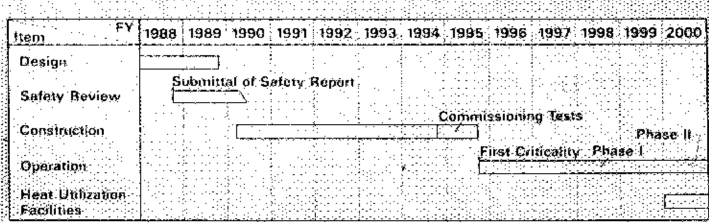
Construction of the High Temperature Engineering Test Reactor (HTTR)

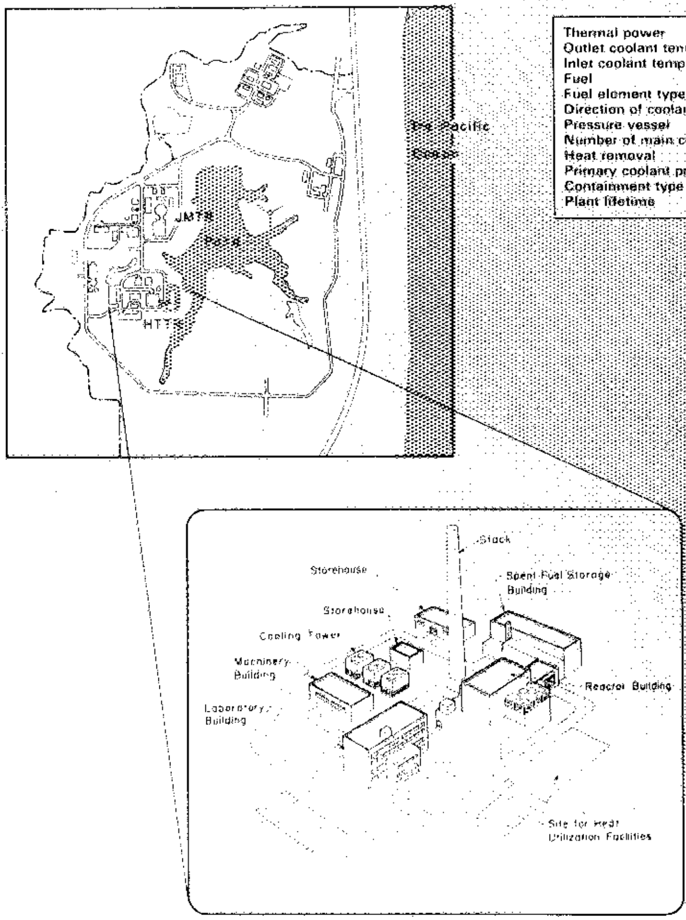
The HTTR aims at establishing and upgrading the technology basis necessary for HTGRs, serving at the same time as a potential tool for new and innovative basic researches.

Major specification of the HTTR

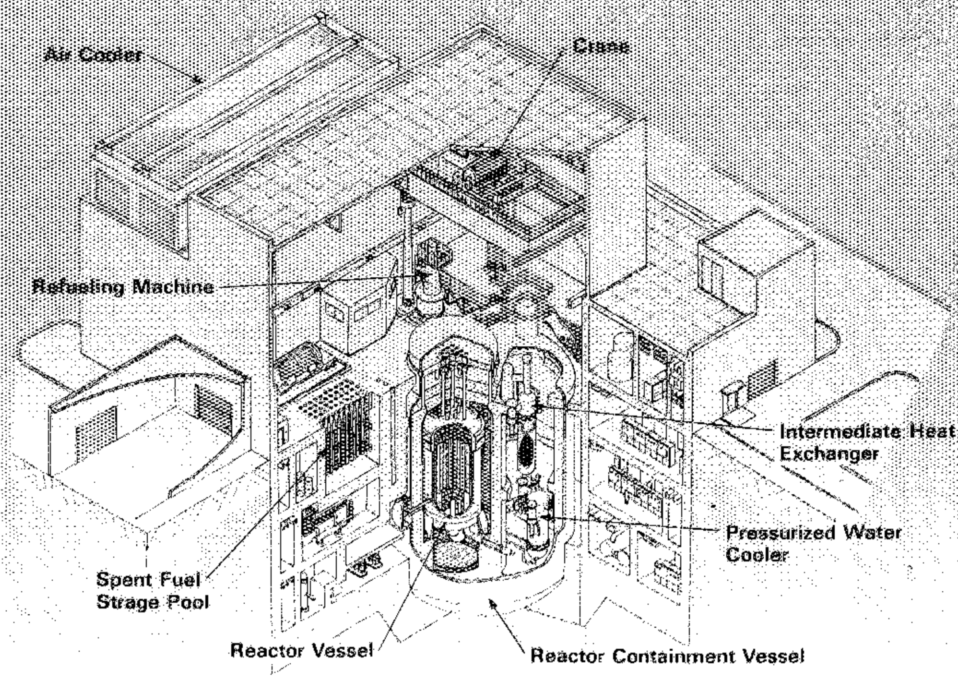
Thermal power	30 MW
Outlet coolant temperature	850°C/950°C
Inlet coolant temperature	395°C
Fuel	Low enriched UO_2
Fuel element type	Prismatic block
Direction of coolant flow	Downward-flow
Pressure vessel	Steel
Number of main cooling loop	1
Heat removal	IHX and PWC (parallel loaded)
Primary coolant pressure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

Schedule of the HTTR program



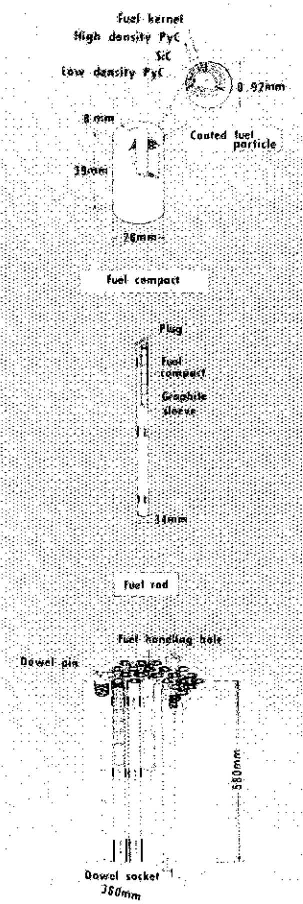


Location and the facility arrangement of the HTTR



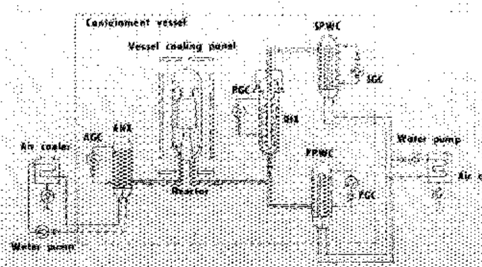
HTTR reactor building

STRUCTURES OF THE HTTR



Helium gas flows through a gap between a vertical hole and a fuel rod to remove heat produced by fission and gamma heating.

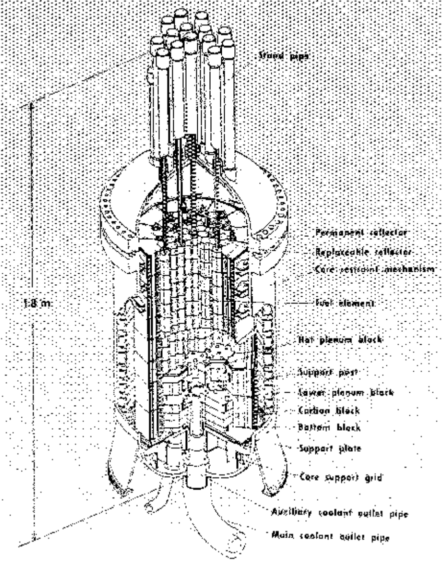
Block type fuel of the HTTR



IHX : Intermediate heat exchanger
PPWC : Primary pressurized water cooler
PGC : Primary gas circulator
SPWC : Secondary pressurized water cooler
SGC : Secondary gas circulator
AHX : Auxiliary heat exchanger
AGC : Auxiliary gas circulator

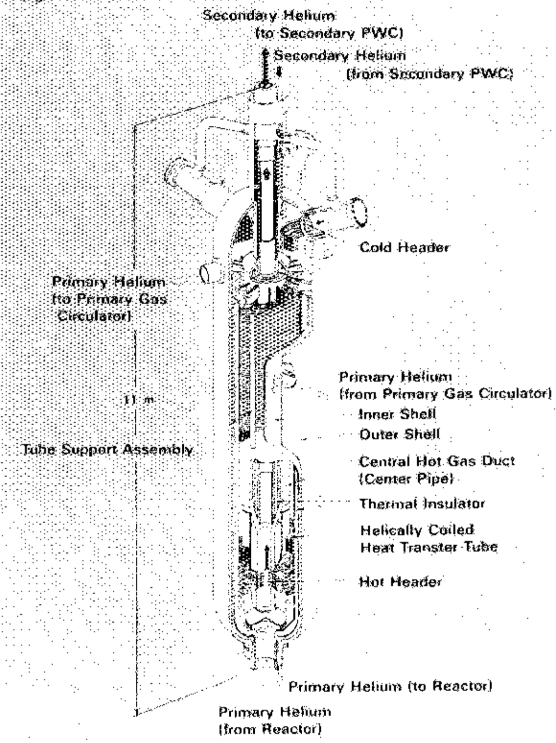
The primary cooling system is separated into two lines outside the reactor vessel. The heated helium gas is cooled in the IHX and the PWC. Coaxial double pipes are used for transferring the hot helium gas.

Cooling system of the HTTR



The reactor core is cooled by helium gas of 395°C at the reactor inlet temperature which flows downward through the core.

Bird's-eye view of the reactor vessel and the core of the HTTR



The intermediate heat exchanger is a vertical helically-coiled counter flow type heat exchanger in which primary coolant flows on the shell side and secondary coolant flows on the tube side.

Bird's-eye view of the He/He intermediate heat exchanger of the HTTR

RESEARCH AND DEVELOPMENT OF HTTR FUEL

[1] HTTR Fuel Specifications

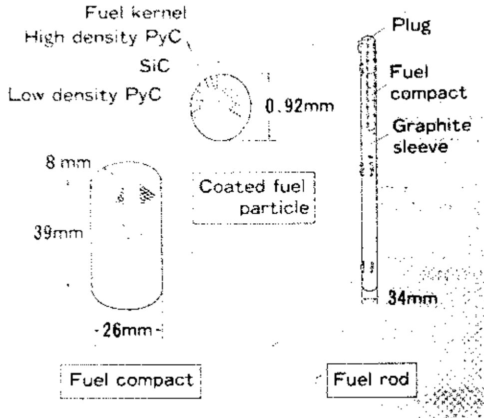


Fig. 1 HTTR fuel : dispersion of coated fuel particles in graphite matrix.

QUALITY

Due to fabrication
- Heavy metal contamination $<1.5\times10^{-4}$
- Defective particle fraction $<1.5\times10^{-3}$

Due to operation
- Defective particle fraction $<2\times10^{-3}$

REACTOR PARAMETERS

Temp. 1350°C max.
Burnup 3.6%FIMA max.
Fast Neutron (E>29fJ) 1.3×10^{25}n/m^2

[2] Fuel Fabrication Tests

FUEL FABRICATION

Nuclear Fuel Industries (NFI)
- Coated Fuel Particles
- Fuel Compacts

Toyo Tanso
- IG-110 graphite components

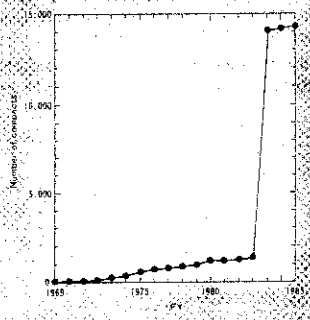


Fig. 2 Cumulative number of fuel compacts in fabrication tests

Present Fuel Quality

	Heavy Metal	Defective Particle
1987	3×10^{-5}	5×10^{-4}
1988	4×10^{-6}	4×10^{-6}

HTTR fuel fabrication is scheduled in 1993–1996 by NFI facility with capacity of 900kgU/year

[3] Fuel Performance Tests

PERFORMANCE UNDER NORMAL CONDITIONS

- Kernel migration, SiC corrosion
- Fuel compacts behavior
- Metal fission product release
- etc.

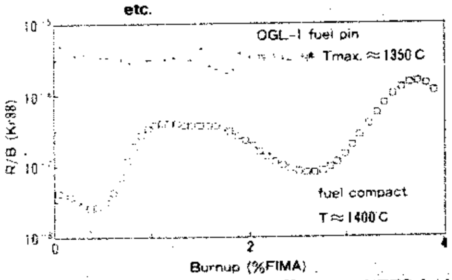


Fig.3 Release rate (R/B) of ^{88}Kr from HTTR fuel

BEHAVIOR UNDER ABNORMAL CONDITIONS

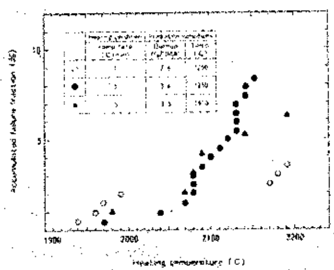


Fig.4 Cumulative failure fraction of coated particles in ramp tests

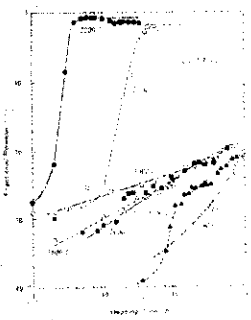


Fig.5 ^{137}Cs release in isothermal heating

MECHANICAL CHARACTERIZATION STUDIES ON GRAPHITES & FERRITIC STEELS FOR HTTR

Materials Strength Laboratory

1. MECHANICAL DESIGN FEATURES OF THE HTTR REACTOR COMPONENTS

1) Graphite Components

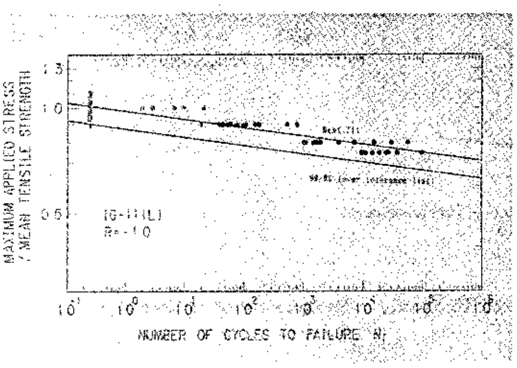
(1) Replaceable core components
: fuel and reflector blocks
- Hexagonal prismatic block
- RT-1250°C
- up to 1.3×10^{25} n/m^2 (E>29fJ)

(2) Permanent support components
: reflector and hot plenum blocks
- large-sized, irregular-shaped block
- RT-1100°C
- Negligible fast fluence

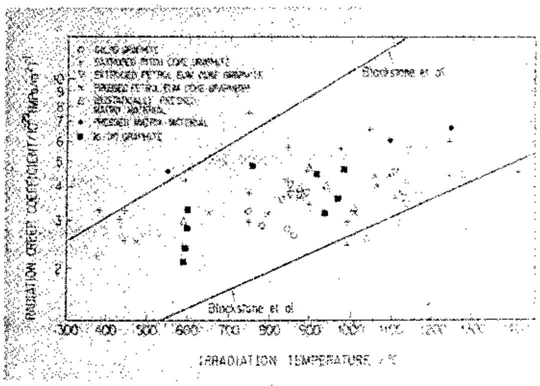
2) Metallic Components
: pressure vessel, primary circuit pipes, core internal components
- 400°C at the cold leg
- $\sim1\times10^{22}$ n/cm^2 (E>160fJ)
- 1×10^5 hours (EFPH)

2. MECHANICAL CHARACTERIZATION OF GRAPHITE MATERIALS

1) Three kinds of graphites and carbon were selected for the individual graphite components : grade IG-110 for core components, grade PGX for support components and grade ASR-ORB carbon for insulation blocks.
2) Extensive mechanical tests have been performed to characterize the selected materials with respect to deformation and fracture behavior.
3) Effects of fast neutron irradiation and oxidation on mechanical properties including creep coefficients have been determined by a series of irradiation tests including the cooperative tests with KFA Jülich GmbH.
4) Test results have been utilized for establishing the graphite structural design codes and the design material properties of the individual graphite materials.



Fatigue test data and the derived S-N curves



Irradiation creep coefficients of various nuclear graphites

3. MECHANICAL CHARACTERIZATION OF FERRITIC STEELS

Design Analysis
Safety Analysis

↔

Alloy Selection & Material Characterization

Ferritic Steels
2¼Cr-1Mo
1 Cr-0.5Mo-0.3V

Heat Treatment condition
: Normalized & Tempered
Alloy composition
: low Si, with P less than 100ppm

Mechanical Characterization Tests

(1) Thermal Aging Tests
400-500°C
~50,000 h

(2) Stress Aging Tests
450°C
~10,000 h

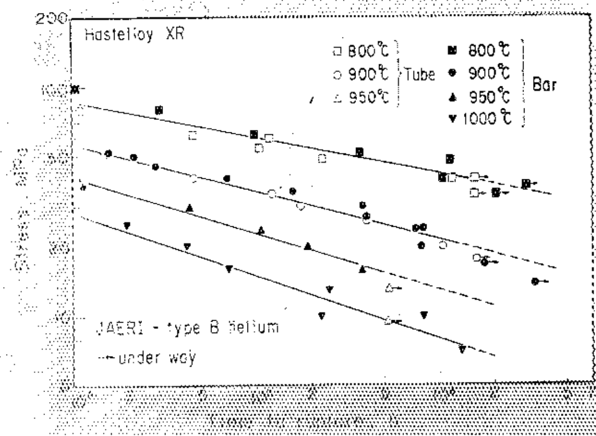
(3) Fatigue Tests
Environmental effects
(Oxidation, Helium)
Creep effects

(4) Neutron Irradiation Tests
300-400°C
$\sim1\times10^{23}$ n/m^2 (E>160fJ)

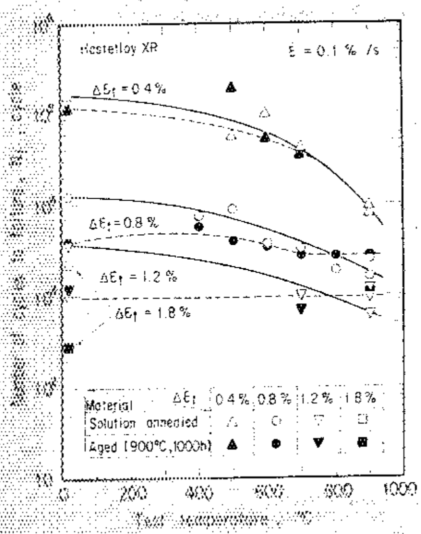
HIGH-TEMPERATURE ALLOYS FOR HTGR APPLICATIONS

Application		Material	Program status
High-temperature structural components	Near-term target (850-950°C)	Hastelloy XR (Developed for HTTR)	Accumulation of design data and quality assurance-oriented studies
	Long-term target (1000°C)	Ni-Cr-W superalloys (Developmental)	Alloy optimization and screening tests
Cladding material of neutron absorber rods		Alloy 800H (Conventional)	Irradiation tests and post-irradiation creep tests

1. Hastelloy XR

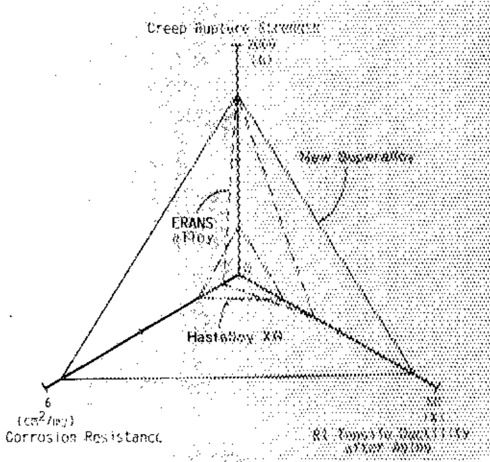


Long-term creep rupture data for Hastelloy XR in simulated HTGR coolant.



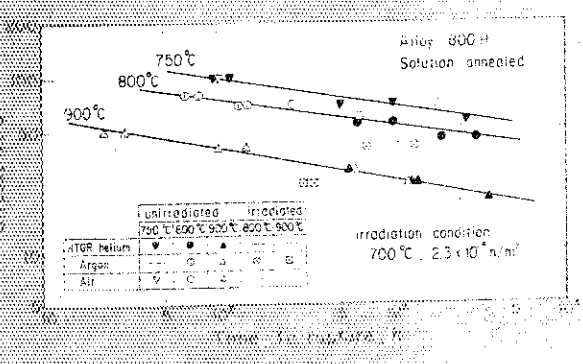
Effect of aging on fatigue properties for Hastelloy XR (50ppmB).

2. Ni-Cr-W superalloy



Comparison of newly developed Ni-Cr-W superalloy (Predicted values) with Hastelloy XR and ERANS alloy.

3. Alloy 800H

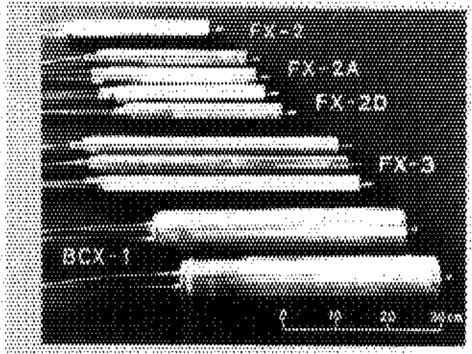


Creep data for Alloy 800H (Comparison between irradiated material and unirradiated one, among three different environments)

REACTOR INSTRUMENTATION

1. NUCLEAR INSTRUMENTATION

The following neutron detectors and monitoring system have been developed for nuclear instrumentation of HTGR.



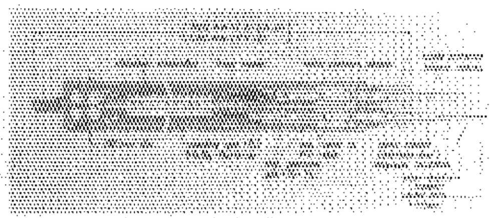
Neutron Detectors developed by JAERI

Wide-Range Monitoring System (WRMS) of HTGR

Neutron Detector :
High-Temperature Fission Counter-Chamber (FX-2A, FX-2D or FX-3),
Operating temp. of 800°C (max.).

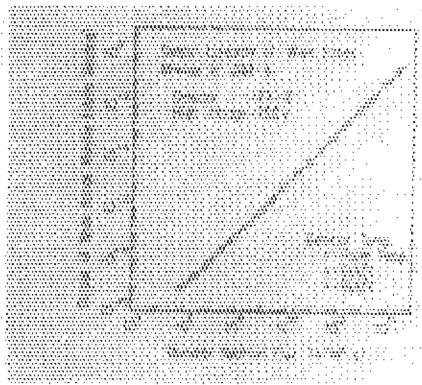
Monitoring Range :
source level $\sim 2.6\times10^{14}$ n/m$^2\cdot$s

The performance of the WRMS was tested in reactors of JMTR, JRR-4(JAERI), HFR (EURATOM) and AVR(Germany) under KFA-JAERI cooperation.


Structure of High-Temperature Fission Counter-Chamber (HTFC)

High-Temperature Gamma Compensated Ionization Chamber for power range monitoring (BCX-1)

Operating Temp. : 600°C (max.)
Measuring Range : $10^{11}\sim10^{15}$ n/m$^2\cdot$s
Neutron Sensitivity : 3×10^{-18} A/(n/m$^2\cdot$s)



High-Sensitive Gamma Uncompensated Ionization Chamber(HSUIC) for reactor protection system

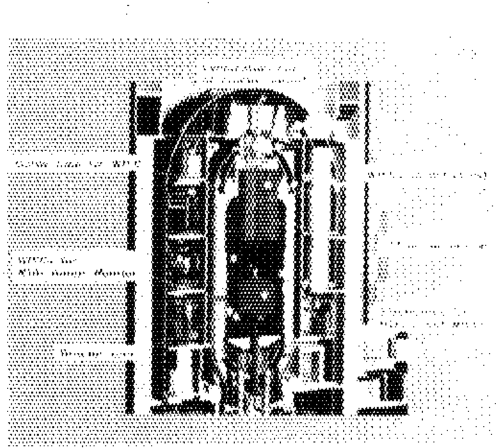
Neutron Sensitivity : 5×10^{-16} A/(n/m$^2\cdot$s)
Measuring Range : $2\times10^{7}\sim3\times10^{12}$ n/m$^2\cdot$s

The sensitivity per unit volume is 10 times as high as that of a conventional HSUIC.

Wide-Range In-Core Fission Counter-Chamber (WIFC) for measurement of neutron flux distribution in HTGR

Operating Temp. : 800°C (max.)
Measuring Range : $10^{6}\sim10^{14}$ n/m$^2\cdot$s
Detector Size : 10mm$\phi\times$413mm

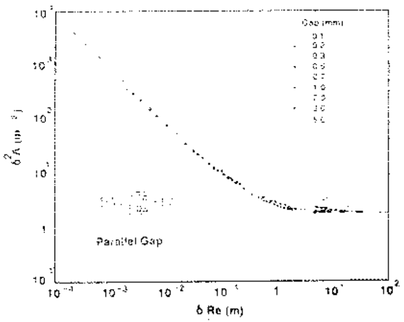
The WIFCs were used for a KFA-JAERI joint experiment for measuring neutron flux in a steam generator region above the AVR core.


Installation position of neutron detectors used for a KFA-JAERI joint experiment in AVR

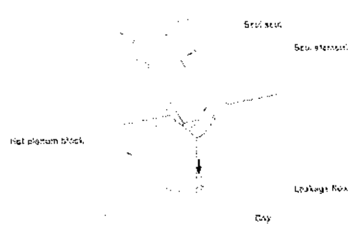
STRUCTURAL INTEGRITY OF CORE & HIGH TEMPERATURE COMPONENTS

Thermal Structure Laboratory

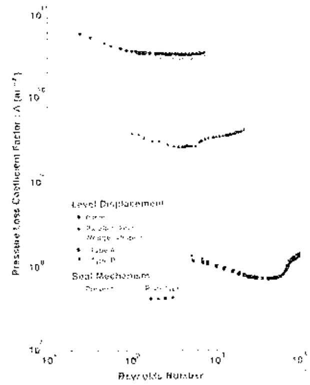
The coolant flow in core
Crossflows through interfacing blocks are assessed by both model tests and theoretical studies over the range of width in expected gaps. Seal elements show a high potential to reduce the leakage flow at the core support blocks.



Crossflow characteristics

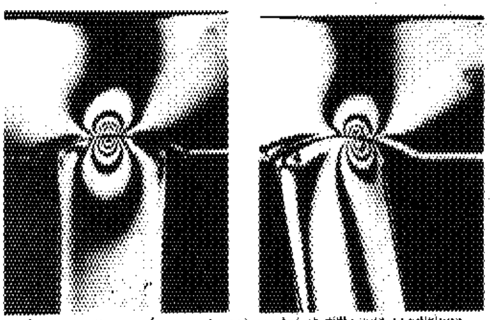


New seal element



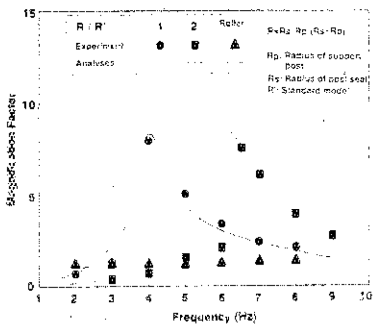
Pressure loss coefficient factor for seal mechanism

The core structure
As well as strength and fracture in the support post, seismic and resonance characteristics in the core support structure are pursued to assess the discontinuity of core structure and the impact force to key-keyways, dowel pins and plenum blocks.

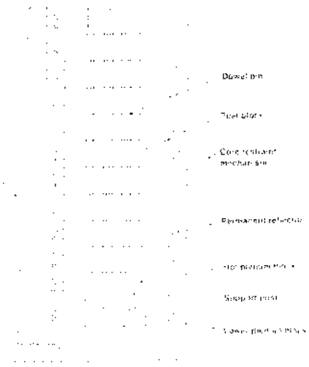


Fringe patterns of support post model at different conditions

Photoelastic experiment

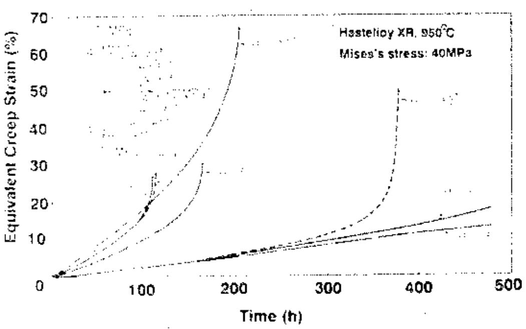


Vibrational characteristics in support post structure

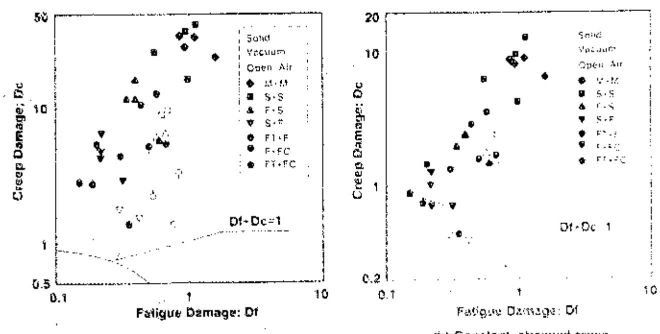


Seismic simulations in HTTR core model

The high temperature structural design
Much of work is directed to the methodology of life prediction in Hastelloy XR at elevated temperature regime, covering creep, fatigue, creep-fatigue interactions, validity of constitutive equation, multiaxiality, weldment, creep racheting, inelastic analyses and fracture mechanics.



Combined-load creep



(a) Constant-loaded creep

(b) Constant-stressed creep

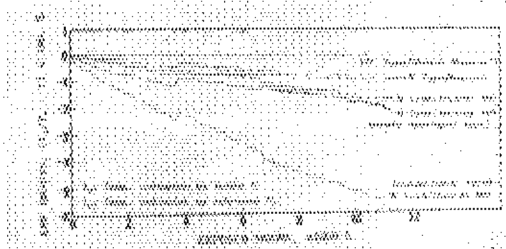
Life prediction in creep-fatigue through time fraction rule

2. HIGH TEMPERATURE MEASURING INSTRUMENTATION

High-temperature Thermocouples for Core-outlet Gas-temperature Measurement

- Requirement ;
  Max. Temp. ; 1100°C (long term)
  1200°C (short term)
  with 100°C margin.
  Service Period ; 20000 hours
- Thermocouples type ;
  (Nicrosil/Nisil element wires)+
  (Best sheath ; what materials?)

EMF drifts of N-type TCs are very small as compared with that of conventional K-type TC.

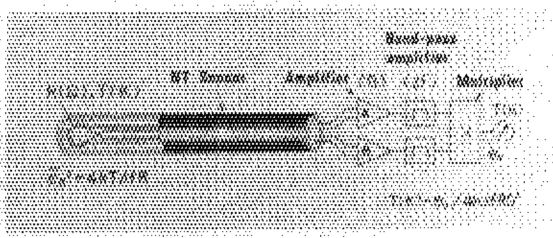


Long-term EMF stability test of N-type thermocouples and of K-type thermocouple in comparison, at 1200°C/He environment

Noise Thermometer (NT) for In-situ Incore-temperature Calibration

KFA/JAERI Joint Experiment ;

- KFA's NT systems were tested in reactor of JMTR. (1984–1986)
- JAERI's NT systems will be tested in KFA. (1990)



Principle of Noise Thermometer

3. FUEL FAILURE DETECTION SYSTEM

FFD Method :

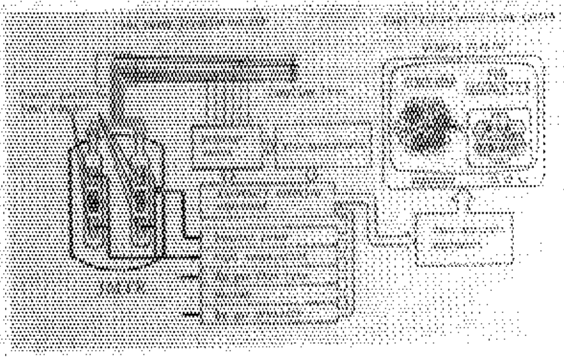
- Selective detection of short-life noble-gas fission products
- A state function for estimating FP concentration in cooling gas
- Fuel failure detection by comparison with measured value and background estimated by the state function

Development Step :

- Fundamental experiments using gas-sweep capsules in the JMTR
- Confirmation tests using OGL-1

Experimenatal Results :

- Sensitivity of cpf failure : 0.04%
- Dependency of released FPs on reactor power and fuel temperature



Schematic diagram of FFD experiment system

THERMOHYDRAULIC STUDY ON SAFETY

I. OUTLINE OF ACTIVITIES

Objects	Items
Safety study on Accidents (1) Air Ingress Accident ·Primary pipe rupture	·Behavior of air ingress into the core ·Mass transfer (graphite corrosion) at temperature higher than 900°C
(2) Loss of Flow Accident ·Blockage of fuel cooling channel ·Inner pipe rupture of primary cooling system ·Secondary cooling pipe rupture ·Water cooling pipe rupture	·Natural convection in the top-cover of the pressure vessel ·Natural and forced combined convection in multi parallel channels of reactor core ·Natural convection on the outside wall of the pressure vessel
Safety Study on Abnormal Transient Event (1) Decrease in Primary Coolant Flow Rate ·Coast-down of helium gas circulator	·Heat transfer in fuel and control-rod cooling channels ·Thermal mixing in plenum block
Backup Study with Visualization Technique	·Visualization of velocity and temperature

II. STUDY ON PRIMARY PIPE RUPTURE

⟨Progress of Accident⟩

(1) Guillotine Rupture of Primary Double Concentric Pipe
(2) Depressurization (4MPa→0.47MPa)
(3) Air Ingress
(4) Natural Circulation $C+xO_2 \rightarrow yCO+zCO_2$
(5) Graphite Corrosion $C+CO_2 = 2CO$
$2CO+O_2 \rightarrow 2CO_2$

⟨Consequences⟩

(a) Reduction in Mechanical Strength of Graphite
(b) Release of Fission Products
(c) Production of Inflammable Gas (CO)

⟨⟨Results of Thermohydraulic Study⟩⟩

(1) Flow Rate of Air Ingress :
Analytical results of flow rate of air entering the core agree with experimental ones. The flow rate in the safety evaluation is overestimated under the conservative conditions.

(2) Graphite Corrosion Rate at High Temperature :
Experimental data for mass transfer coefficients (Sherwood number), from which corrosion rates at high temperature are calculated, are lower than the correlation used for the safety analysis. The correlation is derived from a heat transfer correlation on the assumption of analogy between mass and heat transfer.

··· In HTTR, all the safety criteria for the accident are satisfied by the evaluation based on the conservative analytical conditions.

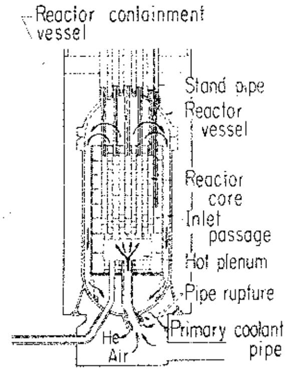


Fig. 1 Cross-sectional View of HTTR

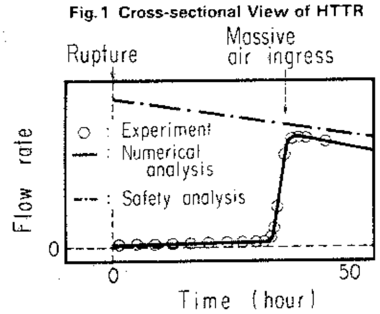


Fig. 2 Flow Rate during Air Ingress

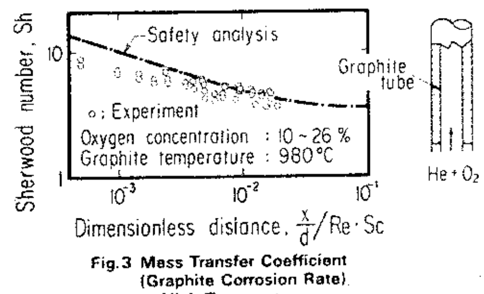
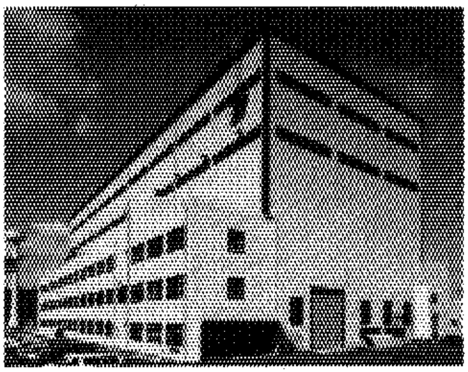


Fig. 3 Mass Transfer Coefficient (Graphite Corrosion Rate) at High Temperature

HELIUM ENGINEERING DEMONSTRATION LOOP (HENDEL)



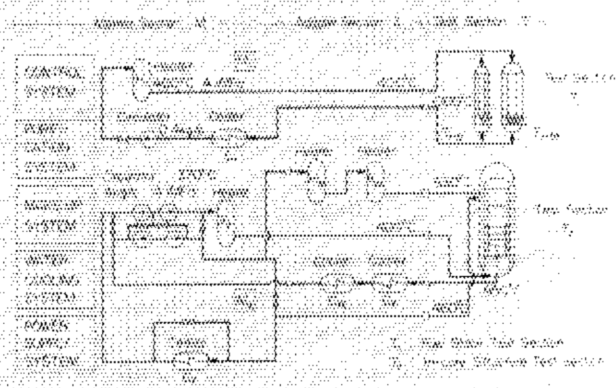
HENDEL Building

I. Purpose

Large Scale Demonstration Tests of
High-Temperature Components of HTTR

II. Major Specification

Items	Specifications
Helium gas	
Temperature	1000 °C (Max.)
Pressure	4 MPa (Max.)
Flow rate	4 kg/s (Max.)
Strage capacity	10,000 Nm^3
Circulator	Gas bearing, centrifugal type
Revolution	3,000~12,000 rpm
Temperature	400 °C (Max.)
Main heater	Electrically heated pipe heater
Material	Incoloy800H (H_{31}) Graphite (H_{32})
Max. power	4.7 MW (H_{31}) 4.36 MW (H_{32})
Main cooler	Shell and tube, segmental baffle type water cooler
Max. power	6.7 MW (C_{31}) 3.5 MW (C_{32})



Schematic Flow Diagram

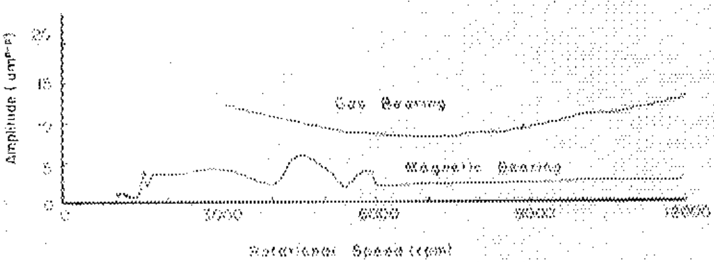
III. Development of Loop Technology and its Components (March, 1982~)

• Development of Helium Circulation Technology at High Temperature, High Pressure and Large Flow Rate

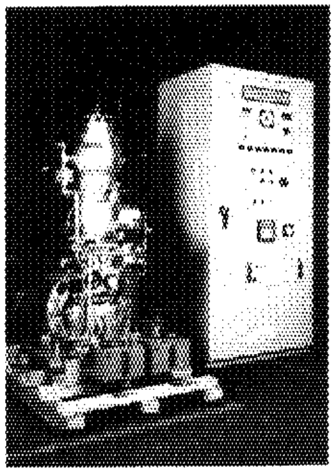
· Studies on rotor dynamics of Gas Bearing Type Circulator

· Research and Development of Magnetic Bearing Type Circulator and reciprocating High Speed Compressor

• Improvement of Helium Purification System



Rotational speed vs Shaft Vibration Amplitude

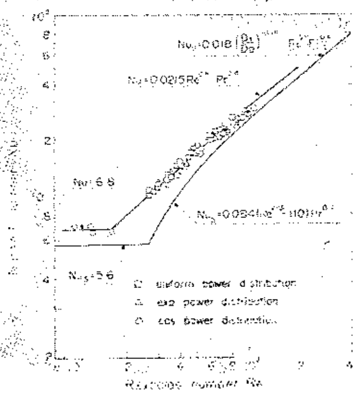


Magnetic Bearing Type
Reciprocating Compressor

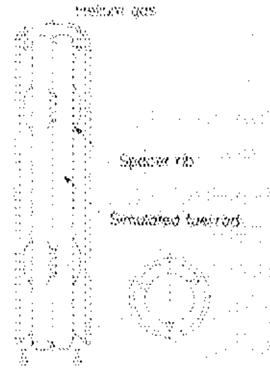
IV. Reactor Component Tests

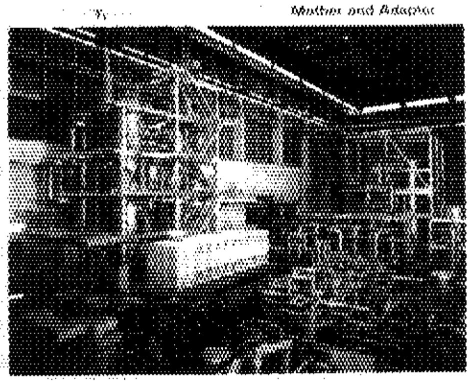
1. Fuel Stack Test Section (T₁) (March, 1983～)

- Thermal and Hydraulic Test of Simulated Fuel Rod, Control rod and Block of HTTR
- Crossflow Test of Fuel Column
- Verification Test of Control Rod Drive Mechanism
- Thermal and Hydraulic Test of Advanced Fuel Element



Heat Transfer Test Results

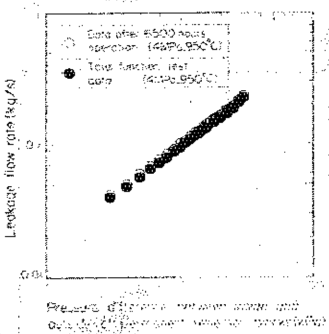




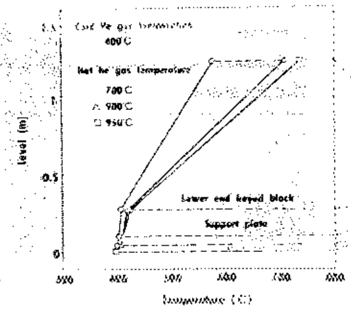
T₁ Test Section

2. In-Core Structure Test Section (T₂) (June, 1986～)

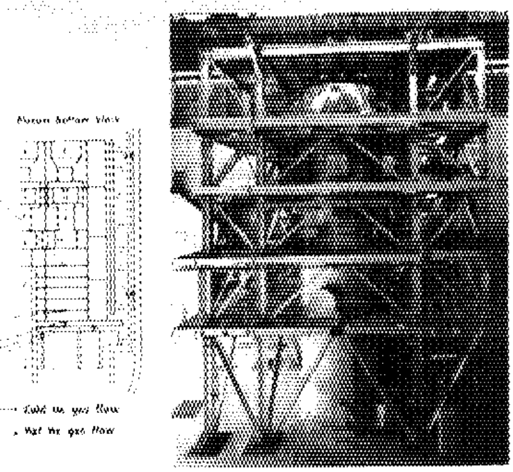
- Demonstration of Large Scale Structure Fabrication
- Sealing Performance Between Large Blocks
- Performance of Thermal Insulation Blocks
- Temperature Mixing Performance of Helium Gas



Leakage Flow Rate through Gaps between Permanent Reflector Blocks



Temperature Distribution in Core-Bottom Structure



T₂ Test Section

REACTOR PHYSICS RESEARCH FOR HTTR

1. Experimental program at Very High Temperature Reactor Critical Assembly (VHTRC)

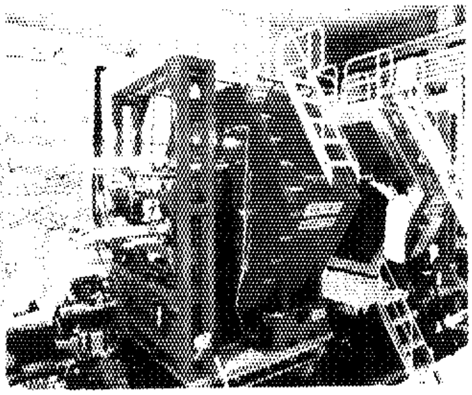
Purpose : Neutronic design verification of HTTR
Schedule : 1985–1987 Basic design verification
1988– Detailed design verification

Main items of experiments

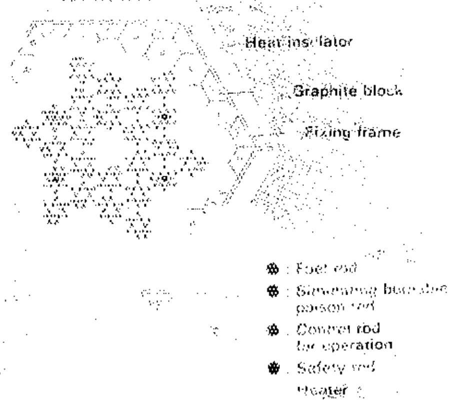
(1) Critical mass
(2) Power distribution
(3) Reactivity worth of control rod and burnable poison rod
(4) Temperature coefficient
· Whole core up to 200°C
· Sample fuel rod up to 800°C
(5) Fuel lattice parameters

2. Main features of VHTRC —— Initial critical in May 1985 ——

Type : Split-table type of hexagonal graphite block structure
Size : 2.4m across flats and 1.2m long in each half
Fuel : Coated particle fuel (2, 4 and 6wt% enriched UO_2)
Core temperature : Room temperature to 200°C
Maximum power : 10W

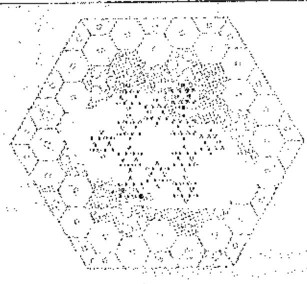
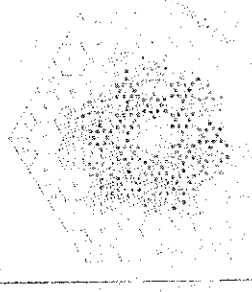


VHTRC



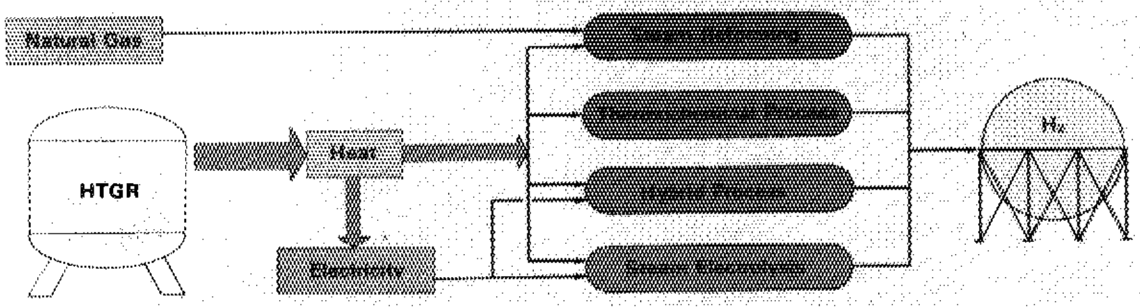
Cross-sectional view (HTTR simulating core)

3. Initial critical experiments at VHTRC-1 core

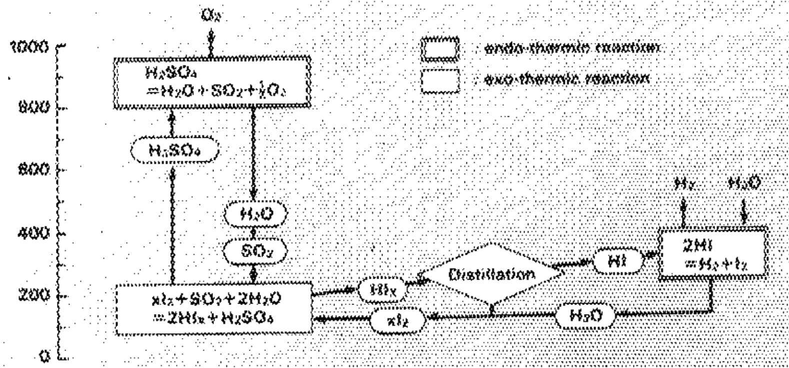
Items	Room temperature	200°C
Fuel rod loading pattern : 4wt% enriched fuel rod : 2wt% enriched fuel rod ⊙ : Heater		
Measured critical mass(^{235}U)	4.66 kg	6.36 kg
(Calc./Meas.−1)×100		
· Critical mass (k_{eff})	−3% (+0.5%Δk)	−1% (+0.1%Δk)
· Flux distribution	≦±3%	≦±3%
· Control rod worth	≦±5%	-----
· Burnable poison rod worth	≦±5%	-----
· Temperature coefficient	+1% (25–200°C)	

Calculation : SRAC code system with ENDF/B-IV nuclear data.

NUCLEAR HYDROGEN PRODUCTION



Thermochemical I-S Process



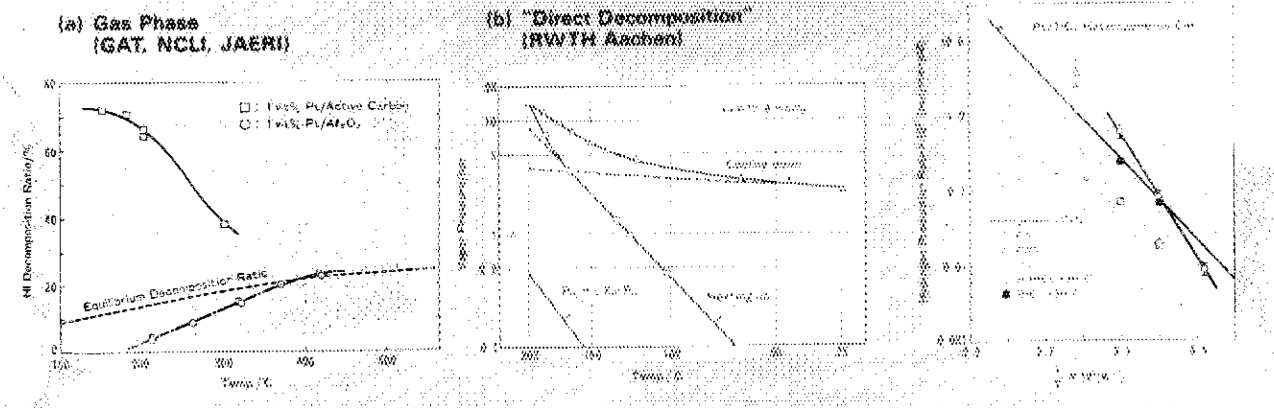
- R & D at
  GAT Inc.
  RWTH Aachen
  JAERI
- International Co-operation at IEA Workshop.

(1) Studies on HI Decomposition

(a) Gas Phase (GAT, NCLI, JAERI)

(b) "Direct Decomposition" (RWTH Aachen)

(c) Liquid Phase (GAT)



(2) Material Screening Test

Environmental	Test Condition	Selected Material
H_2SO_4 decomposition	H_2O, SO_2, SO_3, O_2, 800–900°C 98wt%H_2SO_4 w/woHI, 320°C	Incoloy 800, SUS329J1, SS8113MA, SUS310S Cast Iron (High Si)
Bunsen reaction	50wt%H_2SO_4 with HI, 120°C	Ta, Zr, SiO_2, SiC, Si_3N_4
HI_x distillation	H_2O, HI, I (Feed comp.), 240°C	Ta, Nb, Zr
HI decomposition	HI_x, 200 & 300 & 400°C 57wt%HI, 136°C	Ti, Hastelloy C-276, Inconel 600 Nb, Hastelloy C-276, Ta, Zr