FOREWORD

Technological developments for gas-cooled reactors are at present being carried out on a large scale in the UK for CO₂-cooled Advanced Gas-Cooled Reactors (AGRs) and for High Temperature Gas-Cooled Reactors (HTGRs) in the Federal Republic of Germany, United States of America, Japan and USSR.

All programmes are directed towards construction of plants. The programme in the UK is by far the most advanced effort with several commercial Magnox Reactor and Advanced Gas-Cooled Reactor units in operation and further AGRs in the construction or commissioning phase. HTGRs are in operation in the USA and in the Federal Republic of Germany. Other HTR specific work is being carried out in Austria, China and Switzerland. At the IAEA the International Working Group on Gas-Cooled Reactors (IWGGCR) has been established in 1978 in order to promote an exchange of information on gas-cooled reactor development and to stimulate international cooperation. In the framework of this working group, other countries are following the development of GCR technology in order to investigate its possible benefits for national energy supply. The IWGGCR has recommended to the Agency to convene this Technical Committee on Gas-Cooled Reactors and their Applications, which was attended by more than 200 participants from 25 countries and International Organizations.

The Agency is grateful to the Government of the Federal Republic of Germany and to the Nuclear Research Centre Jülich for their hospitable and efficient arrangements.

This volume contains all papers presented at the meeting.
EDITORIAL NOTE

In preparing this material for the press, staff of the International Atomic Energy Agency have mounted and paginated the original manuscripts as submitted by the authors and given some attention to the presentation.

The views expressed in the papers, the statements made and the general style adopted are the responsibility of the named authors. The views do not necessarily reflect those of the governments of the Member States or organizations under whose auspices the manuscripts were produced.

The use in this book of particular designations of countries or territories does not imply any judgement by the publisher, the IAEA, as to the legal status of such countries or territories, of their authorities and institutions or of the delimitation of their boundaries.

The mention of specific companies or of their products or brand names does not imply any endorsement or recommendation on the part of the IAEA.

Authors are themselves responsible for obtaining the necessary permission to reproduce copyright material from other sources.
Please be aware that all the Missing Pages in this document were originally blank pages.
CONTENTS

Introduction and Summary .......................................................... 7

OVERVIEW OF THE STATUS OF GAS-COOLED REACTORS AND THEIR PROSPECTS (Session A)

Magnox reactors and advanced gas-cooled reactors ................................................. 11
   A.F. Pexton
High temperature gas-cooled reactors ................................................................. 29
   R. Schulten

EXPERIENCE WITH GAS-COOLED REACTORS (Session B)

Development of a high output AGR .................................................... 43
   B.A. Keen
Technical evolution and operation of French CO₂ cooled reactors (UNGG) ............. 53
   Y. Berthion
Fort St. Vrain performance ........................................................................... 61
   H.L. Brey
AVR experience ......................................................................................... 71
   E. Ziermann
THTR operation — The first year ..................................................................... 99
   D. Schwarz

DESCRIPTION OF CURRENT GCR PLANT DESIGNS (Session C)

Restoration of a CEGB Magnox reactor to full power ........................................ 111
   R.J. Paynter, B.J. Roberts, P.T. Sawbridge
The modular high-temperature gas-cooled reactor (MHTGR) ................................ 123
   A.J. Neylan
Status of gas-cooled reactor development in the USSR ..................................... 135
   V.N. Grebennik
The nuclear electricity and heat generation using the VG-400 reactor .................... 141
   V.V. Bulygin, V.E. Vorontsov, V.N. Grebennik, E.M. Ionov, A.N. Protsenko,
Current status of research and development of HTGR in Japan ......................... 151
   T. Hayashi
Plant design of a high temperature engineering test reactor in JAERI .................... 165
   O. Baba
Status of the HTR-Module plant design ....................................................... 179
   I.A. Weisbrodt, W. Steinwarz, W. Klein
Small nuclear power plants: 10 MW GHR gas-cooled heating reactor ................. 199
   H. Schmitt
HTR 100 industrial nuclear power plant for generation of heat and electricity .... 213
   S. Brandes, W. Kohl
HTR 500 — The basic design for commercial HTR power stations .................... 235
   E. Baust, J. Schöning
SAFETY ASPECTS (Session D)

Safety characteristics of current HTR .................................................. 253
   W. Kröger
The safety concept of the modular HTR .............................................. 267
   G.H. Lohnert
MHTGR licensing approach and plant response to off normal events .......... 283
   A.J. Neylan, F.A. Silady
Comparison of regulatory aspects in different countries ....................... 295
   K. Hofmann

GAS-COOLED REACTOR APPLICATIONS (Session E)

HTGR applications, prospectives and future development ....................... 313
   H. Barnert
Perspectives of HTGRs in chemical and iron and steel industries of Japan .... 329
   K. Tsuruoka, T. Katamine, T. Miyasugi, R. Araki
Israeli perspective on HTGRs ......................................................... 343
   A. Barak, A. Beck, E. Greenspan, J. Szabo, L. Blumenau, H. Branover,
   A. El-Boher, E. Spero, S. Sukoriansky

GAS-COOLED REACTOR TECHNOLOGY (Session F)

Overview of US MHTGR base technology development program ................. 371
   J.E. Jones, Jr., P.R. Kasten
Status and development of the German materials programme for the HTGR .... 387
   H. Nickel, F. Schubert, H. Schuster
KVK and status of the high temperature component development ............... 407
   W. Jansing, H. Breitling, R. Candeli, H. Teubner
HTR fuel development and qualification — Treatment of spent fuel ............ 429
   G. Kaiser, K. Hackstein
Characteristics of HTGR spherical fuel elements .................................. 445
   A.S. Chernikov, L.N. Permyakov, L.I. Mikhailichenko, S.D. Kurbakov
Self-sustaining thorium cycle in a high temperature graphite reactor .......... 461
   K. Balakrishnan

USER'S PERSPECTIVES ON GAS-COOLED REACTORS (Session G)

Future applications of the high temperature reactor ............................ 473
   A. Klusmann, M. Stelzer
Financing models for HTR plants — Co-financing, counter trade, joint ventures 483
   J. Bogen, D. Stölzl
Perspectives on the HTGR from utilities in the USA ................................ 499
   H.L. Brey, L.D. Mears
Market prospects of HTRs in newly industrialized and developing countries .... 511
   S. Garriba, C. Vivante

Chairmen and Secretariat ............................................................... 521
List of Participants ......................................................................... 523
INTRODUCTION AND SUMMARY

On the invitation of the Government of the Federal Republic of Germany the International Atomic Energy Agency convened in Jülich, 20–23 October 1986, the Technical Committee on Gas-Cooled Reactors and their Applications. The meeting was hosted by the Nuclear Research Centre Jülich and cosponsored by the Commission of the European Communities. The purpose of the meeting was to review and discuss the current status and recent progress made in the technology and design of gas-cooled reactors and their application for electricity generation, process steam and process heat production. The meeting was attended by more than 200 participants from 25 countries and International Organizations presenting 34 papers.

The technical part of the meeting was subdivided into 7 sessions:

A. Overview of the Status of Gas-Cooled Reactors and their Prospects
B. Experience with Gas-Cooled Reactors
C. Description of Current GCR Plant Designs
D. Safety Aspects
E. Gas-Cooled Reactor Applications
F. Gas-Cooled Reactor Technology
G. User's Perspectives on Gas-Cooled Reactors

At the end of the meeting a round table discussion was organized in order to summarize the meeting and to make recommendations for future activities.

Gas-Cooled Reactors have been under development for more than three decades. Today there are more than 40 gas-cooled reactors in operation or in the commissioning phase in seven Member States of the IAEA. Most of them are CO₂-Cooled Magnox Reactors and Advanced Gas-Cooled Reactors (AGRs) for electricity generation, having accumulated about 800 reactor years of operation. The Helium-Cooled High Temperature Gas-Cooled Reactors (HTGRs) are also for electricity generation, however, the long term development goal for the HTGR is in most countries the production and application of process steam and process heat for chemical industries, coal conversion processes, nuclear steelmaking, etc. Currently there are also gas-cooled reactors of small sizes under design with reduced complexity, passive safety mechanisms and a high potential for modular design and shop fabrication. These reactors are not only appropriate for application in highly industrialized countries but also in countries with less developed industrial infrastructure.

During the meeting the technology of gas-cooled reactors was presented and discussed together with their different applications. Particular emphasis was given to a review of their safety features, and
the perspectives of users on this reactor line. After the meeting a tour took place to the THTR-300, a 300 MWe pebble bed reactor, which is in the commissioning phase and which has successfully reached the 100% power level.

The meeting concluded that gas-cooled reactors have reached a very high level of maturity. In the UK the AGRs, in particular the Hinkley/Hunterston type design, have proven to be economic and safe electricity producers. The successful operation of High Temperature Gas-Cooled Reactors in the USA and the Federal Republic of Germany and the commissioning of the THTR-300 have proven that this technology is available for electricity generation and process steam production. The advantages of the system, i.e., high safety, high temperature potential and diverse applications, environmental acceptability and fuel saving have been confirmed by the operation of prototype and demonstration plants. GCRs are a unique tool for power and steam production, a tool which is based on the margins of safety inherent in the design concept rather than on engineered safeguards, a feature which is of increasing interest in Agency Member States. Because of their specific properties, GCRs are expected to have very good international market chances in particular in the medium sized and small power range.
OVERVIEW OF THE STATUS
OF GAS-COoled REACTORS AND THEIR PROSPECTS

(Session A)
MAGNOX REACTORS AND ADVANCED GAS-COOLED REACTORS

A.F. PEXTON
South of Scotland Electricity Board,
Glasgow, United Kingdom

Abstract

Commercial nuclear power in the U.K. is based on CO₂ cooled graphite moderated reactors. This started around 1950 with the magnox programme having Calder Hall and Chapelcross as the forerunners. Nine twin-reactor commercial magnox stations were then built between 1958 and 1970. The first seven stations have steel pressure vessels with external boilers connected by gas ducts. The last two have an integrated design with core, boilers and gas circulators enclosed in pre-stressed concrete pressure vessels. The magnox reactors have given good service and the older stations are now being subjected to careful review after 20 years operation.

Magnox was superseded by the AGR and five twin reactor stations to three separate and distinct designs, but all with integral gas circuits inside concrete pressure vessels, were ordered in the period 1965 to 1970. The most successful of these, at Hinkley Point 'B' and Hunterston 'B' have now been in operation for over 10 years. Their load factor for the past few years has been of the order of 80%.

In 1979 two further twin reactor AGR stations, Heysham II and Torness, were authorised. These were based on the Hinkley/Hunterston design with detailed improvements arising from operating experience and developments in safety standards. Commissioning tests are now well advanced and the first reactors at each station are scheduled to come into operation in 1987. Work has proceeded close to the original programme and budget.

The Heysham II/Torness design now provides a standard base for on-going development of the AGR. A study is in hand which essentially replicates the main features, but by effective use of margins enables higher output to be achieved.

The future depends on government policy following issue of the report of the inquiry into CEGB's application to build a PWR at Sizewell, as well as on the political environment following Chernobyl. Whatever the outcome, however, the AGR provides an attractive option with many inherently favourable safety characteristics. Whilst it is difficult to see an incentive for the U.K. to change to an alternative form of gas-cooled reactor technology, U.K. gas cooled reactor experience provides scope for transfer of technology to HTGR developments in Europe and the U.S.
The Structure of Electricity Supply

In the UK, production and distribution of public electricity supplies is the responsibility of State-owned monopolies. There are, for planning and commercial purposes, two integrated grid systems, one for England and Wales and one for Scotland but the interconnection capacity between them, some 1000MW, is adequate to allow for power exchanges arising from economic scheduling of the generating plant on both sides of the border.

The Contribution from Nuclear Power

The nuclear stations serving the overall U.K. grid system are shown in Figure 1.

![UK Nuclear Power Stations Diagram](image)

<table>
<thead>
<tr>
<th>Station</th>
<th>Type</th>
<th>Output (MW(e))</th>
</tr>
</thead>
<tbody>
<tr>
<td>Berkeley</td>
<td>Magnox</td>
<td>276</td>
</tr>
<tr>
<td>Bradwell</td>
<td>Magnox</td>
<td>245</td>
</tr>
<tr>
<td>Calder Hall</td>
<td>Magnox</td>
<td>200</td>
</tr>
<tr>
<td>Chapelcross</td>
<td>Magnox</td>
<td>200</td>
</tr>
<tr>
<td>Dungeness A</td>
<td>Magnox</td>
<td>424</td>
</tr>
<tr>
<td>Hinkley Point A</td>
<td>Magnox</td>
<td>430</td>
</tr>
<tr>
<td>Hunterston A</td>
<td>Magnox</td>
<td>300</td>
</tr>
<tr>
<td>Trawsfynydd</td>
<td>Magnox</td>
<td>390</td>
</tr>
<tr>
<td>Sizewell</td>
<td>Magnox</td>
<td>420</td>
</tr>
<tr>
<td>Oldbury</td>
<td>Magnox</td>
<td>434</td>
</tr>
<tr>
<td>Wytta</td>
<td>Magnox</td>
<td>840</td>
</tr>
<tr>
<td>Hinkley Point B</td>
<td>AGR</td>
<td>1040</td>
</tr>
<tr>
<td>Dungeness B</td>
<td>AGR</td>
<td>3300</td>
</tr>
<tr>
<td>Hartlepool</td>
<td>AGR</td>
<td>3300</td>
</tr>
<tr>
<td>Heysham I</td>
<td>AGR</td>
<td>3300</td>
</tr>
<tr>
<td>Heysham II*</td>
<td>AGR</td>
<td>1230</td>
</tr>
<tr>
<td>Hunterston B</td>
<td>AGR</td>
<td>1150</td>
</tr>
<tr>
<td>Torness*</td>
<td>AGR</td>
<td>1360</td>
</tr>
<tr>
<td>Dounreay</td>
<td>PFR</td>
<td>250</td>
</tr>
<tr>
<td>Winfrith</td>
<td>SGHWR</td>
<td>100</td>
</tr>
</tbody>
</table>

FIG. 1. UK NUCLEAR POWER STATIONS.

Commercial nuclear power is based on CO₂-cooled, graphite moderated reactors, starting with the Calder Hall 'Magnox' type, which uses metallic natural uranium fuel with magnesium alloy cladding. Eight twin reactor commercial plants of this type, all to different, progressively larger and more developed designs, were built by The Electricity Supply Industry in England and Wales, and one station, Hunterston ‘A’ was built in Scotland, all between 1958 and 1970, providing a total of 4000MW(e). At one time there were five different industrial consortia competing for the design and construction of nuclear power stations.

The commissioning of a 33MW(e) test bed for the Advanced Gas-Cooled Reactor (AGR) at Windscale in 1962 opened the way two years later to an evaluation of competitive tenders for AGR, PWR and BWR. This led to adoption of the AGR system. Five stations, each with twin reactors of some 600MW(e), were ordered in the period 1965 to 1970, to three separate and distinct designs from the three nuclear design and construction companies.
still active in the UK. Two more twin reactor stations, CEGB's Heysham II and SSEB's Torness were authorised in 1979 based on CEGB's Hinkley Point 'B' and SSEB's Hunterston 'B', the most successful of the three original designs. Start-up of Heysham II and Torness is scheduled in 1987.

At present nuclear power provides 19% of the total energy generated for public electricity in the UK but this will grow, with further plant now building and being commissioned, to more than 25% in the next few years. In Scotland the nuclear share is much higher, the present 45% increasing to over 60% within the next two years. These outputs are compared with that of other countries which have a significant nuclear contribution in Figure 2. Actual information is shown for 1984 together with the OECD estimates for 1990.

![FIG. 2. NUCLEAR SHARES OF ELECTRICITY PRODUCTION.](image)

The annual requirement for increased capacity in the UK is estimated at about 1200MW(e) per year (ie one plant per year) from the mid 1990s. In Scotland alone the corresponding requirement would be only one 1200MW(e) plant per decade but, in view of the long timescale required for planning and approvals, thought is already being given to the next plant.

Based on current cost estimates the advantages of AGR compared with new coal-fired plant for base load operation is at least 30%, expressed in total cost per unit sent out, and could be more depending on the extent of the provisions required for flue gas desulphurization and low production of nitrogen oxides. Clearly in the UK as a whole there is a strong economic incentive to increase the proportion of nuclear in the plant mix. However, if the full benefits of a nuclear ordering
programme are to be achieved, it will be important that this rests on the twin foundations of a reactor design which can be replicated without significant changes and a construction programme in which there is sufficient assurance of successive orders to allow proper planning of resources and career stability for trained and experienced personnel in the design, construction and manufacturing sectors.

The Magnox Programme

Calder Hall, in England, and Chapelcross, in Scotland, the forerunners of the magnox programme each comprise four reactors of 35 MW(e) and operate at 7 bars gas pressure. Each reactor has four separate gas circuits, each containing its own gas circulator and boiler (Fig. 3). Refuelling is carried out off load. The fuel element cans originally had simple circumferential fins but these were later modified to a herringbone design which gave improved heat transfer. This, with other improvements, enabled the net output to be increased to 50MW(e) per reactor and the uprating and the historical load factor of Chapelcross are shown in figure 4. After 25 years of operation, the lifetime load factor of the station, based on rated output, is 85%.

Hunterston 'A' was the third twin reactor commercial magnox station built in the U.K., and its first unit was commissioned in 1964. Each reactor has eight separate gas circuits with configuration shown in figure 5. The carbon dioxide pressure is 10 bars and the design net output is 150MW(e) per unit. Refuelling is carried out at full load using a refuelling machine having access to pressure vessel standpipes below the reactor. The historical load factor of Hunterston 'A' is shown in figure 6. After 22 years of operation it has a lifetime load factor of 82% based on design output.
FIG. 4  ANNUAL LOAD FACTORS OF CHAPELROSS BASED ON RATED OUTPUT

FIG 5  HUNTERSTON 'A' MAGNOX
The first seven U.K. commercial magnox stations have steel reactor pressure vessels with external boilers connected by ducts. A major change was introduced in CEGB's last two magnox stations, Oldbury and Wylfa, which have an integrated design with the whole of the gas circuit, including core, boilers and gas circulators, enclosed in a prestressed concrete pressure vessel.

In 1969 a gas circuit corrosion problem was encountered which affected all magnox stations. The major structural steel components had been manufactured from silicon killed mild steel with good corrosion resistance in moist CO₂ but certain other items had been made of more rapidly corroding rimming steel. The growth of oxide on the faces of corroding washers, for example, had in some areas built up local strains sufficient to fail bolts. Derating of some of the reactors was necessary whilst a systematic programme of monitoring, development and modification was undertaken. At Hunterston 'A', for example, the mean duct outlet temperature was reduced from 390°C to 360°C giving a fall in net station output from the design value of 300MW(e) to less than 280MW(e). Flow tests on a scale model of the reactor demonstrated that peak temperatures in the most critical areas could be reduced by leaving a few peripheral fuel channels empty. As a result the output has now been restored to the original design value.

Magnox fuel is a success story. The design burn-up of commercial magnox fuel was initially 3000MWD/Te, limited by the expected physical life of the fuel elements. The average
channel burnup now being achieved is over 5,500MWD/Te. Over 2.5 million fuel elements have now been loaded into the commercial reactors with an average failure rate less than 0.1%, and the current failure rate is an order of magnitude less than this.

The Nuclear Installations Inspectorate decided that after 20 years operation that the magnox plants should be subject to a comprehensive safety review. The entire range of safety has been studied including structural integrity of the pressure circuit, core and components, the ability of the reactor protection to respond to all types of reactor faults and the ability of structures to withstand external hazards. On Hunterston 'A' alone this review has absorbed over 100 man years of professional effort.

The AGR Programme

The commercial AGR programme adopted the concept of the integral design inside a concrete pressure vessel.

All five of the AGR stations ordered between 1965 and 1970 are now operational and showing encouraging potential. Hinkley Point 'B' and Hunterston 'B' have been generating for over ten years. Figure 7 compares cross sections of Hunterston 'B' and the subsequent design development at Torness.
The rated output of each unit at Hunterston was originally limited to 460 MW primarily by uncertainty as to the corrosion resistance of the 9% Cr. steel boiler tubes and by some minor deficiencies in fuel pin end-caps. With the progressive introduction of detailed improvements the net authorised rating has been raised to 575MW(e), close to the original design target. Further increases will be made as modified blades are fitted to certain of the turbine stages and as improved Stage 2 reactor fuel elements are introduced.

The load factor has been progressively improved, first by eliminating sources of unreliability (mainly in the conventional plant) and, secondly, by establishing on-load refuelling as normal practice. It is necessary to reduce power to 30 per cent for refuelling at present to avoid the risk of damaging the graphite sleeves of the fuel assemblies but the Stage 2 fuel has a more robust graphite sleeve and will allow the refuelling power to be raised. Further reductions of refuelling output losses will therefore be achieved as Stage 2 fuel comes to be discharged.

Figure 8 shows progress in raising the output capability and load factor at Hunterston. Achievement at Hinkley has been similar.

Construction of the earlier AGRs entailed fabrication on site of such large components as the core supports, the gas baffles and the core restraint tanks. However, for Heysham II and Torness, having established an agreed national design, it was decided to secure the benefits of factory manufacture and these components were transported fully assembled to site. New manufacturing facilities were created to produce not only these large fabrications but many others such as graphite blocks, gas circulators, boilers and fuelling equipment, all to very high
standards of quality and with good productivity. Necessarily site-performed operations, such as thermally insulating the pressure vessel, were rationalised using the most up-to-date techniques and full-scale mock-up training facilities were provided for the operatives at site. Figure 9 shows the site construction sequence employed with the large prefabricated components.

Figure 10 shows the achievement in construction of the two units at Torness against the original programme set in 1979. The consent of the Nuclear Installations Inspectorate is expected shortly for the loading of fuel in Reactor 1. The unfuelled commissioning tests have established the integrity of the pressure vessel and the gas circuit and other components of the plant have behaved as designed during hot pressurised gas flow tests.

Hunterston 'B' was a prototype and the many lessons learned from its construction and operation have been incorporated in the Torness project. Measurements of actual operating performance at Hunterston have allowed the rated output of Torness to be increased by 7 per cent without plant modifications. Once the
FIG. 10. TORNES CONSTRUCTION PROGRAM SHOWING COMPARISON OF ORIGINAL PLAN AND ACHIEVED DATES

The plant has settled down, trimming of the operating conditions could permit some further small increase. The expected output unit cost is in accordance with the original estimate.

The Tornes specification incorporates a number of new requirements. Provision is made for improved access to components and structures within the pressure vessel for inspection and maintenance. Extra provision has also been made for remote inspection. Figure 11 illustrates these facilities.

FIG. 11 AGR GAS CIRCUIT - IN SERVICE INSPECTION AND REPAIR
Experience at Hunterston 'B', after 10 years of operation, justifies the expectation that radiation levels within the vessel will permit man entry on a substantial scale throughout the Station's life. Figure 12 shows how the radiation dose acquired by Hunterston 'B' AGR workers compares with typical values in PWR stations.

Cracks appearing in some of the welds of the reactor roof insulation at Hunterston 'B' after ten years' operation have been reported in the Technical Press. The thermal insulation of the pressure vessel around each fuel standpipe consists of mineral fibre blankets interleaved with stainless steel foils, the whole being retained by stainless steel tubes and plates. Figure 13 shows the arrangements at Hunterston and the improved design at Torness. At Hunterston, certain welds, labelled A and B in Figure 14, have suffered cracking by thermal fatigue. A small number of these have been removed for inspection and testing. Research covering structural analysis and flow modelling is now showing that these welds have been subjected to cyclic temperature variations due to instability of the local gas flow. The solution derived is to fit modified thermal plugs.
to the removable fuel assemblies. None of this has safety connotations, at least in the short term, because of the redundancy built into the design of the insulation restraint structure, and normal operations have continued with the consent of the Nuclear Installations Inspector.

**Safety**

Design safety guidelines have been adopted by the UK Electricity Supply Industry which include requirements to meet numerical radiological release targets. The principal target is to aim
for a design which reduces the total probability of an uncontrolled release of activity to the environment to less than $10^{-6}$ per reactor year.

The design of the Torness safety systems resulting from the application of the design safety guidelines is extremely robust, with diverse provisions in the main functions - fault sensing, reactor shutdown and post trip cooling - to maximise reliability and afford protection against common mode failures. There are main and diverse guardlines. The main guardline is a solid state LADDIC system operating on two out of four logic while the diverse guardline is a two out of three system using relays. The primary shutdown system uses control rods inserted into the reactor core under gravity while, to meet the requirement for diversity in all safety functions, a secondary shutdown system is provided which injects nitrogen gas into the core from beneath the reactor. A manually initiated reactor holddown system involving the insertion of boron beads from beneath the core is also incorporated.

Both guardlines will initiate the primary and secondary shutdown systems. However, only one system is required for reactor
shutdown and therefore operation of the secondary shutdown system is automatically inhibited once adequate control rods are inserted into the core.

The post trip cooling systems have a high degree of diversity and redundancy. They operate automatically to bring the reactor into a shutdown mode. The plant is designed so that operator action will not be required within 30 minutes of a fault but the operator can, of course, act to support the operation if a safety system or post trip cooling does not operate correctly.

Due to the high thermal inertia of the core and moderate fuel rating the reactor is very tolerant to failure of the post trip cooling systems. This is illustrated by two examples.

First, loss of electrical supplies leads to gas circulator speed run-down which in turn rapidly reduces gas flow to the reactor core. The reactor trips from gas circulator under-voltage or underspeed signals. The diesel generators start from the reactor trip signal and within 50 seconds bring the gas circulators up to their post-trip speed of 15 per cent. Figure 15 (lower curve) shows the reactor temperature transient for the condition in which only the gas circulators and the decay heat boiler in one of the four quadrants are assumed to operate. It can be seen that temperatures are well controlled.

An investigation of still more severe fault sequences with no restoration of post-trip cooling has shown that, in faults where the reactor remains pressurised, it will take many hours before fuel cladding failure commences and, therefore, correspondingly longer before any activity release to the environment. Figure 15 also shows the long term trends of fuel temperatures in such a sequence and it can be seen that the rate of fuel temperature
rise is modest and that, even after 10 hours, fuel clad temperatures are a long way from their safety limit of 1,350°C.

Secondly, for depressurisation faults, the largest practicable break in the pressure boundary is a fracture in the gas coolant clean-up system. A complete severance of the largest pipe in this system (some 18 cm diameter) would fully depressurise the reactor in about 40 minutes. The reactor would trip from a low gas pressure signal and, in the limiting design fault sequence, post-trip cooling is assumed to be provided by only two of the four quadrants utilising main boilers and gas circulators. The post-trip cooling system progressively speeds up the circulators for this fault to compensate for the falling gas density.

Figure 16 shows the core temperatures initially rising significantly until the reactor trips. This ignores the effect of the auto control system in controlling temperatures at an earlier stage. Even so, the post-trip period temperatures can be seen to be well controlled.

In the same fault the consequences of not initiating any post-trip cooling have been explored. It was found, not surprisingly, that core temperatures increased more rapidly than in the equivalent pressurised reactor fault sequence but nonetheless, as Figure 16 shows, the rate of increase in temperature is still modest. The fuel clad (melting) safety limit would be reached only after about 13 hours. However, with the reactor pressure lost the fuel pin internal pressures will cause the fuel cladding to creep and some failures would be expected about five hours after the fault. Even so, this

![Figure 16: AGR Depressurisation Fault](image-url)
represents a considerable period of time in which the operators could act to restore cooling and, even assuming this conjunction of failures to remove decay heat, it is inconceivable that cooling would not be restored in such a timescale.

In the AGR, with on-load refuelling, it is necessary to consider the potential for creating a radiological release as a consequence of irradiated fuel being dropped during its removal from the reactor. The mechanism leading to such an accident would be failure of the single tie bar which supports the fuel elements. The nuclear safety implications of such an event have been thoroughly investigated and supported by full-scale stringer drop tests. It has been shown that, even at the maximum drop height into the reactor, the automatic trip facilities provided to deal with this situation would maintain the damaged fuel clad well below its melt temperature. The resulting off-site dose to the public would not approach the level at which evacuation procedures would be invoked and indeed it would in all probability represent only a small fraction of this level.

To summarise briefly then, it has been shown that AGR performance is excellent under the most extreme design basis fault conditions, but even more reassuring is the time available to institute remedial measures in those fault sequences, however improbable, in which post-trip cooling systems do not operate.

Towards a New AGR

The reactor design for Torness incorporates substantial margins. It is now clear that these margins can be exploited to increase the heat output from future AGRs.

A contract to produce the design and the appropriate safety assessment of such a reactor has been placed with the National Nuclear Corporation (NNC) with finance from the CEGB, SSEB and NNC itself.

The first phase, confirming the main parameters shown in Figure 17, has been completed. This was aimed mainly at substantiating the higher output but was also necessary to examine any advance in safety criteria emerging from the Industry's practice or the NII's requirements over the years since the last projects were launched. In particular, NII specifically asked for a re-assessment of the gas baffle. This has now been completed and provides a positive demonstration that the integrity is high, but in addition it is tolerant to large defects.

Work is continuing on the second, more detailed phase of the study aimed at the production of a comprehensive design and safety submission.

Overview

Not surprisingly the Chernobyl disaster brought an immediate fall in support for nuclear power in the U.K. Despite explanations of the differences in design and operation of U.K. reactors and differences in licensing procedures, the political
<table>
<thead>
<tr>
<th>PARAMETER</th>
<th>UNITS</th>
<th>HUNTERSTON B (TYPICAL 1986)</th>
<th>TORNESS</th>
<th>PROPOSED A.G.R.</th>
</tr>
</thead>
<tbody>
<tr>
<td>No. OF FUEL CHANNELS.</td>
<td>MW</td>
<td>308</td>
<td>332</td>
<td>332</td>
</tr>
<tr>
<td>REACTOR HEAT</td>
<td>MW</td>
<td>1535</td>
<td>1523</td>
<td>1740</td>
</tr>
<tr>
<td>MEAN CHANNEL RATING</td>
<td>MW</td>
<td>4.98</td>
<td>4.89</td>
<td>5.24</td>
</tr>
<tr>
<td>PEAK CHANNEL RATING</td>
<td>MW</td>
<td>6.44</td>
<td>6.26</td>
<td>6.70</td>
</tr>
<tr>
<td>GAS PRESSURE UNDER TOP SLAB</td>
<td>bar a</td>
<td>39.3</td>
<td>41.0</td>
<td>42.5</td>
</tr>
<tr>
<td>CIRCULATOR GAS OUTLET TEMP.</td>
<td>°C</td>
<td>286</td>
<td>298</td>
<td>297</td>
</tr>
<tr>
<td>BULK CHANNEL GAS OUTLET TEMP.</td>
<td>°C</td>
<td>634</td>
<td>639</td>
<td>639</td>
</tr>
<tr>
<td>CIRCULATOR POWER CONSUMPTION</td>
<td>MW(e)</td>
<td>40.5</td>
<td>42.1</td>
<td>45.2</td>
</tr>
<tr>
<td>BOILER STEAM FLOW</td>
<td>kg/s</td>
<td>515</td>
<td>525</td>
<td>525</td>
</tr>
<tr>
<td>STEAM PRESSURE AT H.P.TURBINE</td>
<td>bar a</td>
<td>155</td>
<td>167</td>
<td>160</td>
</tr>
<tr>
<td>STEAM TEMPERATURE AT H.P.TURBINE</td>
<td>°C</td>
<td>478</td>
<td>538</td>
<td>538</td>
</tr>
<tr>
<td>FEED TEMPERATURE</td>
<td>°C</td>
<td>156</td>
<td>158</td>
<td>142</td>
</tr>
<tr>
<td>GROSS GENERATED POWER</td>
<td>MW(e)</td>
<td>640</td>
<td>705</td>
<td>763</td>
</tr>
<tr>
<td>NET POWER SENT OUT</td>
<td>MW(e)</td>
<td>592</td>
<td>645</td>
<td>700</td>
</tr>
</tbody>
</table>

FIG.17 A G R PARAMETERS

Parties in opposition are calling for either phasing out of nuclear power or a period of re-assessment before authorising new construction.

The report from the Inspector of the Sizewell Inquiry into the proposal by CEGB to build a PWR is expected in the immediate future. Much now depends on the content of this report and how the Government, still strongly supporting nuclear power, decides to proceed. With the likely size of nuclear ordering programme, it would be unrealistic to build more than one nuclear system in the U.K. Even with the most optimistic view of the nuclear ordering programme, this would incur heavy penalties in costs and performance in maintaining the infrastructure.

Clearly there would be no incentive to consider a change in the U.K. from well-tried CO₂ cooled technology to embark on the lengthy business of bringing any new gas-cooled system such as the HTR to commercial success. Notwithstanding the excellent safety characteristics of the AGR which have permitted siting close to urban locations, there has as yet been no serious interest in combining these with heat production for large industrial processes. Nevertheless the continuance of gas-cooled technology in its present form in the UK must be of continuing assistance in the efforts to develop HTGR in Europe and the US, in the better appreciation of what gas-cooled technology has to offer, including the transfer of engineering experience and development know-how, much of which can be directly related to HTGR requirements.
HIGH TEMPERATURE GAS-COOLED REACTORS

R. SCHULTEN
Institut für Reaktorentwicklung,
Kernforschungsanlage Jülich GmbH,
Jülich, Federal Republic of Germany

Abstract

The historical development in energy technology has the goal of realizing high temperatures and high efficiencies for the energy conversion processes. The high temperature reactor (HTR) realizes this goal in the best way compared to other reactor types; it uses ceramic materials and helium as the heat transfer media. The development of the fuel elements on the basis of coated particles is considered to be finished. Future applications are the production of electricity and the production of process heat for other areas of the energy market. Originally the thorium/uranium fuel cycle has been envisaged, today the low enriched uranium fuel cycle is of interest, because of the low prices of uranium. Typical for the HTR is its flexibility with respect to different concepts. On the basis of the existing prototypes AVR, THTR-300 and Forth St.Vrain the following two development lines are in discussion: reactors of small unit size and reactors of medium unit size. The safety qualities have been demonstrated by the operation of the AVR reactor in Jülich, which is now in operation for 19 years. These experiences will be of great importance for future plants.

1. Fundamental concepts

The historical goal of the technology of power station, as well as in energy technology in general, is the realization and application of high temperatures for the processes of energy conversion. The high temperature reactor realizes this goal to the best compared
to other reactor types. With high temperatures in the range between 700 and 1000 °C high efficiencies are realized for the production of electricity and other applications in the energy market are possible. The realization of high temperatures is possible because of the application of ceramic materials. In combination with requirements from the neutronic side the material for the moderator and for the construction of the HTR is graphite. As the coolant the inert gas helium is applied. The pressure of the primary circuit lies between 40 and 60 bars. On the basis of these principles research, development and demonstration work has been done since about 25 years in the Federal Republic of Germany, in the United States, in the USSR, Great Britain and in Japan as well as in some other countries.

One of the most important successes of the R+D-work in HTR technology is the technique of the coated particles, FIG. 1. In the Federal Republic of Germany the pebble bed typ fuel element has been developed and is used. The pebble contains the coated particles in an inner part, which is surrounded by an outer layer free of coated particles. This concept of a fuel element is considered to be an industrial product.

FIG. 1: HTR fuel element
Its development is also a success. An important feature is that very high burn-ups can be achieved. The release of fission products during normal operation is very small. The temperature stability of the fuel elements is also in the range of 1600 to 1800 °C very good. By experiments it has been shown that the release of fission products in this temperature range is minimal. Because of this good temperature stability the concept of the HTR-Modul-reactor and the concept of the HTR-industrial-reactor have realized a particular safety concept.

In this concept practically no radioactivity is released to the surroundings in the heaviest accident. In addition to that this concept realizes repairability after the heaviest accident.

In the Federal Republic of Germany the pebble type fuel element has been developed as well as in the Soviet Union. In contrary to that the bloc type fuel element has been developed in the USA, FIG. 2. The main feature of this fuel element is a hexagonal bloc made from graphite. This bloc contains bore-holes
for the fuel sticks and bore-holes to guide the coolant helium. The bloc type fuel element is applicable for smaller and larger reactor cores.

2. Applications

The HTR is first of all, as well as other reactor types too, developed for the production of electricity. The presently established technology is, to produce steam and to use that steam in turbines for the production of mechanical and electrical energy. The HTR realizes conditions for that steam as they are used in conventional power plants. Fundamentally the application of gas turbines is also possible in direct and indirect loops. This technology has been intensively developed in the seventieths. During this time the further realization of gas turbines could not be continued because the know how on the behaviour of the primary loop was not sufficient. The advantages of the application of gas turbines are remarkable, therefore its application may be of interest in the next decades.

Beside the production of electricity the HTR is also applicable for the production of process heat. In cogeneration of electricity and process steam a broad variety of applications does exist in chemical industry. An area of particular interest in the coming decades is the production of oil via enhanced oil recovery, here injection steam can be produced with the HTR. An intensive research, development and demonstration program has been done for the application of nuclear process heat for the refinement and conversion of coal, oil and gas for the production of environmentally benine liquid energy carriers, which can be used as fuel for motor cars and for heating purposes. The advantages are the following: application of nuclear energy instead of fossil fuels for the process heat, reduction of problems with respect to the environment, particularly the reduction of the
Problems of carbon dioxide. Today the latter applications are not possible because of the low prices of oil. But it is expected that these applications will be economically attractive in the coming decades and it is expected that they will be of great importance in future because liquid energy carriers are required in the energy world supply and because the CO₂-effects on the climate may become a problem.

3. Fuel cycle

The fuel cycle of the HTR is very flexible. The original goal of the development work in the world has been the application of the thorium/uranium-fuel cycle. In calculations it have been shown that with this fuel cycle a near breeder can be realized. In this conference there will be given a report, which shows, that even a breeding process can be realized. Our own calculations, which have been finished five years ago, have shown that a conversion factor of about 0.8 is economically the optimum, this is of course influenced by the costs of the reprocessing of the fuel. A conversion factor of about 0.8 realizes a utilization of the nuclear fuel which is five times higher compared to a fuel cycle without reprocessing.

Today the fuel cycle with low enriched uranium (8 %) without reprocessing and with direct disposal is preferred. The reason is the low price of uranium and its sufficient availability. This fuel cycle has the advantage that almost all plutonium produced in the reactor is used. In addition to that the produced plutonium contains much higher isotopes; therefore it is not dangerous with respect to proliferation. The utilization of uranium with this fuel cycle is almost as high as in the fuel cycle of a light water reactor (LWR) with one reprocessing and recycling of the produced plutonium. In contrary to this fuel cycle proposed in the Federal Republic of Germany,, a fuel cycle with an enrichment of 20 % will be used in the United States.
4. HTR conceptual designs

The pebble type fuel element allows a high flexibility with respect to the dimensions of the reactor core. It is possible to realize reactor cores with small diameters and relatively large heights, as well as cores with a large diameter and smaller heights. The principle of the pebble bed reactor is explained in FIG. 3.

![FIG. 3: The principle of the pebble bed reactor](image)

The pebble bed is cooled by the heat transfer medium helium. The pebble bed type fuel elements are transported by a pneumatic system to the upper part of the core, and they are extracted from the lower part. In principle two fuel loading modes are possible: there is the once through than out mode and the multiple mode, in which the fuel elements are recirculated. Both fuel loading modes have been investigated intensively. During this conference the differences of the two modes will be discussed.

An additional flexibility is realized because two types of reactor pressure vessels are feasible:
the prestressed concrete vessel and the steel vessel. The prototype for HTRs with steel vessel is the AVR in Jülich. This reactor has during its almost 20 years of operation resulted in fundamental know-how on the HTR, FIG. 4.

The German reactor constructors have on the basis of the AVR developed interesting concepts for small HTR units. At first I would like to mention the Modul-HTR of the company Interatom, FIG. 5. In addition to that I would like to mention the HTR-100-concept of the company HRB, FIG. 6. Both proposals are presented
in this conference. It is my opinion that these small HTR units are of interest for all applications in which smaller power rates are of interest. By the combination of several small units power plants in the range of 200 to 500 MWel may be realized. In the USA too the idea of smaller reactors is discussed. This interesting development will also be presented in this conference.
The second concept of the HTR is characterized by the prestressed concrete vessel. A prototype of this kind is the THTR-300 in Schmehausen. This reactor went into operation during this year and has reached 100 % electricity production recently, FIG. 7. This reactor is the base for the next HTR, the HTR-500. The planning of the HTR, which is done by HRB and a group of utilities has started recently. On this conference there will be report about it. Parallel to this there has to be mentioned the Fort StVrain-reactor in the USA. This reactor is in operation since years, about the result a report will be given during this conference.

5. Safety qualities

The AVR-reactor in Jülich is excellently suitable to demonstrate various safety qualities of the HTR. In particular it has been proofen very good as a plant for the tests during the development for fuel elements.
The AVR operates at a mean helium outlet temperature of 950 °C. Even at this high temperature the primary coolant is relatively clean. The contamination is very low. In the AVR-reactor experiments have been performed in which the loss of the coolant without scram has been simulated. The results have been that the negative temperature coefficient of the reactivity is fully effective and that a stabilization of the temperature niveaus in the reactor core takes place. This is due to inherent transport mechanisms for heat in the core. In 1978 there happened an incident by an ingress of water from the steam generator. This incident has produced much know how on effects on steam and water to the reactor core. The operation of the AVR will continue in the years 1987 and 1988. The main purposes are experiments which are safety related.

The concept of the small HTRs in the range of 200 to 250 MWth has been developed on the basis of the experiences gained in the AVR. The most important aspect of the concept of the small HTR is that in heavy accidents the maximum temperature of the core does not exceed 1600 to 1800 °C in the reactor core. Because of this the release of fission products from the fuel elements is rather small. Therefore these reactors have a particular safety quality. In addition to that there is a fundamental possibility of repairability after heavy accidents. It is my opinion that this reactor concept because of its particular safety quality is of great importance for the future.

As mentioned before there are the plans from the utilities and from the company HRB to realize a follower-reactor to the THTR with a power of 500 to 550 MWe. In this concept particular safety qualities of the concrete vessel will be used. Right now an interesting experimental work is done in the KFA Jülich on the behaviour of concrete vessels. The goal is to demonstrate heat resistance of the concrete upto temperature of 1300 °C.
6. Status of the development

The status of the research, development and demonstration work on the HTR is summarized as follows:
- the development of the pebble type fuel element and the bloc type fuel element is almost finished. The high quality of the fuel elements have been demonstrated by the operation of plants as well as by large experimental programs.
- There have been developed concepts for small HTR on the basis of the operational experiences of the AVR-reactor. In the Federal Republic of Germany as well as in the United States there are undertaken efforts for the realization of such small reactors and for the demonstration of their particular safety qualities. The planned small reactors in the USSR also belongs to this category.
- And finally with the successful operation with the THTR-300 it might be possible to offer of follower-reactor with medium unit size.
EXPERIENCE WITH GAS-COOLED REACTORS

(Session B)
DEVELOPMENT OF A HIGH OUTPUT AGR

B.A. KEEN
National Nuclear Corporation Ltd,
Knutsford, United Kingdom

Abstract

The paper outlines the design of a new AGR based on the Heysham II/Torness reactors which are nearing completion of their construction stage. The intent is to replicate as far as possible the Heysham II/Torness design in meeting the target station generator output of 1526 MW(e), and this can be achieved by design changes only to the HP steam pipework.

Two additional design options, a vented containment and a dry refuelling route, are also under consideration.

1. INTRODUCTION

The Advanced Gas-Cooled Reactor Power Stations now being constructed at Heysham in Lancashire, England, and Torness in East Lothian, Scotland, represent the current stage of development of the commercial AGR. Each power station has two reactor turbo-generator units designed for a total station output of 2 x 660 MW(e) gross, although it is currently intended to uprate this as far as possible towards the figures discussed later in this paper.

The construction of these new AGRs has been to programme and within budget. Fuel loading for the first reactor at Heysham is expected late in 1986 with the other three reactors following over the subsequent twelve months. The design of both stations has been based on the successful operating AGRs at Hinkley Point and Hunterston which have now been in service for almost ten years, although minor changes were made to meet new safety requirements and to make improvements suggested by operating experience.

At the Sizewell 'B' Public Inquiry, the CEGB gave a commitment to retain the AGR option. NNC is engaged in the definition of the improved AGR design. In order to use the invaluable experience gained since the last two AGR stations were ordered, it was decided that the new designs should be developed from the Heysham II/Torness design. As part of this study, it was necessary to address the
specific safety requirements for future AGRs presented to the PWR Inquiry by the Nuclear Installations Inspectorate.

In the interest of continued safety improvement aimed at mitigating the consequences of accidents however unlikely these may be, consideration is also being given to a design of vented containment. The move by the CEGB to construct a national dry fuel store has created interest in a dry refuelling route as another possible development. It should be noted that the design study is not yet complete and no decisions have yet been taken on these options.

2. REACTOR DESIGN

The new design of power station has two identical reactors of 1740 MW thermal output driving individual turbo-generators giving a station gross electrical output of 2 x 763 MW(e). The additional rating of the turbing/generators will be accommodated by small changes to the design of the Heysham II/Torness units. Principal dimensions and design parameters are given in Table 1.

The reactor is contained in a single cavity prestressed concrete pressure vessel (PCPV), a section through the vessel being shown in Figure 1. The reactor core is supported on a diagrid and enveloped by the gas baffle, which is welded to the pressure vessel liner floor to prevent movement in the event of an earthquake. The gas baffle enables the graphite moderator and other core structural components to be adequately cooled during normal operation.

The reactor fuel is made from slightly enriched uranium in the form of sintered uranium oxide pellets. The pellets are assembled in multi-start spirally ribbed stainless steel cans to form fuel pins, 36 of which are enclosed in a graphite sleeve to form a fuel element. The fuel elements are to the new Stage II design with single thick sleeves to improve their resistance to fuel handling loadings, and with a lower flow resistance. Each fuel assembly comprises a bottom support unit and reflector, eight fuel elements, a top reflector and a lower gag unit threaded onto a tie-bar which is suspended from a plug unit which incorporates a flow control gag unit.

Eight gas circulators deliver carbon dioxide gas at 42.5 bar a and 297°C into the plenum below the diagrid. Heat is transferred to the gas from the fuel raising its temperature to 645°C. Main boiler units are arranged circumferentially in the annulus formed by the gas baffle and the reactor pressure vessel liner. Each boiler has high pressure and
### TABLE I

**PRINCIPAL DIMENSIONS AND PARAMETERS**

<table>
<thead>
<tr>
<th><strong>Pressure Vessel</strong></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Internal height</td>
<td>21.9m</td>
</tr>
<tr>
<td>Internal diameter</td>
<td>20.3m</td>
</tr>
<tr>
<td>Wall thickness</td>
<td>5.8m</td>
</tr>
<tr>
<td>Top slab centre thickness</td>
<td>5.4m</td>
</tr>
<tr>
<td>Bottom slab centre thickness</td>
<td>7.5m</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th><strong>Reactor Primary Circuit</strong></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Gas baffle internal diameter</td>
<td>13.9m</td>
</tr>
<tr>
<td>Gas baffle dome height above liner floor</td>
<td>19.9m</td>
</tr>
<tr>
<td>Core restraint tank outside diameter</td>
<td>13.6m</td>
</tr>
<tr>
<td>Type of main boiler</td>
<td>2 start, once through Serpentine platen rectangular unit</td>
</tr>
<tr>
<td>Number of main boilers</td>
<td>4</td>
</tr>
<tr>
<td>Number of boiler units per main boiler</td>
<td>3</td>
</tr>
<tr>
<td>Type of decay heat boiler</td>
<td>1 start, once through Serpentine platen 8 (2/quadrant) Centrifugal, single stage encapsulated, variable inlet guide vanes</td>
</tr>
<tr>
<td>Number of gas circulators</td>
<td></td>
</tr>
<tr>
<td>Type of gas circulators</td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th><strong>Reactor Core</strong></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Number of fuel assemblies</td>
<td>332</td>
</tr>
<tr>
<td>Number of control assemblies</td>
<td>89</td>
</tr>
<tr>
<td>Number of secondary shutdown system channels</td>
<td>163</td>
</tr>
<tr>
<td>Height of core (including neutron shields)</td>
<td>12.8m</td>
</tr>
<tr>
<td>Overall width of core across flats</td>
<td>12.4m (regular 16 sided polygon)</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th><strong>Reactor Parameters</strong></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor heat</td>
<td>1740 MW</td>
</tr>
<tr>
<td>Bulk circulator outlet gas temperature</td>
<td>297°C</td>
</tr>
<tr>
<td>Bulk fuel channel outlet gas temperature</td>
<td>645°C at top of fuel temperature stack</td>
</tr>
<tr>
<td>Net circulator flow</td>
<td></td>
</tr>
<tr>
<td>Gas pressure at circulator outlet</td>
<td>4697 kg/s</td>
</tr>
<tr>
<td>Pressure difference across gas baffle dome</td>
<td>45.2 bar</td>
</tr>
<tr>
<td>Total primary circuit pressure drop</td>
<td>3.02 bar</td>
</tr>
<tr>
<td>Peak channel power</td>
<td>6.7 MW</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th><strong>Boiler Parameters</strong></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Feedwater flow</td>
<td>552 kg/s</td>
</tr>
<tr>
<td>Feedwater temperature</td>
<td>142°C</td>
</tr>
<tr>
<td>HP steam temperature</td>
<td>538°C</td>
</tr>
<tr>
<td>HP steam pressure</td>
<td>160 bar a</td>
</tr>
<tr>
<td>Reheater steam temperature</td>
<td>539°C</td>
</tr>
<tr>
<td>Reheater steam pressure</td>
<td>40.7 bar</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th><strong>Turbine - Generator Parameters</strong></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Turbine heat rate</td>
<td>2.31 kJ/kJ(e)</td>
</tr>
<tr>
<td>Nominal gross electrical output/reactor</td>
<td>763 MW</td>
</tr>
<tr>
<td>Nominal net electrical output for station</td>
<td>1400 MW</td>
</tr>
<tr>
<td>Design lifetime</td>
<td>30 full power years</td>
</tr>
</tbody>
</table>
reheat sections with separate feed and steam penetrations through the outer cylindrical surface of the pressure vessel. The main boilers and gas circulators are physically divided into four separate quadrants by division plates at the circulator inlet plenum below the boiler seal. During operation, feed water is supplied to the boilers at 142°C and 217 bar giving steam temperatures of 538°C. The gas circulators are driven directly from grid supplies using 45 MW from the station output. Gas flow control is achieved by adjusting the angle of guide vanes at the inlets of each circulator. A totally diverse secondary system for the removal of decay heat is provided via the decay heat boilers (DHB). These are commissioned automatically following a trip and have their own feed system and heat sink.

The primary system for control and shutdown of the reactor comprises absorber rods and drives housed in standpipes in the top cap of the PCPV. The actuator and chain store is mounted between a closure unit and radiological shield plug through which the chain passes and from which hang the articulated tubular control rods.
A secondary shutdown system (SSD) designed to inject nitrogen into the core is provided as a diverse means of shutting the reactor down in the remote fault situation, should a predetermined number of control rods fail to enter into the core when required. This system operates from the SSD room beneath the PCPV. Boron glass beads can be injected by the operator into the core to provide a long-term reactivity hold-down capability in the event of continued failure to release the primary shutdown rods before the reactor is depressurised.

Both reactors are served by the fuel handling facilities located between them, a single fuelling machine and other shared equipment. These facilities provide for the storage and assembly of new fuel, the transfer of fuel in and out of the reactors and decay storage tubes, and subsequent dismantling of the fuel, servicing of reusable plug unit components and disposal of debris, the subsequent pond storage of irradiated fuel and finally its despatch from site.

3. DESIGN CONSTRAINTS

The main constraint on the development of the design was to replicate as closely as possible the Heysham/Torness design. Only changes to meet new safety requirements or simplification of the design which would not lead to significant alterations to station layout were considered. Operating constraints were also applied to ensure that the experience of the existing AGRs were still valid and that the safety case prepared for Heysham/Torness remained applicable.

Within these constraints, and with the exceptions given below, it has been shown that sufficient flexibility remains in the design parameters to allow for uncertainties and to ensure that the target station generator output of 1526 MW(e) will be achieved for the operating lifetime of the plant of the equivalent of 30 full power years. It is possible that greater output might be achieved after commissioning has demonstrated the expected available margins.

4 HAZARD DESIGN

4.1 Earthquakes

A new enveloping Safe Shutdown Earthquake (SSE) has been defined for all future nuclear stations in the UK. This SSE is based on historical evidence of seismic activity on the British mainland and has a peak acceleration of 0.25g. The effect of this earthquake on the buildings and plant will be dependant on the soil structure at the site. The SSE loadings have been derived for the two extremes of UK
site, with the most onerous conditions being used as the design basis. All essential plant and structures are designed to survive this design basis SSE without preventing the shutdown and post-trip cooling of the plant. In order to prevent a sudden loss of capability at earthquakes with accelerations just beyond the design basis, the aim of the design has been to provide a conservative response at the SSE such that adequate shutdown and cooling would be available for earthquakes up to 0.35g and beyond.

4.2 Aircraft crash

The frequency of aircraft crash on the vulnerable areas of the station has been assessed for typical sites "outside" and "within" Areas of Intense Aerial Activity. The study confirmed the impact frequency was very low for both areas and that with the high degree of separation and segregation of plant in the Heysham II/Torness layout, the risk of causing a radiological hazard was not significant. The addition of a fire fighting system in the reactor pile cap area to contain aviation fuel fires reduces the risk further.

5. DESIGN CHANGES

The effect of the higher output on the design of Heysham and Torness has necessitated significant design changes only to the HP steam pipework to accommodate the higher steam flows without excessive pressure drops, and to meet the extended station lifetime.

Manufacturing and construction experience gained during the Heysham and Torness projects has been used to make detailed changes in various areas of the design to simplify manufacture, ease construction and/or to reduce costs. None of the changes adopted will significantly affect the object of replication.

The safety case for on-load refuelling at Heysham II and Torness has only been presented for refuelling in batches as part power levels with consequential loss of output. Simplification of the protection systems will enable the time taken for each batch to be reduced, while further generic work on refuelling is expected to provide the means for increasing the present limit on refuelling power towards full power and single channel refuelling. These improvements will give a significant increase in reactor availability.
6. VENTED CONTAINMENT

AGR and Magnox Safety Cases have always been made on the basis that no credible accident can be identified which would require secondary containment of the reactor. Until recently, for Heysham II/Torness and earlier AGRs, the probability of a dropped fuel assembly during refuelling had been considered to be so low that it was not necessary to take account of it in the safety cases. However, should it occur, vented containment of the sub-pile cap volume would reduce the activity release from such an accident.

Other potential release areas in the reactor building have also been considered, even though no accidents have been postulated which would require such provision. The key areas identified for CO₂ discharges are:

- Pile Cap
- Quadrants
- SSD Room
- CO₂ Bypass and Clean-up Circuit
- Fuel Decay Store

Each of the five areas will have its own high integrity ventilation system, broadly based upon those which currently exist on Heysham II/Torness AGRs.

In the event of a breach of the pressure circuit, the normally running ventilation plant is isolated and a hot gas release route is opened to discharge the escaping gas directly to atmosphere.

The new plant connected to the above ventilation routes by normally closed dampers, would comprise heat exchanger, high efficiency particulate air and charcoal filters, and extract fans discharging to a new stack. The plant would start up automatically on high temperature or CO₂ concentration and when the discharge flow through the hot gas route reduces to the capacity of the new plant, dampers would close to route all the discharge through the new plant. For discharges within the capacity of the filter system, isolation would be immediate.

In the unlikely event of a dropped assembly, fuel damage and activity release would be immediate, therefore the new plant would considerably reduce the consequences. Other breaches within the design basis, although initially exceeding the capacity of the plant, would not give rise to early fuel failures. For faults beyond the design basis, the plant would mitigate the consequences by reducing the total release of activity.
7. DRY FUEL ROUTE

Recently the CEGB have decided to build a large central dry store away from the reactors, capable of accepting irradiated fuel and acting as a buffer between them and the final reprocessing at Sellafield. This new development makes it logical to review the possibility of adopting a dry fuel route for any future AGR.

The present sequence at Heysham/Torness is for irradiated fuel to be placed in one of 30 buffer storage tubes after removal from the reactor by the fuelling machine. The buffer storage tube is water cooled, the heat being transferred from the fuel stringer to the water by natural circulation of pressurised CO₂. After 28 days the decay heat has reached a level where cooling by air at atmospheric pressure is adequate and the fuel is transferred to one of two Irradiated Fuel Disposal Cells to be dismantled. The elements are then transferred to the pond where it is stored for about 100 days before dispatch.

The dry route would follow the same sequence, but the number of buffer storage tubes would need to be increased to 50 to allow storage for 100 days. The increased decay time is required to permit the decay heat to reduce to acceptable levels before the fuel is placed in the dry store after passing through the IFD cell. In the dry store, the fuel will remain for a minimum of 265 days before it can be despatched from site in a dry fuel flask.

The dry store is separate from the reactor building with loading and unloading cells at one end so that the store may be extended during station life should this become necessary. The initial capacity proposed is 1280 elements, stacked 16 high in 80 tubes, cooled externally by natural circulation of air.

There are no technical obstacles and a dry route would remove any doubts about the long-term storage of AGR fuel after passing through a pond system, although development of a dry road transport flask would be required.

8 CONSTRUCTION PROGRAMME

Based on the experience gained at Heysham and Torness the construction period from start of permanent works to commercial load of the first reactor would be reduced by 3 months to 75 months.
9 LICENSING PROGRESS

As the high output AGR is very similar to Heysham and Torness, their Station Safety Report, SSR, forms the basis for the licensing of any future AGR in the UK. The SSR was submitted to the NII in 1986 in preparation for start-up of the first reactor at Heysham. An assessment of the extra requirements raised by the NII at the Sizewell Inquiry has shown that while extra safety studies will be required, no significant design changes are expected to be made.

10 CONCLUSIONS

The successful progress of the Heysham and Torness projects in the UK coupled with the good operating experience of the similar Hinkley Point and Hunterston reactors forms a strong base for the design of a high output AGR. In producing the new design, the major aim has been to replicate the Heysham and Torness AGRs with the minimum of changes. The increase in output of 15% to 1526 MW(e) gross generator output coupled with a reduction in the construction programme and costs improves the competitiveness of the new design.

With a modest investment, two further developments, vented containment and a dry refuelling route, could be added. The former would mitigate the release from faults which are beyond the present design basis. The latter would be a logical step in view of the intention to build a national dry fuel store.

An assessment of the NII requirements for a future AGR in the UK as stated at the Sizewell Public Inquiry has shown that no major difficulties are expected in obtaining an operating license, although design substantiation will be required.
TECHNICAL EVOLUTION AND OPERATION OF FRENCH CO\textsubscript{2} COOLED REACTORS (UNGG)

Y. BERTHION
CEA, Centre d'études nucléaires de Saclay,
Gif-sur-Yvette, France

Abstract

The technical evolution of the five French CO\textsubscript{2} cooled reactors (UNGG) from 1981 to 1986 needs to be outlined. These technical evolutions concerned the fuel element of Bugey 1 which is now slightly enriched, as well as the load reduction operation required by the grid. In addition work in underway to increase the safety at the two St Laurent units, or to repair the hot steel upper-structures of Chinon-3 unit.

1. INTRODUCTION

The technical evolution of the five French CO\textsubscript{2} cooled reactors (UNGG) from 1981 to 1986 needs to be outlined. This evolution will be illustrated by examples taken from a survey made at each nuclear power plant. I will speak of the steel corrosion and the repairs (ISIS) at the Chinon-3 unit, the erosion - corrosion of heat exchanger and safety at the two St Laurent units, the graphite corrosion and the load-reduction operation at Bugey.

Evolution in the last years is illustrated by figure 1 which gives the cumulated load factors of each unit and the predicted cumulated load factor quoted PEON. One can see the shutdown for repair at Chinon-3 unit started on May 4, 1984. But figure 1 does

![Figure 1: UNGG cumulated load factors versus time](image-url)
not give a correct idea of the availability factor of the last two years; from now on, the predominance of nuclear generation in the French electrical network will require the nuclear units to take part in load-following and frequency adjustment of the grid. In France the PWR units equipped with the control mode G allowing power variations are cheaper than UNGG units. Also the grid requires load reduction of the UNGG reactors.

2. THE CHINON NUCLEAR POWER PLANT

Three UNGG nuclear reactors are located in Chinon.

- The CHINON-A1 reactor which shutdown definitively in 1973 now serves as a museum and this since February 3, 1986.

- The CHINON-A2 reactor was shutdown for economical reasons in June 1985 after 20 years in operation. Its cumulative load factor was 73.7 % i.e. close to the PBON prediction (figure 1), and have a last run of 869 days without scram. The management of the fuel load was optimized and during the last 8 months no new fuel element was loaded. The radial shuffling of fuel elements and the overmoderation obtained with the feeding of the fuel channel by graphite log provided the needed reactivity to operate and produced 970 GWeh of extra electricity. The average availability factor of the last five years was 86 % with a maximum of 94.5 % in 1984. During the life of this reactor, 133,374 fuel elements were loaded with only 6 clad failures. Some shuffled fuel elements are now in hot cells for examination. Good results are expected and the radial shuffling will be planned for the end-of-life of other UNGG reactors in order to save fuel. Table I shows the reliability of UNGG fuel elements.

<table>
<thead>
<tr>
<th>Number of</th>
<th>Graphite Core fuel elements</th>
<th>Annular fuel element</th>
</tr>
</thead>
<tbody>
<tr>
<td>unloaded</td>
<td>SICRAL F1</td>
<td></td>
</tr>
<tr>
<td>fuel</td>
<td>369,020</td>
<td>75,550</td>
</tr>
<tr>
<td>elements</td>
<td></td>
<td></td>
</tr>
<tr>
<td>loaded</td>
<td>542,250</td>
<td>88,140</td>
</tr>
<tr>
<td>fuel</td>
<td></td>
<td></td>
</tr>
<tr>
<td>elements</td>
<td>10</td>
<td>3</td>
</tr>
<tr>
<td>clad</td>
<td></td>
<td></td>
</tr>
<tr>
<td>failures</td>
<td></td>
<td></td>
</tr>
<tr>
<td>failure</td>
<td>&lt; 2 \times 10^{-5}</td>
<td>&lt; 4 \times 10^{-5}</td>
</tr>
<tr>
<td>rate</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

- The CHINON-3 reactor has been shutdown since May 4, 1984 in order to repair the steel corroded upper structure of the reactor. Figure 2 shows a vertical section of this unit. The corrosion did not have the same consequences on the Chinon-A2 reactor whose structures were bolted instead of welded (figure 3), nor on the subsequent reactors because the CO2 steel corrosion was already known. This operation of structure repairing is called ISIS and the cost up to mid-1987 will amount to 300 MF. The
FIGURE 2: CHINON-3 reactor vertical section

FIGURE 3: Mild steels corrosion mechanisms by CO2

erection of a building was necessary to shelter a true-scale model of a sixth of the upper structure. Rigorous correspondence between the model and the upper structure is obtained by a laser telemetry done on the reactor. Five robots are necessary: three on the reactor, i.e., one for the camera, one for the tools and one for the metal to be welded. The two others are on the model for learning purposes. The reactor will operate again in mid-1987. Steel corrosion was responsible for
the nominal power lowering: 360 MWe instead of 480 MWe (see Table II). On the other reactors, the nominal power was reduced for different reasons. But local defects where steel corrosion is involved have been noted and their evolution is being carefully watched.

TABLE II. POWER OF UNGG REACTOR

<table>
<thead>
<tr>
<th>UNIT</th>
<th>Design Power (MWe)</th>
<th>Effective Power (MWe)</th>
<th>Difficulty</th>
</tr>
</thead>
<tbody>
<tr>
<td>CHINON A-2</td>
<td>480</td>
<td>360</td>
<td>Stell corrosion</td>
</tr>
<tr>
<td>CHINON A-3</td>
<td>480</td>
<td>390</td>
<td>erosion-corrosion</td>
</tr>
<tr>
<td>SL1</td>
<td>480</td>
<td>390</td>
<td>erosion-corrosion</td>
</tr>
<tr>
<td>SL2</td>
<td>515</td>
<td>515</td>
<td>on heat exchanger</td>
</tr>
<tr>
<td>Bugey</td>
<td>540</td>
<td>540</td>
<td>graphite corrosion</td>
</tr>
</tbody>
</table>

3. THE BUGEY NUCLEAR POWER PLANT

A UNGG unit is located on the Bugey nuclear power plant. This unit differs from the others in its fuel elements which are externally and internally cooled, as well as in the CO₂ pressure which is 43 bars instead of 29 bars. These two factors allow for a greater specific power. But high specific power and high pressure mean high rate of graphite corrosion. The maximum local graphite corrosion measured is 22.4 % at 8.58 aep (year equivalent full power). The mechanical behavior of such a corroded block is poor, and mechanical and seismic calculations are under way in order to evaluate a maximum acceptable level of corrosion and to determine the remaining lifetime of the reactor. To day the lifetime is estimated to be 2 aep. So this unit is now considered as an energy tank. In France the marginal cost of PWR is the cheapest and these reactors operate in load following mode. So from an economical point of view the grid necessitates a load reduction of these UNGG reactors. Therefore, the optimal management of this energy tank prompts the operator to define several operation levels acceptable by the fuel and the reactor (Table III).

TABLE III. OPERATION LEVEL

<table>
<thead>
<tr>
<th>Power (MWe)</th>
<th>Comment</th>
</tr>
</thead>
<tbody>
<tr>
<td>540</td>
<td>Exceptional grid demand</td>
</tr>
<tr>
<td>470</td>
<td>No more than 500 hours/year</td>
</tr>
<tr>
<td>400, 300, 200</td>
<td>Normal operation level upon grid demand</td>
</tr>
<tr>
<td></td>
<td>Ten times a year.</td>
</tr>
</tbody>
</table>
This operating mode requires more from the fuel elements, so in order to increase the load-reduction number the fuel behavior is examined in hot cells. To reduce the graphite corrosion, methane is injected into the CO₂ with a 450 VPM level. The decomposition of methane produces a carboxyhydrogenated deposit which reduces the heat transfer from the fuel, and the reactivity with the neutron capture on hydrogen. The high hydrogen content in the stack has made necessary the use of a slightly enriched uranium (0.76 % instead of 0.72 %) since June 28, 1984. This enriched fuel gives better power distribution, and flexibility. The low increase in cost is compensated for by saving fuel in the outer part of the reactor. To increase the burnup of the outer fuel radial shuffling is not used at the moment.

4. THE SAINT-LAURENT NUCLEAR POWER PLANT

Two UNGG units called SL1 and SL2 are located on this site. Figure 4 shows a vertical section of the SL1 reactor. The SL2 unit is very similar, as is the Vandéllos unit in Spain. The nominal power was reduced (Table II) because of an erosion-corrosion problem of the heat exchanger. This phenomenon originated from a conception flaw, and the vaporization level was changed in reducing the power. Meanwhile, during the first operating years, the exchanger was affected by erosion-corrosion, and today some leaks still occur.

The heat exchangers are divided into two separate parts and the leaks affected mainly one of the two halves in both SL1 and Vandéllos. So, some other phenomenon may also be involved. The water quality is important, and at Vandéllos better results were recently obtained with AMP than with morpholine.

Over the last five years, however, work has been focussed on a reexamination of the Safety Report in the light of feedback experience. Among these works are:

. EAR* use as ultimate emergency system.
. The building of an independant emergency control room named BUS.
. Increasing blowing availability by reducing the common-causes failure with fire retarding bulkheads and the separation of cableway, and by supplying auxiliary sets with main steam.
. The possibility, in case of scram, of feeding turboblower with low quality main steam. This forced running provides more than half an hour of blowing.
. The use of false tulip-shaped fittings on each fuel channel in order to prevent plugging.

In addition to the erosion-corrosion problem and work on safety, a third point concerning the two Saint-Laurent units is a thirty-year extension of their life. To achieve this extension, the consequences of steel and graphite corrosion will have to be studied.

* heat exchanger cooling the outage unit.
Nota : la turbosoufflante est ramenée dans le plan de coupe.

Tension appliquée à chaque câble : 185 t. effc.
Longueur totale des câbles : 191.5 km
Poids au mètre de chaque câble : 12.3 kg
Ceint. sup. + ceint. inf. : 11.4 km
Dalle sup. + dalle inf. : 37.8 km

FIGURE 4 : Saint-Laurent 1 reactor vertical section
TABLE IV. HEAT EXCHANGER LEAK NUMBER

<table>
<thead>
<tr>
<th>UNIT</th>
<th>leak number</th>
</tr>
</thead>
<tbody>
<tr>
<td>SL1</td>
<td>39</td>
</tr>
<tr>
<td>SL2</td>
<td>15</td>
</tr>
<tr>
<td>Vandelles</td>
<td>82</td>
</tr>
</tbody>
</table>

5. CONCLUSION

Nuclear UNGG power plants today are somewhat overshadowed by PWR reactors which, in France, produce more than 65 % of all electricity generated at the lowest cost.

Nevertheless, there is reason to be proud of the fact that these UNGG reactors, with which France entered the nuclear age, have aged well despite their early flaws and discontinued commercial development. These flaws, steel and graphite corrosion, erosion-corrosion of heat exchangers, were compensated by an acceptable reduction of the nominal power. ISIS work showed the realism of a large-scale undertaking on upper reactor internals.

The Chernobyl event has widened the field of our safety research on the UNGG reactor type. Indeed, we now need to find the right things to do in case of a severe accident. For instance, the use of lead would be ill-advised because it would create an eutectic with the magnesium fuel clads.
FORT ST. VRAIN PERFORMANCE

H.L. BREY
Public Service Company of Colorado,
Denver, Colorado,
United States of America

Abstract

Fort St. Vrain, on the system of Public Service Company of Colorado, is the only high temperature gas-cooled power reactor in the United States. This plant was designed by General Atomic Company and utilizes helium as the primary coolant. The core consists of triso-coated uranium and thorium fuel particles cast into cylindrical rods within prismatic graphite blocks. The primary coolant system consists of the reactor, twelve steam generator modules, and four helium circulators contained within a prestressed concrete reactor vessel. The once-through steam generators provide high quality 1000°F, 2400 psi main steam and 1000°F reheat steam to the main turbine. The plant has generated 3,334,000 MWH of electricity and has undergone 3 reactor refuelings.

Many of the primary system components at Fort St. Vrain are proto-typical in nature and this plant has undergone an extensive and elaborate testing program. Difficulties primarily with the helium circulators and fuel element column movements particularly at high power levels have resulted in significant modifications.

Power operation at 100% was achieved in late 1981 with a net thermal efficiency of nearly 39%. Excellent performance of the coated fuel particles has contributed to a full power circulating activity sixty times lower than the design with subsequent cleanliness throughout the plant. Except for 1985, when a combined total radiological exposure for all personnel reached 35 manrem due to the necessity to completely refurbish all of the control rod drive mechanisms, the average total combined exposure for each of the past five years was under 3 manrem. This exemplary radiological cleanliness has, with the exception of tritium, allowed Fort St. Vrain to consistently operate with noble gas airborne and liquid effluent releases more than an order of magnitude lower than the average for the U.S. nuclear power industry.

Fort St. Vrain (FSV), on the system of Public Service Company of Colorado, is the only high temperature gas-cooled (HTGR) power reactor in the United States. This 330 megawatt net electric generating plant features a helium-cooled reactor with an uranium-thorium fuel cycle.

PLANT DESIGN

A pre-stressed concrete reactor vessel (PCRV) contains the total primary coolant system including the reactor, steam generators, and helium circulators (Figure 1). The reactor is graphite moderated and reflected with an active core consisting of 1482 hexagonal...
fuel elements loaded with triso-coated uranium and thorium particles cast into cylindrical rods (Figure 2). Helium, at 700 psia (100% of rated power), is discharged from four helium circulators and passes down through the core picking up heat from the fission process. This heat is then distributed equally through twelve steam generator modules.

With the exception of the unique once-through steam generators, the secondary side of the plant is similar to conventional fossil-fueled units (Fig. 3). Main steam leaves the steam generator modules and enters the throttle of the high-pressure turbine with 2400 psi, 1000°F conditions. Cold reheat steam then leaves the high-pressure turbine and provides the driving force for the four helium circulators, with subsequent reheat to 1000°F in the upper (reheater) section of each steam generator module. This reheated steam is then directed back to the intermediate- and low-pressure turbines and finally to the condenser.

The conventional condensate system features a full-flow demineralizer, three low-pressure heaters, and deaerator. Three boiler feed pumps (two turbine-driven and one motor-driven) direct the feedwater through two high-pressure heaters back to the steam generator modules, thus completing the cycle. The rated steam conditions help provide for a full-load plant thermal efficiency of approximately 39% (Ref. 1).
Many of the primary system components of FSV are prototypical in nature and this plant has undergone an extensive and elaborate testing program including:

- pre-operational evaluation of individual systems prior to initial reactor fueling in 1974
- low power physics tests to qualify the design and operability of the reactor and some primary coolant components, at low power levels and
an initial rise-to-power program to evaluate and verify
all plant systems through a step-by-step power escalation
program.

The unique design of the components within the primary coolant
system, especially with the helium circulators, resulted in some
delay in initial plant start-up. Major examples of equipment
deficiencies involves two situations concerning the helium circulator
pelton wheels. Each circulator uses a pelton wheel as its back-up
power drive in the event main steam is unavailable. It became
evident in the initial testing program that corrective measures
were needed to prevent both cavitation and stress cracking of these
wheels. The corrective measure to prevent cavitation involved
the design and installation of a pelton cavity pressurization system.
Prevention of stress cracking involved replacement of the cast
Inconel wheels with wrought Inconel material.

Reactor performance was closely evaluated throughout the low
power and rise-to-power testing programs. With the exception of
a phenomena known as "core fluctuations", reactor performance was
in close agreement with design requirements. Core fluctuations
became prevalent as power level was increased towards 70% with
a corresponding increase in reactor differential pressure. This
phenomenon was to be the major cause of testing and the design
and installation of special instrumentation both ex- and in-core
which required 2½ years. The core fluctuation phenomenon was caused
by small movements in fuel and reflector elements (moving as a
column) caused by combined thermal-hydraulic effects. These small
column movements resulted in redistribution of the core coolant, and
corresponding significant changes in individual fuel region outlet
temperatures and steam generator module inlet temperatures.

Consider, in Figure 1, that the core is made up of separate
refuelling regions where the helium flow through each region can
be individually regulated (orificed). This, coupled with a design
where the elements are fixed in place at the bottom of the core
but not at the top, with clearances between elements for thermal
expansion and refuelling, set up the probability of lateral internal
core movement. Non-uniform temperatures and helium flows primarily
between refuelling regions caused localized thermal bowing of the
fuel columns which resulted in redistribution of the coolant flow.
This redistribution would then cause a further localized change
in the temperature profile and, over a time span (averaging about
10 min), column bowing would occur in a new pattern thus resulting
in fluctuations of region outlet and steam generator module
temperatures. These core fluctuations were corrected in late 1979
when constraining devices were installed on the upper elements
between each refuelling region of the core.

Subsequent testing to 100% power has shown that the FSV core
experiences a small movement resulting in a slight redistribution
of some individual fuel region coolant outlet temperatures at high
power levels. However, with installation of the constraining
devices, this redistribution is not cyclic in nature and does not
represent a plant safety issue (Ref. 2).

The rise-to-power testing program continued after installation
of the constraining devices and on November 6, 1981, FSV obtained
100% power operation. Plant performance at 100% power was generally trouble free and a thermal efficiency of 38.5% was achieved.

The majority of the problems at FSV have been associated with mechanical hardware such as valves and pumps. System complexity, improper valve selection, inadequate pump and compressor capacity, or electrical malfunctions have been the principal reasons for unfavorable plant performance; whereas, with the exception of the core fluctuations, the nuclear reactor and its auxiliaries have performed quite well (Ref. 3).

**PLANT OPERATION**

The first-of-a-kind nature of many systems at FSV has resulted in reduced plant reliability. Although the helium circulators themselves have performed well and have operated at their full design values, the complexity of the helium circulator auxiliaries and corresponding operational problems have been major concerns. Unfortunately, a large number of plant transients have been initiated by improper response/interaction between the helium circulators and their auxiliaries. These transients, as well as operator errors, have resulted in sizable quantities of circulator bearing water entering the primary coolant helium or contaminated primary coolant helium entering the auxiliary components, with possible release into the reactor building (Ref. 4). Cleaning of moisture from the primary coolant has been a principal contributor towards low plant availability.

Figure 4 is a graphical representation of plant power output from the beginning of 1980. Plant operation to 1984 was generally good with shutdowns for two refuelings and a major modification to the circulator auxiliaries which essentially provided isolation so that a circulator upset in one loop would not affect the circulators of the opposite loop. In mid-1984 an upset occurred on the plant where six of the thirty-seven control rod pairs required
manual operator action for insertion after receipt of a reactor scram command. As a result, all thirty-seven control rod drive mechanisms were subsequently extensively refurbished and brought to a "near-new" condition. The control rod drive refurbishment program was completed in July, 1985. However, extensive discussions had been in progress since January, 1985, with our Nuclear Regulatory Commission (NRC) pertaining to the ability of FSV to meet regulatory requirements (10CFR50.49) associated with Environmental Qualification of electrical equipment (EQ).

In the late 1970's, a sizable effort was undertaken to qualify plant electrical equipment to the harsh environment involving a high-energy steam line break. This effort was documented and submitted to the NRC in the 1979-1980 time frame. A subsequent review by the NRC of our EQ Program in late 1984 led to three areas of concern. These included equipment aging, equipment operability time, and operator response time. In conjunction with operator response time, the testing program to qualify equipment in the late 1970's was based upon the operator's ability to isolate a high energy steam line break within four minutes. The subsequent temperature profile in the reactor and turbine buildings resulting from four minutes of blowing steam was utilized in the equipment testing program. The NRC has subsequently determined that credit could not be given for proper operator response in less than ten minutes. The good steam conditions afforded by FSV's design made the temperature profile for a ten-minute isolation time prohibitive in qualifying our equipment and a program to design and install a very fast steam line break detection and isolation system was undertaken. Installation of this system allows us to take credit for previously tested equipment as environment temperatures now resulting from a steam line break are substantially lower.

The plant is currently shut down to complete the Environmental Qualification effort. Start-up is anticipated for January, 1987.

RADIOLOGICAL PERFORMANCE

The radiological consequences of operating FSV have been exceptionally low. The personnel exposure rates and discharges to the environment have been consistently orders of magnitude below those of the average light water reactor (LWR) power plant in the United States. The major reason for this performance is inherent in the design of the core and primary coolant system. Release of contamination into the reactor building environment is minimized by (1) the unique graphite core structure, with ceramic-coated fuel particles, (2) the use of inert primary coolant gas, helium, and (3) the PCRV with its double sealed pressurized (with purified helium) penetrations.

The equilibrium radioactivity circulating in the primary coolant was conservatively calculated to be 30,900 Ci (design level) with an expected value of 2630 Ci. Actual measurements at 100% power show 439 Ci with the third cycle core (Ref. 5).

Figure 5 provides a comparison of FSV's personnel exposure history with that of the U.S. light water reactors. The radiological exposures for each of the years 1980 through 1983 was less than 1 man-rem grand total for all personnel badge at FSV. Total
exposure levels increased to 35 man-rem in 1985 due almost entirely to the control rod refurbishment program. So far, in 1986 total personnel exposure has been less than 1 man-rem.

A total of 150 m³ of low-level solid waste has been accumulated at FSV in the past 12 years. This includes waste generated as a result of (1) initial fuel loading, (2) three refuelings, each with approximately one-sixth of the reactor core replaced, and (3) numerous plant modifications and maintenance operations, including major work performed on components within the primary coolant system.

With the exception of tritium, noble gas airborne and liquid effluent releases from FSV have been greater than an order of magnitude below the average of the U.S. nuclear power industry. Higher than anticipated amounts of tritium have been experienced at FSV primarily because of upsets in the helium circulator auxiliary system, which have allowed small amounts of bearing water to enter and circulate in the primary coolant system. The tritiated moisture is removed by the purification system and is discharged from the plant as liquid waste (Ref. 6).

FUTURE

Although the last two years have been difficult for FSV because of the control rod drive and EQ programs, Public Service Company of Colorado (PSC) has been committed to making FSV a viable electricity-generating plant. To help accomplish this, the FSV Improvement Committee was formed to investigate possible plant modifications which would enhance plant availability. Moisture ingress emanating from the high pressure bearing water within the existing helium circulators has been the primary cause for low plant availability and is therefore the prime candidate for review by this Committee. Possible actions currently being considered by the Committee include investigation of the following possibilities:
Replacement of the existing circulators with machines that utilize magnetic thrust bearings while continuing to incorporate the existing pelton wheel and single stage steam turbine drives. The use of magnetic thrust bearings would allow the elimination of high pressure bearing water; however, very low pressure water would be used as both the lubricating medium on the journal bearings as well as a sealant to prevent the primary coolant from flowing down the circulator shaft.

Evaluate the possibility of extensively modifying the existing circulator auxiliaries in the following manner:

- complete separation of bearing water and buffer helium systems for each of the four circulators
- elimination of back-up and emergency bearing water sources by incorporating individual bearing water pumps for each circulator powered by Class 1 electrical systems.

We also may elect to evaluate the possibility of installing hydrostatic seals on the existing circulator shafts which would tend to seal the bearing cartridge from the primary system upon a pressure differential reversal.

If the decision by the Committee is to proceed on these evaluation efforts, they will continue over the next 18 months. Final selection of circulator improvements would not be anticipated until these reviews have been completed (Ref. 7).

Poor operation of FSV throughout the past two years has resulted in PSC's recent financial evaluation of this Plant with respect to overall Company economics. This evaluation resulted in the following actions:

- A write down of PSC's asset in its Fort St. Vrain Plant involving approximately two thirds of the current capital cost associated with the plant.
- Removal of Fort St. Vrain from the rate base. However, a partially escalatable rate of 4.8¢/kwh is allowed for all net generation from Fort St. Vrain.
- An extensive study will be undertaken to consider numerous alternatives relative to the future of Fort St. Vrain.

These actions are intended to shield the rate payer from all FSV operating consequences. They are not meant to imply that PSC has lost faith in the future of FSV or the HTGR, or intends to shut the Plant down. They provide the nuclear staff with a well-defined financial operating climate in that successful operation will be rewarded, and if good performance does not materialize, these actions limit the risk placed on the Company and our shareholders.
CONCLUSIONS

Considerable effort has been expended over the past two years to bring the plant to a high degree of readiness. This includes enhancement in the areas of management, training, increased technical staffing, preventive maintenance, and procedure controls. PSC is confident that successful continuous plant operation will be achieved beginning in early 1987.

REFERENCES

AVR EXPERIENCE

E. ZIERMANN
Arbeitsgemeinschaft Versuchsreaktor (AVR) GmbH,
Jülich, Federal Republic of Germany

Abstract

The paper contains a description of the Arbeitsgemeinschaft Versuchsreaktor (AVR), a small pebble bed HTGR located in Jülich, Federal Republic of Germany. Besides a description of plant design, the experience made with main components, the operating performance and operational results are presented together with an outlook to possible future plant improvements.

1. Introduction

The Arbeitsgemeinschaft Versuchsreaktor (AVR) GmbH is an association of 15 electric utilities established in 1959. The aim of this association was to obtain scientific, technological and economic information about the reactor system from the construction and operation of an experimental nuclear power plant with a high-temperature reactor. In August 1959, the order was placed with the company partnership BBC/Krupp for the construction of the experimental power station. Work on the plant began in 1961. In August 1966, the reactor became first critical, and on December 17th, 1967, electricity was supplied to the public supply grid for the first time. In February 1974, the mean gas outlet temperature, which had initially been set at 850 °C, was increased to 950 °C.

The initial aim of experimental operation was to demonstrate that a helium-cooled pebble-bed reactor can be constructed and operated safely. In addition to continuous component testing and the collection of operating experience and results, the plant was also used in the following period for the implementation of experiments. In particular, these included the testing of various types of fuel elements, investigations of fission product behaviour in the circuit as well as experiments on the chemistry of coolant gas impurities.
FIG. 1. View of the AVR reactor building.

2. Structure of the plant

The core of the reactor consists of a pebble-bed containing 100,000 fuel spheres with a diameter of 6 cm in a graphite reflector pot. The steam generator is located above this core and is shielded against radiation from the core by a 500 mm thick top reflector made of graphite and two additional 500 mm thick layers of carbon brick. The coolant is helium which is pressurized at 10 bar and circulated by two blowers located in the lower section of the reactor vessel. The helium is heated in the core from 270 °C to 950 °C and then flows through the steam generator, where it transfers the energy to the water-steam circuit. The generated steam has a pressure of 73 bar and a temperature of 505 °C. The core, steam generator and blowers are surrounded by two concentrically arranged reactor vessels. The interspace between the vessels is filled with pure
helium at a pressure slightly above the coolant gas pressure. An initial biological shield is located between the cylindrical sections of the reactor vessel. This double vessel system is again surrounded by a safety containment and a 1.5 m thick concrete tower.

The reflector, made from graphite blocks, has four bored bulges, which project into the core space. One shutdown rod is fitted into each of these bores.

These shutdown rods have the task of compensating for the difference in reactivity between the hot and cold critical condition, and to keep the reactor subcritical in the cold
condition by at least 0.5%. During power operation, the fuel elements are continuously circulated and measured regarding their burn-up. Spent fuel elements are removed and replaced by fresh fuel.

In the purification plant, the helium is purified of dust as well as active and inactive impurities.

3. Experience with the main components

3.1 Fuel elements

The fuel elements have an external diameter of 6 cm. In an internal zone of 50 mm diameter, the fuel is located in the form of 30,000 coated particles. These particles have a kernel of several tenths of a millimetre in diameter consisting of mixed crystals of uranium and thorium carbide, or uranium and thorium oxide. The kernels are surrounded by several layers of pyrolytically deposited carbon. The inner layer has a relatively slight density and is compressible.
By compressing this layer, storage space for the fission gases is produced. The external layer is the pressure-bearing encapsulation of the particle. An additional diffusion barrier layer is partially provided between these layers. With only a few exceptions, the fuel elements contain 1 g uranium-235 and different amounts of thorium and uranium-238.

Overall, fuel elements of different compositions and particle sizes were tested in large numbers under operating conditions up to high burn-ups. This showed that fuel elements with excellent fission product retention capacity can be produced. The average burn-up is noticeably above 100,000 MWd per tonne heavy metal, reaching a maximum of slightly more than 180,000 MWd per tonne heavy metal. Our very low coolant gas activity of about 25 Ci, at the present time, is a consequence of these very dense fuel elements.

3.2 Reflector

The inner side reflector and the top reflector consist of needle coke graphite ARS/AMT manufactured by Sigri-Elektrographit. The side reflector consists of extruded and the

FIG. 4. AVR — Top reflector.
top reflector of rammed blocks. Since the upper side reflector and the top reflector, next to the fuel elements, are exposed to high neutron radiation, there were fears that due to neutron-induced deformation stressing and cracks could have occurred in the reflector. It was also feared that there would be corrosive material erosion due to coolant gas impurities. These concerns led to a visual inspection of the top reflector and the upper side reflector in 1984. This inspection of the top reflektor and the visible sections of the upper side reflector showed that the reflector was in perfect condition. Cracks were not found, and corrosion wear was not detectable.

3.3 Fuelling system

The fuelling system permits insertion of the spheres against a helium pressure of 10 bar, the removal as well as the circulation of the spheres during power operation. The transfer devices are formed by ball valves and retaining latches. The circulation route starts with the sphere discharge tube, which has an internal diameter of 500 mm to avoid bridge formation. This is closed by a slotted disc, the so-called diminisher. By turning this disc, spheres fall in groups into the storage container of the singulizer. This singulizer is a scoop wheel, which raises the spheres out of the container and supplies them to the scrap separator. This is where defective spheres and fragments are separated. At present, the fracture rate is about one defective sphere per 30,000 circulated spheres. The intact spheres are passed to the conveyer via a dosing wheel. At this conveying point, the type and burn-up of the spheres is determined. On the basis of these results and input control data, a computer then decides in which of the five sectors the spheres are to be resupplied to the core, or whether they must be removed from circulation. This system has the considerable advantage that the burn-up can be continuously balanced by the addition of fresh fissile material. In addition, there is no necessity to shut down at a specific time for refuelling.
This system was developed from scratch for our power station and incorporates errors typical of a pioneering system of this type. Thus the different friction

a) Top reflector 1965.

b) Top reflector 1984.  

(VIDEO INSPECTION)

FIG. 5.
conditions under helium and air were not taken into adequate account, and the design of the individual components was not optimized. Despite these understandable deficiencies, the system performs satisfactorily. Up to now over 2.2 million spheres have been circulated. All defects were rectified in an acceptable time with low radiation exposure for the repair personnel. In part, these repairs were
carried out without interrupting power operation. Only less than 3% of the non-availability time was caused by this fuelling system.

3.4 Shutdown rods

The reactor has four shutdown rods, which are inserted into the reflector core from below. They are arranged symmetrically in a circle with a diameter of 2 m, whose centre point coincides with the vertical axis of the reactor core. The rods move in bores of the four graphite noses.

Each shutdown rod is connected to a heavier counter-rod via a rack- and -pinion, so that when the coupling between the pinion and drive is opened, the absorber section is pushed into the core from below by gravity. The shutdown rods as such consist of concentric tubes with boron carbide filled in between them.

FIG. 7. Sectional view of the AVR reactor building.
The drop and motion times of the shutdown rods are measured. Up to now no changes in the motion and drop characteristics of the rods have been noted. The rods themselves have operated maintenance- and trouble-free since the commissioning of the system. The gearing of the racks and pinions was checked in 1980. Here again, no defects were noted. In the region of the rotary penetrations, we have replaced the metal bellows, as the theoretically expected service life had been reached. As a precaution, all bearings in the drive section were also replaced at the same time.

3.5 Steam generator

The steam generator consists of four parallel Benson (once through) systems. The temperature is controlled by water injection outside the reactor vessel. In front of this injection, each system consists of five parallel tubings.

FIG. 8. AVR — Steam generator.
The superheating section behind the injection consists of ten parallel tubings per system. Thus the steam generator consists of a total of 60 separate tube loops. The proportion of superheater tube surface is 5% of the total heating surface. To avoid power losses through possible failures of tube loops, the steam generator was designed with a standby heating surface of 10%. The tubes of the steam generator are made from ferritic steels, as are normally used in conventional boiler construction.

After 72,000 hours of trouble-free operation, a leak occurred in one final superheater tube. Even though the cross-section of the hole was finally only about 3 mm², 27 tonnes of water entered into the primary system, flowed through the core and collected in the lower vessel section.

The inflow of such a large amount of water was possible, because firstly the chemical composition of the leakage water resulted in wrong conclusions as to the source of the leakage, and secondly no measuring instruments were available which could have indicated the volume of penetrating water.

Whilst the location and closure of the defective steam generator tube took place without particular difficulty, the removal of the penetrated water from the primary circuit and the drying of the system required a considerable amount of time. The difficulties during the removal of the water were mainly caused by the lack of drainage possibilities. Thus a residue of 2 m³ water could only be removed by evacuation.

The cause of the leak is still unclear at present, since we have not removed the steam generator due to the expense it would incur. However, failure due to creep is excluded, since the steam generator was so conservatively designed that up to now only a slight portion of the theoretical life expectancy has been utilized. Since completion of the repair work and recommissioning of the plant, no further damage has occurred. Up to now the service life has been slightly in excess of 111,000 operating hours.
3.6 Blowers

The two coolant gas blowers in the lower section of the reactor vessel each consists of an asynchronous motor, mounted in oil-lubricated plain bearings, and an impeller mounted in an overhung position on the motor shaft. Labyrinths and a seal gas system prevent oil from entering the coolant gas circuit and coolant gas from entering the motor casing. The speed is adjusted via a frequency changer. The blowers are designed for maintenance-free operation. With the exception of one bearing exchange in 1978, which required the removal of one blower, these machines have been operating maintenance-free since 1966, even though after the steam generator defect both machines were submerged in water above their axles.
FIG. 10. AVR — Cut steam generator pipes.

FIG. 11. AVR — Welding of steam generator pipe to steam collector.
The removal of this one blower was necessary because, after maintenance work on the lubrication system, cooling water had penetrated into the bearing oil system through a leaking screw union.

Removal of the blower and replacement of the bearing did not cause any major problems. During the work, it was noted that the blower was in very good condition, and that it showed no damage due to ageing.

![Diagram of cooling gas circulator](image)

**FIG. 12. AVR — Cooling gas circulator.**

3.7 **Vessels and supporting structures**

The inner reactor vessel is made of 19 Mn 5 + Mo. It is subjected to virtually no stress during operation, as the interspace between the vessels filled with pure helium, is subject to almost the same pressure as the coolant gas in the inner vessel as such. Slight differential pressures only occur during the start-up and shutdown processes, which are far below the design pressure. Up to now, the inner vessel has only been actually stressed twice during pressure testing.
FIG. 13. AVR — Dismounting of the blower.

FIG. 14. Damaged radial bearing.
The outer reactor vessel is made of fine-grain steel HSB 45 with a thermal yield point of 250 N/mm$^2$ at 200 °C. The true load is below 200 °C at 127 N/mm$^2$, so that no limitation applies here either.

The supporting structure for the core and the graphite and carbon brick internals is made of 15 Mo 3. It is designed for a temperature of 450 °C. The actually measured temperature is 270 °C. Due to dimensional stability requirements, the limit load of 470 t should not deflect the structure by more than 1.5 mm. This requirement leads to exceptionally low stress loads. The supporting structure is protected against radiation from the core and the sphere discharge tube by a 15 cm thick steel base plate and shielded by a

FIG. 15. Sectional view of the AVR reactor building.
boron carbide shield so that the neutron dose is significantly below $10^{18}$ cm$^{-2}$ even after 19 years of operation. This means that this component is also operated well below its design values, so that a service life limitation is not expected.

3.8 Gas purification system

Through the addition of fresh fuel elements, through minor leaks in cooling equipment and experiments, as well as during maintenance and repair work on primary circuit components, air and moisture can enter into the primary system. To limit graphite corrosion, the level of impurities is kept low by continuous purification of the primary circuit. The suspended matter is retained in a gas prepurification system, which consists of two parallel gravel bed filters and coolers. The throughput of approx. $800 \text{ m}^3/\text{h}$ is produced by the delivery pressure of the coolant gas blowers.

From the helium prepurified in this manner, a bypass flow of $50 \text{ m}^3/\text{h}$ is extracted and passed through two purification systems arranged in tandem. The first of the systems consists of silicagel driers and activated charcoal adsorbers, which are kept at low temperatures using liquid nitrogen.
This section is operated in state of adsorption for approx. 3,000 hours, so that mainly noble fission gases are retained here.

In a catalytic oxidation stage, hydrogen and CO is oxidized. The subsequent gas purification stage also consists of liquid nitrogen-cooled adsorbers, which are operated in state of adsorption for a mean time of approx. 10 days. In addition to the oxidation products, mainly nitrogen is retained here. Both gas purification stages are duplicated so that when one stage is loaded, a changeover to the unit connected in parallel is possible. The loaded stage is desorbed by heating, and the desorbed substance is retained in delay tanks until the vast majority of the short-lived noble gases have decayed.

Initially, we had problems with the diaphragm compressors which pump the gas through these systems. After some of these compressors had been replaced by dry-running piston compressors, the systems now operate to our full satisfaction.

4. Operating performance

4.1 Power control

Our reactor has no control rods for power control. The temperature of the reactor is adjusted via the fuel loading, and the power is adjusted via the blower speed, i.e. via the helium mass flow, and thus through the energy extraction from the core. The power control is effected automatically via the negative temperature coefficient.

The steam produced by the steam generator is supplied to the turbine, which is fitted with an initial pressure regulator. It thus takes precisely the power which it is offered by the steam generator. The steam temperature is adjusted through the feedwater mass flow. The system operates in such a stable manner that we can practically do without the planned temperature control through water injection.
4.2 **Start-up procedure**

When starting up from the cold condition, two limitations must be mainly observed. The first limitation is set by the tensile strength of the carbon brick bridge above the core. So as to prevent the tensile stresses inside the bricks from exceeding the permissible values, due to the temperature gradients during heating, the hot gas temperature may only be raised by max. 3 K/min.

The second limitation arises from stability considerations for the steam generator. To avoid vaporization at the wrong places in the steam generator, steam production should only start when the helium entering the steam generator has a mean temperature exceeding 600 °C. From these limiting conditions, a start-up procedure has been devised which we divide into five stages. In the first stage, the reactor is operated until it is critical by withdrawal of the shutdown rods.

![Diagram](image)

**Legend:**
- $P_{TH}$ = THERMAL POWER
- $\bar{R}_{BLower}$ = SPEED OF BLOWERS
- $S_R$ = AVERAGE POSITION OF SHUT-DOWN RODS
- $\dot{m}_{FW}$ = MASS-FLOW OF FEED-WATER
- $S_{HG}$ = HOT-GAS-OUTLET TEMPERATURE
- $\dot{f}_w$ = FEED-WATER-TEMPERATURE
- $X$ = WET-STEAM

**FIG. 17. AVR — Start-up diagram.**
In the second stage, the power is increased so far that the steam generator can just be kept in water operation. With this power, the temperature is then raised to 600 °C in the third stage in accordance with the limitation specified above. When the temperature has reached 600 °C, vaporization in the steam generator is started by increasing the power and reducing the feedwater mass flow in the fourth stage. In the fifth stage, the power, coolant gas temperature and steam temperature are set to the required values, the turbine is put into operation and the generator is synchronized with the mains. When running-up the turbine to full capacity, limits again arise from the turbine material stressing. We need four and a half hours for start-up, from the beginning of rod withdrawal up to the synchronization of the generator.

4.3 Shutdown procedures

Three different procedures are available for shutdown:

4.3.1 Scram

If the max. permissible neutron flux or the max. admissible neutron flux change is exceeded, an automatic scram takes place. The rods drop into place, the coolant gas blowers are switched off and the generator is disconnected from the mains. After such an automatic shutdown, the rods are manually withdrawn to half their length again so as to reduce thermal stresses. To cool down the reactor system, the feedwater mass flow is set at approx. 40%.

4.3.2 Insertion of the shutdown rods by motor drive is the second procedure

All other activations from the reactor protection system cause an automatic insertion of the shutdown rods, combined with a shutdown of the coolant gas blowers and the shutdown of the turbine. Even after this shutdown, the shutdown rods are withdrawn again to half their length, and the feedwater mass flow is set at approx. 40%. After about three hours, the blowers are then started up again in stages, so as to accelerate the heat removal.
4.3.3 The third procedure is switching off the coolant gas blowers

Switching off the coolant gas blowers is the most gentle and slowest shutdown process. It is always used if the reactor is to be shut down specifically, with the rods initially remaining in position. Only after some time they are inserted into the core position, when the reactor has considerably cooled down. As with the other shutdown procedures, the feedwater mass flow is set at approx. 40%, and the blowers are put into operation in stages after three hours.
4.4 Load changes

Since load changes only involve slight temperature variations in the core and reflector, the limitation for the rate of change only arises from the max. possible change in blower speed.

We have carried out load reductions from 100% to 50% in two minutes. When raising the power, equally rapid changes are basically possible.

However, it must be noted that when changing the blower speed the removed amount of energy is set. The core power is automatically readjusted via the negative temperature coefficient. The inertia in the system causes the power to overshoot. Where the power is carelessly raised, this overshoot could result in a shutdown of the reactor, due to reaching the limiting value for the max. admissible power.

NEUTRONFLUX RECORD DATE: 02.14.1984

FIG. 19. AVR — Dynamic experiment.
4.5 Performance during failure of the afterheat removal system and the shutdown rods

We have demonstrated by way of experiment that after failure of the active heat removal during power operation, with simultaneous failure of the shutdown device, the reactor shuts down automatically and remains subcritical for about one day. The temperatures that arise in the fuel remain within the range of temperatures prevailing during operation. Calculations by computer show that with a simultaneous loss of coolant gas the max. fuel temperatures would not reach critical levels.

As described above, we have taken this event, which was originally considered to be a serious emergency condition, as one of our standard shutdown procedures.

---

FIG. 20. AVR — Simulated failure of shut-down equipment and interrupted decay heat removal for that time.
5. **Operational results**

5.1 **Availability**

Since the first supply of electricity on December 17th, 1967 until today, 1.5 thousand million kW hours of electric power have been produced. This is equivalent to 4,100 full-load days. The mean time utilization is on average approx. 68%.

We achieved the best figure in 1976 with 92%. The number of shutdowns due to incidents was noticeably reduced during the first few years, and since then has fluctuated at figures of less than 5.

Since we have focused our attention not only on the demonstration of good operating results and component testing, but for some time increasingly on the implementation of experiments, which we will still intensify in future, the time utilization in the coming years will be slightly reduced.

![Legend: Operation Factor, Load Factor](image)

**FIG. 21. Availability of the AVR power plant over the years 1968–85.**
5.2 Radiation exposure to the personnel

The collective annual dose for all the personnel employed varies between 0.5 and 0.6 Sv (50 to 60 man x rem). All in all it has a decreasing tendency.

During the 19 years of operation the permissible individual doses have so far never been exceeded.

5.3 Emission of radioactive substances

The emission of radioactive substances into the atmosphere is slight. It ranges between 15 and 30 Ci/a for noble gases, about 50 Ci/a tritium and 1 to 2 Ci C-14.

The emission of aerosols is within the μCi range, whilst iodine is virtually not present in the exhaust air.
FIG. 23. Doses to personnel in AVR power plant over the years 1968–85.

6. **Possible improvements as seen by the operator**

Naturally, we can see from our plant that improvements to both the design as well as the individual components are possible. Thus the division of the biological shield into two sections is a definite disadvantage. The consequence of this is that numerous facilities which require maintenance, such as measuring equipment, valves, compressors, etc., are located in areas exposed to increased radiation. The relatively poor top shielding of the reactor building is also a disadvantage. Due to reflection of the radiation on the atmosphere (sky-shine effect) this leads to areas of slightly increased radiation fields at the power station site.
The occasional difficulties with the fuelling system have already been mentioned. Similarly, we initially had difficulties with the diaphragm compressors, solenoid valves, the frequency changer for the coolant gas blower, etc.

However, we are also aware that our experience has been fully taken into account in the design of subsequent plants, so that these items are AVR-specific, and are by no means characteristic of an HTR.

7. **Summary**

Our AVR experimental nuclear power station is the first plant with a helium-cooled, high-temperature reactor using spherical fuel elements. Despite the resultant imperfections and difficulties, the plant was built in a relatively short time, and has now been successfully operated for over 19 years. It has been found that the reactor has excellent safety characteristics, and that the radiation exposure for the personnel as well as for the environment is very low. Operation of the plant is unproblematic and does not make any increased demands on the operating personnel. All disturbances that have arisen were mainly rectified by our own operating crew.

The condition of the main components does not indicate any changes due to ageing. Our operating experience and the pre-exposure of the plant do not suggest any limitations to the further use of the plant.
THTR OPERATION — THE FIRST YEAR

D. SCHWARZ
Vereinigte Elektrizitätswerke Westfalen AG,
Dortmund, Federal Republic of Germany

Abstract

The paper contains a description of the performance of the THTR-300 since its first criticality in 1985. The THTR-300 is a pebble bed HTGR of 300 MWe located in Schmehausen/Uentrop in the Federal Republic of Germany. The main steps and events during the start-up and commissioning phase are presented and discussed in detail.

Main features and overall valuation

One year ago, Sept. 6, 1985 the THTR 300 had reached first criticality.

One year later (by Sept. 6, 1986), the reactor has been critical for 5,153 hours. First current was sent out on Nov. 16, 1985 at a power level of 40 %; 60 % have been reached on March 19, 1986, 80 % on April 10, 1986. The power plant has produced a total of 420 million KWh within 3,071 hours by Sept. 6, 1986.

In the meantime, on Sept. 23, 1986 THTR has achieved the 100 % power level. It has been operated at that level till Oct. 3, 1986. Then it has been shut down for a 6 weeks-revision, after having produced a total of 540 million KWh.

Comparing THTR commissioning with that of other nuclear and conventional power plants we can state a very satisfactory lack of any big problem. Yet commissioning took much more time than with plants that had big problems. The main reason for this seeming contradiction is the fact that THTR 300 is a prototype. This is expressed in

the great number of experimental transients required, by far exceeding one hundred
the problems of staying between very narrow design margins of
temperature, pressure, flow etc. during those transients, lea-
ding to changes in procedures and repetitions of the experimen-
tal program

small, however annoying problems with components, control, and
operator error, and

special carefulness of the authority and of experts acting on
behalf of the authority, leading to time consuming persistence
on formal procedures, on the experimental program, and on tech-
nical margins. Practically every activity needs expert opinion
and the authority's consent, for major steps in the commissioning
program we have to get a release, and for changes that have more
than zero relevance with respect to safety we need a new license.
Needless to say that much time is needed for preparing and getting
decisions and for the paperwork before and after every activity.

In addition, Chernobyl cost us a total of 61 days, 22 due to an in-
terruption of the experimental program (operation at steady power)
39 due to a politically motivated shutdown.

In view of these circumstances, the progress in commissioning THTR
as it has been achieved up to now can be valued as very satisfactory.

THTR operation from Sept. 1985 to Sept. 1986

Let us now analyse the power raising history of THTR in some more
detail, as given in fig. 1 and 2. There have been 37 events (35 shut-
downs and 2 load reductions) since first criticality (the 37th at the
beginning of Sept. 1986): 13 shutdowns and 1 load reduction can be
attributed to instrumentation and control (hard- and software); 10 ca-
eses have been due to operator error, often involving control problems;
in another 10 cases the plant had to be shut down for mechanical rea-
sons in the conventional part of the plant, mostly leaks. 2 shutdowns
resulted from planned experiments, 1 load reduction had an external
cause.
FIG. 1. Performance diagram during commissioning (THTR 300)
(Phase LI) — February 1986

FIG. 2. Performance diagram during commissioning (THTR 300).
These are normal numbers, as they do occur also with our coal fired power plants - with other words: they are very good for a prototype plant. The same judgement can be drawn from the fact that the majority of unplanned shutdowns happened during experimental transients; the number of which exceeded the number of ensuing shutdowns by far.

The experimental program included load following, shutoff of one of the two turbine generators for the motor driven gas circulators, shutoff of one of the two turbine driven feed-water pumps and switch to the stand-by motor driven pump, the so-called fast cool down program (not so fast as the so-called aftercooling mode which functioned excellently from the beginning), load shedding with ensuing station power production, switching between different station power supplies (station transformer, grid, diesel engines), shutoff of one steam generator and putting it back into operation (for this, the load of the other five steam generators has to be reduced), and so on. Several experiments had to be repeated at different power levels. A minor number of repetitions resulted from unsuccessful trials.

This experimental program accounted for most of the commissioning time. In addition, we had a rather long shutdown period, from Nov. 22, 1985 to Jan. 19, 1986, that was necessary for several plant improvements and the ensuing paperwork. The activities included: Inspection of turbine bearings and control rods, improvement of the fuelling system with respect to hardware (components) and software (computer program), adding thermal insulation to the feed-water and steam lines above the reactor pressure vessel, cleaning of the cooler in the decay heat removing system, exchange of flexible into fixed hydraulic tubing for the feed-water and steam tripping valves, finishing instrumentation and control etc. Later on shutdown times were also used for inspection and minor improvements and sometimes determined the duration of a standstill. The period from Febr. 7 to Febr. 17, 1986 was mainly used for modifications of instrumentation and control, resulting from operating experience.

The second major interruption, costing us another two months, was a consequence of the Chernobyl accident. Thereafter, judging the general climate, we decided by ourselves to stop the experimental program and to go on with the production of electricity at a safe level.
(40 %). We did so from May 5 to May 27, 1986 with only one interruption (malfunction during a transient due to a steam generator check-up). On May 27 we resumed our commissioning program. However, on May 30 - such things always happen late on Fridays, because they are then broadly covered in weekend media and politicians cannot ask their staff - the minister responsible for the supervision of THTR was caught on not being informed about a minor release of aerosols that had happened already on May 4. The release amounted to 1.2 m Ci and added 0.1 Bq/m² to the 50 000 Bq/m² from Chernobyl that had rained down near THTR one day earlier. These 1.2 m Ci were below the allowed release of 2 m Ci/d (10 m Ci/a) which is already a very low value amounting to about 1/30 of the specific values per MW installed German LWRs may emit. The staff of the ministry, the supervisory experts (TÜV) and the THTR safety council had all been informed about the release and had correctly valued it as unimportant. After the first excitement, we tried to reach an agreement with the minister to go on with the operation in parallel with investigating the incident and improving plant and procedures. However, political forces were too strongly against such a compromise. So we had a 39 day shutdown. One third of the time was needed for the formulation of requirements which had either been fulfilled in advance or were fulfilled shortly thereafter. The rest of the time was spent for getting expert opinion and the final consent to restart the plant. We tried to make the best of it, carrying through some changes that also needed the consent of experts and authority.

After restart on July 10, 1986 we did not resume power production at 80 % as already achieved in April; instead, for about six weeks, we operated the plant at 40 % and raised power to 60 % several times to repeat experimental transients from that level. This became necessary because experience with the 80 % operation had led to modifications of procedures; in addition, one of the transients had to be repeated two more times until it was successful.

The power level of basically 40 % during that period had several reasons. One originated from a near zero availability of the refuelling equipment, due to defective switch bearings, wrong labelling of pebbles etc. This resulted in a dense core inducing us to risk as little shutdowns as possible and to run the plant at a well-known power level.
Another consequence was that we could not change the composition of the core which was then characterised by high excess reactivity (enhanced by previous protactinium decay), deeply inserted core rods, and power production shifted to the lower, hot end of the core; in this situation the safest way was to burn down excess reactivity at a reduced power level.

Another reason for power reduction, which alone would have been sufficient at least at day-time, has been hot summer weather that impaired cooling in the steam pipe rooms above the reactor pressure vessel. There, according to our own application, temperature margins have been licensed that are lower than necessary from a safety point of view. Trying to adhere to those margins with additional insulation did not prove successful enough under all ambient conditions. Cooling the air with cold drinking water, a simple solution, was pursued for some time but was ruled out because of time consuming law of water procedures. Now we have applied for a temporary license to operate with higher temperatures that are still low enough for instruments, leak detection and concrete. The application for the final solution, an air recooling equipment, shall be submitted thereafter. At the end of August, the weather has turned unusually cold, so we have ample time to settle that question. Altogether load reductions for the reasons named above cost us several full power days.

Some more days were lost for a formally similar reason, i.e., we had to stick to margins and procedures that we had applied for and that had been so licensed, but that turned out to be over-conservative. This was the case, when - on several occasions with high excess reactivity and low xenon - we had to build up new xenon after a restart at about 30% thermal power for about one and a half days without producing electricity. We shall apply for a change of the license, but the authority has already indicated that this will only be granted after extended operating experience.

Please note that length of explanation is not equivalent to importance. To bring things back into perspective I have summarised THTR power production history in table 1.
Table 1  Synopsis of THTR power production

<table>
<thead>
<tr>
<th>Event Description</th>
<th>Days</th>
</tr>
</thead>
<tbody>
<tr>
<td>Time reviewed</td>
<td></td>
</tr>
<tr>
<td>16.11.1985 - 06.09.1986</td>
<td>294</td>
</tr>
<tr>
<td>Plant improvement, revision</td>
<td>58</td>
</tr>
<tr>
<td>22.11.1985 - 19.01.1986</td>
<td></td>
</tr>
<tr>
<td>Shutdown ordered by authority</td>
<td>39</td>
</tr>
<tr>
<td>01.06. - 10.07.1986</td>
<td></td>
</tr>
<tr>
<td>Shutdown or nuclear power only</td>
<td>69</td>
</tr>
<tr>
<td>For other reasons:</td>
<td></td>
</tr>
<tr>
<td>- Instrumentation and control, including improvements</td>
<td></td>
</tr>
<tr>
<td>- Leaks and other reasons in the balance of plant</td>
<td></td>
</tr>
<tr>
<td>- Xenon buildup, inspection, trial shutdowns and others</td>
<td></td>
</tr>
<tr>
<td>Power production</td>
<td>128</td>
</tr>
<tr>
<td>Reduced power production after Chernobyl</td>
<td>20</td>
</tr>
<tr>
<td>Without experiments 05. - 27.05.1986 minus 2 days shutdown</td>
<td></td>
</tr>
<tr>
<td>Reduced power production for various reasons (see text)</td>
<td>30</td>
</tr>
<tr>
<td>With experiments 10.07. - 21.08.1986 minus 12 days shutdown or nuclear power only</td>
<td></td>
</tr>
<tr>
<td>Power production as planned with experiments</td>
<td>78</td>
</tr>
</tbody>
</table>

Summing up, I wish to repeat the valuation given above, that in view of the circumstances the progress achieved up to now is very satisfactory.
The main reason of success is the faultless function of all essential components, which must not be forgotten over minor problems, however annoying they may be.

Experience with the first large pebble bed reactor

In addition to last year's history, the behaviour of the first large pebble bed reactor will be of special interest:

Core physics are not easy, because there is no incore instrumentation. Generally such an instrumentation is not necessary due to the large difference between operating and limiting temperatures. Only in special situations of the first core as in July 1986 we might wish to have it. We have fairly good knowledge about reactivity changes with insertion depths of individual rods and rod groups and about the radial distribution of axially integrated power production from temperature measurements under the core. These values, together with neutron flux measurements and results from pebble flow experiments are put into computer programs and compared to reality, finally ending up with a good understanding of the core behaviour. Computer modelling of the core with deeply inserted rods with ensuing shifts of flux and pebble flow, which we only will have with the first core, still needs improvement. Even under these conditions short term core calculations can be done with sufficient accuracy.

Fission product retention is very good. The measured values of noble gas release are practically identical with those anticipated by calculation as shown in fig 3. No influence can be seen from broken pebbles which is a result of the fact that fission products are mainly retained by the coated particles.

There have been more broken pebbles than anticipated, $10^3$ by an order of magnitude out of a total of 675 000. This has to be attributed to harsh experimental conditions: Lack of lubricating gas ($\text{NH}_3$) and a very dense core due to many core rod movements following the experimental program and due to relatively small amounts of pebbles withdrawn at the bottom. Those conditions will not prevail in normal operation. We thus expect that the number of broken pebbles will go back to values near zero.
The shutdown system has been amply discussed on German TV. It was said that the reflector rods cannot shut down the reactor (they do) and that the core rods alone cannot guarantee long-time cold subcriticality (they do, even with two or three of them totally outside the core). These facts have been confirmed once more by recent jurisdiction. On one occasion, 7 out of 42 core rods failed to reach their full insertion depth by a few inches. On a second move, however, they did go all the way down. This happened on a unique experimental occasion (dense core, all circulators running adding more density, and no NH$_3$) and even then the large reactivity margin already mentioned was not noticeably affected. In normal operation we have not had nor do we expect any trouble.

The fuelling equipment was the source of the small release of May 4, 1986. When switching from manual to automatic operation the gas content of the fuel entry lock was released to the atmosphere unfiltered. In the meantime, operational modes have been changed and filters have been backfitted.

A major problem with the fuelling equipment is the fact that the number of pebbles that can leave the core decreases above a power level of 40%. The higher the power, the higher the pressure difference at the gas circulators; this also affects the pressure drop.
at the lowermost end of the fuel discharge tube, where the cooling
gas that has come down the inner wall of the discharge penetration
changes its way to move up through the downcoming pebbles. Above
40 % power, the cooling gas blowing against the sliding pebbles
makes it increasingly difficult for them to come out of the box at
the end of the discharge tube. The problem will be solved by cutting
some extra holes into that box, allowing part of the gas to search
another way. The right size of these holes is investigated in a mock-
up, the remote cutting is trained in another mock-up. The cutting will
be done next year. Until then, it is possible to run the plant at 100 %
power on working days and do all the required refuelling at 40 % power
on weekends. For this operation mode we need a high availability of
the refuelling equipment. This was already achieved on the weekend of
Sept. 6/7, 1986, when we operated the plant at 80/40/80 % power as
planned.

Summing up, we can say that THTR 300 has proved the feasibility of large
pebble bed reactors with respect to all essential components. The nature
of the problems left is not generic, they are a question of layout. So
we can take full advantage of the properties of the pebble - high
stability, high fission product retention, low temperature differen-
ce between fuel and gas (opening the window to very high temperature
applications), easy handling, and underground storage of spent fuel
without or with very little conditioning. The possibility to go to
very high temperature is enhanced by a stratified core, that can be
easily brought about with the pebble bed by means of a once through
core. In addition we can take advantage of on-load refuelling - no
burnable poison at equilibrium, i. e. higher fuel utilisation, plus
the chance to reach high availability as with the heavy water reac-
tors of recent years. Already with THTR we expect short shutdowns
for inspection in the coming years, not yet, however, in this year.

Looking into the future, we can say that the technical and licensing
lessons learnt with the THTR 300 has led to the very satisfactory con-
cept of the HTR 500. We hope that we will soon be in a position to
start the second (detail) planning phase of this reactor. The planning
contract has already been finally negotiated and signed provisionally.
What we now need is a unanimous decision of a large user group. This
provided, we hope that the anticipated reorganisation of the user group
during the planning phase can be brought about and that the HTR 500 will
finally be erected.
DESCRIPTION OF CURRENT GCR PLANT DESIGNS

(Session C)
RESTORATION OF A CEGB MAGNOX REACTOR TO FULL POWER

R.J. PAYNTER, B.J. ROBERTS, P.T. SAWBRIDGE
Berkeley Nuclear Laboratories, Central Electricity Generating Board, Berkeley, United Kingdom

Abstract

The Hinkley Point 'A' Magnox Reactor was commissioned in 1965 and recently completed its 21st calendar year of operation. The paper contains a description of the operating experience of this plant together with the main technical problems and their solutions.

1. INTRODUCTION

The CEGB has operated Magnox reactors commercially since the early 1960s. These are natural uranium fuelled reactors cooled by pressurised CO₂. This paper examines the operating experience for one of these plants over an extended period. The station in question is the Hinkley Point 'A' plant which was commissioned in 1965 and has recently completed its 21st calendar year of operation with a mean availability of 73%.

A number of technical problems have arisen over this extended period of operation which have led to a reduction in the potential electrical generation capacity of the station. In the present paper we show how these problems have been overcome. The strategy has been to carefully monitor the plant and analyse its performance, supporting this with an R&D capability which enables a fundamental understanding of the processes involved to be obtained. This has led to the development of whole plant models which offer substantial benefits to the operators in optimising the performance of the plant.

2. STATION DESIGN

The Hinkley 'A' station consists of two magnox reactors housed in spherical steel pressure vessels. Each reactor has six dual pressure boilers, each of which has its own gas circulator, which draws the coolant gas down through the boiler (Figures 1 and 2). The steam supplies six main turbo-generators, each with a maximum continuous rating of 93.5 MWe, plus three variable speed turbo-generators of 33MWe which provide electrical power for the gas circulators. The net electrical generation of the station was designed to be 500MWe.
3. STATION OPERATING HISTORY

When the station was commissioned a net electrical generation of 500 MWe was achieved consistently with a maximum reactor gas outlet temperature of 373°C, a value 5°C below design level. After four year generation it was recognised that bolts securing the steel core support structure were oxidising at unacceptably high rate. Lifetime considerations necessitated a limitation on the maximum reactor gas outlet temperature of 360°C.

At about this time failure of one of the main turbines occurred. It was recognised that this was a generic fault, which required modifications to be made to all six of the main turbines over the following four years. This constrained the station to a reduced power output over this period with the boilers operating under 'off-design' conditions.
FIGURE 2  Sectional View of a Hinkley 'A' Boiler
When all the turbine modifications had been made the reactors were returned to full power, albeit with the restriction on gas outlet temperatures dictated by oxidation. It gradually became apparent that there had been a gradual deterioration in electrical generation capability of 30 MW to 40 MW, when one compared reactor performance pre and post the period of low power operation. It was at first assumed that this was part of a station ageing process. However by the later 1970s it was recognised that the decline in performance was greater than would be expected on these grounds. Plant monitoring showed that an increase in boiler gas outlet temperature of between 9 - 10°C was observed, for operation at a constant circulator speed. This suggested that the overall boiler heat transfer coefficient had been reduced.

Detailed analysis of station performance during steady state operation, confirmed that this was consistent with the net station generation capability had dropped by approximately 40 MW (Figure 3). In order to do this it was necessary to normalise the generation in 1965 and 1975 to a fixed set of boundary conditions via a steady state whole plant model.

<table>
<thead>
<tr>
<th>PARAMETER</th>
<th>VALUE IN 1965</th>
<th>NORMALISED</th>
<th>VALUE IN 1975</th>
<th>NORMALISED</th>
</tr>
</thead>
<tbody>
<tr>
<td>Boiler gas inlet temperature (°C)</td>
<td>371</td>
<td>365</td>
<td>360</td>
<td>365</td>
</tr>
<tr>
<td>Condenser pressure (Pa)</td>
<td>3000</td>
<td>3500</td>
<td>3300</td>
<td>3500</td>
</tr>
<tr>
<td>Circulator speed (rpm)</td>
<td>2840</td>
<td>2700</td>
<td>2650</td>
<td>2700</td>
</tr>
<tr>
<td>Boiler gas outlet temperature (°C)</td>
<td>181</td>
<td>175</td>
<td>182</td>
<td>183</td>
</tr>
<tr>
<td>Net electrical generation (MWe)</td>
<td>518</td>
<td>490</td>
<td>432</td>
<td>450</td>
</tr>
</tbody>
</table>

FIGURE 3 Normalised Operating Parameter Comparison 1965 and 1975

4. EXAMINATION OF BOILER HEAT TRANSFER SURFACES

This resulted in detailed examination of the boiler heat transfer surfaces. Whilst on the gas side a thin black uniform deposit was observed, examination of the waterside revealed the presence of substantial magnetite deposition. This was concentrated in unheated sections of the boiler tube, i.e. outside the boiler ducts. However after acid cleaning only a small improvement in station performance was observed.

Microscopic examination was thus carried out on gas side heat transfer surfaces, on studs removed from the finned boiler tubes. A special tool was built to do this without disturbing the deposit. A carbonaceous deposit was observed to cover most of the tube surfaces to a depth of between 100 μm and 200 μm.
Deposit thickness was measured using a scanning electron microscope.

**FIGURE 4 SEM Examination of Deposit Structure**

Figure 4 shows the deposit on a stud examined by SEM. Compositional analysis showed it to be mostly carbon (by volume), but by mass, approximately 50% consisted of various iron oxides. The deposits also formed in plumes which grew vertically out of the surface (Figure 5). The volume of the plumes was approximately 10% of the deposit envelope and the plumes themselves has porosities of approximately 90%. The overall density of material within the deposit envelope was calculated to be approximately $25 \text{ Kg m}^{-3}$.

No evidence was found within the deposit or substrate for catalytic growth. What evidence there is, suggests that the deposit forms homogenously within the core due to polymerisation of CO under irradiation. It also appears that deposit formation requires low hydrogen levels in the gas. The Hinkley A gas circulators had air bearings and thus the circuit had lower hydrogen levels than other magnox reactors, which had oil lubricated circulator bearings.

5. **THEORETICAL CONSIDERATIONS OF DEPOSIT HEAT TRANSFER**

A theoretical assessment of the effects of coolant flow through the deposit suggested:

(i) The forces on the deposit were unlikely to affect the plume orientation or morphology
Higher magnification using transmission electron microscopy showed that the columns were porous and composed of microspheres.

Overall deposit envelope porosity shown to be around 99%.

FIGURE 5 Transmission Electron Microscopy of Deposit Morphology

(ii) The gas flow rates would be sufficiently low through the deposit to be essentially laminar. It was therefore suggested that the primary effect of the deposit would be to extend the fluid boundary layer adjacent to the surface (Figure 6). Paynter and Roberts [1] developed a

FIGURE 6 Laminar Flow at the Tube Surface for Clean and Deposited Tubes
simple model on this hypothesis. Gas flow in a boiler tube during normal operation is essentially turbulent, with a thin boundary layer of laminar flow. In a turbulent magnox reactor during normal operation this layer is approximately 200 μm thick. Paynter and Roberts' calculations [1] based on the assumption that the deposit essentially extended the laminar layer concluded that a 40% reduction in film coefficient of heat transfer. This implied that the effective thermal conductivity of the deposit was 0.05 W/m°K.

If these low thermal conductivities could be substantiated, it was clear that most of the station loss of generation capacity resulted from these gas side boiler deposits.

6. EXPERIMENTAL VERIFICATION OF THE LOW THERMAL CONDUCTIVITY OF THE DEPOSIT

Direct thermal conductivity measurements were ruled out because of doubts concerning the relevance of measurements made on deposit in stagnant rather than flowing gas. Consequently a comparative measurement on clean and deposited tube was made under representative boiler operating conditions (Figure 7).

The rig used an electrically heated tube 0.3m in length, which had been removed from the reactor. A pressurised rig meant that the correct range of Reynolds numbers could be obtained without excessive gas velocities, which could have damaged or removed the deposit.

A pressurised rig was constructed to measure the deposit heat transfer resistance.

FIGURE 7 Experimental Measurement of Deposit Thermal Conductivity
The results are shown in terms of the dimensionless heat transfer co-efficient (Nusselt number) \( \text{Nu}_f \) and the gas Reynolds number and Prandtl numbers (\( R \text{ Pr} \))

\[
\text{Nu}_f = 0.186 \times R^{0.66} \frac{1}{\text{Pr}^{1/3}}
\]

In Figure 7 regression lines are shown for the clean and deposited tubes. The regression line for the deposited tube suggested a boundary layer extended by 85 \( \mu \text{m} \) (in terms of Paynter and Roberts' theory) compared to an actual measured deposit thickness of 110 \( \mu \text{m} \). This was felt to be a reasonable agreement, since the surface of the deposit was rougher than the clean tube and some conduction must occur through the deposit plumes. At least it gave sufficient confidence to deploy the necessary resources on deposit removal techniques.

7. DEPOSIT REMOVAL

It was obviously economic to devise a cleaning method which did not require removal of the boilers from service. Laboratory experiments on air, CO and oxygen dosed coolants suggested that oxygen dosing was the most effective way of removing the deposit by oxidation. Sufficient oxygen has to be introduced to oxidise any free CO in the coolant before deposit oxidation occurred. The deposit oxidation was very temperature sensitive (Figure 8) thus removal from the upper boiler superheater banks was expected to occur in a few days whereas the economiser sections, operating at 130°C, were expected to take many months (Figure 9).

![Figure 8](image)

**FIGURE 8** Effect of Temperature on Deposit Burn-off in 900 vpm \( \text{O}_2 \) and \( \text{CO}_2 \)

The oxidation essentially removed the graphitic part of the deposit and freed the iron oxide particles, which were gradually removed by filters in the coolant treatment by-pass circuit. However, a small increase in the activity around the circuit was measured. This indicated that there had been significant deposition in the core, which when removed by oxidation of the carbon released activated iron oxide particles.
A comprehensive safety assessment of the effects of the oxygen injection on fuel performance, graphite moderator lifetime and the performance other key reactor components, showed that the proposed method was feasible and safe. Oxygen injection commenced in reactor between January and May 1985. Plant monitoring showed that within days deposit removal commended in the higher temperature regions of the boiler. This was evidenced by a rapid change in superheater steam outlet temperatures, followed by a progressive increase in HP steam water flow (Figure 10). However the LP water/steam flow initially decreased and overall station electrical generation was little improved. It is thought that deposit removed from upper banks initially deposited on the lower banks and also the gas inlet temperature to the lower LP banks reduced, thus limiting the LP water/steam flow rate. Only when substantial cleaning of the lower banks was effected did total water/steam flow rates, and hence electrical generation increase.

Table 1 shows how boiler parameters have improved following the oxygen injection. Overall the electrical generation capacity of the station has been increased by approximately 20 MWe, with a benefit to the CEGB of approximately £2 x 10^6 per annum in replacement fossil fuel costs. Recent relaxation in the gas outlet temperature restriction due to oxidation limits to 370°C, together with the improved boiler performance, has contributed to making Hinkley Point 'A' the United Kingdom's leading Magnox station, and fifth position in the world in terms of culminative electrical generation. Thus in 1984/85 the station generated more electricity than in all but two of its previous years of operation. There is a possibility that in 1986/87 it may possibly challenge its annual generation record in the 21st year of operation.
TIME DEPENDENCE OF CHANGE IN STATION PERFORMANCE
DURING OXYGEN INJECTION
INTO GAS CIRCUITS OF REACTOR 1

FIGURE 10 Progressive Changes in Boiler HP and LP
Water/Steam Flow and Station Electrical Generation During Oxygen Injection

TABLE 1 CHANGE IN BOILER AND STATION PERFORMANCE
DUE TO OXYGEN INJECTION INTO THE
REACTOR GAS CIRCUITS

<table>
<thead>
<tr>
<th>PARAMETER</th>
<th>VALUE PRIOR TO OXYGEN INJECTION</th>
<th>VALUE POST OXYGEN INJECTION</th>
</tr>
</thead>
<tbody>
<tr>
<td>Boiler Gas Inlet Temperature (°C)</td>
<td>370.0</td>
<td>370.0</td>
</tr>
<tr>
<td>Boiler Gas Outlet Temperature (°C)</td>
<td>184.1</td>
<td>178.7</td>
</tr>
<tr>
<td>Boiler Gas Flow (kg/s)</td>
<td>787</td>
<td>800</td>
</tr>
<tr>
<td>Feed Water Temperature (°C)</td>
<td>68.5</td>
<td>70.0</td>
</tr>
<tr>
<td>HP Steam Temperature (°C)</td>
<td>359.8</td>
<td>361.9</td>
</tr>
<tr>
<td>LP Steam Temperature (°C)</td>
<td>344.8</td>
<td>348.3</td>
</tr>
<tr>
<td>Boiler HP Water/Steam Flow (kg/s)</td>
<td>32.7</td>
<td>36.4</td>
</tr>
<tr>
<td>Boiler LP Water/Steam Flow (kg/s)</td>
<td>21.8</td>
<td>21.0</td>
</tr>
<tr>
<td>Ratio of HP and LP Flows (-)</td>
<td>1.50</td>
<td>1.73</td>
</tr>
<tr>
<td>LP Drum Pressure (N/m²)</td>
<td>17.3x10⁵</td>
<td>13.5x10⁵</td>
</tr>
<tr>
<td>HP Drum Pressure (N/m²)</td>
<td>43.1x10⁵</td>
<td>45.4x10⁵</td>
</tr>
<tr>
<td>Station Net Electrical Generation at a Condenser (MWc)</td>
<td>458</td>
<td>479</td>
</tr>
</tbody>
</table>

The values quoted are averages for the station
ACKNOWLEDGEMENTS

This paper is published by the permission of the Central Electricity Generating Board.

The authors wish to express thanks to all their colleagues at Hinkley Point Power Station and in the South West Region Scientific Services Department who took part in this project.
THE MODULAR HIGH-TEMPERATURE GAS-COOLED REACTOR (MHTGR)

A.J. NEYLAN
GA Technologies, Inc.,
San Diego, California,
United States of America

Abstract

The MHTGR is an advanced reactor concept being developed in the USA under a cooperative program involving the US Government, the nuclear industry and the utilities. The design utilizes basic HTGR features of ceramic fuel, helium coolant and a graphite moderator. However the specific size and configuration is selected to utilize the inherently safe characteristics associated with these standard features coupled with passive safety systems to provide a significantly higher margin of safety and investment protection than current generation reactors. Evacuation or sheltering of the public is not required. The major components of the nuclear steam supply, with special emphasis on the core, are described. Safety assessments of the concept are discussed.

The Modular High-Temperature Gas-Cooled Reactor (MHTGR) is being developed in the United States under a cooperative program involving the U.S. Government's Department of Energy (DOE); the nuclear reactor industry led by GA Technologies Inc. (GA), the industrial pioneer of the HTGR concept; and the utility/user industry represented by Gas-Cooled Reactor Associates (GCRA). The design is an advanced version of the basic High Temperature Gas-Cooled Reactor (HTGR) concept. The overall goal is to provide safe economical power with a standardized, pre-licensed factory fabricated reactor.

This modular design uses the following standard HTGR features:

- refractory coated particle fuel capable of retaining fission products at very high temperatures,
graphite moderator which remains stable to very high temperatures and has a high heat capacity, and

helium coolant which is inert, non-corrosive, and remains as a gas under all operating conditions.

As in previous concepts these standard HTGR features provide the capability to operate at temperature for high efficiency electricity generation and offer the potential to access the cogeneration and high temperature process heat markets. The basic characteristics also provide inherent advantages associated with safety, investment protection, and environmental compatibility which are greatly enhanced by the special design features of the MHTGR reactors.

DESIGN DESCRIPTION:

In the MHTGR, the low thermal output, the annular geometry of the core, the use of a totally passive ultimate heat removal system and the installation of the reactor system in a below grade silo are design features incorporated in the design to provide safety which is not dependent on active safety systems, on operator actions or on the evacuation or sheltering of the public.

The reference MHTGR plant design consists of four reactor modules, each rated at 350 MW(t), coupled to two steam turbine generators yielding a net power output of about 550 MW(e). An added advantage of the modular concept is the flexibility of providing sequential deployment of individual reactor modules to match the end user's load growth requirements or financing constraints.
The four-module plant is divided into two major areas, the Nuclear Island comprising the reactor enclosures, reactor modules, and the safety related systems, and the balance of the plant housing the electric power generating systems. Each reactor module is housed in a vertical cylindrical concrete enclosure which is fully embedded in the earth. Each reactor enclosure serves as an independent confinement structure having a vented and filtered exhaust system. The Nuclear Island portion of the plant also includes auxiliary structures which house common systems for fuel handling, helium processing, and other essential services.

The modular reactor components are contained within three steel vessels: a reactor vessel, a steam generator/circulator vessel, and a connecting concentric crossduct vessel. The vessels use existing LWR technology and have weight and dimensions within previously demonstrated fabrication and transportation limits.

The reactor vessel, 6.9m (22.5 ft.) in diameter and 22m (72 ft.) in length, contains the core, reflector, and associated supports. A shutdown heat exchanger and a shutdown cooling circulator are located at the bottom of the reactor vessel. Eighteen top-mounted standpipes contain the control rod drive mechanisms and hoppers containing boron carbide pellets for reserve shutdown. Six of these standpipes are also the access ports for refueling and in-reactor inspection.

A helical coil steam generator and the electric motor-driven main helium circulator are contained in the steam generator vessel which is 4.3m (14 ft.) in diameter and 26m (85 ft.) in length. Feedwater enters the steam generator through a bottom-mounted
header and superheated steam exits through a side-mounted nozzle. The steam generator is similar to, but simpler than, the Fort St. Vrain (FSV) design. The main helium circulator is located above the steam generator.

The reference design circulator utilizes magnetic bearings and a submerged motor to eliminate the water ingress problems that have plagued Fort St. Vrain (FSV). Although this is an important and preferred design selection to meet availability goals, it would also be possible to meet the design requirements with a greatly simplified third generation water bearing system that has already been full scale tested at GA. The single stage axial circulator, driven by a variable speed 3.6 MW electric motor, delivers 158.0 Kg/s (1,254,000 lb/hr) of helium at 6.378 MPa (925 psia.)

In the primary flow circuit, the helium coolant, at approximately 260°C (500°F), flows downward through the core, where it is heated by the nuclear reactions. The hot helium, at approximately 688°C (1270°F), then flows through the inner cross-duct and downward over the steam generator bundle where its heat is transferred to the water to make steam. The cooled gas then flows upward in an annulus between the steam generator bundle and the vessel, is recompressed by the circulator and driven into the annulus between the inner and outer crossduct. The cool gas entering the reactor vessel flows up through channels in the lateral core restraint structure to the top of the core to complete the circuit. All surfaces of the steel pressure vessels are in contact with helium at the lower temperature.

In the secondary flow circuit, feedwater is transformed to superheated steam in the once-through, uphill boiling steam
generator. The steam generator delivers 137.3 Kg/s (1,089,700 lbs/hr.) of superheated steam at 17.340 MPa (2515 psia) and 540°C (1005°F) to a two-turbine generator unit. Exhaust steam from the turbine is condensed and returned to the steam generator through a standard condensate/feedwater system. Control of operations for the complete plant is performed from a single control room in the control building located adjacent to one end of the reactor services building.

A completely independent shutdown cooling loop, comprising a heat exchanger and circulator, is provided at the bottom of the reactor vessel for maintenance purposes. Provision of this capability is directly in response to user requirements to minimize downtime and meet availability goals. In the shutdown cooling mode hot helium enters the shutdown helium to water heat exchanger through a flow-activated shut-off valve. After transferring its heat, the cool helium is compressed by the shutdown circulator. The compressed helium is then discharged into a plenum at the bottom of the reactor vessel and returns to the top of the core through the annular flow passage between the core and the reactor vessel.

The key to the MHTGR performance and capability is the refractory coated fuel. As in the FSV fuel, fissile and fertile fuel particles are blended into 1.27cms (1/2") dia x 5.08cms (2") long fuel rods which are in turn loaded into prismatic fuel elements approximately 78cms (31") in height by 35.6cms (14") across flats. Vertical coolant holes are provided in the fuel element blocks. Major differences from the FSV fuel are the adoption of Low Enriched Uranium (≤19.9%) to meet National policy guidance on proliferation resistance and improvements in the fuel manufactur-
ing process to produce a fifty fold improvement in fuel quality --- a remarkable objective considering the excellent operating experience and performance of fuel in FSV.

The fuel elements are stacked in columns to make up an annular-shaped core having a radial thickness of about 0.9m (3 ft.), an average outer diameter of 3.5m (11.5 ft.) and a height of 7.6m (25 ft.) Unfueled graphite blocks surround the active core to form replaceable inner and outer radial and upper and lower axial reflectors. Permanent reflector blocks are located at the outer periphery of the replaceable graphite blocks.

The total assembly is restrained in a steel core barrel within the reactor pressure vessel. Lessons learned at FSV on preventing core fluctuations have been factored into the design. Although the core is taller both ends of each column are pinned (the design fix at FSV) and the core pressure drop (the driving force) is considerably lower due to the elimination of orifice valves and the six year core region fuel loading used on FSV.

Refueling of a module is accomplished with the reactor shutdown and depressurized. After removal of the required control rods and drive mechanisms by an auxiliary service machine, the refueling machine is located over the desired refueling port by the crane in the reactor services building and fuel elements are removed from the core and transported in shielded casks to the fuel storage area in the fuel building. One half of the fuel is replaced each 20 months resulting in a 3.3 year fuel cycle.

Two independent and diverse reactivity control systems are provided. Reactor power is controlled by 6 inner and 12 outer
articulated control rods that travel in vertical channels located in the inner and outer graphite reflector regions. Twelve reserve shutdown channels are provided which, when activated, permits small boron carbide spheres to enter channels located in the innermost row of fuel columns.

The annular core geometry and power density of 5.91 w/cc were selected to limit peak temperatures and hence fuel failure during loss of forced circulation events such that the user requirement of not exceeding US Environmental Protection Agency’s Protective Action Guidelines (PAG) dose limits at the site boundary would be achieved.

Decay heat removal during shutdown, inspection or maintenance periods can be accomplished using the main heat transport loop with the steam generated in the secondary circuit bypassing the turbine and condensing in the condenser. The decay heat can also be removed by the shutdown cooling system located at the bottom of the reactor vessel.

In addition to the two independent active systems used to remove heat from the core i.e. the main loop (steam generator/circulator) and the shutdown cooling loop (heat exchanger/circulator) a third totally passive system is located in the reactor cavity adjacent to the reactor vessel. This system, designated the Reactor Cavity Cooling System (RCCS), removes radiant heat from the reactor vessel under normal and accident conditions using naturally circulating ambient air in specially designed cooling panels. The system is totally passive; no valves, fans or mechanisms are utilized. It is the only system needed to remove heat from the core to meet regulatory safety requirements.
SAFETY ASSESSMENT:

The safety characteristics of the MHTGR are a result of the ability to effectively utilize the inert coolant; the capability of the refractory coated fuel to withstand high temperatures; and the high thermal inertia and high temperature stability inherent in the graphite core and support structures. These unique features have been exploited in the MHTGR to yield a reactor that does not depend on active engineered safety features or human actions for safety of the public or the investment.

Helium gas, the reactor coolant, is inert so there are no chemical reactions between the coolant and the fuel, graphite core structure, or reactor vessel which could lead to fires or explosions. Further, helium has no effect on the nuclear reaction.

The graphite fuel elements and reactor internals which make up the reactor core have a high heat capacity and maintain strength to temperatures beyond 2760°C (5000°F). As a result, temperature changes of the core occur slowly and without damage to the core structure even in the event of accidents where the capability to remove heat is impaired.

As stated previously the annular core geometry, core power density, and the total module power of the MHTGR have been chosen such that the decay heat generated within the core can be passively removed by means of conduction, radiation, and natural convection without the fuel reaching a temperature at which the refractory coating would fail during an accident.
Primary cooling systems and shutdown cooling systems are available to remove heat and the core temperatures do not exceed normal levels. Even if both active cooling systems are unavailable, decay heat is dissipated by conduction and radiation to the reactor cavity cooling system (RCCS) in the reactor enclosure. The heat transfer is sufficient to avoid damage to the fuel or reactor components. The maximum fuel temperature is limited to about 1600°C, well below the failure temperature.

The high temperature stability and slow heatup characteristics of the graphite reactor core and fuel not only retains fission products within the coated particles but provide a user friendly and forgiving reactor design that allows many many hours for operator corrective action.

The inherent characteristics of the MHTGR also assure the safety of the system against other types of accidents.

- Anticipated Transients Without Scram (ATWS) are mitigated by the strong negative temperature coefficient and zero coolant void coefficient of the reactor core. These inherent characteristics cause the reactor to automatically shutdown in the event reactor temperature rises above normal as a result of equipment failure or operator error.

- The reactivity control systems provide considerable margin over the most adverse reactivity insertion accident including unlimited water ingress.
Chemical reactions between steam and graphite can only occur as a result of accidents or equipment failures allowing the steam or water to come into contact with the graphite. At normal operating temperatures the reaction rate is insignificant. The reaction rate increases with temperature but, in any event, is highly endothermic (heat absorbing) and therefore self terminating.

Air/oxygen reactions with graphite are exothermic (heat releasing) but can only be sustained if multiple breaches of the reactor vessel occur and if the graphite is heated to temperatures much higher than normal. The silo installation limits the air available to support combustion in the extent of any reaction would be minor. Even were the graphite to be burned away the fission product activity is retained by the refractory particle coatings.

No public evacuation or sheltering is required even for severe low probability accidents because the consequences are mitigated by the inherent passive features of the MHTGR.

CONCLUSION:

The Modular High Temperature Gas-Cooled Reactor (MHTGR) is a second generation nuclear power system which can satisfy the concerns of the public, the government, the utilities and the investor community about nuclear safety and investment protection. Based on technology developed and demonstrated in the U.S. and Germany, the unique system makes use of refractory
coated nuclear fuel, helium gas as an inert coolant and graphite as a stable core structural material. The safety and protection of the plant investment is provided by inherent and passive features and is not dependent upon operator actions or the activation of engineered systems. The high performance MHTGR provides flexibility in power output and siting, competitive energy costs, and can serve diverse energy needs both domestically and internationally.

The work described in this paper reflects the combined efforts of all the US program participants under contract to the Department of Energy. I wish to express my appreciation to DOE for permission to publish this paper.

ACKNOWLEDGEMENTS

The author would like to thank Mr. A. C. Millunzi, Acting Director, Division of HTGR, U.S. Department of Energy, for approval to publish this paper. Thanks are also expressed to the management of GA Technologies Inc. for permission to write and present this work, which was supported by the Department of Energy, San Francisco Operations Office Contract DE-AC03-84SF11963.
STATUS OF GAS-COOLED REACTOR DEVELOPMENT IN THE USSR

V.N. GREBENNİK
I.V. Kurchatov Institute of Atomic Energy, Moscow, Union of Soviet Socialist Republics

Abstract

The paper contains a description of HTGR concepts as being developed in the USSR for electricity generation and process steam and heat production. The HTGR-M, a module type reactor, and the VG-400 are presented together with their main design features and their applications.

In the USSR the nuclear energy is currently developed predominantly in the direction of electricity generation where it competes successfully with fossil-fuel power plants.

Efforts are also being made for utilization of nuclear energy sources for low-temperature heat production on the basis of nuclear district heating plants (NDHP) and nuclear co-generation plants (NCGP). For these purposes use of commercial light water reactors.

At the same time more actual is becoming the problem of extending fields of nuclear energy application for industrial and district heating and industrial processes consuming great amounts of oil and gas. In the conditions of large-scale development of the nuclear energy ensuring of higher nuclear and radiation safety and further improvement of operation reliability of the nuclear energy sources is a task of great importance.

To a significant degree these requirements may be met on account of realization of a new nuclear energy direction introduction of the high-temperature gas-cooled reactors (HTGR) whose advantageous features are high nuclear and radiation safety, effective utilization of the nuclear fuel and capability to generate high-temperature heat.
These features are accounted for by use of the helium coolant which is chemically inert and has no phase transitions, and use of graphite as the structural material for the core.

In industry HTGR can be used for co-generation of steam, high-temperature heat and electricity. As the steam conversion and other high-temperature processes are being mastered on the basis of power from HTGR, a possibility will open for wide and effective application of the nuclear energy for substitution of oil and gas in various power-consuming industry fields.

The analysis of the scale and structure of power consumption in the industry shows that there both powerful plants (about 1000 MW(th) and higher) for power supply of large industrial centers and units of relatively low power capacity can find application /1/.

In the USSR R and D works are being carried out on some pilot scale plants of various capacities as well as on application of HTGRs for process steam and heat supply.

The HTGR-M reactor of modular type with a unit power of 200-300 MW(th) is considered as a small pilot plant.

Taking into account the experience gained earlier, development of the HTGR-M reactor is based on the following main statements:

- use of a pebble-bed core with spherical fuel elements and realization of various principles of their movement in the reactor;
- compensation for burnup excess reactivity by spherical absorbers;
- use of the control and safety rods only in the side reflector;
- ensuring of high safety on account of the negative temperature coefficient of reactivity, no phase transitions and chemically inert coolant;
use of the current technology of manufacturing metal pressure vessels and application of materials available in reactor building.

The HTBR-M reactor unit is designed for verification of basic technical solutions and technology of works on HTGR as well as for experimental tests of high-temperature heat, steam and electricity production processes.

The HTGR-M core is installed inside a reinforced steel vessel about 5 m in diameter.

The primary circuit of the reactor can be designed in two versions: with the steam generator or with the steam reformer. The HTGR-M with the steam generator can be realized at the first stage since it proposes a lower helium temperature (750°C) and a simpler heat exchange equipment. The second version proposes testing of steam conversion of methane, with the reactor unit operated at a temperature as high as 950°C.

An important factor for realization of the modular-type plant is unification and commercialization of standard equipment. Large industrial plants (about 1000 MW(th) can be created by integration of several one-type modular low-power units. In this case the problems of redundancy and reliable power supply of continuous technological productions would be favorably solved.

The pilot industrial-plant VG-400 is designed for co-generation of high-temperature heat (up to 950°C), steam and electricity, which will enable it to be used in various power-consuming industries such as chemical, petrochemical, metallurgical, etc.).

In the course of VG-400 realization the design and technical solutions of the main equipment and systems of an industrial plant with the HTGR reactor will be confirmed and tested in the scale of industrial experiments.

In accordance with the objective of the plant the NPH schemes and parameters were chosen, which enables attainment of characteristics suitable for transition to an extensive and effective industri-
rial utilization of the nuclear energy for process steam and heat production in industry.

Since the plant is a pilot-scale by its nature, NPH is designed so that the rated parameters are reached step-by-step with continuous rise in the reactor temperature and power levels and simultaneous tests of the steam generating and heat power equipment, core and high-temperature energotechnological station.

Two schemes of the reactor unit were considered: conventional one with the heat exchanger, steam generator and gas blower connected in series, and proposed one with bypass valves in the heat exchange loops (bypass scheme) /2/.

The reactor has the following main parameters:

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor power</td>
<td>1060 MW(th)</td>
</tr>
<tr>
<td>Helium pressure in the primary circuit</td>
<td>4.9 MPa</td>
</tr>
<tr>
<td>Helium temperature at the core outlet</td>
<td>950°C</td>
</tr>
<tr>
<td>Helium temperature at the core inlet</td>
<td>350°C</td>
</tr>
</tbody>
</table>

For the VG-400 reactor the integrated arrangement of the primary circuit equipment installed inside the prestressed concrete many-chamber reactor vessel has been adopted /3/.

The reactor unit has the intermediate helium circuit which prevents penetration of radioactive fission products into the process circuit and contamination of the primary circuit by products of the technological process. The intermediate circuit increases the unit safety and makes it flexible to application of various technologies. One of the most important and strained elements of the intermediate circuit is the high-temperature intermediate heat exchanger (IHE) which serves for transfer of high-temperature heat from the reactor to the process circuit. The VG-400 reactor is equipped with a cassette-type heat exchanger which is installed inside the cylindrical cavity of the prestressed concrete reactor vessel. The heat exchanger is designed for normal operation at 950-750°C.

In the core the once-through-then-out (OTTO) principle of spherical fuel element movement is realized, which will permit the fuel
element temperature to be reduced and the refuelling to be performed with the reactor in operation. The core design with a system of discharge holes and radially homogeneous loading of fuel elements has been adopted as the basic version.

Practically all components of the main equipment of the plant are prototype, which enables, basing on the technical solution on pilot plant VG-400, the advance to the HTGR-equipped plants of various capacities and objectives.

A full-scale development of the nuclear energy will require to solve the problem of reproduction of the nuclear fuel resources. The analysis shows that this problem could be solved by means of the fast breeders which have a high rate of excess plutonium buildup and short doubling time. In the USSR investigations on the helium breeder, which is considered an alternative of the sodium-cooled breeder, are in progress.

Application of HTGR as an effective higher-safety nuclear source of the new generation will be an important factor in power supply of the national economy and in saving scarce fossil fuels.

REFERENCES

THE NUCLEAR ELECTRICITY AND HEAT GENERATION USING THE VG-400 REACTOR

I.V. Kurchatov Institute of Atomic Energy, Moscow, Union of Soviet Socialist Republics

Abstract

The paper contains a description of HTGR concepts as being developed in the USSR for electricity generation and process steam and heat production. The HTGR-M, a module type reactor, and the VG-400 are presented together with their main design features and their applications.

At the present stage of the nuclear energy (NE) development more actual is becoming the problem of expanding fields of reactor energy application for industrial and district heating and for industrial processes consuming large amounts of oil and natural gas. Against this background particularly important is further improvement of the nuclear and radiation safety of the nuclear energy sources.

These problems could be solved to a significant degree by introduction of the high-temperature helium-cooled reactors which are currently under development in the USSR. One of the nuclear designs which represents this new nuclear energy direction is the VG-400 high-temperature gas-cooled reactor.

The pilot energotechnological plant with the VG-400 reactor is intended for co-generation of high-temperature heat, process steam and electricity for chemical, petrochemical and other industries /1/.

Schematic diagram of the plant is shown in Fig.1. It consists of the reactor unit (RU), steam turbine unit (STU) and chemical technological production (CTP) where heat is supplied through the high-temperature intermediate heat exchangers (IHES) /2/.
FIG. 1. Schematic diagram of power technological plant using VG-400 reactor.

1. VG-400 reactor
2. High-temperature intermediate heat exchanger (IHB)
3. Steam generator
4. Gas-blower of the primary circuit
5. Bypass valves
6. Conversion apparatus
7. Steam-gas mixture preheater
8. Steam generator
9. Gas blower of the intermediate circuit
10. Turbogenerator
11. Separator-steam-preheater
12. Condensate pump
13. Low-pressure preheater
14. Deaerator
15. Feed pump
16. High-pressure preheater
Creation of the VG-400-equipped plant will permit experience of designing, manufacturing and operating the helium-cooled reactors designed for various energy and technological purposes to be gained.

The plant's main characteristics are listed in Table I.

<table>
<thead>
<tr>
<th>No.</th>
<th>Parameter</th>
<th>Unit</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Reactor thermal power</td>
<td>MW</td>
<td>1060</td>
</tr>
<tr>
<td>2</td>
<td>Thermal power supplied to the technological production</td>
<td>MW</td>
<td>440</td>
</tr>
<tr>
<td>3</td>
<td>Thermal power supplied for steam generation</td>
<td>MW</td>
<td>635</td>
</tr>
<tr>
<td>4</td>
<td>Core</td>
<td></td>
<td>pebble-bed</td>
</tr>
<tr>
<td>5</td>
<td>Fuel element</td>
<td></td>
<td>ball-type</td>
</tr>
<tr>
<td>6</td>
<td>Fuel element size</td>
<td>mm</td>
<td>60</td>
</tr>
<tr>
<td>7</td>
<td>Core power density</td>
<td>MW/m³</td>
<td>6.9</td>
</tr>
<tr>
<td>8</td>
<td>Core dimensions:</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>diameter</td>
<td>m</td>
<td>6.4</td>
</tr>
<tr>
<td></td>
<td>height</td>
<td>m</td>
<td>4.5</td>
</tr>
<tr>
<td>9</td>
<td>Reactor coolant</td>
<td></td>
<td>helium</td>
</tr>
<tr>
<td>10</td>
<td>Pressure</td>
<td>MPa</td>
<td>5.0</td>
</tr>
<tr>
<td>11</td>
<td>Temperature:</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>at core outlet</td>
<td>K</td>
<td>623</td>
</tr>
<tr>
<td></td>
<td>at core inlet</td>
<td>K</td>
<td>1223</td>
</tr>
<tr>
<td>12</td>
<td>Number of heat exchange loops</td>
<td>pcs</td>
<td>4</td>
</tr>
<tr>
<td>13</td>
<td>Power for coolant pumping</td>
<td>MW</td>
<td>15</td>
</tr>
<tr>
<td>14</td>
<td>Intermediate circuit coolant</td>
<td></td>
<td>helium</td>
</tr>
<tr>
<td>15</td>
<td>Intermediate circuit parameters</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>pressure</td>
<td>MPa</td>
<td>5.5</td>
</tr>
<tr>
<td></td>
<td>temperature</td>
<td>K</td>
<td>623</td>
</tr>
<tr>
<td></td>
<td>at INE inlet</td>
<td>K</td>
<td>1173</td>
</tr>
<tr>
<td>16</td>
<td>Electrical power</td>
<td>MW</td>
<td>265</td>
</tr>
<tr>
<td>17</td>
<td>Steam pressure</td>
<td>MPa</td>
<td>17.0</td>
</tr>
<tr>
<td>18</td>
<td>Steam temperature</td>
<td>K</td>
<td>813.</td>
</tr>
</tbody>
</table>
The pebble-bed core with spherical fuel elements and graphite reflectors is installed in the center of the prestressed concrete reactor vessel and the heat removal equipment (IHE and steam generator (SG)) is located in the side cavities of the reactor vessel /3/. The bypass valves are in the lower part of the IHE cavities and the main gas blowers - in the lower part of the SG cavities.

The reactor unit layout is presented in Fig. 2.

**FIG. 2. VG-400 reactor unit layout.**

1. Reactor vessel
2. Core
3. Gas blower
4. Steam generator
5. High-temperature intermediate heat exchanger
6. Ionization chamber
7. Control rod (side)
8. Control rod (central)
9. Load tubes
10. Bypass valve
11. Unload tube
The main building is a complex of the main and auxiliary services of the nuclear power plant, including the reactor building, auxiliary building, steam turbine hall, spent fuel storage, etc.

In the reactor building are also installed the loading-unloading complex, auxiliary systems of the primary and intermediate circuits, a part of the CTP heat-exchange equipment. The auxiliary building is adjacent to the reactor building and serves for accommodation of disassembled IHE, SG, GCG and other equipment.

The layout of the main building of the plant is shown in Figs. 3 and 4/4/.

The schematic diagram and layout of the reactor unit have been developed taking into account different stages in designing, manufacturing and testing of the plant and CTP elements. The objective
of the first stage is the reactor unit tests at the medium temperature level and with minimum number of the power complex elements. For this purpose it is envisaged to operate the reactor at a helium outlet temperature of 1023K, with heat transferred to SG for steam production and electricity generation. At the second stage the core and the reactor elements will be tested at the designed temperatures when the core outlet temperature is increased up to 1223K and the SG outlet temperature is kept at 1023K by adding some cold coolant bypassed from the gas blower head line to the SG inlet /5/. Tests at the first and second stages are performed with the mastered steam turbine. At the third stage the intermediate circuit will be put into operation for tests of the CTP equipment and its interaction with the reactor. This three-stage approach permits some additional time to be saved for development of high-temperature materials, IHE design and CTP elements. The schedule of bringing the core to the temperature 1223K with heat removal to SG is shown in Fig. 5.
FIG. 5. Primary circuit parameters when controlling by means of by-pass valve.

1 $M_6g$ - flow rate of four gas blowers
2 $6g$ - gas blower speed
3 $P_{as}$ - temperature difference in the core
4 $T_i$ - temperature at the core outlet
5 $T_i'$ - temperature at the steam generator inlet
6 $a_3$ - flow rate through the core

When the valve is opened, the flow rate through the core reduces: $G_{core} = G_i - G_{bv}$ and the core outlet temperature rises

\[ T = \left( \frac{N_{core}}{C_R} \right) (G_i - G_{bv}) + T_2 \]

reaching 1223K at $G_{bv} = 0.33_{core}$.

The analysis of some operation regimes of the reactor unit with bypass showed that apart from realization of the stage-test program the bypass scheme reveals some positive functional possibilities in the main operation regimes as well.

The safety of the VG-400 plant is ensured by a reliable monitoring and control of the reactor under all operation conditions, precautions preventing failures of the reactor elements and equipment in design accidents and limitation of magnitudes of possible losses of leak-tightness in the primary circuit.
The safe operation of the plant is also ensured by the inherent features of the high-temperature reactor itself, which are: use of chemically inert helium having no phase transitions as the coolant and graphite as the core structural material, integrated arrangement of the equipment in the prestressed concrete reactor vessel and some others. One of the important conditions is creation of a reliable system for heat removal from the reactor. In accordance with the current specifications the emergency core cooling system (ECCS) must consist of several independent cooling channels and each such channel must ensure heat removal from the reactor even in the case of failure of any channel, independent of the initial accident event.

A schematic diagram of ECCS is shown in Fig. 6. The system must ensure heat removal in loss of outside power supply sources, loss of feed water, shutdown of the turbogenerator and heat consumers as well as in the case of the primary circuit depressurization. The coolant system has four loops for main reactor heat removal, which permits a part of the normal equipment to be used for ECCS, therefore the latter consists of independent channels connected with the main heat removal loops. The capability of each ECCS channel is sufficient for the residual heat to be removed in the above misoperations of the plant, except for the accident of depressurization in the primary circuit, when simultaneous operation of two channels is needed. It is the equipment of the heat removal loops and not the core that imposes limitations on the increase in the reactor coolant temperature, therefore each ECCS channel has two subsystems: that of immediate control and of durable cooldown.

The first subsystem serves for heat removal in the initial period of the accidental situation and duration of its operation (3-5 min) depends on the time of startup of the emergency diesel generator and durable cooldown subsystem. The second subsystem repre-
FIG. 6. Schematic diagram of VG-400 reactor emergency cooling system.

1. Reactor
2. High-temperature intermediate heat exchanger
3. Steam generator
4. Gas blower of the primary circuit
5. Condenser
6. Condensate pump

sents a closed circulation loop designed for removal of the reactor heat to a safe level that practically corresponds to the amount of losses into the environment.
REFERENCES


CURRENT STATUS OF RESEARCH AND DEVELOPMENT OF HTGR IN JAPAN

T. HAYASHI
Japan Atomic Energy Research Institute,
Tokai-mura, Ibaraki-ken,
Japan

Abstract

In Japan, the very high-temperature reactor (VHTR) for a nuclear process heat application has been developed by Japan Atomic Energy Research Institute (JAERI) since 1969.

Looking back upon the activities, there are considerably accumulated technologies enough to construct the experimental VHTR.

Recently, however, economic structure in Japan has been changed with the low growth rate of economy and the nation's energy demands have been relieved so rapidly that the needs for nuclear process heat application have been declined. This trend also has been accelerated by introducing firmly the energy-saving technologies in industry and with increasing of the electric power generation by light water reactors.

In this situation, the Special Committee of Japan Atomic Energy Commission (JAEC) has recognized newly the significance of the VHTR development from the viewpoint of long future and has recommended the construction of a high-temperature engineering test reactor (HTTR) to upgrade the current VHTR technologies.

JAERI has started new deployment of the HTTR as a research facility to prepare for the realization of nuclear process heat application in the 21st century. This paper deals with technical background and the current design concept of the HTTR.

1. Introduction

In Japan, since 1969 research and development for very high temperature reactors (VHTRs) has been carried out for about seventeen years.
In 1970s, Japan Atomic Energy Research Institute (JAERI) conducted the design study and development of the experimental VHTR which had the output power of 50MWe with the reactor coolant outlet temperature of 1000°C in order to demonstrate as a heat source of reducing gas production for iron and steelmaking industries. In parallel with these activities in JAERI, the development on utilization system of the nuclear heat was started in 1973 by Engineering Research Association of Nuclear Steelmaking (ERANS) as the big project in Ministry of International Trade and Industry (MITI). The results accomplished in this project were transferred to JAERI after the termination of the project in 1980, when the phase-I program was completed.

In 1980s, economic fluctuation in Japan has been induced with the low growth of economy, and the energy-saving technologies have been introduced thoroughly into industry. As the result of these situations, nation's energy demands have been relieved so rapidly that the needs for nuclear process heat application in industry have been declined.

In April, 1986 the Committee on Nuclear Process Heat Utilization of the Japan Atomic Industrial Forum (JAIF) issued the report on perspectives of the nuclear heat application in industry, in which the Committee emphasized the necessity of continuation to develop the indigenous technology of VHTRs in preparing for the coming century.

In connection with the view of industry, Japan Atomic Energy Commission (JAEC) started the Special Committee on HTGR Research and Development Program in April, 1986 to investigate clearly a significance of the VHTR development in Japan.

After serious discussion for four months, the Special Committee submitted a interim report to the JAEC. In this report the Committee recognized the definite significance of the VHTR development in Japan and recommended that it was appropriate to construct a high temperature engineering test reactor (HTTR) as a research facility to replace the experimental VHTR schemed by JAERI.
Based on this conclusion, JAERI has set about the construction program of HTTR to use it for the purpose of not only upgrading the current technologies but also basic researches in cooperation with universities and national institutes. JAERI is now about to solidify a design concept of HTTR by qualifying the current technologies developed for seventeen years.

2. Technical Background

Looking back upon the development activities in Japan, main achievements are outlined as follows:

2.1 Research and Development in JAERI

In the beginning, the efforts were concentrated on the design of the experimental VHTR supplying the thermal output of 50MW with the reactor coolant outlet temperature at 1000°C. The functions of the experimental VHTR were as follows:

1) To demonstrate the technologies for nuclear process heat applications,
2) To confirm the safety of VHTR plants,
3) To develop the technologies for proto-type VHTRs.

However, following the recommendation of the Long-Term Program for Nuclear Energy Development revised by JAEC in 1982, the reactor coolant outlet temperature was changed from 1000°C to 950°C since the big project for the iron and steelmaking application was terminated in 1980.

Afterwards, the design efforts have been continued and the detail design of the experimental VHTR has been completed in 1985.

In parallel with the design works, research and development has been carried out on fuels and graphite materials, heat resistant alloys, components and structures, reactor engineering, irradiation tests at a high temperature, and even thermochemical hydrogen production.
To serve for the irradiation tests of the coated particle fuels and other materials at the temperature up to 1000°C, the in-pile helium gas loop OGL-1, installed at the Japan Material Testing Reactor (JMTR), was put into operation in 1976. After numerous irradiation works in the OGL-1, the 10th irradiation test of the fuel compacts was finished in May, 1986.

For the purpose of clarifying the behaviors of the graphite materials in the core environment, extensive studies have been in progress on chemical and physical properties as well as mechanical properties. The results through these studies have been presented in the IAEA Specialist Meeting.

As for the heat resistant alloys for structural applications, the long-term performance tests including corrosion, carburization, creep, stress-rupture and thermal aging characteristics on the Hastelloy-XR have been conducted. For the development of second generation superalloys, the various tests to assess the feasibility of nuclear reactor applications on the five kinds of Ni-Cr-W alloys with different Cr/W ratio have been examined for exploring the optimum ratio for balancing between creep strength and corrosion resistivity.

Concerning the development of components and structures at a high temperature, the Helium Engineering Demonstration Loop (HENDEL), which consists of the Mother (M), Adapter (A) and Test (T) sections, was put into operation partially in 1982.

The M+A section of the HENDEL, which provides the helium gas at a maximum temperature of 1000°C with a maximum flow of 4kg/s, has been operated for more than 6,300 hours so far. In the fuel stack test section (T₁), the thermo-hydraulic tests of the mock-up fuel and control rods in reactor operating conditions were finished in 1985.

The construction of the test section T₂ for in-core structure tests was completed in June, 1986, and test operation for about 43 hours at a temperature of 1050°C was successfully proceeded.
The test items in the T2 test section are:
1) Coolant leakage through the graphite block structures,
2) Performance of the core support structure,
3) Coolant mixing behavior at the high-temperature plenum.

Other test sections for the high-temperature components of the heat removal system such as an intermediate heat exchanger and a steam generator are to be constructed in the near future.

In the area of reactor engineering, critical experiments on the 20% enriched uranium graphite-moderated cores simulated to VHTR have been carried out in the critical assembly (VHTRC) which was reconstructed from the experimental facility, Semi-Homogeneous Experiment (SHE).

For the reactor instrumentation, the high-temperature fission counter and gamma-compensated ionization chambers have been developed to withstand at a temperature up to 800°C.

In addition, the miscellaneous developments have been carried out on the thorium fuel, fuel reprocessing, coaxial double pipe, plate out of fission products and thermochemical hydrogen production, etc.

2.2 Research and Development Activity in ERANS

The purpose of the project was to establish the fundamental engineering technologies required for the construction of the nuclear steelmaking pilot plant, which would be connected with the experimental VHTR being developed by JAERI to demonstrate the first application of nuclear heat.

Main items and obtained results were as follows:
1) High temperature heat exchanging system
   An intermediate heat exchanger (1.5MWt) and a high temperature helium test loop were constructed and operated successfully from 1978 to 1980 by Ishikawajima-Harima Heavy Industries.
2) Heat resistant superalloy

Several kinds of alloy were tested with respect to the required properties, and two alloys, KSN and SSS113MA, were selected as the candidate for the heat transfer tube of the intermediate heat exchanger. These were concerned by Sumitomo Metal Industries, Hitachi Metals Industries, and Kobe Steel Industries.

3) High temperature heat insulating materials

Development works were conducted on alumino-silicate fibrous materials and fused silica type refractories by Isolite Babcock Refractories Co. and Toshiba Ceramics Industries.

4) Reducing gas making unit

Research and development was carried out on making the reducing gas by steam reforming of light hydrocarbons and gasification of the pitch, which were obtained by steam cracking. Steam reforming test plant and pitch gasification test plant were constructed and operated satisfactorily by Chiyoda Chemical Engineering & Construction Co. Ltd.

In addition, the conceptual design of the nuclear steel-making pilot plants, FM-50 system and UC-50 system, was carried out sufficiently as prearranged by the Group of System Analysis.

These results achieved in the program were transferred from ERANS to JAERI after the termination of the program in 1980.

2.3 Development in Industry

Most of nuclear industries in Japan have been concerned the development of VHTR technologies in cooperation with JAERI and/or ERANS.

At Mitsubishi Heavy Industries a high-temperature helium gas loop was installed in 1977 to test the performance of com-
ponents, behaviors of metallic and graphite materials at a temperature up to 1000°C and a pressure up to 49kg/cm²G.

In 1978, a high-temperature helium test loop (EM-1.5) was constructed at Ishikawajima-Harima Heavy Industries to develop He/He intermediate heat exchangers and steam generators under the contract with ERANS.

In 1979, a high-temperature and high-pressure helium gas loop (KH-200) was installed at Kawasaki Heavy Industries to test the dynamic performance of helium flow, endurance of components and verification of dynamic analysis calculation codes.

Babcock-Hitachi Industries also built a high-temperature helium gas loop (BHK) in 1984 to develop He/He heat exchangers connected with side-by-side piping, steam generators and steam reformers.

The data attained in these developments are used individually for design of the reactor systems conducted under the contracts between JAERI and industries.

As for fuels and graphite materials, Nuclear Fuel Industries (NFI) fabricated the mass production system of the coated particle fuel in 1973 and the nuclear grade graphites IG-11 and IG-110 have been produced by Toyo Carbon Co. since 1975. These products are provided for the various experiments in OGL-1, VHTRC, HENDEL, etc.

3. Perspectives on VHTR Development

Special committee of JAEC issued the Interim Report in August, 1986.

Outline of the Report is summarised as follows:
3.1 Significance of Research and Development on VHTRs in Japan

In general, it is essentially important for promotion of nuclear energy development to improve the economy and to expand an application of nuclear energy.

Nowadays in Japan, electric power generation by light water reactors is established well so that the total installed capacity is 24,686 MWe with 34 units, which the gross generated output takes the share of 26% in total electricity and in 2030 the share of nuclear power supply is projected to be 58%.

Looking at a future situation, if the thermal efficiency of plant is more increased by producing high temperature nuclear heat, it can be expected to improve the economy and to expand applications of nuclear energy, which will be resulted in improvement of our living environment.

The VHTR is considered as one of the most promising nuclear reactors to solve these problems because it has the characteristics of highly-inherent safety as well as high-temperature nuclear heat to be useful for broad range.

In Japan, however, the definite time to introduce the nuclear process heat in industry is so uncertain at present that the steady promotion of the VHTR research and development is very significant from the viewpoint of long future.

In addition, Japan is now facing the time to encourage creative researches for innovation in science and technology so that national research institutes and universities as well as industries should promote basic researches for producing seeds of new technologies.

Along these trends, it is expected that the research and development on VHTRs will give a remarkable effect on technical innovation in the future.
3.2 Future Perspectives

Based on the significance of research and development on VHTRs and the review of technologies attained up to now, it is recommended that a high-temperature engineering test reactor (HTTR) should be constructed in JAERI as early as possible and more efforts should be devoted to upgrade a VHTR technology on the basis of the accumulated technologies and human resources so far in Japan.

Main subjects to be upgraded are as follows:
1) To establish fully the indigenous technology of VHTRs,
2) To demonstrate the highly-inherent safety of VHTRs such as passive safety,
3) To improve the core performance such as higher coolant temperature, fuel burnup, power density, etc.,
4) To improve the thermal efficiency by use of high-temperature nuclear heat,
5) To serve for irradiation tests.

Moreover, it is instructed that the operation of HTTR should be carried out in cooperation with national institutes and universities which have the requirements to use the HTTR for basic researches.

Consequently, the core structure of HTTR is to be a prismatic block type that is suitable for conducting the various irradiation in the core and then the thermal output is to be 30MW at least for setting the irradiation regions in the core.

4. New Program in JAERI

In accordance with the Interim Report, JAERI has shifted the research and development program from the previous planning of experimental VHTR to the new deployment of the HTTR as a research facility.

Forcusing on the construction of HTTR, it is reconsidered to modify the previous design concept of the experimental VHTR.
Implementing a design of HTTR, there should be taken account of on the items as follows:

1) To accomplish upgrading technologies of VHTRs,
2) To serve for the various irradiation tests,
3) To prove the usefulness of the nuclear heat application in future.

Therefore, regarding a current concept of HTTR, the basic specifications are shown in Table I, in which the reactor output power is 30MWt with the helium outlet temperature up to

<table>
<thead>
<tr>
<th>TABLE I BASIC SPECIFICATIONS OF HTTR</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>MAIN ITEM</strong></td>
</tr>
<tr>
<td>1 REACTOR THERMAL OUTPUT</td>
</tr>
<tr>
<td>2 REACTOR OUTLET COOLANT TEMP (AVG)</td>
</tr>
<tr>
<td>3 REACTOR INLET COOLANT TEMP (AVG)</td>
</tr>
<tr>
<td>4 FUEL</td>
</tr>
<tr>
<td>5 FUEL ELEMENT TYPE</td>
</tr>
<tr>
<td>6 DIRECTION OF COOLANT FLOW</td>
</tr>
<tr>
<td>7 PRESSURG VESSEL</td>
</tr>
<tr>
<td>8 NUMBER OF PRIMARY COOLANT CIRCUIT</td>
</tr>
<tr>
<td>9 HEAT TRANSMISSION</td>
</tr>
<tr>
<td>10 PRIMARY COOLANT PRESSURE</td>
</tr>
<tr>
<td>11 SECONDARY COOLANT PRESSURE</td>
</tr>
<tr>
<td>12 COMPONENTS IN THE SECONDARY CIRCUIT</td>
</tr>
<tr>
<td>13 CONTAINMENT TYPE</td>
</tr>
<tr>
<td>14 PLANT SERVICE LIFE</td>
</tr>
</tbody>
</table>
950°C. Fig. 1 shows coolant diagram of the system. As shown in Fig. 2, the core of HTTR is composed of the hexagonal fuel blocks and control blocks. Fig. 3 shows the vertical view of the core contained in the pressure vessel. The reason why we choose the core of prismatic block type is based on
availability and flexibility for various irradiation tests. As for the circumstance of neutron irradiation in the reactor, there are three regions, namely the regions of core, replaceable reflectors and permanent reflectors as shown in Table II, in which it can be estimated that the maximum flux of thermal neutron is around $8 \times 10^{13} \text{n/cm}^2 \cdot \text{s}$ and the maximum helium temperature is about 1100°C at the assumed core.
TABLE II IRRADIATION CHARACTERISTICS OF HTTR CORE REGION

<table>
<thead>
<tr>
<th>Irradiation Region</th>
<th>Core Region</th>
<th>Replaceable Reflector</th>
<th>Permanent Reflector Column</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Fuel Rod Column</td>
<td>Control Rod Column</td>
<td>Repl. Ref. Column</td>
</tr>
<tr>
<td>Irradiation</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Condition</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Style</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Size</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Number of Specimens (Max)</td>
<td>24</td>
<td>7</td>
<td>48</td>
</tr>
<tr>
<td>Max. Neutron Flux (Fast &gt; 0.1 MeV)</td>
<td>~2 x 10^12 n/cm²·s</td>
<td>→</td>
<td>~1 x 10^12 n/cm²·s</td>
</tr>
<tr>
<td>Thermal (&lt;= 0.4 eV)</td>
<td>~6 x 10^13 n/cm²·s</td>
<td>→</td>
<td>~7 x 10^12 n/cm²·s</td>
</tr>
<tr>
<td>Temperature at the Region</td>
<td>400 ~ 1,100°C</td>
<td>400 ~ 1,000°C</td>
<td>400 ~ 900°C</td>
</tr>
<tr>
<td>Instrumentation installed in the Core</td>
<td>Measurements of Temperature, Thermal Neutron Flux, Neutron Monitor, and Fuel Failed Detector</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Example of Irradiation Specimens</td>
<td>Block Type Fuel, Pebble Type Fuel</td>
<td>Fuel Compact, Graphite, Metal, Ceramics</td>
<td>→</td>
</tr>
</tbody>
</table>

Concerning the construction schedule of HTTR, as shown in Table III, licensing review by Nuclear Safety Commission is to be conducted during FY1988 and the construction of the HTTR is projected from FY1989 to FY1994.

TABLE III CONSTRUCTION SCHEDULE OF HTTR

<table>
<thead>
<tr>
<th>FISCAL YEAR</th>
<th>'86</th>
<th>'87</th>
<th>'88</th>
<th>'89</th>
<th>'90</th>
<th>'91</th>
<th>'92</th>
<th>'93</th>
<th>'94</th>
</tr>
</thead>
<tbody>
<tr>
<td>Modified Design</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Final Design</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Geological Survey</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Licensing</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Construction</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fuel Fabrication</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

5. Conclusion

Based on the recommendation of the Special Committee, JAEC recognized the significance of the VHTR development from the viewpoint of long future in Japan.
At present no one can predict definitely an economic stability under the low price of oil in the 21st century. Although it is assumed in Japan that there is no expectation to make a practical use of the nuclear process heat within this century, the VHTR is considered as one of the most promising reactors since it has highly-inherent safety and potentiality to expand the market of the nuclear process heat application. These senses on the VHTRs are being emphasized because much more it is very important to realize an intrinsically safe reactor now that many peoples have experienced two severe accidents such as TMI-2 in 1979 and Chernobyl in last April. The Russian "Chernobyl" means, as you know, "Wormwood" which is indicated in the Revelation to John of the New Testament, namely that "The name of the star is Wormwood. A third of the waters became wormwood, and many men died for the water, because it was made bitter." The prediction has come true.

Nowadays it seems that we are about to get into new age that the small is beautiful since small and medium reactors are strongly expected for not only electric power generation but also process heat application in industry.

Therefore, in Japan, it should be emphasized that the importance of the VHTR development is not so much how to investigate the economy of the nuclear process heat application as how to establish the indigenous technologies as early as possible in order to cooperate with the advanced countries, so that it is actively expected for the HTTR to play an important part at the frontier of the VHTR development toward the 21st century.
PLANT DESIGN OF A HIGH TEMPERATURE ENGINEERING TEST REACTOR IN JAERI

O. Baba
Japan Atomic Energy Research Institute,
Tokai-mura, Ibaraki-ken,
Japan

Abstract

Special Committee on HTGR Research and Development Program submitted an interim report after serious discussion which recommended that a high temperature engineering test reactor (HTTR in this report) should be built to make possible irradiation tests not only for HTGR fuel but also for various materials in high temperature.

The required functions to HTTR and its basic design condition are described together with selecting progress. A preliminary design of a high temperature testing reactor, which makes possible HTGR fuel irradiation tests and demonstration tests of HTGR inherent safety, has been done and the results are also explained.

1. Introduction

In March 1986, Special Committee on HTGR Research and Development Program, which was set up by Japan Atomic Energy Commission (JAEC), started the review on the results of past activities on VHTR development and energy situation in Japan. The committee has submitted an interim report to JAEC in August. The report recommends that it is appropriate to construct a high temperature gas-cooled testing research reactor as a research facility.

According to the recommendation JAERI has started the design of a high temperature engineering test reactor (HTTR) in place of the Multi-purpose Very High Temperature Experimental Reactor (VHTR-Ex), which JAERI has developed since 1969.

2. Functions of HTTR

Major functions which were required to HTTR in the interim report of the Special Committee are as follows:

Category A. For upgrading present HTGR technology,
   a. Irradiation test of HTGR fuel and graphite,
   b. Demonstration test of HTGR inherent safety,
   c. Demonstration test of high temperature heat utilization system,
Category B. For various progressive fundamental researches in high temperature,
a. Irradiation test in the field of material science and technology,
b. Irradiation test in the field of fusion technology,
c. Irradiation test in the field of radio-chemistry.

The former functions (A. above) have been attached to the VHTR-Ex. The latter functions (B. above) are newly added reflecting the needs in Japan. These haven't been taken into account for the design of the VHTR-Ex, therefore, JAERI has to change its design to make possible these irradiation tests. As it is not clear yet what kind of irradiation tests are possible in what condition, JAERI is going to decide irradiation condition, capacity and limit in HTTR in detail after feasibility study.

The irradiation tests which are subjected to the feasibility study at present are listed up below:

a. Irradiation of high temperature alloys, carbon and ceramic materials,
b. Irradiation of fusion reactor blanket,
c. Irradiation test of high temperature radiation resistant detectors,
d. Demonstration test of continuous Tritium production and recovery technology,
e. Production of radio-isotopes,
f. Various materials irradiation tests for radiochemical hydrogen production, thermal splitting of tar, etc..

3. Basic Design Conditions

In June 1986, JAERI has finished a preliminary design of a high temperature testing reactor which has the functions of category A. As mentioned above, this design should be changed in detail after feasibility study in order to fulfill the functions of category B.

However, as it is expected that the basic design conditions will not be changed, here is written the selecting progress of design condition.

(1) Thermal Power

Reactor core performance such as core life, shut down margin including rod-stuck margin and maximum fuel temperature was examined for three cases, 50Mw, 30Mw and 10Mw of thermal power. Irradiation test capability of HTGR fuel element and graphite block was also examined.
The target for core life was set up to 660 effective full power days and 1300°C for the maximum fuel temperature in operation condition of core average power density and core outlet coolant temperature at 2.5 w/cc and 850°C respectively.

The reactor core of 10Mw did not fulfill the target values of core life and reactor shut down margin. The maximum fuel temperature was lower than 1300°C, but higher than that of 30Mw core.

The core of 30Mw didn’t meet the irradiation conditions of VHTR fuel, but it became clear that 30Mw core can give better irradiation condition than 50Mw core.

(2) Reactor Outlet Coolant Temperature

The outlet temperature was surveyed from the view point of defining the lowest temperature which can be used for hydrogen production as a case of nuclear process heat direct application. For various hydrogen production processes with thermochemical reaction, the highest temperatures of each processes were compared. The UT-3 process, which is developed in Tokyo University in Japan, was found to have the lowest reaction temperature of 730°C at the highest point in the process.

Reactor outlet coolant temperature, therefore, was set to be 850°C taking into account primary and secondary coolant temperature difference at an intermediate heat exchanger (IHX) and temperature drop at piping.

(3) Configuration of Reactor Cooling System

The configuration of reactor cooling system was examined from the aspects of safety, required function, plant controllability and construction cost.

There were mainly two possibilities in the selection of core cooling method in case of failures of main cooling system. One of them was that the reactor vessel cooling system (VCS) cools reactor vessel and core in all cases of main cooling system failure. The other one was that a forced direct core cooling loop in safety class cools the reactor core when flow path is sound and VCS cools the core in other case. After core cooling analyses, the latter was selected to prevent damage of control rod.

Numbers of main cooling loop, safety class forced cooling loop and VCS were set 1, 1 and 2 respectively to minimize construction cost.

There were many possibilities with an IHX. Heat exchange capacity was one of the important discussion points. IHXs of 25Mw have been
designed for long time in VHTR-Ex in order to have two symmetry main cooling loops. An IHX of 10Mw was selected as a variation for it considering a scale of process heat plant and heat loss in the piping.

Another possibility with IHX was construction schedule. One of the options was to build it from the first of reactor construction and another was to build after operation of the reactor for certain duration in order to decrease initial construction cost. Cases which had no IHX for whole reactor life were also discussed. Final decision for this discussion will be done after comparison between whole construction cost and budget.

(4) Containment Type

Licensability of exclusion the rapid depressurization accident in primary cooling system was examined.

The materials used in reactor coolant pressure boundary are well known their properties and well experienced in nuclear and chemical industries. However the possibility of ductile ustable rupture may be very low, JAERI came to the conclusion that it might be very difficult to prove the probability enough low and to establish leak before break criteria because of lack of back data.

After irradiation dose analyses for public in case of rapid depressurization accident, containment vessel was selected.

(5) Selected Design Conditions

Typical cases with combinations of design parameters described above are shown in Table I. Construction cost for each case was roughly estimated and compared. The numbers show the order from lower cost.

Finally, the case 2 was selected and other important design conditions were decided as shown in Table-II.

3. Preliminary Design Result

(1) Cooling System

The cooling system is composed of a main cooling system (MCS), an auxiliary cooling system (ACS) and two VCSs. MCS is non-safety class and ACS and VCSs are safety class as explained above.
Table I  Typical Combination of Design Parameters

<table>
<thead>
<tr>
<th>Case</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
<th>5</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal Power (Mw)</td>
<td>50</td>
<td>30</td>
<td>30</td>
<td>30</td>
<td>10</td>
</tr>
<tr>
<td>Outlet Temp. (°C)</td>
<td>950</td>
<td>850</td>
<td>850</td>
<td>850</td>
<td>850</td>
</tr>
<tr>
<td>IHX</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>(Mwt)</td>
<td>10</td>
<td>10</td>
<td>10</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Set up Schedule</td>
<td>after operation</td>
<td>with reactor</td>
<td>after operation</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Cooling System</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Construction Cost (number from low)</td>
<td>5</td>
<td>3</td>
<td>4</td>
<td>2</td>
<td>1</td>
</tr>
</tbody>
</table>

Table II  Basic Design Conditions of High Temperature Testing Reactor

<table>
<thead>
<tr>
<th>Item</th>
<th>Basic condition</th>
</tr>
</thead>
<tbody>
<tr>
<td>1 Reactor thermal output</td>
<td>30 Mw</td>
</tr>
<tr>
<td>2 Reactor outlet coolant temperature (average)</td>
<td>850°C, 950°C</td>
</tr>
<tr>
<td>3 Reactor inlet coolant temperature (average)</td>
<td>395°C</td>
</tr>
<tr>
<td>4 Fuel</td>
<td>Low enriched, UO₂</td>
</tr>
<tr>
<td>5 Fuel element type</td>
<td>Prismatic block</td>
</tr>
<tr>
<td>6 Direction of coolant flow</td>
<td>Downward-flow through the core</td>
</tr>
<tr>
<td>7 Pressure vessel</td>
<td>Steel</td>
</tr>
<tr>
<td>8 Number of main cooling loop</td>
<td>1</td>
</tr>
<tr>
<td>9 Heat transmission</td>
<td>IHX and PWC (parallel loaded)</td>
</tr>
<tr>
<td>10 Primary coolant pressure</td>
<td>4.02 MPa</td>
</tr>
<tr>
<td>11 Secondary coolant pressure</td>
<td>He pressure is higher than primary coolant pressure</td>
</tr>
<tr>
<td>12 Containment type</td>
<td>Steel containment</td>
</tr>
<tr>
<td>13 Plant life</td>
<td>20 years</td>
</tr>
</tbody>
</table>
MCS has two heat transfer paths from reactor core to the environment air. One of them is from reactor core to air cooler via primary pressurized water cooler (PPWC) and the other one is from reactor core to air cooler via IHX and secondary pressurized water cooler (SPWC). The air cooler for both paths is common. IHX and SPWC have the same heat transfer capacity of 10Mw. PPWC has two modes of operation which are 30Mw and 20Mw. When IHX and secondary helium loop are operated, PPWC is operated at 20Mw in the same temperatures at primary helium inlet and outlet of PPWC as operated at 30Mw. In order to make possible these two modes operation, PPWC has two primary helium outlet nozzles each of which is used for 20Mw or 30Mw operation. Main reasons for two modes operation are,

(a) to operate reactor at 30Mw even in case the IHX is not built at the same time with reactor,

(b) to get simpler and clearer conditions in safety experiments by operating PPWC at 30Mw, in which thermal transient might be much severer than normal operation.

Reactor inlet temperature is controled by changing inlet water temperature at PPWC and secondary helium flow rate for IHX. All of the gas circulators and water pumps are operated with commercial electricity without emergency back up.

ACS consists of an auxiliary heat exchanger (AHX), auxiliary gas circulators (AGCs) and an air cooler. ACS is independent system from MCS and is normally in stand-by condition. ACS is operated only when MCS has some failures or the reactor is shut down and MCS is in overhaul. All of the active components such as AGC and water pump are redundant and have emergency back up power supply.

A flow diagram of cooling system is shown in Fig. 1, and a heat and mass balance of MCS at parallel operation with IHX and PPWC is shown in Fig. 2.

The two VCSs have the same heat removal capacity of 100% to cool reactor vessel and core in emergency. Both VCSs are normally operated at 100% flow rate in order cool biological shield around the reactor vessel, therefore no start-up motion is required at the accident.
FIG. 1. Cooling System

FIG. 2. Mass and Heat Balance (1)
(2) Reactor Building

The reactor building contains a containment vessel, sub-systems for cooling systems, ventilation and air conditioning systems, reactor control room, fuel handling and storage facility, etc., in order to simplify asseismic design. The reactor vessel is put at almost the centre of the reactor building as many LWRs is Japan, so that the centre of gravity of reactor building is almost in the reactor vessel and a vertical motion of the reactor core caused by rocking motion of the building during earthquake is minimized. As shown in Fig. 3, the containment vessel is very small and it has a big nozzle above reactor vessel. The cover of the nozzle is removed when fuels are exchanged. The big room over the operating floor is so designed that pressure inside is maintained slightly lower than outside.

(3) Reactor Core and Vessel

Reactor fuel is a prismatic block type so as to serve irradiation space for HTGR fuel and graphite block. An active core is surrounded by

FIG. 3. Cross-Section of Reactor Building
replaceable and permanent reflector blocks. The reactor pressure vessel is made by a normalized and tempered 2 1/4 Cr-1Mo steel. Reactor coolant flows into the pressure vessel through a nozzle attached at the bottom and flows up inside of the vessel in the mean time it cools core restraint ring. It flows downward through the core and all coolant is guided into a hot plenum below the core and it flows out through the same nozzle at the bottom but inside of the co-axial double nozzle. Coolant of ACS flows almost the same path with that of MCS but through different nozzle. A vertical cut down cross section is shown in Fig. 4.

FIG. 4. Reactor Vessel and Core
(4) Components in Cooling Systems

The IHX is a vertical helical coil counter flow type heat exchanger as shown in Fig. 5. Primary coolant flows shell side and secondary coolant tube side. Materials for cold pressure boundary and very high temperature structures are 2 1/4 Cr-1Mo steel and Hastelloy-XR respectively. Primary hot gas flows into IHX through inside of a co-
axial double nozzle at the bottom and flows up outside of heat transfer tubes exchanging heat with secondary coolant in the tubes. Primary hot gas flows out through a nozzle at the top of IHX vessel side wall and pressurized by MGC and flows in again into the IHX vessel. It flows down ring space between inner and outer vessels and goes out through the bottom nozzle. The flow path is more complexed than the IHX tested in ERANS (Engineering Research Association of Nuclear Steelmaking), but it was thought safer for the failures at thermal insulator and inner vessel.

PWCs and AHX have very similar structure which is a vertical U tube type heat exchanger. Helium gas, which is primary coolant for PPWC and AHX and secondary coolant for SPWC, flows outside of heat transfer tube and pressurized water flows inside of the tube. Flow path of helium gas is similar to that of IHX on the same reason. The PPWC has two kinds of helium outlet nozzles to adjust heat transfer area to 20Mw and 30Mw as explained before. (Fig. 6)

FIG. 6. Pressurized Water Cooler (PWC) Heat Transfer: 30KW
Co-axial double pipe is adopted for high temperature gas transfer which is shown in Fig. 7. Cold gas at about 400°C flows the gap between inner and outer pipes toward reactor inlet and hot gas from the reactor flows inside of the inner pipe. Temperature of inner pipe is almost the same as that of outer pipe because its inside is thermally insulated. Difference of thermal expansion between these pipes is absorbed by a flexibility of inner pipe. Therefore no compensator as bellows joint is used. This is also the reason why coolant nozzle of reactor pressure vessel is at the bottom.

![Diagram of Co-axial Double Pipe]

4. Conclusion

A preliminary design of a high temperature testing reactor has been done successfully which has the functions of category A. As described in 2. Functions of HTTR, the HTTR has to have the functions of category B which includes irradiation tests of various kind of materials for individual purpose.
JAERI has to define the irradiation condition which can be given in the HTTR after feasibility study. Afterwards JAERI is going to give necessary change to the basic conditions in detail and the preliminary design written in this report.

5. Acknowledgement

All of the works written in this report was done by the members in VHTR Designing Laboratory of JAERI in co-operation with the members in Fuji Electric Company and Mitsubishi Heavy Industry.

Author thanks all of them.

Reference

STATUS OF THE HTR-MODULE PLANT DESIGN

I.A. WEISBRODT, W. STEINWARZ
Internationale Atomreaktorbau GmbH,
Bergisch Gladbach

W. KLEIN
Kraftwerk Union AG,
Erlangen
Federal Republic of Germany

Abstract

Based on the demand for a technically clear, safe and economic reactor system, the KWU Group consistently used the extensive know-how obtained from German light water reactor technology and successful operation of the AVR for development of the HTR-Module. With its operating and safety features and its flexible application possibilities, this reactor concept fulfills current market demands to a substantial extent.

1. Introduction

After Chernobyl the discussions concerning inherently safe or forgiving reactors have been increasingly intensified, not only among reactor safety experts but also in the general public. The HTR-Module, developed as a flexible small heat source by Kraftwerk Union AG (KWU) and its affiliated company Interatom GmbH, exhibits a high degree of passive safety by consequently using the inherent safety features of small high-temperature reactors.

Detailed information on this aspect will be presented in one of the next lectures by Dr. Lohnert. This report deals with the status of the technical design as well
as the prospects of the short-term and long-term market applications.

2. **Design Features**

The concept of a modular high-temperature reactor with low unit power is a new approach for designing a nuclear energy source for the direct and indirect application of process heat in industrial countries (Fig. 1). Furthermore, the modular power plant system is well suited for small and medium grid capacities, which are usually required for electricity generation in developing and threshold countries. However, the utilization of nuclear energy in these countries is not restricted to electricity generation alone. In the future,

![Fig. 1 Applications of the HTR-Module](image-url)
increasingly vital facilities for the drinking water supply or for irrigation could be realized by means of desalination plants combined with HTR-Module plants. A further potential application is the extraction of oil from oilsand and residues in oil deposits by flooding these with process steam (so-called tertiary oil extraction).

In this case the coupling with hydrogen production could be interesting. In the longer term, the production of hydrogen and synthetic fuels from coal or natural gas using nuclear process heat will gain importance for countries with domestic fossile resources.

Maximum flexibility regarding plant power rating and type of application is achieved by combining standardized modular reactor units; therefore this system permits step-by-step adjustment to the increasing energy demand for Third World countries by simply adding on more modules to existing plants. These units operate in parallel, giving high overall plant availability.

The main design criteria and design elements of the HTR-Module are (Fig. 2):

- Extensive reference to the construction and operational experience with the 40 MJ/s AVR experimental power plant.

- Extensive use of KWU light water reactor technology: In particular this includes the use of steel pressure vessels - under quite comparable operating conditions - as well as the
Fig. 2 Cross Section of a Modular Unit with Steam Generator (Primary Circuit)
adaption of techniques and processes for in-service inspections, maintenance control and instrumentation. Last but not least, the buildings, their arrangement and the secondary and auxiliary systems are in accordance with the technology proven in the course of light water reactor construction.

- Standardized well-proven graphite balls with low enriched uranium as fuel elements (pebble bed core). The waste disposal management is based on direct final storage without reprocessing.

- By selecting a relatively small core diameter of 3 m, the control rods and absorbers can be located in the graphite reflector. This permits gravity insertion of the rods and absorber balls.

- Due to the small core diameter in combination with a low core power density, it is possible to dissipate the decay heat from the reactor core in a passive way to 3-fold redundant surface coolers located outside of the reactor pressure vessel via heat conduction and radiation in the event of failure of the main heat sink. The maximum fuel temperature does not exceed 1,600 °C during all accident conditions. This in turn precludes a significant failure of fuel element particles and a high release of fission products from the fuel.
Active, redundant primary cooling loops, with associated redundant secondary cooling chains and emergency power supplies can therefore be omitted.

The aforementioned boundary conditions lead to a power rating of 200 MJ/s per module for the steam generating plant with 700 °C core outlet gas temperature, and to a power rating of 170 MJ/s for the process heat application with 950 °C core outlet gas temperature.

Reactor core and steam generator are installed in ferritic steel pressure vessels of the proven LWR-quality in side-by-side arrangement with lowered steam generator. Due to this arrangement thermohydraulic decoupling of the reactor core and the operational heat sink is realized after disruption of active gas circulation.

Thermal loads of the steam generator due to natural convection are therefore inherently avoided. Furthermore direct water ingress into the core in the case of a tube failure is not possible.

The arrangement of the components results in easy accessibility for inspection, maintenance and repair and to low dose rates.

In the light of the Chernobyl catastrophe, some of these features, such as limitation of the fuel temperature and passive decay heat removal, appear
to us to be especially important for nuclear power plants intended for sites in highly populated or industrial areas. In the extreme case, the HTR-Module plants could even be left without interference and would not have any detrimental effect on the environment. As a result, administrative measures (evacuation etc.) after accidents are not required.

3. **Technical Layout of the Nuclear Steam Generation System**

The arrangement of the modular reactor unit as a nuclear steam generation system is shown in Fig. 2. Cold helium with an inlet temperature of 250 °C enters the core from the top via reflector boreholes. Flowing downwards through the core the helium is heated up to 700 °C. It passes the bottom reflector which is designed as a mixing chamber and flows from the hot gas plenum into the horizontal hot gas duct leading to the steam generator. To achieve upflow evaporation, the hot helium flows from the top of the steam generator to the bottom. From there the cooled-down helium flows upwards through the annular cold gas section around the steam generator bundle to the blower placed above the steam generator.

The blower pumps the coolant back to the reactor via the annulus in the horizontal pressure vessel and through channels in the reflector to the cold gas plenum on top of the core.
Feed-water and steam flow upwards through the heat-exchanger tubes of the steam generator in counterflow to the helium. The live steam produced in this way has a temperature of 530 °C and a pressure of up to 220 bar.

The reactor pressure vessel and the steam generator pressure vessel together with the connecting vessel form the pressure enclosure of the modular reactor. This only comes into contact with cold gas, due to the selected gas ducting, thus permitting application of the pressure vessel technology proven in light-water reactors.

Six absorber rods, which are positioned in the boreholes of the side reflector, are provided for control and hot shutdown of the reactor core.

For all possible incidents the reactor safety system always initiates the same procedures as follows: The six reflector rods are dropped, the blower is stopped and the blower flap is closed. In the case of a loss of pressure accident and a steam generator heating tube rupture, the steam generator is quickly drained in addition.

In the cold condition, the core is shut down by way of absorber spheres, which are contained in vessels above the thermal shield during normal operation. If required, the small absorber spheres are released and drop into 18 elongated holes provided for this purpose in the side reflector.
Each HTR-Module unit is installed in a primary cell, the concrete walls of which support the reactor and steam generator pressure vessels and their internals. Surface coolers are positioned around the reactor pressure vessel for the removal of heat losses during normal operation and for decay heat removal when the reactor is shut down.

However, their main task is to protect the reactor pressure vessel and the concrete walls of the reactor cell. No environmental impacts are to be anticipated even if they should fail.

The steam produced can either be used in a back-pressure turbine or in a condensing extraction turbine for the generation of electric power.

The main data of the nuclear heat generating system of the HTR-Module with steam generator are listed in Fig. 3.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Unit</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal power</td>
<td>MW</td>
<td>200</td>
</tr>
<tr>
<td>Core diameter</td>
<td>m</td>
<td>3.0</td>
</tr>
<tr>
<td>Core height</td>
<td>m</td>
<td>9.6</td>
</tr>
<tr>
<td>Mean power density</td>
<td>MW/m³</td>
<td>3.0</td>
</tr>
<tr>
<td>Helium temperatures (inlet/outlet)</td>
<td>°C</td>
<td>250/700</td>
</tr>
<tr>
<td>System pressure</td>
<td>bar</td>
<td>60</td>
</tr>
<tr>
<td>Number of control rods</td>
<td></td>
<td>6</td>
</tr>
<tr>
<td>Number of absorber ball systems</td>
<td></td>
<td>18</td>
</tr>
<tr>
<td>Loading scheme</td>
<td></td>
<td>Reshuffling ∼ 15 times</td>
</tr>
<tr>
<td>Fuel cycle</td>
<td></td>
<td>U.Pu</td>
</tr>
<tr>
<td>Heavy metal loading of fuel element</td>
<td>g</td>
<td>7</td>
</tr>
<tr>
<td>Enrichment</td>
<td>%</td>
<td>7.8</td>
</tr>
<tr>
<td>Burnup</td>
<td>MWd/t</td>
<td>80000</td>
</tr>
<tr>
<td>Fuel incore time</td>
<td>days</td>
<td>∼ 1000</td>
</tr>
</tbody>
</table>

Fig. 3 Main data for the KWU modular HTR-core for co-generation of steam and electricity
The single HTR-Modules of a multi-modular plant can be operated independently. Each module can be individually shut down, run up and connected to the overall plant once it has achieved the operating conditions. Thus, even in the case of failure of a single module, operation of the overall plant can be continued at a reduced power level.

Fig. 4 shows a cross section of the reactor building, which is designed as a vented containment and provides full protection against all external events.
Fig. 5 shows a site plan of an industrial power plant with two HTR-Modules as an example. The reactor building is arranged in replicated serial modular units. The turbine building is located in front of the reactor building and houses the two replicated plant sections of the water system cycle with turbo-set and the connected process steam cycle.

The inherent safety features of the HTR-Module allow the entire conventional steam power plant with all secondary components and systems to be designed and managed in line with generally applicable guidelines for conventional plant construction.
4. **The HTR-Module for the Direct Utilization of Process Heat**

The HTR-Module is also particularly qualified for the direct utilization of process heat because of its capability to provide heat at a high temperature level.

Depending on the field of application, the steam generator of the HTR-Module generating electric power or process steam is replaced by a helium/helium intermediate heat exchanger or by a steam reformer (see Fig. 1). As the steam reformer uses process heat only in the upper temperature range from approx. 950 °C to 700 °C, a steam generator is installed in series to utilize the lower temperature range from 700 °C to 300 °C for the production of process steam.

By reducing the power rating of the core from 200 MJ/s to 170 MJ/s the passive inherent safety features of the modular reactor are fully retained.

A plant with steam reformer for steam reforming of methane is described as an example of direct process heat utilization (Fig. 6).

5. **Barge-mounted energy station**

Especially for use in developing and threshold countries, a barge-mounted power plant has been designed as a self-sufficient station for the production of electric power or, alternatively, for the co-generation of electricity and process steam, or of electricity and drinking water or
water for irrigation purposes. This is especially suitable for countries with a low grid capacity and arid areas.

Fig. 7 shows the general arrangement of a 2-Module Plant for electricity and drinking water production as an example.

The prelicensed power plant will be completely mounted on a barge in the shipyard for shipping to the plant site by sea. At the site the barge will be secured in a coastal position or in a prepared lagoon near the coast. In principle, the barge can also be anchored in the off-shore area as a floating energy station.
Fig. 7 General Arrangement of a 2-Module Plant for Electricity and Drinking Water Production
The convincing features of this concept are simplified project management, workshop manufacture, cost and time advantages due to prefabrication, standardization and hence possible unification of the licensing procedure.

Significant economic advantages result from the combination of electricity production and desalination using the reverse osmosis (RO) technique: Depending on the conditions, different product mixing ratios can be realized in a flexible manner. Fig. 8 shows a simplified flow scheme of such a plant. Hereby part of the electrical energy generated by the turbo-sets is branched off for the desalination plant. The main consumers are the high-pressure pumps for overcoming the osmotic pressure and covering the pressure losses.

Fig. 8  Simplified Flow Scheme of a 2-Module Plant for the Co-Generation of Electricity and Drinking Water
A further interesting field of application is the generation of high-pressure steam for tertiary oil extraction or the combination of \( \text{H}_2 \)-production from oil-gas (with a steam reformer) and steam production for injection in the oil field (with a steam generator connected in series).

6. **Economics**

In order to underline the expectations for a potential, economic application, one example of our analyses for electricity generation alone is offered by Fig. 9 as a suitable comparison measure. The calculation is based on the economic conditions and parameters prevailing in Germany.

It shows the results of an economic comparison of a nuclear power plant with 4 respectively 8 HTR-Modules and a hard-coal-fired power plant of equal size.

![Diagram showing power generating costs for hard coal power stations v.s. modular HTR's averaged over a 20-year rateoff period (in 1986 currency) (as per 8/86)]
The hard-coal-fired power plants are calculated for different coal prices

- DM 260.--/t  (German coal price)
- DM 210.--/t  (mixed price of imported coal and German coal)

The other main input data for the economic analyses are indicated on the right side of this figure.

From this analysis it can be concluded that, in Germany, power plants with an HTR-Module can easily compete with fossil-fired power plants of equal size. This evaluation improves significantly in the application field of co-generation of heat and electricity. For other countries, especially countries of the Third World, this result has to be proved case by case.

Fig. 10 shows the costs for the production of drinking water using a barge-mounted energy station with a selected electricity/drinking water ratio according to Fig. 8. When the results are considered in view of the current market situation, it is evident that the application chances of such plants can be evaluated as being very good in the potential user countries.

With reference to the financing, plants with a low power rating and hence lower investment costs are particularly interesting, as they can be financed in a simpler way. In the case of the barge-mounted plant, the fundamental possibility of remobilization reduces the financial risk significantly.
7. **Aspects of realization**

From KWU's point of view, the technology of the HTR-Module plants has reached such a status that application for the generation of electrical power and process steam (up to 530 °C) is possible at once. The application in the chemical industry or similar energy-intensive industries offers the best conditions in industrial countries. Today, an immediate application is also possible and economic for tertiary oil extraction, especially in combination with the production of hydrogen. For developing and threshold countries seawater desalination and electricity generation are the most promising applications.
The detailed planning and licensing procedure will be carried out during a 2 - 3 year phase according to the prevailing local conditions. In this context it is noteworthy that an advisory panel of the Federal Minister of the Interior has already evaluated and fully confirmed the safety concept. Furthermore, the panel has emphasized its particular suitability for industrial sites and sites near towns. The next step of realization has been initiated with the application for licensing of the technical and safety concept in 1986. The construction period of an HTR-Module power plant (land-based or barge-mounted, in this case including towing to site) is 36 - 48 months depending on size. This corresponds to the construction period of a coal-fired power plant of equivalent capacity.

Especially after Chernobyl the introduction of nuclear power plants into the market presupposes acceptance by politicians and the general public. HTR-Module plants with their high degree of inherent safety features seem to have a very good chance for such public acceptance in the future.
SMALL NUCLEAR POWER PLANTS: 10 MW GHR
GAS-COOLED HEATING REACTOR

H. SCHMITT
Hochtemperatur-Reaktorbau GmbH,
Mannheim, Federal Republic of Germany

Abstract

The GHR is a simple, safe and economic heating reactor that is designed on the basis of the proven technology of the Gas-Cooled High-Temperature Reactor. The main benefits derive directly from the favourable technical and safety related HTR-characteristics. These are

- low radiation exposure
- high inherent safety, even in the event of hypothetical accidents
- few active systems

The HRB/BBC-design of the GHR makes a wide use of developed and proven HTR components, such as

- proven fuel elements containing coated particles for an extremely favourable fission product retention
- helium as a coolant and ceramic structures that insure stability of the structures under all conditions
- components manufactured for AVR and THTR and tested up to service life loads.

The reactor pressure vessel, the intermediate circuit, the control room and all safety related systems are located underground which provides protection against all external impacts, sabotage etc. The remaining overground building is of small, conventional design and only houses auxiliary systems.
The whole nuclear heat generating system (primary system) is integrated into a prestressed concrete reactor pressure vessel. It is composed of the core, the absorber rods, a circulator and a heat exchanger to transfer the heat via the intermediate circuit to the consumer circuit.

The reactor control philosophy, the decay heat removal systems and the fission product retention barriers follow HTR-principles, which as a whole characterizes the GHR heating plant by the following features

- simply operating
- passive safety
- less maintenance and operating effort
- very low radiation exposure
- fuel elements for long operating periods
- low energy generating costs
- shop prefabrication
- short construction period.

**Plant Design**

From 1982 on first analyses of the heat market were conducted in Switzerland about the application of the nuclear energy from small reactors. Due to the specific Swiss residential structure a nuclear short-distance heat supply rated between 10 and 50 MW power output was envisaged. The relevant annual and daily heat requirements are presented in Fig. 1.

This initiative received a positive echo from the public as well as from industry. The preparatory project on the basis of BWR technology was called SHR. Two further projects joined: a second LWR type operating without active components under hydrostatic pressure (GEYSER) and the gas cooled heating reactor
concept (GHR) described in this paper. The main design data of the GHR are listed in Fig. 2.

This concept benefits from the safety and operating HTR characteristics which also results in an economic solution. Thus the GHR heat generating costs are competitive with those of conventional heat supply systems.
Primary System (Helium Circuit):

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal power</td>
<td>10 MW</td>
</tr>
<tr>
<td>Core volume</td>
<td>5 m³</td>
</tr>
<tr>
<td>Power density</td>
<td>2 MW/m³</td>
</tr>
<tr>
<td>Number of spherical elements</td>
<td>28.350</td>
</tr>
<tr>
<td>Inlet/outlet temperatures</td>
<td>250/450 °C</td>
</tr>
<tr>
<td>Mass flow</td>
<td>10 kg/s</td>
</tr>
<tr>
<td>Pressure</td>
<td>15 bar</td>
</tr>
<tr>
<td>Circulator power</td>
<td>200 kW</td>
</tr>
</tbody>
</table>

Secondary System (Intermediate Circuit):

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Temperatures</td>
<td>95 - 135 °C</td>
</tr>
<tr>
<td>Pressure</td>
<td>10 bar</td>
</tr>
<tr>
<td>Water flow rate</td>
<td>60 kg/s</td>
</tr>
</tbody>
</table>

District Heating System (Consumer Circuits):

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Temperatures</td>
<td>60 - 120 °C</td>
</tr>
<tr>
<td>Pressure</td>
<td>15 bar</td>
</tr>
<tr>
<td>Water flow rate</td>
<td>40 kg/s</td>
</tr>
</tbody>
</table>

FIG. 2. GHR — Main design data.

The nuclear heat generating system (primary system) is integrated into a prestressed concrete reactor vessel equipped with a gas-tight liner, see Fig. 3. The liner is anchored in the concrete. The primary components are the reactor core and internals, the primary coolant circulator and the liner cooling system. The primary circuit is operated with helium at a pressure of 15 bar. The hot gas and cold gas temperatures are 450 and 250 °C, respectively.

The heat is transferred from the primary system to the two intermediate circuits via the liner cooling system i.e. the primary heat exchanger. The liner cooling system is operated at 95 °C inlet and 135 °C outlet waterside temperatures with a system pressure of about 10 bar. The heat from the intermediate circuits is transferred to one or several consumer circuits (local heating systems) operating at a water supply temperature of 120 °C (at 100 % load) and 60 °C return temperature at a pressure of 16 bar (cf. Fig. 4).
The GHR is flexible with regard to these parameters and can easily be adapted to modified consumer requirements.

The helium coolant flows downward through the core (Fig. 3). After flowing through the core the hot gas is directed radially outward along the inner surface of the reactor pressure vessel and then passes upward within the annular clearance between the liner.
and the thermal shield before it enters the circulator as a cold gas. An upper gas flow guidance separates the suction and compression sides of the circulator. The cooling pipes are welded to the outer surface of the liner and are grouted in the concrete. The liner cooling system is designed to absorb and convey the total thermal power.

The core is formed by a bed of 28,350 spherical elements of 60 mm diameter each. The pebble bed is stationary, i.e., the elements are not circulated during reactor operation. The fuel loading is designed for the longest possible residence time, i.e. 14 years.

The temperature coefficient is strongly negative, which results in a stable neutron kinetic plant behaviour.

The reactor is operated in a low enriched fuel cycle (LEU). The fuel uses the proven TRISO-coated $\text{UO}_2$-particles.
The core is completely enclosed by a graphite reflector with a minimum thickness of 1.0 m. The side reflector is equipped with vertical boreholes to accommodate the absorber rods.

The radial circulator is of vertical design equipped with active magnetic bearings. It is electrically driven. The circulator unit is integrated into the top plate of the PCRV.

The prestressed concrete reactor vessel, the intermediate cooling loops including the components provided for decay heat removal as well as the safety-relevant electrical equipment and instrumentation including the control room are located below ground. They are protected by a plate which is safe against effects from all external events.

The entire part of the building arranged above ground is of conventional design.

The reactor building is equipped with a ventilation system. Using this system various levels of negative pressure are maintained in the different rooms of the plant. The waste air is monitored for activity before it is released to the environment through the ventilation stack.

**Shutdown Concept**

The GHR is provided with two shutdown systems consisting in total of 24 absorber rods accommodated in boreholes in the side reflector.

The first shutdown system consists of 6 rods. They are held by their upper ends and are automatically released on reactor scram. They drop into the reflector by gravity at a rate of about 10 cm/s. This system is capable on its own to take the reactor to
a safe condition under all events of maloperation and design basis accidents also for a prolonged period.

The second shutdown system comprises the remaining 18 control rods. They are subdivided into three groups. 6 of these rods each are additionally provided for trimming and control.

The rods of the second shutdown system are also automatically released in the event of reactor scram. In that case they drop by gravity at a rate of about 1 cm/s.

The rods of the second shutdown system are equipped with a drive system which is diverse from that of the first shutdown system.

Withdrawal of the rods of the second shutdown system is only possible if all rods of the first shutdown system are in their upper end positions.

The second shutdown system is more effective than the first shutdown system. The second shutdown system is capable to take the reactor to a safe condition under all events of maloperation and design basis accidents for a prolonged period.

Decay Heat Removal Concept (Fig. 5)

The decay heat and the stored heat can be removed by one of the following systems:

- Plant main cooling system using the local heating system as a heat sink:

When operable the main cooling system is used for operational decay heat removal and for decay heat removal in the event of accidents. No specific requirements are however made to its availability.
The decay heat is removed by natural convection via the intermediate circuits to the district heating system.

- Two-loop decay heat removal system:

In the event of failure of the plant main cooling system the decay heat is removed via two decay heat removal loops with decay heat removal coolers. Each decay heat removal loop is connected to an intermediate circuit.

The decay heat removal cooler consists of a storage tank filled with demineralized water. It is so arranged as to ensure sufficient natural convection for decay heat removal.

Each loop of the decay heat removal circuits is so designed that in the event of damage in the other loop and failure of the external electricity supply the total decay heat can be removed. This applies to the pressurized reactor as well as to depressurization of the reactor.
Cooling of the storage tank is generally effected by specifically provided air loops when external electricity supply is in operation.

In the event of failure of the air circulator the tank cooling is effected at a reduced pace by natural convection of air. In this case evaporation of the water in the decay heat removal storage tank may occur in the long run. The capacity of one storage tank is so designed that the decay heat can be removed for a period of more than 100 hours after the occurrence of an accident.

- Decay heat removal through the PCRV wall:

Even in the extremely improbable event of total failure of both intermediate cooling loops the decay heat can be removed through the wall of the prestressed concrete reactor vessel with the reactor pressurized as well as under depressurized conditions.

The relevant hot gas temperatures and helium pressures are presented in Fig. 6.

**Activity Containment**

The concept of activity containment is based on the multibarrier principle

- coated fuel particle
- graphite matrix of the fuel element
- prestressed concrete reactor vessel
- reactor building (vented containment).

The first and the second barriers are provided by the spherical graphite fuel elements containing the fuel in the form of coated TRISO particles which are embedded in a graphite matrix.
Decay heat removal by natural convection using the auxiliary systems

Decay heat removal by convection and radiation in the event of total failure of all auxiliary systems

This ensures an extremely low release of fission products under all operating conditions. This applies also to accident conditions since the relatively low temperatures of the fuel elements do not result in a considerable increase in fission product release.

The third barrier is provided by the leaktight failsafe prestressed concrete reactor vessel.
For pressure limitation and prevention of over-stress the prestressed concrete reactor vessel is provided with redundant depressurization systems (safety valves and rupture discs).

The reactor building together with the auxiliary facilities for retention and filtering of potential leakages of reactor coolant represent the fourth barrier of the activity containment.

Due to the low coolant gas activity coolant leakages resulting from accidents can be released to the environment through the ventilation channels.

**Accident Behaviour**

The GHR safety concept makes extensive use of the general HTR passive safety characteristics. The high level of passive safety in the GHR is mainly obtained by the limitation of unit power and power density to low values so that even in the event of extremely improbable accidents an impermissible release of fission products from the core must not be expected, see Fig. 7.

**The GHR Fuel Element**

**Design Values and Design Limits**

<table>
<thead>
<tr>
<th></th>
<th>Design values</th>
<th>Design limits</th>
</tr>
</thead>
<tbody>
<tr>
<td>Average power</td>
<td>0,6 kW</td>
<td></td>
</tr>
<tr>
<td>Maximum power</td>
<td>1,7 kW</td>
<td>4,5 kW</td>
</tr>
<tr>
<td>Burn-up</td>
<td>75 GWd/to</td>
<td>115 GWd/to</td>
</tr>
<tr>
<td>Central temperature</td>
<td>684 °C</td>
<td>1,250 °C</td>
</tr>
<tr>
<td></td>
<td>(&lt;800 °C)</td>
<td></td>
</tr>
</tbody>
</table>

FIG. 7. The GHR fuel element: design values and design limits.
Also in the events exceeding design base accidents the GHR offers high safety:

- Due to the inherent temperature limitation under all accident conditions important time reserves are available to take appropriate measures.

- Even in the event of severe accidents access to the reactor building is possible.

- Even in the event of total failure of the decay heat removal system the core temperatures remain relatively low limiting the release of fission products.

Even in the event of total failure of the decay heat removal systems it is possible to remove the decay heat through the vessel walls. Under such conditions the maximum core temperature of about 850 °C (locally) in a depressurized reactor remains low. Due to the low temperatures and the core structure composed of ceramic materials core damage and core meltdown can be absolutely ruled out. There is only a transient release of iodine 131 which results

1. Normal operation
   Normal annual release 0,2 mrem/a

2. Refueling 0,5 mrem

3. Depressurization accident
   - Without failure of forced circulation 3,0 mrem
   - With failure of forced circulation 6,5 mrem

FIG. 8. GHR — Radiation exposure to the environment (effective whole body dose) (at 100 m distance).
in a maximum emission of 0.03 Ci. The maximum equivalent whole body dose resulting from the release of radioactive iodine and radioactive noble gases amounts to not more than 6.5 mrem (immersion, ground radiation and inhalation), Fig. 8.

This favourable situation 6.5 results in a high flexibility with respect to site selection. GHR plants can be sited near urban centers.
HTR 100 INDUSTRIAL NUCLEAR POWER PLANT FOR GENERATION OF HEAT AND ELECTRICITY

S. BRANDES
Hochtemperatur-Reaktorbau GmbH

W. KOHL
Brown, Boveri und Cie AG
Mannheim, Federal Republic of Germany

Abstract

Nuclear power plants operated with small high-temperature reactors provide an economic, environmentally favourable and reliable electricity and heat supply for industrial plants and communities.

Based on their proven high-temperature reactor with pebble bed core, BBC/HRB have developed an HTR 100 plant which combines favourable capital costs and high availability. Due to the high HTR-specific standards this plant is especially well suited for siting near the end user.

The safety concept is designed as to permit further operation of the plant or decay heat removal via the operational heat sinks in the event of maloperation and design basis accidents having a higher probability of occurrence. Thus high loads are avoided, especially in the reactor pressure vessel. In the event of hypothetical accidents the decay heat is removed from the reactor pressure vessel by radiation, conduction and convection to a concrete cooling system operating in natural convection.

As an example of the new HTR 100 plant concept a twin-block plant design for extraction of industrial steam is presented.

The reactor building is located in the center of the overall plant. It is a safety containment
building with its inner structure detached from the outer shell for safe control of any external impacts. Its inner structure is designed identically for two reactor units. The nuclear steam supply system is included in a steel reactor pressure vessel in which all the primary components are integrated. All the safety-relevant systems are installed in the reactor building.

The building configuration of the plant is characterized by clear functional structuring, short distances for systems and personnel, economic design, short construction period and low space requirement.

1. Introduction

In the early 80ies small and medium-size reactors gained more attention. Especially for the supply of heat and electricity to industrial factories only small power plants are suitable. Since 1980 Brown Boveri & Cie and the subsidiary company Hochtemperatur-Reaktorbau GmbH have developed a small High-Temperature Reactor designed for 258 MWth thermal power and a net electrical output of 100 MW (HTR-100) on the basis of the AVR concept and making use of the advanced technology of THTR for components and systems.

This reactor type is particularly suited to supply electric power and heat for application in industry, communities and industrial threshold countries. Siting near industrial centers and near centers with high population density are permissible because of the high safety standard of the reactor with passive safety characteristics. To hold cost down special emphasis has been placed on
the greatest possible extent of prefabrication and preassembly of the plant.

Early in 1985 a Development Consortium was formed to initiate the construction of a reference plant and optimize the technical, safety engineering and economic aspects of the HTR-100 designed by BBC/HRB.

Members of this consortium are:

Hochtemperatur-Reaktorbau GmbH (HRB) Dortmund
Brown, Boveri & Cie AG (BBC) Mannheim
Deutsche Babcock Maschinenbau AG (DBM) Ratingen
Mannesmann Anlagenbau AG (MAB) Düsseldorf
Strabag Bau-AG Cologne
INNOTEC Gesellschaft für Spitzen-technologien und Technologietransfer mbH & Co. Energietechnik KG Essen

with INNOTEC KG acting as the central coordinator.

2. Nuclear Steam Supply System

The nuclear steam supply system of the HTR-100 consists of the reactor core, the steam generator, the circulators and the shutdown facilities, all integrated within a steel reactor pressure vessel (FIG. 1). The reactor vessel is located in the center of the reactor building surrounded by a cylindrical biological shield. The inner surface of the biological shield is equipped with a concrete cooling system to protect the concrete from high temperatures (FIG. 2).

In normal operating conditions the reactor core is cooled by an upward coolant flow at a pressure of 70 bar. The hot gas leaves the core at a temperature of about 740 °C, flows through the top reflector and mixes with various bypass flows which results
FIG. 1: Reactor pressure vessel with internals
in a mean steam generator inlet temperature of 700 °C. During normal operation a thermal power of 260 MJ/s is transferred and steam is generated at 530 °C and 190 bar. After transferring the heat to the steam generator the coolant gas enters the circulators at a cold gas temperature of 250 °C. The circulators force the coolant through the annular clearance between the inner wall of the pressure vessel and the steam generator and the core, respectively, into the lower part of the reactor pressure vessel where it enters the pebble bed through openings in the bottom reflector.
2.1 Reactor Vessel

The reactor vessel is the pressure containment for the overall primary coolant circuit (FIG. 1). The total height of the vessel is about 30 m. The maximum diameter is 6.1 m. The vessel is supported approximately at its centre of gravity. This permits the control of expansion and shifts resulting from high temperatures. The vessel is manufactured from creep-resistant fine-grain steel 20 MnMoNi 55 forgings in compliance with the design and inspection requirements proven in pressure vessel construction.

2.2 Reactor Core

The reactor core is a cylindrical bed of spherical fuel elements of about 3.5 m diameter and a mean height of 8 m. It is embedded into a graphite reflector comprising a bottom reflector, side reflector and top reflector. The unit is vertically mounted on a star-shaped metal support, which is itself positioned on the support ring of the reactor pressure vessel. This permits thermal expansion and prevents torsional effects. The side reflector is interlocked with the thermal side shield.

The reactor core is fueled with spherical fuel elements of the TRISO type with multiple passage of the spheres through the core using the same type of fuel elements with low enriched fuel which have been tested in the AVR in Juelich. The fuel elements are introduced continuously into the core through five refueling pipes in the top reflector. They pass downward through the reactor core before being finally withdrawn through four fuel element discharge pipes located below the core in the support structure. A burnup measuring facility equipped with a computer determines the burnup of the fuel elements.
<table>
<thead>
<tr>
<th>TABLE I: Reactor Core Main Data</th>
</tr>
</thead>
<tbody>
<tr>
<td>Diameter</td>
</tr>
<tr>
<td>Pebble bed height</td>
</tr>
<tr>
<td>Number of spherical elements</td>
</tr>
<tr>
<td>Power density</td>
</tr>
<tr>
<td>Cold gas temperature</td>
</tr>
<tr>
<td>Hot gas temperature</td>
</tr>
<tr>
<td>Helium pressure</td>
</tr>
<tr>
<td>Helium mass flow</td>
</tr>
<tr>
<td>Initial core</td>
</tr>
<tr>
<td>U235 initial enrichment</td>
</tr>
<tr>
<td>Heavy metal content per fuel element</td>
</tr>
<tr>
<td>Graphite elements</td>
</tr>
<tr>
<td>Equilibrium cycle</td>
</tr>
<tr>
<td>U235 initial enrichment</td>
</tr>
<tr>
<td>Heavy metal content per fuel element</td>
</tr>
<tr>
<td>Graphite elements</td>
</tr>
<tr>
<td>Residence time of fuel elements (full power)</td>
</tr>
<tr>
<td>Mean burnup</td>
</tr>
<tr>
<td>Heavy metal content of the core</td>
</tr>
</tbody>
</table>

Fuel element spheres with a high burnup are discharged from the cycle and are replaced by fresh fuel elements, while fuel elements which are not yet completely spent are pneumatically recirculated into the reactor core.

The reactor core (TABLE I) having a mean core power density of 4.2 MW/m³ consists of 317500 spherical elements (60 mm in diameter). In the equilibrium cycle only 55% of the spherical elements are fuel elements, the remainder are graphite elements.

At a load factor of 0.9 the mean residence time of the fuel elements in the equilibrium cycle is three years. This amounts to five passages through
the core and a mean burnup of appr. 100 GWd/t of heavy metal. A two-zone core is established by using different burnup ages in both zones. The initial enrichment is 9 % U235.

The initial core consists of a mixture of graphite spheres, absorber elements and fuel elements with 6 % initial enrichment. The initial core fuel elements are replaced by equilibrium fuel elements during the first years of operation.

The reactor core is controlled by two shutdown systems. 24 absorber rods inserted in holes of the side reflector are used for control and reactor scram. For longterm shutdown small absorber spheres (app. 9.5 mm) are filled into vertical holes of the four graphite buttresses.

8 absorber rods are used for load control. They compensate the necessary excess reactivity to allow load-following between 50 and 100 % load. FIG. 3 shows lateral power distributions at full power in

FIG. 3: Lateral power distribution in the reactor core
the upper and lower part of the core (equilibrium cycle). The control rods are only inserted in the upper part of the core to compensate the excess reactivity. The figure shows also the boundary between the inner and the outer core.

FIG. 4 shows the radial gas outlet temperature distribution. The maximum gas outlet temperature exceeds the average value of 740 °C only by 25 K.

![Core gas outlet temperature distribution](image)

**FIG. 4:** Core gas outlet temperature distribution

2.3 Shutdown System

For tripping the reactor 16 absorber rods of THTR design are automatically inserted into bores in the side reflector. The reactivity value of these rods is at least 3.4 % $\Delta k$ for compensating reactivity effects of unlimited water ingress. Under normal conditions the core temperature after reactor trip is reduced only by approximately 100 K to ensure a quick restart.

The small absorber spheres, providing the second shutdown system (FIG. 5), are manually released from containers into the channels in the buttresses.
The small absorber sphere shutdown system is used, when shutdown periods are longer than one day and for long-term shutdown at ambient temperature.

The containers are installed outside the reactor pressure vessel. The release mechanism is activated releasing the spheres into the channels in the graphite buttresses by gravity.

For re-starting the reactor, the boron spheres are recirculated to the containers under gas pressure via exit valves below the core, feed chambers and return piping.

The reactivity value of the small absorber sphere system is greater than 15 % $\Delta K$, which compensates the maximum reactivity demand occurring in the initial core (13 % $\Delta K$).
2.4 Internals

The cavity holding the spherical fuel elements is formed by the ceramic internals, the bottom reflector, the side reflector and the top reflector (FIG. 6).

FIG. 6: Core structure
Mechanical loads from dead weight and spheres pressure as well as seismic loads are compensated by the graphite components and transferred to the adjacent metal component parts.

The rigidity requirements of the graphite structure are not very strict, since jamming of fuel elements cannot occur in the pebble bed core and flowing of the spheres is ensured even in case of irregularities in the reflector structure. The side reflector is formed by an inner and an outer cylindrical wall consisting of reflector blocks piled up to form columns. The blocks are vertically fixed by dowels, horizontally they are connected by keys. The keys simultaneously serve as seals, limiting the bypass flow through the vertical gaps between the columns to the permissible amount.

The four graphite buttresses protruding into the reactor core are connected with the side reflector.

The top reflector consists of layers of cantilevered segment-shaped graphite blocks and a central plug. The cantilevers are supported by the inner layer of the side reflector.

For the gas flow and for the addition of fuel elements and the small absorber spheres suitable holes are provided in the segment-shaped blocks.

2.5 Steam Generator

The hot gas leaving the core flows upwards in the steam generator transferring its heat to the feedwater (FIG. 7).

On the water side, the steam generator consists of two tube bundles of about 160 heating tubes each,
FIG. 7: Steam generator

Manufactured from Incoloy 800, which are wound into each other in helicoils. On the coolant gas side, the two bundles form one unit. In transient conditions as well as during startup and shutdown procedures and in feeding one tube bundle only, the steam generator tubes are never exposed to impermissible load.

The feedwater inlets and main steam outlets are located laterally at the upper cold end of the steam generator. The pipes penetrate the reactor pressure vessel horizontally immediately above the steam generator.
2.6 Circulators

After transferring the heat to the steam generator, the helium coolant enters the two circulators at a cold gas temperature of about 250 °C (FIG. 8).

![Helium circulator diagram](image)

FIG. 8: Helium circulator

The coolant gas circulators are single stage radial blowers driven by a speed-controlled asynchronous motor. The circulator shaft is held in active magnetic bearings so that there is no mechanical contact between rotating and static parts. The bearings are free of wear requiring no lubricants.

2.7 Fuel Circulating System

The fuel circulating system permits continuous loading and withdrawal of spheres during reactor operation. The greater part of the fuel circulating system is located below the reactor vessel. The flow diagram (FIG. 9) shows the arrangement. The
spherical elements - fuel, graphite and absorber elements - are delivered to the power station in transport casks. The casks are transported from the fresh fuel storage to the fueling station. From here they are transferred to the distributing station and then forwarded pneumatically to the core via 5 pipes allowing to build up a two-zone core.

After passing through the core by gravity, the elements leave the core at the bottom and are directed to the singulizer which is coupled with a separator for damaged spheres. Damaged spheres are eliminated. Intact spheres are led to a burnup measurement device. Spent fuel elements are discharged. Fuel elements which are not fully spent, are recirculated into the core.

2.8 Helium Purification System

The helium purification system removes chemical impurities such as \( \text{H}_2, \text{CO}, \text{H}_2\text{O}, \text{CO}_2 \) using molecular sieves to minimize corrosion of the core graphite.
Graphite dust is also removed. However, neither noble gases nor nitrogen are removed. The molecular sieves are regenerated with helium.

3. Safety Concepts

3.1 Shutdown

Reactor scram is automatically achieved by absorber rods which are inserted into holes in the side reflector by gravity. The reactor remains in hot subcritical condition for at least 20 h.

For longterm cold shutdown the small absorber spheres are released into the channels in the reflector buttresses by gravity.

3.2 Decay Heat Removal

After reactor scram the decay heat generated in the core has to be removed. For this purpose two independent and diverse systems are provided in the HTR-100.

During shutdown of the power plant from normal operation the power of the nuclear steam supply system and the steam-feedwater circuit is slowly reduced from 100 % to about 10 %. Then the operation is switched to one of the two loops of the operational startup and cooldown system (FIG. 10) by opening its shutoff valves and closing the main steam shutoff valve. The wet steam is condensed in the steam condenser of the operational startup and cooldown system and cooled by the auxiliary cooling water system in section-type coolers. The condensate is recirculated to the steam generator through the feedwater line.
FIG. 10: HTR 100 industrial nuclear power plant heat flow diagram

Under upset conditions, and under most accident conditions resulting in reactor scram, the heat is removed by the auxiliary feed-water system.

The auxiliary system including the cooling chain, has builtin redundancy. The steam generator is designed with two loops and each reactor is equipped with two circulators. An additional auxiliary power supply is provided by two diesels. The measures result in a high availability of the auxiliary feedwater system. This system alone can be used to remove decay heat in the event of upset conditions and during most accidents resulting in reactor scram without undue load to the components.

Accidents such as turbine scram, or pipe rupture are completely controlled by the auxiliary feedwater system. In the event of failure of one circulator,
plant operation is continued at reduced power until a scheduled shutdown. Even in the event of failure of both circulators, heat transfer from the reactor to the steam generator by natural convection within the primary circuits is sufficient for heat removal using the auxiliary feedwater system.

Depressurization is controlled by one circulator. In case of failure of one steam generator loop, decay heat is removed by the second loop.

The high heat capacity and the low power density of the HTR-100 permit a limited interruption of decay heat removal offering time for manual action.

In the rare event of an accident and simultaneous failure of the auxiliary feedwater system the heat is removed through the concrete cooling system (FIG. 1). This system is independent of the auxiliary feedwater system. The heat is transported from the reactor core by heat conduction, radiation, and convection through the outer wall of the reactor pressure vessel to the surface coolers of the concrete cooling system. The cooling water in the concrete cooling system is heated, flows upward by natural convection and discharges its heat to the atmosphere by evaporation. Cold water flows into the surface coolers from the reservoirs. The capacity of the reservoirs is sufficient for up to 2 days heat removal.

3.3 Activity Release

During normal operation of the reactor the radiation exposure results in a small environmental dose near 0.01 mSv/y.

In the hypothetical event of a depressurization accident with longterm interruption of decay heat removal by the operational startup and cooldown system, the fuel element temperatures in a small
core region rise to a maximum of 1680 °C after about 60 hours for a short period. In this case the heat is solely transported by heat conduction and radiation to the concrete cooling system outside the reactor pressure vessel. If the concrete cooling system is not available, the fuel temperatures will nevertheless remain below 1680 °C. In this accident the radiation exposure at the most unfavourable spot results in

<table>
<thead>
<tr>
<th>Component</th>
<th>Dose</th>
<th>(% of Legal)</th>
</tr>
</thead>
<tbody>
<tr>
<td>whole body</td>
<td>0.16 mSv</td>
<td>0.3%</td>
</tr>
<tr>
<td>thyroid (infant)</td>
<td>3.4 mSv</td>
<td>2.3%</td>
</tr>
</tbody>
</table>

Figures in brackets give percentages of the values legally permitted in the Federal Republic of Germany under all accident conditions.

4. Balance of Plant Description

The forementioned NSSS can be used for the generation of heat, electric power, and steam up to 500 °C for application in industry and communities. A single or twin-block station is offered. As an example a twin-plant version for cogeneration of process steam and electricity is described below.

4.1 Nuclear Island Arrangement

The general arrangement of the buildings for the HTR-100 power plant is shown in FIG. 2.

The buildings are arranged in a compact configuration with the reactor building in the center. On one side of the reactor are the buildings containing control areas such as the reactor auxiliary building and the spent fuel storage building. On the other side there is the conventional plant, the connection building, the turbine hall and the building housing
the industrial steam supply plant. The electrical equipment building and the access and security building are arranged perpendicular to the main axis of the power plant buildings. The overall length of the building is 148 m, the total width is 82 m, and the reactor building has a height of 44 m.

4.2 Steam Power Plant

FIG. 10 shows the major components of the main steam turbine generators and condensate/feedwater systems for the twin plant.

The steam supply system is designed according to the principles of cogeneration in order to produce industrial steam at 270 °C, 16 bar and at a mass flow of 170 - 500 t/h together with about 100 MW - 175 MW gross electric output.

The main steam from the two generators flows to a common header from where a high-pressure (hp) backpressure turbine is supplied at a temperature of 530 °C and a pressure of 190 bar. The exhaust steam passes to a mp/lp extraction condensing turbine. The overall gross electric output of the two turbosets is 127 MW at 400 t/h industrial steam extraction.

The exhaust steam from the mp/lp turbine is condensed in an air-cooled condenser. After passing through a three-stage regenerative feed water header, the condensate is divided between two loops and fed into one of two deaerator-heating stages. The feedwater is then pumped into the two steam generators through the hp preheaters.

To achieve high availability, the industrial steam supply plant is also designed as a two-loop system. The condensate flowing back from the indus-
trial steam network is preheated in three stages and converted into steam at 270 °C in a steam converting and superheating station.

The heating steam for the superheater and converter is extracted from bleeds in the turbines. The collected condensate cascades from one stage to the next, transfers its residual heat, and is finally recirculated into the steam/feedwater circuit. The industrial steam circuit is separated from the steam feedwater circuit by the heat exchangers so that no activity from the steam/feedwater circuit will reach the industrial steam circuit if a leakage occurs.

This two-loop design system permits continuous operation of the plant even during inspection of individual parts of the plant. Thus, for example, one reactor and its can be shut down for inspection purposes, while the second reactor continues to operate.

In the event of failure of the hp turbine, the mp/lp turboset is supplied directly with main steam through a reducing station. In case of failure of the mp/lp turbine, the steam expands in the hp turbine to the bleed pressure for the industrial steam supply plant. In both cases a power output of about 90 MW is obtained, more than half the total output of both turbines. This shaft asymmetric turbine connection has a high efficiency during both normal operation and under upset conditions. Compared to other multi-shaft turbine connections, it requires less space in the turbine building and less instrumentation and control equipment.

The heat flow for the HTR-100 power plant is designed to achieve:

- High availability of industrial steam and electric power
A range of capacity for industrial steam of 170-500 t/h;

- Gross electrical output of 100 MW-175 MW;

- Constant supply of at least 170 t/h of industrial steam and 80 MW of electrical power;

- Up to 90 percent net efficiency for heat and power.
HTR 500 — THE BASIC DESIGN FOR COMMERCIAL HTR POWER STATIONS

E. BAUST, J. SCHÖNING
Hochtemperatur-Reaktorbau GmbH,
Mannheim, Federal Republic of Germany

Abstract

With the completion of the THTR 300 and the development of the THTR follow-on plant HTR 500, BBC/HRB have taken the pebble bed high-temperature reactor to the threshold of commercialization.

The THTR 300 is the reference plant for the entire high-temperature reactor line in the power range between 100 and 600 MW. In addition, the experience is available which has been acquired with the successful 19 years operation of the 15 MW AVR in Juelich, also constructed by BBC/HRB.

As a next step of market penetration in the Federal Republic of Germany the construction of the HTR 500 is intended, a power station for electricity generation and the possibility to extract process heat and district heat. The power size of 550 MW takes account of the modified demand on the nuclear power plant market which for reasons of grid size, financing expenses and the lower increase in electricity demand internationally shows a trend towards smaller power units in the range between 400 – 600 MW.

Due to the use of HTR-specific safety characteristics and extensive application of THTR 300 technology, the 550 MW power station is economically competitive with large PWRs.

For special applications in industry (cogeneration of steam and electric power) and for special siting conditions (near industrial centers) the small 100 MW HTR has been developed which can also be constructed as a 200 MW twin-plant under economic conditions, competitive with conventional power plants of the same size without involving the problem of air pollution.

As a further HTR application, small heating reactors of about 10 MW can be designed on the basis of the pebble bed reactor concept.

After the Chernobyl reactor disaster the high inherent safety of the HTR has been under discussion with increased emphasis. In the present situation characterized by sensitive attitude toward nuclear energy, the HTR with its high safety potential offers favourable conditions regarding its acceptance by the public and the politicians.
The HTR is a decisive innovation in the field of reactor technology which represents an important contribution to the intermediate and long-term supply of safe, environmental and economic energy to the entire electricity and heat market.

The HTR is particularly suited for Third World countries turning to the peaceful usages of nuclear energy for the first time.

1. PRESENT SITUATION

On November 16, 1985 the THTR 300 prototype nuclear power plant in Hamm-Uentrop supplied first electricity to the public grid, on September 23, 1986 full power was reached. With this the first large high-temperature reactor with spherical fuel elements started electric power generation. The nuclear power plant was constructed by Brown Boveri & Cie AG (Mannheim) in a consortium with the subsidiary company Hochtemperatur-Reaktorbau GmbH (Dortmund) and the fuel element manufacturer Nukem GmbH (Hanau) on the plant site of the Westfalen power plant of VEW Dortmund. The plant owner is Hochtemperatur-Kernkraftwerk GmbH (HKG), an association of regional and local public utilities.

The first step of the German HTR development was the construction of a 15 MW electric power experimental reactor in Juelich in the Federal State of Northrhine Westphalia. This reactor, called AVR, was taken into operation by the users group "Arbeitsgemeinschaft Versuchsreaktor" (abbreviated AVR) in 1967 and has demonstrated an extremely satisfactory operating behaviour for 19 years (Fig. 1). Long years of operation of the experimental reactor at a maximum coolant gas temperature of 950 °C have verified the suitability of the HTR for generation of nuclear process heat.

The THTR 300 as the second step of the HTR line represents the essential basis of commercialization of the High-Temperature Reactor in the Federal Republic of Germany (Fig. 2). It is the reference plant for the entire high-temperature reactor line with all follow-on plants in the power range between 100 and 600 MW el. As the prototype includes large design reserves, higher power levels up to about 600 MW can be achieved with almost identical overall dimensions and structural volumes. In a twin plant design, a power range up to 1200 MW can be covered.

With the THTR 300 the nuclear licensing procedure for high-temperature reactors has been introduced in the Federal Republic of Germany establishing the re-
FIG. 1. 15 MW AVR Nuclear Power Plant Jülich.

FIG. 2. 300 MWe Nuclear Power Plant Hamm-Uentrop.
quired know-how with experts and authorities. In addition, a competent manufacturer and supplier industry has been built up. Finally, also the competence on the user side has been substantially increased.

The commissioning results of the THTR 300 have fully confirmed the design of the large pebble bed high-temperature reactor and have verified the mechanics of the pebble bed core on original scale.

Thus BBC/HRB have the know-how required for designing, licensing, constructing high-temperature reactors. For the construction of the first-of-its kind plants HTR 500 and HTR 100 the only requirement remaining is a research and development R&D program to confirm the design. These R&D activities conducted in parallel have been performed for the LWR to a much larger extent, since they have been continued and substantially sponsored from public funds even after the construction of several LWR plants.

Thus the HTR is at the threshold of commercialization and is ready to be practically used by public utilities and other potential customers on the thermal energy market.

2. HTR LINE WITH ITS MAIN CORNERSTONES HTR 100 AND HTR 500 (FIG. 3)

Based on the constructed AVR and THTR 300 plants, BBC/HRB developed HTR plant concepts in the power levels between 100 (HTR 100) and 550 MW (HTR 500). These plants can be used for

- electric power generation (also in arid areas by using dry cooling towers)
- combined generation of electricity and process steam up to 530 °C by co-generation of heat and power
- combined generation of electricity and district heat.

For an economic utilization of domestic coal the HTR represents an environmentally favorable and resource-saving solution, when it operates combined with conventional gasification procedures and especially in direct application of the nuclear process heat at high gas temperatures of 950 °C. The application of the HTR development which is being pursued within the nuclear process heat project (PNP) under substantial sponsorship from the Federal Ministry for Research and Technology and the Northrhine Westphalia Ministry for Economics and Technology.
FIG. 3. Power Spectrum of the HTR line.

The generation of SNG (substitute natural gas) offers the possibility to make use of the available domestic natural gas piping systems and markets for SNG generated from German lignite and hard coal. At the same time, imports can be reduced by the SNG quantity produced, which results in a noticeable relief of the balance of payments. Another possibility is the production of liquid hydrocarbons which are capable to replace mineral oil in case of necessity.

Also for this long-term development objective, the direct application of the high gas temperatures up to 950 °C e.g. for coal gasification or for the application of the gas turbine, the main components of the nuclear heat generating system of the HTR 500 and the HTR 100 can be essentially adopted. However, further development is required for the heat-exchanging components, hot gas valving etc. until a commercial application will be possible. This development will be successfully completed by the end of this century provided that the required R&D support will be provided from public funds also in future.

Based on the present energy situation, application of the HTR for coal gasification is to be expected not before the beginning of the next century.
3. THE HTR 500 POWER PLANT

3.1 The Technical Concept

In Fig. 4 the general arrangement of the power plant is compared with the THTR 300 on identical scale. It is optically quite evident that the plant dimensions are practically the same for a power output which has almost been doubled.

In the center of the HTR 500 power plant complex the reactor containment building is situated housing the prestressed concrete reactor vessel including the primary system, the shutdown facilities, parts of the decay heat removal system and other safety-relevant components. In addition the reactor containment building acts as a protection against external impacts. All the component parts carrying activity are arranged inside the reactor containment building except the fuel element storage, which is, however, also equipped with a protection against external impact. The reactor service building on the one side of the reactor containment building and the turbine building on the other side form a line of buildings to which the electrical equipment building is arranged in an opposite position.
Fig. 5 shows a vertical section of the prestressed concrete reactor vessel with its internals. The fuel elements contain low enriched uranium and pass through the reactor core only once (OTTO fueling). The spent fuel elements are discharged from the core through three fuel element discharge pipes without interruption of power operation. The spent fuel elements are directly discharged into casks and transported to the storage. No special conditioning for spent fuel is necessary for longterm interim storage and for final storage.

![Diagram of HTR 500 Reactor Pressure Vessel with Internals](image)

FIG. 5. HTR 500 — Reactor pressure vessel with internals.

The waste management concept is adapted to the special characteristics of the HTR line. A 10 years interim storage of the spent fuel elements in fuel storage casks is intended followed by transport to a nuclear waste repository for ultimate disposal. Due to the limited quantity of spent fuel elements, closing of the HTR fuel cycle is no suitable solution, so that — as for other prototype nuclear power stations — direct ultimate disposal is envisaged in the long run.

This concept corresponds to the AVR and THTR 300 waste management concept, as licensed by the authorities.
The reactor pressure vessel is designed as a large cavity prestressed concrete reactor vessel (PCRV) as in the THTR 300. The complete primary system is integrated in the PCRV.

The HTR 500 is equipped with 6 steam generators of helicoil design operating in countercurrent. Each steam generator is associated with a circulator. The circulator design is the same as that of the THTR circulator, it is however, equipped with active magnetic bearings and arranged in vertical position.

The helium coolant flows downward through the reactor core, then upward through the steam generators to the circulators from where it is recirculated to the cold gas plenum, cooling on its way the outer surfaces of the primary system.

Because of the availability of the separate two-loop decay heat removal system the main cooling system consisting of 6 steam generator/circulator units has no safety function. Therefore the main cooling system is of a purely conventional design outside the reactor containment building, i.e. it is designed to the usual standards applied to modern conventional power plants.

![Flow diagram of HTR 500](image)

**FIG. 6. HTR 500 — Flow diagram.**

Components supplied by emergency power:

1. Reactor core
2. Steam/feedwater circuit
3. Main heat sink
4. Decay heat removal system
5. Liner cooling system
6. Cooling water system
7. PCRV safety valve
8. Reactor containment pressure relief system
9. Exhaust air system
10. Ventilation stack
3.2 Safety Concept for Accident Control (Fig. 6)

The accident control concept of the HTR 500 is based on the following principles:

- Multi-barrier principle for activity enclosure
- Safety systems for reactor shutdown and decay heat removal
- Use of the HTR-inherent safety characteristics for taking manual measures in the event of failure of active safety systems.

Multi-BARRIER Principle

The following barriers are provided against activity release:

- Coated particles (TRISO)
- Fuel elements
- Prestressed concrete reactor vessel with leak-tight liner
- Reactor containment building

The design of the fuel element and especially the build-up of the coated particles ensures a low release of activity during normal operation and as a result of accidents. Even in the event of hypothetical accidents an increased release of activity from the fuel elements can only occur if the operating temperatures are substantially exceeded. Such high temperatures would only be reached as a result of failure of all decay heat removal systems and only several hours after the occurrence of the accident. Therefore sufficient time is available to take manual measures to interrupt the accident sequence.

The reactor pressure vessel is a burst-safe prestressed concrete reactor vessel equipped with an inner liner, a liner cooling system and a thermal insulation. Large-scale failure of the primary system containment can be ruled out.

The greater part of the vessel closures is arranged in the top head of the vessel. Penetrations for the helium purification system and the fuel circulating system pass through the bottom plate of the prestressed concrete reactor vessel. Vessel closures and penetrations are so designed that large leakages from the primary system can be safely ruled out, e.g. by installation of redundant structural elements. The leakages to be assumed are improbable and limited to a maximum of 33 cm².
The reactor containment building is designed to withstand earthquakes, external shock wave and airplane crash.

Primary gas leakages due to operation or accidents up to leak sizes of approximately 2 cm$^2$ are vented to the environment under controlled conditions through filter systems and the stack.

The maximum leak cross section of the PCRV has been limited to 33 cm$^2$ by structural measures as in the THTR. As a result of this design, there can only be a slow leakage of the coolant; the time for pressure equalization is approximately 90 minutes. In this case, the coolant gas is passed straight into the stack until the depressurization system of the reactor containment building closes again.

As a result of such low coolant gas activity, the potential radiation exposure of the environment of the nuclear power plant originating from the radioactive substances released with the vent air is less than 1 mrem/a wholebody dose. The environmental load in the event of accidents is about 10 mrem wholebody dose.

### 3.3 Safety Systems

Reactor scram following accidents is automatically effected by the reflector rods which drop into bores in the side reflector by gravity. In the relatively rare events of long-term shutdown only the incore rods are inserted directly into the pebble bed core. Long-term shutdown is exclusively actuated by manual release being generally required only after 24 hours in order to reach the cold subcritical state.

The decay heat removal concept (Fig. 7) is characterized by

- the use of the main heat removal system
- a separate and redundant decay heat removal system (two loops) with particular circulators and heat exchangers in the primary system
- utilization of natural convection for decay heat removal in the primary system
- possibility of a prolonged failure of the decay heat removal system and subsequent resumption of operation following manual measures
- integration of the liner cooling of the prestressed concrete reactor vessel into the decay heat removal concept without activity release in the event of total failure of the decay heat removal systems listed above
FIG. 7. HTR 500 — Multiple measures for accident control.

- in the hypothetical case of total failure of all decay heat removal possibilities \((< 10^{-6}/a)\) only limited radioactivity release to the environment.

3.4 Accident Behaviour of the HTR 500

Accident analyses performed for the THTR 300 as well as risk analyses have shown that the design criteria and the effectiveness of the safety systems are mainly determined by the following accident categories:

- Failure of individual decay heat removal systems or loops, i.e., reduced decay heat removal capacity.
- Reactivity accidents, i.e., higher core power to be removed.
- Depressurization accidents, i.e., reduced primary circuit heat removal capability.
- Tube rupture in the steam generator, i.e., penetration of water into the primary circuit, followed by a rise in primary gas pressure an increased corrosion of the graphite internals.
Moreover, the control of failures of total systems was investigated:

- failure of the scram system
- failure of the decay heat removal system.

As a summary result of these analyses Fig. 8 shows the time curves of the maximum fuel element temperatures for the accidents listed above compared to the load and failure limits of the fuel.

In the event of design basis accidents, i.e. accidents whose controllability has to be verified in the licensing procedure, forced-circulation cooling of the core is maintained. In such events the fuel element temperatures are reduced to temperatures below 500 °C within a few hours, so that even most improbable accident-failure combinations involving water or air ingress can be controlled without any problems. (Below some 500 °C the reaction rates of steam or oxygen with the graphite of the fuel elements are negligibly small).

In addition, Fig. 8 shows that even in the event of improbable accident sequences such as total failure of the decay heat removal systems or failure of the scram system (ATWS accidents) the fuel element temperature will rise for a short period, but will remain clearly below the design temperature permitted for con-
tinuous operation. This ensures that an increased release of activity from the fuel elements will not occur.

### 3.5 Use of the HTR-Inherent Safety Characteristics for Minimizing the Residual Risk

The inherent safety characteristics of the HTR, i.e. the low power density, the high failure temperature of the coated particles and the integration into a prestressed concrete reactor pressure vessel are used in the design.

The prestressed concrete reactor vessel maintains its geometrical form and acts as an important barrier against fission product release, since part of the fission products are deposited and thus retained within the vessel. The permissible failure period of the decay heat removal systems is 10 h.

Accessibility of the external components of the decay heat removal system is provided so as to ensure that the required measures can be performed with a sufficient reliability within a period of 10 hours.

It has been demonstrated in a great number of analyses on the accident behaviour of HTR plants that there is a possibility of an even further reduction of the anyway low accident risk of the HTR, if the liner cooling system of the prestressed concrete reactor vessel, operating in normal conditions, is integrated in the decay heat removal concept. It is a very simple and therefore very reliable system for decay heat removal which can rapidly be taken back into operation by simple measures such as mobile aggregates.

Therefore it has been designed so as to ensure that even in the event of a long-term total failure of the decay heat removal loops with the reactor pressurized no activity is released. The pressure rise in the primary system resulting from the temperature increase is limited by overflow of helium coolant into the storage tanks of the helium purification system. Core temperatures remain below 1200 °C.

In the extreme event of additional failure of the liner cooling system the inert behaviour of the system provides sufficient time to take measures for restarting the liner cooling system by emergency feed.

Analyses performed by the Institute for Nuclear Safety Research in the Nuclear Research Center Juelich have shown that even in such hypothetical events there will be no immediate effects and the collective risk of late radiation damage is very low.
3.6 Cost and Economics

An HTR with a power level of 550 MW has the advantage of a better correspondance with the increase in power demand and thus lower investments to be made in advance as compared to the large LWRs.

Additional advantages result with regard to power reserve and mains loading. These advantages may be decisive in case of otherwise almost identical electricity generating costs.

The consequent utilization of HTR-specific characteristics as well as an optimization of the design of components and circuits leads to the result that the electricity generating costs of an HTR 550 are competitive to those of the large LWRs.

3.7 State of the HTR 500 Project

In 1983 the users group Arbeitsgemeinschaft Hochtemperaturreaktor (AHR), representing 16 power industry companies, placed an order with BBC/HRB to perform conceptual design studies on the HTR 500 on a private business basis. These studies have been successfully completed.

AHR has verified the results of the HTR 500 conceptual design project. In this evaluation the positive statements on
- licensability
- technical feasibility
- economics and competitiveness with the large 1300 MW convoy pressurized water reactor

have been confirmed also from the users side. This evaluation was documented by AHR in a summary report in July 1984 and distributed to the utility companies.

In addition, the technical and safety concept of the HTR-500 has been confirmed by an experts committee invoked by the Federal Ministry of the Interior, which was composed of members of the Reactor Safety Commission, the technical supervisory association (TÜV) and the licensing authorities.

Based on these positive results, the German companies operating nuclear power plants took a fundamental decision to continue the HTR 500 project design. According to the state of the negotiations with the users group Hochtemperaturreaktor Gesellschaft (HRG) it can be expected that BBC/HRB will be awarded the two years' contract for the design of the HTR 500 in the near future. This period includes the nuclear licensing procedure for the concept of the HTR 500.
Immediately afterwards, Part B of the design contract is to be placed which covers the complete nuclear licensing procedure so as to rule out delays during construction resulting from the procedure.

Because of the possibility of future application of the HTR for process steam and process heat, the Union of Chemical Industry and Ruhrkohle AG will participate in the project.

4. STRATEGY OF HTR COMMERCIALIZATION

Based on the mentioned facts the following strategy of the commercialization of the HTR is envisaged:

- Design and construction of the first HTR 500 plant, i.e. a plant for electricity generation with the possibility of extracting process steam and district heat in the Federal Republic of Germany. The power output of about 550 MW meets the modified demand of the power plant market, which for reasons of grid size, scope of financing and the lower increase in electricity demand internationally shows a trend towards power units in the range of 300 - 600 MW.

The possible start of construction of the HTR 500 will be 1991. Design and erection of the HTR 500 are financed on a commercial basis. The design work is supported by a R+D program sponsored from public funds.

The realization of the first HTR 500 will render the HTR 500 competitive as an electricity and process steam generating plant. Further contracts can be accepted on a worldwide scale from the early 90ies onward.

For the HTR 500 a great part of the plant components can be fabricated in the Third World countries by their domestic industry, including the erection of a prestressed concrete reactor vessel. Joint ventures have recently become common practice. Potential customers are Third World countries having a corresponding electricity demand, such as China, Taiwan, Indonesia, Turkey, and others.

- The small HTR designed to a power output of 100 MW is envisaged for special application in industry (steam and electricity as combined products). The HTR 100 MW can also be constructed as a 200 MW twin plant which is economically competitive with fossil-fueled power plants and hence competitive with conventional plants of identical size without involving the problem of air pollution.
For this purpose an industry consortium HTR 100 was founded by the companies BBC, HRB, Deutsche Babcock, Mannesmann Anlagenbau, Strabag and Innotech. It is the objective of this consortium to establish a detailed design of a competitive plant by simplification, standardization, series production, shop fabrication and development of a transport and assembly system for prefabricated plant components. Also for this project a supporting R&D program is conducted with the sponsorship from public funds.

Currently a users consortium for the HTR 100 is being established with the objective of awarding a contract for the design of the first-of-its-kind plant in the Federal Republic of Germany; construction of the plant will be possible in the early 90s.

For Third World countries having a lower electricity demand and few fossil resources, the HTR 100 represents an excellent possibility for generating electric power and heat. Extensive participation of the domestic industry is ensured. In our opinion there is a great number of countries that will be potential users of the HTR 100 from the early 90s onwards.

- The small 10 MW heating reactor offers an excellent opportunity for providing densely populated areas with space heat which eliminates their problems of air pollution. Swiss industry envisages the construction of a prototype plant in Switzerland in the next years. A successful use of these plants is possible in many countries of the world.

- From the mid-nineties onward also the construction of plants for the generation of nuclear process heat up to 950 °C will be possible. This date will, however, finally be determined by the market. Based on the present energy situation, this application is to be expected not before the beginning of the next century.
SAFETY ASPECTS
(Session D)
SAFEY CHARACTERISTICS OF CURRENT HTR

W. KROGER
Institut für Nucleare Sicherheitsforschung,
Kernforschungsanlage Jülich GmbH,
Jülich, Federal Republic of Germany

Abstract

The paper contains a description of HTR-safety features and results of safety calculations for the HTR-500, a 550 MW(e) pebble bed HTGR. A set of advanced safety criteria is proposed and discussed.

The objectives of statutory regulations in W. Germany are

. to avoid accidents by means of high technical and human quality
. to control accidents by means of reliable engineered safety features and
. to mitigate the consequences of severe accidents by technical and administrative measures, e.g. emergency planning.

Limits for radiation exposure during normal operation and incidents as well as for design-relevant accidents are set up by the Radiation Protection Ordinance (30 mrem/a and 5 rem whole body dose of reference person at most adverse location outside the plant). For severe accidents radiological limits are not established; however, there are reference values for protective actions planned in advance and remote siting criteria mainly to facilitate evacuation.

On the basis of these regulations a set of design relevant accidents and safety requirements have been developed for light water reactors in a far reaching deterministic manner. The result is

. a high safety standard assured by a highly active complex safety system with three or four redundant
trains and no need for operator actions within the first 30 minutes,

- a small possibility for (beyond design) severe accidents but for which catastrophic activity releases and consequences cannot be excluded, and

- a small risk considered justifiable by the Supreme Court, but hardly accepted in its structure by the public.

For non-standard reactor systems, like HTR, no firmly established design rules and standards exist. Literally fulfillment of LWR requirements is possible, as shown in the licensing procedure of the THTR-300, but it leads to an in part overloaded safety system if the passive safety features are different or more favourable, respectively.

Therefore effort has been made in recent HTR concepts to develop a more specific safety concept based on the general sense of legal requirements and, according to their nature, making use of Probabilistic Safety Assessment. This concept (see Table I) has been applied to the HTR-500 and HTR-Modul under development and judged positively by a group of experts established by the upper licensing authority /1/.

**TABLE I**

Proposed Concept for Frequency-Oriented Safety Requirements for Advanced Reactors

<table>
<thead>
<tr>
<th>Event Classification, Explanation of Frequencies</th>
<th>Radiological Limits (Whole-Body/Thyroid Dose Risk)</th>
</tr>
</thead>
</table>
| Accidents relevant for design
  >3 x 10^{-2}/yr
  3 x 10^{-2}...10^{-4}/yr
  10^{-4}...10^{-1}/yr
| Event not to be excluded during lifetime of the plant
  Event not to be excluded during lifetime of several plants
  Event not to be expected during lifetime of several plants |
| 30/90 mrem/yr (§45 Strl SchV)* |
| Event not to be excluded during lifetime of the plant
  Event not to be expected during lifetime of several plants
| Risk distinctively below a PWR of comparable size |
| Severe accidents |
| <10^{-1}...10^{-3}/yr |
| Event can be excluded practically (out-of-design range; mitigating properties + protective actions) |
| <5/15 rem/event |
| 5/15 rem/event (§28.3 Strl SchV) |

*Strl SchV: Strahlenschutzverordnung = radiation protection ordinance.
Very promising passive safety features of HTR (see Table II) on one hand and a need for urban siting on the other encouraged us to think informally about more restrictive safety requirements. In the light of Chernobyl-desaster they may become more important as a general proposal for advanced safety criteria (ASC) and may help to overcome public resistance against nuclear power. In any way, they should be used as starting point for a discussion on an international level aiming at common or comparable safety criteria for super-safe advanced reactors.

**TABLE II**

<table>
<thead>
<tr>
<th>Important Safety Characteristics of HTRs Relative to Standard LWRs (Values for HTR-500)</th>
</tr>
</thead>
<tbody>
<tr>
<td>• Larger heat capacity (~10^3 tons of graphite), much lower power density (6 MW/m^3)</td>
</tr>
<tr>
<td>→ Slow reaction on loss of cooling (500°C heatup after 5 to 10 h)</td>
</tr>
<tr>
<td>• Distinct negative temperature coefficient of reactivity</td>
</tr>
<tr>
<td>→ Fast shutdown system not necessary</td>
</tr>
<tr>
<td>• Ceramic core, ceramic fuel coating with high retention capacity even under core heatup conditions</td>
</tr>
<tr>
<td>→ No fuel melt, fission product release small and delayed</td>
</tr>
<tr>
<td>• Single-phase inert coolant (helium), lower pressure level (~50 bar)</td>
</tr>
<tr>
<td>→ Loss-of-coolant accident with fewer problems, need for (forced) convection, low potential for mechanical destruction</td>
</tr>
<tr>
<td>• Potential for massive graphite corrosion if T &gt; 500°C and O_2 &gt; 1 ton</td>
</tr>
<tr>
<td>→ Air/water ingress accidents significant</td>
</tr>
<tr>
<td>• Coolant gas activity insignificant, plateout and dustborne activity relevant</td>
</tr>
</tbody>
</table>

The basic idea for the proposed ASC (Table III) is to limit the activity release resulting from severe accidents in such a way that emergency measures do not need to be considered for the protection of the public and the financial damage can be coped with by the society. Emergency measures include

- acute measures like evacuation, distribution of stable iodine tablets and early relocation as well as
TABLE III
Advanced Safety Criteria (ASC)
proposed by the Institute

The aim is to limit activity release of severe accident in such a way that emergency measures do not need to be considered (no evacuation, relocation, decontamination) and the financial damage can be coped with.

The way is to establish short-term and long-term dose limits for individuals at worst location, ranges of
- 1 to 25 rem for 7 days exposure
- 10 to 250 rem for 30 years exposure

Reactor with passive inherent safety characteristics is most promising for fulfilment.

Für this purpose a dose limit for short-term (7 days) and long-term exposure (30 a) after severe accidents is proposed for individuals at worst location. Reasonable values have been derived from intervention levels for protective actions showing dose ranges of
- 1 to 25 rem for short-term exposure and
- 10 to 250 rem for long-term exposure.

For the conversion of release into doses calculation principles should be applied which have originally only been valid for design-relevant accidents /2/ and which are slightly reduced in their degree of conservatism for this purpose.

Severe accidents include all events and event sequences of such a low probability that the plants do not need to be designed against them. They normally dominate the risk and need to be identified by comprehensive probabilistic safety analysis (PSA). Frequencies in the range of lower than $10^{-7}$ ... $10^{-8}$ per reactor...
year are proposed as cut-off line. Vulnerability in relation to acts of sabotage and catastrophic rare events should be as small as possible.

Assuming 2.5 rem as dose limit for 7 days and 25 rem as an even more stringent limit value for 30 a exposure time as well as the proposed conservative principles for back-calculation the tolerable release of the representative nuclide Cs-137 is in the range of 20 to 1000 Ci depending on technical parameters like emission height and time.

We do not intend to dictate the technical measures by which these ASC can be fulfilled. But we know from PSA experience and risk reduction studies for mitigation features that a reactor design with passive inherent safety characteristics is the most promising way:

The reactor may not lose its retention capability even under total loss of cooling conditions due to physical reasons; the attempt to exclude practically loss of cooling conditions by providing an additional active safety system is misleading.

The course of severe accidents must be slow with time for mitigating countermeasures and for the reversal of wrong human actions.

The question is whether the safety characteristic of current HTR concepts of small (HTR-Modul of the KWU-group, HTR-100 of the BBC-group) or medium power (HTR-500 of BBC-group) are appropriate to meet these ASC.

As long as massive corrosion attack can be excluded temperature induced coated particle failure is the dominant release mechanism for fission products. THERMIX core heatup calculations show (Fig. 1) that for small HTR the maximum temperatures for fuel elements are below or slightly in the range of increased
Fig. 1: Maximum temperatures of fuel elements for core heatup accidents in pebble-bed HTGR.

Fig. 2: Cooling system and confinement of HTR-500.
permeability, whereas they reach the range of total particle failure in case of the depressurized uncooled HTR-500. Therefore, I want to limit the subsequent discussion mainly to the example of HTR-500. The original plant design is shown in Fig. 2, meanwhile it has been further developed by the vendor.

The HTR-500 MW (electric) has a core of spherical fuel elements cooled by helium in a downward flow. A prestressed-concrete reactor vessel (PCRV) with a steel liner contains all the components of the primary circuit; a thermal insulation and a redundant liner cooling system (2 x 100%) protect the vessel against excessively high temperatures. Apart from the main cooling system, a two-train auxiliary cooling system is available for afterheat removal. It is started automatically and one train is sufficient even with an unpressurized reactor. On the primary side, the afterheat can also be removed by natural convection at full system pressure. This requires a special operator action from the control room after half an hour. The final activity barrier to the environment is a vented confinement that filters small leaks from the primary

<table>
<thead>
<tr>
<th>TABLE IV</th>
<th>Core Heatup Release Categories for HTR-500*</th>
</tr>
</thead>
<tbody>
<tr>
<td>Category</td>
<td>Frequency 1/yr</td>
</tr>
<tr>
<td></td>
<td></td>
</tr>
<tr>
<td>K1 core heatup</td>
<td>10^-1</td>
</tr>
<tr>
<td>PC open</td>
<td>0 to 200 (LoPS - 100%)</td>
</tr>
<tr>
<td>Failure of LCS</td>
<td>200</td>
</tr>
<tr>
<td>RPB stack</td>
<td>&gt;200</td>
</tr>
<tr>
<td>K2 core heatup</td>
<td>3 x 10^-4</td>
</tr>
<tr>
<td>PC open</td>
<td>35</td>
</tr>
<tr>
<td>RPB stack</td>
<td>&gt;35</td>
</tr>
<tr>
<td>K3 core heatup</td>
<td>10^-5</td>
</tr>
<tr>
<td>PC open</td>
<td>15</td>
</tr>
<tr>
<td>RPB filter stack</td>
<td>&gt;15</td>
</tr>
<tr>
<td>K4 core heatup</td>
<td>10^-4</td>
</tr>
<tr>
<td>PC short term open</td>
<td>(LMCS 85%)</td>
</tr>
</tbody>
</table>

Note: PC = primary circuit, RPB = reactor protection building (confinement), ( ) = contribution to frequency, LoPS = loss of preferred power supply, and LMCS = loss of main loop cooling system.

*Initial inventories of grouped nuclides are in parenthesis.
Loss of preferred power supply (LoPS) $4 \times 10^{-7}$/yr

Start diesel or restoration of grid supply $\leq 5$ h

Core heatup reaching set point of PCRV-RV after $\sim 5$ h and failure to close, partial depressurization

Restoration of energy supply $\leq 10$ h

Massive core damage $\sim 150$ h

RPB-RV

CORE HEATUP PC OPEN FAILURE OF ECS RPB-FILTER-STOCK

K1 $2 \times 10^{-7}$/yr

RPB = Reactor protection building

RV = Relief valve

AHRS = Afterheat removal system

MCS = Main cooling system

LCS = Liner cooling system

PC = Primary circuit

Fig. 3: Dominant event sequences for core heatup categories of HTR-500
circuit and releases larger leaks up to a maximum rate of 11.5 kg/s unfiltered into the environment, in both cases via the stack.

To establish the accident topology, parallel to the planning work, more than three years ago a small effort PSA /3/ has been carried out by the Institute. Dominant initiating events are not so much leaks in the primary circuit as so-called transient events. The release category which dominates the risk and results in the highest release figures is initiated by loss of preferred power, followed by reactor scram, failure of diesels to start and non-restoration of grid supply within five hours for the afterheat-removal-systems, non-restoration of energy supply for the liner cooling system and no emergency feeding within 10 hours (Fig. 3).

The slow temperature rise is illustrated by Fig. 4. A period of at least 5 h remains for the repair of afterheat removal system or restart of power supply and at least 10 h to restore liner cooling.

Only a few percent by volume of the core reach temperatures of a maximum of 2250°C. Large sections remain below 1600°C and do not contribute to temperature-induced fission product release which explains even for this low frequency event sequence (≤ 2·10^-7/reactor-year) relatively low release values.

Release into the environment does not begin until after ~5 h when the response pressure of the PCRV relief valves has been reached. If they fail to close the first phase of release is finished after 35 h when the depressurization is completed. An additional 'activity thrust' had to be expected in the long term caused by the thermal degradation of unprotected concrete of the PCRV in the top area.
In total 1% of the core inventory of cesium, this means $3 \times 10^4$ Ci of Cs-137, would be released unfiltered via stack. In comparison to tolerable release values derived from the proposed ASC they are too high by one or two orders of magnitude. But nevertheless, it seems to be feasible to meet these stringent requirements even with the HTR-500 if we take new results of safety research and design options for passive systems into account.

The retention of fission products by absorption in the graphite of the core and the reflectors have been treated inadequately in the past, that means in the small-effort PSA. Taking absorption of metallics beyond 1500°C into account by extrapolating available data according to known physical laws, this would lead to a reduction of cesium release by a factor of 5 up to a potential of more than 10. Experimental work has been initiated to improve the data base for the matrix and reflector graphite and to confirm that absorption process can be described by steady-state assumptions.

More sophisticated thermohydraulic calculations have shown that the afterheat can be dissipated into the massive PCRV-structure (Fig. 5, maximum core temperature below 1600°C after half a year) without jeopardizing its consistancy.
The temperatures of the liner including its insulation keep below 1200°C even if the heatup and vaporization of the water is neglected conservatively. In the past we have assumed that the liner would fail at temperatures of 1000°C, but current experiments have shown more favourable values, may be much higher than 1200°C; the steam produced is externally released through the pores and cracks of the concrete. If this can be confirmed by more representative experiments with segments cut from the PCRV thermal degradation of the concrete (beginning after more than one week) has no longer to be assumed. The second phase of release would be eliminated leading quantitatively to a reduction of cesium release by a factor of 3.

Even if this failure mode would not be impossible the amount of water released from heated up concrete and entering the core is expected to be much less than assumed in the past. The maximum hole in the PCRV is of 33 cm² in size (relief train of the safety valve); it limits the mass flow into the reactor building to values which allow filtering of the exhaust. According to recent studies metal fibre filters /5/ are available which are characterized by their temperature resistance up to 300°C and pressure difference resistance up to
1 bar; decontamination factors of $10^{-3}$ can be reached for cesium aerosols. A passively arrangement of these filters within the reactor building is possible. In the case of a rapid depressurization these filters made from metal fibres can be overloaded by a factor of 10 as compared to the normal design.

In conclusion, it is clearly possible to meet non-catastrophic release requirements, technically formulated by the proposed ASC, by small and potentially even by medium HTR. Although some restrictions have to be made:

- More comprehensive safety studies including the whole spectrum of initiating events and event sequences, including changes in reactivity or air- and water ingress, have to confirm the results of small-effort assessments; the trends derived from current research work have to be verified.

- The proof of safety for passive components and physical features, which largely determine the advanced safety standard, must be further raised in quality. It must be shown that a catastrophic failure of the pressure vessel (PCRV may be more advantageous the steel vessels), large closures or pipes can be excluded or does not lead to higher consequences caused by massive graphite corrosion.

- Safety concepts which do not only rely on the retention capability of the fuel elements, but on controlled release via filters, like for the modified HTR-500, have to make sure that the filters cannot be bypassed under severe accident conditions.

These reservations should motivate further work and cooperation, but they do not doubt the attractiveness of HTR from the safety point of view.
REFERENCES


/2/ Empfehlung von RSK/SSK: "Störfallberechnungsgrundlagen für die Leitlinien des BMI zur Beurteilung der Auslegung von KKW mit DWR gemäß § 28 Abs. 3 StrSchV, Bundesanzeiger, 35, 245a (1983)


THE SAFETY CONCEPT OF THE MODULAR HTR

G.H. LOHNERT
Internationale Atomreaktorbau GmbH,
Bergisch Gladbach, Federal Republic of Germany

Abstract

The paper contains the most important design features of the modular HTGR with pebble bed fuel and a description of modular HTGR safety features and safety concepts.

Components of the nuclear steam supply system are described together with their impact on the plant's safety behaviour.

1 INTRODUCTION

The modular HTR power plant is a universally applicable energy source for the co-generation of electricity, process steam or district heating.

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 (HTR) are also available at higher power plant ratings.

Due to its universal applicability and excellent safety features, the modular HTR power plant is suitable for erection on any site, but particularly on sites near other industrial plants or in densely populated areas.

The principle 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 is due to the selected reactor design which guarantees that the modular HTR power plant does not present any hazard to the environment either during normal operation or in the case of any accidents.

The most important design features are as follows:

- The use of spherical fuel elements, which are capable of retaining all radiologically relevant fission products up to fuel element temperatures of approx. 1600 °C.
- The reactor core is designed such that a maximum fuel element temperature of 1600 °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 transport mechanism (such as heat conduction, radiation, natural convection) to a simple surface cooler with water flowing
through it. This is installed outside the reactor pressure vessel in the primary cell.
- Reactor shutdown solely by absorber elements in the reflector, which, on demand, can drop freely into boreholes.
- Sole use of Graphite in core areas with high temperatures (fuel elements, core internals). Temperature-incurred failure of this material is impossible at the maximum occurring temperature of 1600 °C.
- The single phase noble gas helium, which is neutral from a chemical and neutron physical viewpoint, is used as coolant.
- Due to the high activity retention in the fuel elements, a pressure-tight reactor building is not necessary. The reactor building is accessible for repair work at 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 case of failure of the primary circuit cooling. This installation also increases the accessibility of the components for maintenance and repair.

2 NUCLEAR STEAM GENERATING SYSTEM

Construction and Design Features

The nuclear steam generating system (HTR-Module) basically consists of:
- The reactor pressure vessel with core, core internals, shutdown systems and facilities for the charging and discharging of fuel elements
- The connecting pressure vessel with hot gas 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 (see Fig. 1). Pipes with water flowing through them (surface coolers) are installed on the inside wall of the primary cell near the reactor pressure vessel. These surface 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 beside each other.

This characteristic HTR-Module layout offers substantial advantages:
- After a reactor shutdown there is no harmful natural circulation of hot helium through the primary circuit thanks to the thermohydraulic decoupling of heat source and heat sink. Consequently there is no need to cool the steam generator after shutdown. Thus it can be shut off and left in a hot condition.
FIG. 1. Vertical cross section of the reactor building

1 Reactor pressure vessel
2 Steam generator pressure vessel
3 Connecting pressure vessel
4 Primary circuit blower
5 Primary cell
6 Protective shell
7 Surface cooler

FIG. 2. Primary circuit of an HTR-Module

1 Pebble bed
2 Pressure vessel
3 Fuel element discharge
4 Small sphere shutdown unit
5 Reflector rod
6 Fuel element loading
7 Steam generator tube bundle
8 Steam generator shroud
9 Feed water line
10 Live steam line
11 Primary circuit blower
12 Hot gas duct
13 Surface cooler
The positioning of the steam generator beside and lower than the reactor permits simple and operationally favorable upward evaporation.

The 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 accessibility problems.

The staggered arrangement leads to a favorable height-to-width ratio of the reactor building. This simplifies verification of its stability in the case of loads resulting from external impacts (e.g. earthquake, explosion blast wave).

The HTR Fuel Element

Fissile uranium is present in the HTR-Module in spherical fuel elements. Each fuel element has a diameter of six centimetres and contains approximately 11600 coated particles within the inner graphite matrix.

Each of these coated particles consists of a fuel kernel (uranium dioxide $\text{UO}_2$) with a diameter of about 0.5 mm, which is coated with several layers of pyrocarbon (PyC) and silicon carbide (SiC). These layers enclose the fuel kernel, thus preventing a fission product release from the fuel kernel.

One fuel element contains a total of 7 g of uranium with a $\text{U}^{235}$ enrichment of 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 practically completely retaining all radiologically relevant fission products in the intact coated particles up to fuel element temperatures of around 1600 °C.

For this reason, the HTR-Module was designed in such a way that the maximum fuel element temperatures limit themselves (i.e. without the intervention of active systems) to values of approximately 1600 °C during all accidents.

At temperatures of less than 1600 °C there is only a slight activity release from fuel particles with defective coatings. Numerous experiments have shown, however, that the number of defective coated particles to be expected is extremely low. This fact also explains the low coolant activity and the low environmental exposure in the case of loss of coolant accidents.

Reactor Core

The reactor core, which is located inside the reactor pressure vessel, 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 limited to 3 MW/m$^3$ and the mean core outlet temperature to 700 °C.
The design of the reactor core is based on the following principles:

- In all accidents and accident combinations, a maximum fuel element temperature of approx. 1600 °C is not exceeded even without active removal of the decay heat from the core. Decay heat removal can be effected solely by heat conduction, heat radiation and natural convection to the surface coolers positioned outside the reactor pressure vessel.
- The reactor can be shut down purely by dropping the absorbers into the reflector boreholes.
- The uranium loading of the fuel elements amounting to 7 g of uranium is selected such that water ingress into the primary circuit as the result of an accident will cause a lower reactivity increase than the accidental withdrawal of all reflector rods. Therefore the faulty withdrawal of all reflector rods is a covering reactivity accident. This accident is controlled by simply switching off the primary circuit blower, whereby the permissible fuel element temperature of approx. 1600 °C is not exceeded.

In order to obtain the most uniform possible power density distribution, the spherical fuel elements pass through the core approximately 15 times before reaching their final burn-up.

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 up of single graphite and carbon blocks. The top reflector consists of several layers of 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 reactor pressure vessel at a point above the connecting pressure vessel, acts as supporting structure for the ceramic core internals. The cover of the core vessel is constructed as a radiation shield (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 for the shutdown and control systems are mounted on the thermal shield. These consists of the reflector rods and the small sphere shutdown units.

The 6 reflector rods are for the reactor control and hot shutdown of the core. Each rod consists of several individual sections holding the absorber material B\textsubscript{4}C.

In order to initiate a reactor scram, the power supply to the drive is interrupted, thus causing the rod to drop freely into its lowest position in the boreholes of the side reflector due to gravity.

18 small sphere shutdown units serve to compensate the increase in reactivity when running down the reactor to a cold condition. Graphite spheres with an approx. 10 % B\textsubscript{4}C 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 over the side reflector, drop freely into the reflector boreholes on demand.
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 downwards 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.

Steam Generation

The steam generator is designed as a once-through, helical tube steam generator.
The hot helium is fed into the steam generator above the heating tube bundle. While flowing around the steam generator heating tubes, the helium releases its heat to the water/steam side, whereby it cools down from approx. 700 °C to approx. 250 °C.
On leaving the heating tube bundle, the flow is deflected through 180 °C. The cold helium then flows upwards between the steam generator shroud and the inner wall of the pressure vessel so that the pressure vessel is cooled by the cold gas.

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 feedwater enters the feedwater nozzle with a temperature of 170 °C. The water flows through the helical tubes from the bottom to the top, during which process it evaporates and is subsequently superheated. Via a compensating bundle above the heating tube bundle, the steam which is 530 °C hot enters the live-steam nozzle and passes through the live-steam line to the machine hall.

Primary Circuit Blower

The primary circuit 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 blower increases the pressure of the helium sucked in from the steam generator outlet side by approx. 1.5 bar.

Cavity Cooler

The cavity cooler surrounds the reactor pressure vessel at a distance of about 1.5 m. It is installed as a closed tube wall in front of the concrete walls of the reactor cell with a clearance distance of approximately 10 cm. Water at a temperature of about 40 °C flows through the surface 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 case of failure of the main heat sink.

Nuclear Ventilation System

In the reactor building and reactor auxiliary building of the modular HTR power plant, all areas are accessible during normal operation with the exception of the primary cells. Treated outside air is fed into the reactor building to remove the dissipated heat and to retain the necessary quality of the air. The desired subpressure is maintained by regulating the supply air flow while keeping the exhaust air quantity constant. During normal operation the vent air is released to the environment unfiltered via the vent stack due to the low activity pollution.

Dust-bound radioactive materials may enter the vent air during repair or maintenance work on the primary circuit or connected systems. For this reason, the vent air filter system (aerosol filter) is switched on during such work as a precautionary measure.

If, as the result of leakages in components containing primary coolant, gas-borne activity manages to enter the reactor building or the reactor auxiliary building, the facility is automatically switched over to the subpressure maintenance system with its activated carbon and aerosol filters.

3 SAFETY OF THE MODULAR HTR POWER PLANT

3.1 GENERAL

The overriding aim of the safety design is the in-plant retention of the radioactive materials which unavoidably form due to nuclear fission such that any danger to the environment can be excluded both during normal operation and after accidents.

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 radioactivity releases. This is primarily due to the optimum exploitation of the favorable inherent features of small gas/graphite reactors.

Inherent Safety

"Inherent (i.e. intrinsic) safety" is a specialized term 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 can be reached. These reactions are governed by the laws of nature, i.e. they always function regardless of the condition of active systems. This means that they cannot malfunction or fail.

The technical and nuclear physical design of the HTR-Module is such that the fuel element temperature always stabilizes itself at approx. 1600 °C even in the case of assumed failure of all active shutdown and decay heat removal systems.
On the one hand this is achieved by the fact that, in the HTR-Module, there is a temperature span of approx. 700 K between the maximum permissible of the fuel element temperature and the maximum operating temperature of the fuel elements. This temperature span ensures that the reactor core shuts itself down via the negative temperature coefficients of reactivity, even after accident-incurred introduction of the existing surplus reactivity, before the above limit temperature of 1600 °C is reached.

On the other hand, 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 ensure that the decay heat can escape from the reactor core to the surrounding components and structures solely by means of physical processes (heat conduction, radiation, convection).

Activity Release Barriers

A characteristic safety feature of the HTR-Module is to be found in the fact that, in all operating and accident situations, the radioactive materials formed during nuclear fission are enclosed in the coated particles in such a way that a significant activity release from the coated particles can be excluded.

This safe activity enclosure is guaranteed by the design of the fuel particle coatings and the inherent limitation of the maximum possible accident temperature of the fuel elements to approx. 1600 °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 is distributed in the primary circuit.

The gas-borne activity in the primary circuit is reduced by radioactive decay, by separation in 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 a release of radioactive materials. Leaks in the connecting pipes which cannot be isolated are very improbable due to the planned quality assurance measures.

In a nonetheless postulated leakage, in principle only the very low gas-borne primary coolant activity and a small fraction of the activity deposited on the primary circuit surface could be released into the reactor building. For this reason, and basically because of the high retention capability of the coated particles, no tightness requirements are imposed on the reactor building of the HTR-Module for adherence to the permissible accident dose limit values of Article 28 of the German Radiation Protection Ordinance, i.e. the released activity can be discharged to the environment without any danger.

3.2 ACCIDENTS

Measures for Accident Control

Even though the nuclear power plant is designed and constructed in accordance with the highest quality requirements, it is impossible to completely exclude failure of technical systems.
Due to the selected design of the HTR-Module, only a few protective actions have to be taken after accidents to shut the reactor down and to keep it in a safe condition. After being shut down, the HTR-Module can be left standing in a hot condition as, in this condition, the maximum fuel element temperatures only increase to approx. 1100 °C at a normal operating pressure even without active core cooling, and to approx. 1600 °C after loss of pressure.

The steam generator need not be cooled due to the thermohydraulic decoupling of reactor and steam generator after shutdown. 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 an accident, the decay heat is removed from the HTR-Module via heat conduction and heat radiation to a cavity cooler installed outside the reactor pressure vessel. This cooler is capable of protecting the core internals, the shutdown systems, the reactor pressure vessel and the concrete structure of the reactor cell from inadmissible temperatures during all accidents. During normal operation, the cavity cooler removes the dissipated heat from the reactor cell. It is therefore continuously in operation and need not be specially started up when the accident occurs.

The water throughput and water temperature of the cooler are so low that, for an assumed failure of all pumps or recooling chains, water supplied by a hose pipe after approx. 15 hours (e.g. by the fire department) would suffice to ensure adequate cooling of the reactor cell and the reactor pressure vessel without any damage having been caused in the plant that would not allow further operation. In principle, in the case of a postulated primary circuit leakage, it is possible to directly discharge the primary coolant unfiltered to the environment. Nevertheless, in the case of depressurization accidents with leakage cross-sections of up to 2 cm², it is planned to pass the leaking primary coolant through the subpressure maintenance system and to discharge it filtered via the stack in order to minimize the radiological exposure of the environment. In the case of larger assumed leakage cross-sections which would cause a notable pressure increase in the reactor building, the building is shortly depressurized with direct discharge to the environment.

Reactor Protective Actions

Accidents cause characteristic changes in the process variables which are detected by the reactor protection system and which 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 the feedwater and live-steam valves) separates the reactor plant from the steam power plant. In the case of a loss of pressure accident and a steam generator heating tube rupture, the primary circuit isolation valves are additionally closed resp. the steam generator is quickly drained.

3.3 BASIC DESIGN ACCIDENTS

In order to assess the efficiency of the safety design and the planned protective measures, the occurrence of accidents is assumed in analyses and the resulting accident courses are examined.

Reactivity Accidents

In order to define a covering reactivity accident it is postulated that all 6 shutdown rods withdraw with maximum speed during full load operation due to an assumed fault in the control system.

The reactor protection system detects the accident due to the change in the neutron flux or the increase in the hot gas temperature and shuts the reactor down once it reaches the corresponding limit values. This does not result in any notable fuel element temperature increase in the core. No activity is released to the reactor building or the environment.

All other events, in which a reactivity increase is possible, e.g. faulty operation of a small sphere shutdown unit, water ingress, erroneous raising of the primary coolant throughput or erroneous lowering of the cold gas temperature, are covered by this accident.

Disturbances in the Primary Heat Transfer System

This covers all events leading to a disturbance of the heat transport form the reactor core via the primary circuit and the water/steam circuit to the condenser or the process steam converter.

Such events can be caused e.g. by an interruption of the flow through the primary circuit (e.g. failure of the primary circuit blower) or through the secondary circuit (e.g. pump failure) or by possible changes in the heat balance (e.g. faults in the live-steam or process steam extraction).

Loss of power (emergency power case) also leads to a disturbance of the heat transfer, as the primary circuit blower and the feedwater pumps simultaneously lose their driving energy.

The mass flows of the primary and secondary circuits are monitored by the reactor protection system. If the values exceed or fall short of a given flow ratio the plant is automatically shut down. Sudden changes in the heat removed are also made evident by an increase in the cold gas temperature. Once a specified limit value is reached, the plant is also shut down via the reactor protection system.
The sole consequence of all these disturbances and accidents is the temperature loading of components, but these loadings are lower than the permissible limit values. The shutdown reactor remains in a hot condition until the disturbance is eliminated. The maximum fuel element temperatures are around 1100 °C. None of these events results in an activity release to the reactor building or the environment.

Leakages and Ruptures on the Secondary Side

These events include all leakages and ruptures in tubes of the water/steam circuit. The rupture of a live-steam line in the reactor building is the covering accident. The rapidly sinking live-steam pressure due to the rupture causes a fast increase in the feedwater transport and, as a result, the flow ratio between primary and secondary circuit is disturbed. The reactor protection system initiates the protective actions dropping of the reflector rods, shutdown of the primary circuit blower and isolation of the steam generator once the limit value has been reached.

In the case of a leakage which cannot be isolated, the steam generator slowly steams out. The outflowing steam escapes from the building via the pressure relief openings. There are no inadmissible pressure or temperature loadings of the reactor building or its internals. The steam generator and its internals are designed to withstand the loadings resulting from this event. Consequently a heating tube rupture is not to be assumed as a subsequent failure. Leakages and ruptures on the secondary side therefore only lead to temperatures and pressure loadings of the components and the building. There is no activity release to the reactor building or the environment.

Steam Generator Leakages (Water Ingress)

In the assumed rupture of a steam generator heating tube water or steam forces its way into the primary circuit due to the overpressure on the secondary side (190 bar to 60 bar). The water ingress is detected by means of a moisture measurement.

If the reactor protection limit value is exceeded, the reactor is shut down and, in addition, the steam generator is rapidly drained to limit the quantity of water entering the primary circuit. The water which has leaked into the primary circuit is removed from it using the helium purification facility with the water separator. This is switched on manually. Water entering the primary circuit can cause corrosion of the fuel elements due to the reaction \( H_2O + C \rightarrow H_2 + CO \). Of the approx. 600 kg of water, which at most enter the primary circuit during this accident sequence, approx. 100 kg convert to water gas in the course of the counter-measures described above. The remaining quantity of water and the forming water gas are removed from the primary graphite burnup at the fuel elements as a result of the water gas reaction amounts to approx. 0.7 % This is non-problematic both with regard to fuel element corrosion, stability and fission product release.
If failure of the water separator is postulated, the primary circuit pressure control alone is also capable of preventing an increase in pressure resulting from the additional formation of water gas. Consequently the pressure relief system of the primary circuit is not actuated in this case either.

The sequences described above which take place due to the ingress of water do not result in any activity release from the primary circuit to the reactor building or the environment.

If the improbable accident combination: "Failure of the water separator and failure of pressure control" is assumed, a pressure rise in the primary circuit can occur. This actuates the pressure relief system after about 5 hours. As a result, some of the primary coolant mixed with steam and water gas is blown off into the reactor building. It is then filtered and released to the environment from this building.

The radiological consequences are covered by the consequences described for the accident "Loss of pressure in the primary circuit". They remain under the values stipulated in Article 28 of the German Radiation Protection Ordinance by approximately four orders of magnitude.

**Loss of Pressure in the Primary Circuit With Subsequent Core Heat-Up**

Possible events initiating this accident can primarily be leakages in a system connected to the pressure vessel, as the result of which primary coolant can flow into the reactor building.

Primary coolant leakages are detected by the reactor protection system by way of the decreasing primary circuit pressure. The reactor protective actions dropping of the reflector rods, shutdown of the primary circuit blower, closure of the primary circuit isolation valves are initiated as counter-measures. Closure of these primary circuit isolation valves, which are installed in the connecting lines to the pressure vessel outside the primary cells, prevents a further discharge of primary coolant.

If helium resp. activity is measured in the vent air, the plant is automatically switched over to the subpressure maintenance system to ensure that small leakages are released in filtered form. This measure is taken to minimize the environmental exposure.

In order to determine the maximum consequences, a total non-isolable rupture of the largest connecting line (33 cm²) directly at the pressure vessel is assumed as covering accident.

Rupture of the line is followed by the complete depressurization of the primary circuit until it has reached the same pressure as the environment after several minutes. In order to prevent an inadmissibly high internal pressure in the reactor building, the primary coolant is discharged to the environment via pressure relief openings with flaps in the reactor building.

The environmental exposure caused by the radioactive materials released with the escaping primary coolant is much lower than
the permissible accident limit values given in the German Radiation Protection Ordinance.

After depressurization, the core slowly begins to heat up if the decay heat cannot be removed by the main heat sink. After approx. 32 hours, a maximum fuel element temperature of approx. 1600 °C is reached in one small sector of the reactor core, as shown in Fig. 3.

Less than 2% of all fuel elements are exposed to a temperature exceeding 1500 °C. The temperatures of the other fuel elements are much lower (see Fig. 4).

FIG. 3: Temperature variations after depressurization accident with core heat up

FIG. 4: Time-dependent fraction of fuel element for various temperatures (depressurization with core heat up)
Due to the heating up of the core and the related expansion of the cooling gas, radioactive materials can be released into the reactor building from the primary circuit. However, as the limit temperature for fuel elements of approx. 1600 °C is not exceeded, there is no massive fission product release from the core.

If activity is measured in the vent air, the subpressure maintenance system with its filters is automatically started. Consequently the radiological consequences of the activity release are kept at a level which is much lower than the limit values given in the german Radiation Protection Ordinance (see Table 1).

In addition, the fact that the core only heats up slowly provides adequate time for the implementation of counter-measures. The reactor building can always be entered to start repair measures both during and after core heat-up.

Table 1 shows the maximum radiation exposure of single organs in comparison to the limit values listed in Article 28 of the German Radiation Protection Ordinance. The calculation rules of this article include all exposure paths like submersion, inhalation and ingestion (ingestion over a period of 50 years!).

The table shows that, even in this covering accident with activity release, the values are much lower than the limit values of the German Radiation Protection Ordinance and that there is no danger to the population.
3.4 PROTECTION AGAINST EXTERNAL IMPACTS

In addition to accident events, which are caused by operation of the reactor itself, other events are observed. These have an external impact on the plant and can cause damage.

Earthquake

Important safety-relevant systems of the modular HTR power plant are designed for a maximum seismic intensity. According to scientific knowledge it is extremely improbable that this intensity will be exceeded.

The maximum possible loading as the result of an earthquake is taken into consideration in the design of the reactor building and the central control building.

Inside the reactor building basically only the pressure vessel with primary and secondary circuit isolation valves, the cavity coolers with safeguarded recooling and all systems required for correct functioning of the reactor protection system are designed to withstand an earthquake. The systems installed in the central control building such as the reactor protection system, the emergency power system, the control room etc. also function after an earthquake due to suitable dimensioning. This means that all measures required for minimization of the activity release can be taken.

Aircraft Crash

The reactor building is constructed such that a crashing aircraft cannot penetrate the roof and the outer walls.

The pressure vessel including the primary circuit isolation valves and the cavity coolers are designed to withstand the vibrations transferred to the building as the result of a crash and remain intact.

If defects arise in the systems connected to the primary circuit after the aircraft crash, the reactors are automatically shut down and isolated by the reactor protection system as the result of the disturbance, and the decay heat is removed via the cavity coolers.

If recooling of these coolers were to fail, water could be supplied to the surface coolers via fire department hose pipes. More than 10 hours time is available for the initiation of this measure.

If the central control building is destroyed by the aircraft crash, the necessary reactor protective actions will nevertheless be actuated due to the design (e.g. closed circuit current principle) and the reactor will be brought to a safe condition. The plant can subsequently be monitored from a emergency control room located in side the reactor building.

Explosion Blast Wave

The reactor building together with the safety-relevant systems such as pressure vessel with primary circuit isolation valves and cavity coolers is designed to withstand the loadings resulting from an explosion blast wave.

If other parts of the plant are damaged the reactor is automatically shut down by the reactor protection system.
If the central control building is inoperative as the result of the explosion blast wave, the plant can be monitored from the emergency control room located inside the reactor building. In the case of failure of the recooling of the cavity coolers, water can be supplied to the surface coolers via fire department hose pipes. More than 10 hours time is available for the initiation of this measures.

### 4 MAIN DATA

**Overall plant (2 reactors)**

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value 1</th>
<th>Unit(s)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor thermal power</td>
<td>400</td>
<td>MW&lt;sub&gt;th&lt;/sub&gt;</td>
</tr>
<tr>
<td>Generator terminal power</td>
<td></td>
<td></td>
</tr>
<tr>
<td>- for max. process steam generation</td>
<td>72</td>
<td>MW</td>
</tr>
<tr>
<td>- for min. process steam generation</td>
<td>124</td>
<td>MW</td>
</tr>
<tr>
<td>Quantity of process steam</td>
<td></td>
<td></td>
</tr>
<tr>
<td>max.</td>
<td>115</td>
<td>kg/s</td>
</tr>
<tr>
<td>min.</td>
<td>47</td>
<td>kg/s</td>
</tr>
<tr>
<td>Process steam pressure/temperature</td>
<td>17 bar/272 °C</td>
<td></td>
</tr>
</tbody>
</table>

**Reactor**

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value 1</th>
<th>Unit(s)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Number of fuel elements</td>
<td>360 000</td>
<td></td>
</tr>
<tr>
<td>Fuel element diameter</td>
<td>6</td>
<td>cm</td>
</tr>
<tr>
<td>Fuel element charging process</td>
<td>Multiple passage</td>
<td></td>
</tr>
<tr>
<td>Primary coolant</td>
<td>Helium</td>
<td></td>
</tr>
<tr>
<td>Helium flow at 100 % power</td>
<td>85</td>
<td>kg/s</td>
</tr>
<tr>
<td>Helium temperatures</td>
<td>250/700</td>
<td>°C</td>
</tr>
<tr>
<td>Mean helium pressure</td>
<td>60</td>
<td>bar</td>
</tr>
<tr>
<td>Mean power density</td>
<td>3.0</td>
<td>MW/m&lt;sup&gt;3&lt;/sup&gt;</td>
</tr>
</tbody>
</table>

**Steam Generator**

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value 1</th>
<th>Unit(s)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Live-steam pressure at SG outlet</td>
<td>190</td>
<td>bar</td>
</tr>
<tr>
<td>Live-steam temperature at SG outlet</td>
<td>530</td>
<td>°C</td>
</tr>
<tr>
<td>Live-steam mass flow</td>
<td>77</td>
<td>kg/s</td>
</tr>
<tr>
<td>Feedwater inlet temperature</td>
<td>170</td>
<td>°C</td>
</tr>
</tbody>
</table>
MHTGR LICENSING APPROACH AND PLANT RESPONSE TO OFF NORMAL EVENTS

A.J. NEYLAN, F.A. SILADY
GA Technologies, Inc.,
San Diego, California,
United States of America

Abstract

The MHTGR design meets stringent top-level regulatory and user safety requirements that require that the normal and off-normal operation of the plant not disturb the public's day-to-day activities. Quantitative, generic top-level regulatory criteria have been specified from US NRC and EPA sources to guide the design. The user has further specified that these be met at the plant boundary. A probabilistic risk assessment has been utilized to select licensing basis events that cover a wide spectrum of events. The events have been grouped into three frequency regimes governed by the 10CFR50 App. I, 10CFR100, and PAG risk and dose criteria.

The MHTGR fuel has been designed to limit the primary circuit activities to levels that, even if completely released, are within those allowed by 10CFR100. The focus of the safety approach has then been centered on retaining the radionuclide inventory within the fuel by removing core heat, controlling chemical attack, and by controlling heat generation. The core geometry, core power, core power density, heat removal geometry, and heat sinks are designed to provide passive core temperatures during accidents. To limit the potential for chemical attack of the MHTGR core, the primary coolant boundary is designed to make large ingress of air and water very unlikely. Furthermore, the fuel particle coatings are highly impervious to oxidizing agents that do enter the primary coolant. The core heat generation is controlled by the large negative temperature coefficient and by insertion of control material to maintain a subcritical core configuration during the span of the accidents. Thus, the three key safety functions have been accomplished by relying on the inherent characteristics of the MHTGR so that the response is largely passive and relatively simple.

SAFETY PHILOSOPHY

The overall safety philosophy guiding the design of the MHTGR is to produce a safe, economical plant design which meets NRC and user requirements by providing defense-in-depth through the pursuit of four goals: 1) Maintain Plant Operation, 2) Maintain Plant Protection, 3) Maintain Control of Radionuclide Release, and 4) Maintain Emergency Preparedness.

With regard to the achievement of NRC criteria for the accomplishment of the first two goals, measures are taken in the design of the MHTGR to minimize defects in the fuel so that normal operational releases or any accidental releases of primary circuit activity are low and worker exposures are minimized.
The unique aspect of the MHTGR, however, is the approach which has been taken to achieve the third goal and thereby minimize the design requirements from the fourth goal. To accomplish this with high assurance, the design of the MHTGR has been guided by the additional philosophy that control of radionuclide releases be accomplished by retention of radionuclides within the fuel particles with minimal reliance on active design features or operator actions. The overall intent is to provide a simple safety case that will provide high confidence that the safety criteria are met. This approach is consistent with the NRC's Policy on Advanced Reactors (Ref. 1). There are two key elements to this philosophy which have had a profound impact on the design of the MHTGR (especially in the selection of core size and geometry, power density, and vessel type); the basis for each element is described below.

First, the philosophy requires that control of radionuclides be accomplished with minimal reliance on active systems or operator actions. By minimizing the need to rely on active systems or operator actions, the safety case centers on the behavior of the laws of physics and on the integrity of passive design features. Arguments need not center on an assessment of the reliability of pumps, valves and their associated services or on the probability of an operator taking various actions, given the associated uncertainties involved in such assessments.

Second, the philosophy requires control of releases by the retention of radionuclides within the coated fuel particle rather than reliance on secondary barriers (such as the primary coolant boundary or the reactor building). Proof of containment is dramatically simplified if arguments can center on issues associated with fuel particle coating integrity alone.

TOP-LEVEL REGULATORY CRITERIA AND USER SAFETY REQUIREMENTS

Top-level criteria and requirements are defined primarily from two sources: the regulator, whose concern is primarily public health and safety, and the user, whose concern is all encompassing (e.g., safety, performance, availability, and economics). Each of the four goals has been quantified by a series of top-level criteria and requirements (Ref. 2, 3). The top-level regulatory criteria are the basis for plant licensability with the Preliminary Safety Information Document (PSID) and the Probabilistic Risk Assessment (PRA) presenting how the design meets these criteria.
The following bases were adopted for the selection of top-level regulatory criteria.

1. Top-level regulatory criteria should be a necessary and sufficient set of direct statements of acceptable health and safety consequences or risks to individuals or the public.

2. Top-level regulatory criteria should be independent of reactor type and site.

3. Top-level regulatory criteria should be quantifiable.

The first basis ensures that the criteria are fundamental to the protection of the public and the environment. The second, consistent with the first, requires that the criteria be stated in terms which do not discriminate among reactor types and sites. Finally, the third basis ensures that compliance with the selected criteria can be demonstrated through measurement or calculation.

Through comparison with the selection bases, the following regulatory sources have been found to contain numerically-expressed criteria or limits which appropriately form top-level regulatory criteria.

1. 51 FR 28044 - Policy Statement on Safety Goals for the Operation of Nuclear Power Plants

2. 10CFR20 - Standards for Protection Against Radiation

3. 10CFR50, Appendix I - Numerical Guides for Design Objectives ... to Meet the Criteria "As Low As Reasonably Achievable" for Radioactive Material ... in Effluents

4. 40CFR190 - Environmental Radiation Protection Standards for Nuclear Power Operations

5. 10CFR100 - Reactor Site Criteria

6. EPA-520/1-75-001 - Manual of Protective Action Guides for Protective Actions for Nuclear Incidents

The numerical consequence or risk values contained within the above regulatory sources are the top-level regulatory criteria for the MHTGR (Ref. 1).
The utility/user group has specified an additional safety requirement (Ref. 2) that is more restrictive in that item 6 above of the top-level regulatory criteria is to be satisfied at the plant boundary. In this way the emergency planning zone, which is generally 16000 m (10 miles) for US LWRs, is reduced to the MHTGR's 425 m Exclusion Area Boundary (EAB). This allows the utility/user to limit emergency drills to the area and personnel within its control. The need for offsite sheltering and evacuation is obviated, the public's normal day-to-day activities are not disturbed by the proximity of the MHTGR plant. The specific quantitative user requirements are the EPA Protective Action Guidelines (PAGs) of 5 rem thyroid and 1 rem whole body doses evaluated at the 425 m.

**LICENSING BASIS EVENTS**

For the purpose of deriving the regulatory licensing bases of the MHTGR, the probabilistic bases for the design have been cast in a framework and format similar to that of traditional licensing approaches. Postulation of a set of bounding licensing basis events is one of the key elements in the traditional US regulatory process. Licensing basis events are used to demonstrate compliance with dose criteria for a spectrum of off-normal events.

![FIG. 1. Licensing basis regions.](image-url)
For the MHTGR, selection of licensing basis events (LBEs) is based on the probabilistic risk assessment. The use of the PRA for LBE selection provides a basis for judging, in a quantitative manner, the frequency of the entire event sequence and, therefore, the appropriate dose or risk criteria to be applied.

The initial step in the selection of LBEs is to establish a frequency-consequence risk plot defining three regions bounded in frequency by three agreed-upon mean frequencies and in consequence, by allocated dose limits related to 10CFR50 Appendix I, 10CFR100, or the PAGs. Figure 1 provides this plot as established for the MHTGR.

Those families of events in the PRA having the potential for radionuclide releases or consequences in excess of those allowed by the top-level regulatory criteria were it not for design selections that function to control the release of radionuclides are those selected as LBEs. Depending upon their predicted frequency, the selected events are encompassed by one of the following three categories:

1. Anticipated Operational Occurrences (AOOs): These are families of events expected to occur once or more in the plant lifetime. Their dose consequences are realistically analyzed in the PSID to demonstrate compliance with 10CFR50 Appendix I.

2. Design Basis Events (DBEs): These are families of events lower in frequency than AOOs that are not expected to occur in the lifetime of one plant but which might occur in a large population of MHTGRs (approximately 200). The families of events selected as DBEs at this stage in the MHTGR design are listed in Table 1. These DBEs are evaluated conservatively in the PSID against the 10CFR100 dose criteria.

3. Emergency Planning Basis Events (EPBEs): These are families of events lower in frequency than DBEs that are not expected to occur in the lifetime of a large number of MHTGRs. The EPBE consequences are analyzed realistically in the PRA for emergency planning purposes and environmental protection assessments.

In addition to demonstrating compliance with the dose limits of the top-level regulatory criteria and the user safety requirements, the LBEs are considered collectively to show compliance with the NRC Policy Statement on Safety Goals (Ref. 4).
TABLE 1
MHTGR DESIGN BASIS EVENTS

<table>
<thead>
<tr>
<th>Design Basis Event</th>
</tr>
</thead>
<tbody>
<tr>
<td>Loss of forced core cooling</td>
</tr>
<tr>
<td>Main loop transient without control rod trip</td>
</tr>
<tr>
<td>Control rod withdrawal without main loop cooling</td>
</tr>
<tr>
<td>Control rod withdrawal without forced core cooling</td>
</tr>
<tr>
<td>Earthquake</td>
</tr>
<tr>
<td>Moisture inleakage</td>
</tr>
<tr>
<td>Moisture inleakage without forced core cooling</td>
</tr>
<tr>
<td>Moisture inleakage with moisture monitor failure</td>
</tr>
<tr>
<td>Moisture inleakage with steam generator dump failure</td>
</tr>
<tr>
<td>Primary coolant leak</td>
</tr>
<tr>
<td>Primary coolant leak without forced core cooling</td>
</tr>
</tbody>
</table>

PLANT RESPONSE TO OFF-NORMAL EVENTS

As an introduction to the plant response to off normal events an illustration of the MHTGR safety characteristics is made. To do this, the available fission product inventory is compared to the allowable activity release from the plant that would satisfy the long-term 10CFR100 and PAG thyroid dose limits. To keep the example simple, only the dominant contributor, I-131, is considered. The allowables for I-131 release are obtained from the dose limits using a conservative USNRC Regulatory Guide 1.4 weather dispersion factor. The result is an allowable I-131 release of 250 Ci and 8 Ci for the 10CFR100 and PAG thyroid doses of 150 and 5 rem, respectively.
In comparison, the MHTGR fission product inventories of I-131 are given below for an equilibrium core:

<table>
<thead>
<tr>
<th>Location</th>
<th>I-131 Inventories (Ci)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core Within fuel particles</td>
<td>9.0 x 10^6</td>
</tr>
<tr>
<td>Vessels Plateout in circuit</td>
<td>20</td>
</tr>
<tr>
<td>Circulating</td>
<td>0.018</td>
</tr>
</tbody>
</table>

As shown above even if all of the plateout and circulating activity is released, the total release is an order of magnitude lower than the 10CFR100 limits. Therefore, the safety approach is focused logically on limiting the fractional release from fuel particles to less than 3 x 10^-5 (250 divided by 9 x 10^6) and to less than 1 x 10^-6 (8 divided by 9 x 10^6) for the 10CFR100 and PAG doses, respectively.

The approach taken in the design of the MHTGR is to rely on the coated fuel particles for meeting the 10CFR100 doses and on other additional largely passive retention barriers for meeting the more restrictive PAG doses. Three functions have been identified which, when accomplished, assure that radionuclide retention within the fuel remains acceptable:

1. Remove core heat
2. Control chemical attack
3. Control heat generation

There are many ways these functions can be accomplished, and the various LBEs utilize different design selections to perform the same function depending upon the accident scenario. Generally, the less frequent LBEs rely more heavily on passive design features. For example, the MHTGR has three independent and diverse cooling systems, any one of which can, and in certain LBEs do, perform the function of removing core heat. However, while this multiplicity of systems capable of performing these functions contributes to increasing the margin of safety for the MHTGR and is considered in the LBE analyses, the MHTGR safety design approach emphasizes a minimum set of largely passive design features which, by themselves, are sufficient to accomplish these functions. How the MHTGR meets each of the three key safety functions is now briefly discussed by examining selected LBEs and the fractional release from the fuel.
FIG. 2. 350 MW(t) transients during depressurized and pressurized conduction cooldown.

FIG. 3. Coated fuel particles maintain integrity at high temperatures.
Remove Core Heat

The inherent features for heat removal include the intrinsic core dimensions and power densities of the reactor core, internals, and vessel and the passive cooling pathway from the core to the environment. Figure 2 presents the temperature transients for two DBEs, one with the primary system pressurized and one depressurized, in which the first two independent means of forced cooling are unavailable. Passive heat removal by conduction, radiation and natural convection from the core through the vessel to the reactor cavity cooling system limit fuel temperatures to acceptable levels. Passive heat removal is possible due to the large thermal margins in the fuel. As shown in Figure 3 the fuel must exceed approximately 2100°C before thermal decomposition of the silicon carbide coating results in significant failure. The normal peak fuel temperature is much lower at 1100°C.

Therefore, in the case of the depressurized conduction cooldown DBE, the release of I-131 from the fuel particles is limited by the passive heat removal to $2 \times 10^{-5}$ of the fuel inventory. This fraction is further reduced to $6 \times 10^{-6}$ since the release is slow so that the overall release fraction considering only fuel retention meets the $3 \times 10^{-5}$ 10CFR100 allowable. Additional passive retention mechanisms within the vessel and building reduce this further to $6 \times 10^{-10}$, thereby also easily meeting the $1 \times 10^{-6}$ PAG allowable.

Control Chemical Attack

The inherent features for controlling chemical attack of the fuel by water include the non-reacting cooling, a water-graphite reaction that is endothermic and requires temperatures above the average normal operating conditions, and the silicon carbide coatings on the fuel itself. The MHTGR design features that limit water ingress and its consequences include the limited sources of water, reliable detection and isolation systems, and two forced core cooling systems.

The high quality of the fuel particle coatings limits the I-131 inventory available for release due to water chemical attack for any LBE to those particles with initially failed particles (from either in-service failure or manufacturing defects). The fraction available is $2 \times 10^{-4}$ of the full inventory. This is further reduced by a factor of .015 since not all the iodine will be released if the fuel kernel is hydrolyzed. The total retention due to the fuel is then $3 \times 10^{-6}$, meeting the $3 \times 10^{-5}$ 10CFR100 allowable.
Other additional retention mechanisms limit I-131 release by limiting the amount of water entering the core and by retention in the vessel and surrounding buildings to give an overall release fraction of $6 \times 10^{-8}$, which meets the $1 \times 10^{-6}$ PAG allowable.

The inherent features for controlling chemical attack of the fuel by air include the non-reacting coolant, the embedded ceramic fuel particles, the nuclear grade vessel, and the below grade vessel silo. Figure 4 presents the fraction of the core graphite reacted for two sizes of primary coolant leaks without forced core cooling (one DBE and one EPBE) considering the amount of air available in the reactor cavity silo. As shown the fraction reacted is less than $10^{-4}$ of the core. No oxidation of the embedded fuel particles is predicted for any LBE.

![Figure 4. Limited air-graphite reaction retains radionuclides in core.](image)

Control Heat Generation

The inherent features that control reactivity include a strong negative temperature coefficient, a single phase (no void coefficient) and neutronically inert coolant and large thermal margins. These inherent characteristics cause the reactor to inherently shutdown. As shown in Fig. 5 for a pressurized conduction cooldown without any control insertion, fuel temperatures remain low. Furthermore, the plant protection system, which is separate from the operational system, includes two diverse reactivity control systems that are gravity inserted and highly reliable to protect against even rarer events. The reactivity systems can maintain the core subcritical with margin for the maximum water ingress.
In conclusion, the passive safety features of the design prevent and mitigate radionuclide release over a wide spectrum of off-normal events. Preliminary analysis has shown that for the release mechanisms and the associated severe accidents considered, the regulatory and user criteria can be met by relying largely on the HTGR fuel performance attributes. Attenuation of release by other barriers such as the reactor vessel and reactor building provides additional margin in meeting the 10CFR100 and PAG dose requirements.

REFERENCES


ACKNOWLEDGEMENTS

The authors would like to thank Mr. A. C. Millunzi, Acting Director, Division of HTGR, U.S. Department of Energy, for approval to publish this paper. Thanks are also expressed to the management of GA Technologies Inc. for permission to write and present this work, which was supported by the Department of Energy, San Francisco Operations Office Contract DE-AC03-84SF11963.
COMPARISON OF REGULATORY ASPECTS
IN DIFFERENT COUNTRIES

K. HOFMANN
Rheinisch-Westfälischer Technischer
Überwachungs-Verein e.V.,
Essen, Federal Republic of Germany

Abstract

High Temperature Gas Cooled Reactors (HTGR) for the
generation of electricity are at present only opera-
ted in the United States of America (USA) and the
Federal Republic of Germany (FRG). Major development
work is also evident in the High Temperature Reactor
sector in these countries. Aspects of licensing prac-
tice in the USA and the FRG are therefore dealt with
you to give possible impetus for the procedure to be
adopted in countries where the HTGR technology will
be introduced.

The design criteria and the technical codes must be
supplemented with regard to the requirements of High
Temperature Gas Cooled Reactors. Moves in this direc-
tion and experience in the application of the rele-
vant codes to High Temperature Gas Cooled Reactors,
codes which are mostly oriented to Light Water React-
tors (LWR), are to be found in the USA and the FRG in
conduct of HTGR licensing procedures. Some criteria
have to be applied in an analogous manner. The analog-
gous application of criteria for HTGRs depends on the
HTGR-design and refers mainly to the criterion for
testability with respect to periodic inspections and
tests, the criteria for shutdown systems, residual
heat removal systems, reactor coolant boundary and
the reactor containment.

The IAEA standard "Design for Safety of Nuclear Power
Plants, A Code of Practice" is applicable to HTGRs
without major difficulties.
1. Introduction

High Temperature Gas Cooled Reactors (HTGR) used for the generation of electricity are at present only operated in the United States of America (USA) and the Federal Republic of Germany (FRG). Major development work is also evident in the HTGR sector in these countries. It is therefore worthwhile examining aspects of licensing practice in the USA and the FRG to give some idea of the procedure which could be adopted in countries where the HTGR technology is to be introduced.

What follows gives an overview of licensing practice in the aforementioned countries and it deals with particular regulatory and design aspects of High Temperature Gas Cooled Reactors.

2. Nuclear Licensing in the USA

The federal government in the USA has the responsibility for virtually all aspects of nuclear licensing. The U.S. Nuclear Regulatory Commission (NRC) is the centralized regulatory body for the nuclear licensing process. Of particular interest are the Office of Nuclear Reactor Regulation and the Office of Standards Development, which deal with licensing technical reviews and development of technical requirements respectively.

The licensing of a nuclear power plant in the USA comprises essentially two steps: construction permit (CP) proceedings and operating license (OL) proceedings. The two stages are centered on submittal of the Preliminary Safety Analysis Report and the Final Safety Analysis Report for the CP and the OL proceedings respectively.
The fundamental statute implementing the Atomic Energy Act is Title 10 of the Code of Federal Regulations (10CFR). Parts of particular interest are: 10CFR Part 20 "Rules for Protection Against Radiation", 10CFR Part 50 "Licensing of Production and Utilization Facilities", and 10CFR Part 100 "Reactor Site Criteria". 10CFR Part 50 provides the basic technical guidance for the design of nuclear power plants, both in its body and in a series of appendices. 10CFR Part 50 Appendix A is a listing of the "General Design Criteria for Nuclear Power Plants".

The basic requirements are expanded and/or interpreted by other documents issued by the NRC. The NRC Regulatory Guides are guidelines without the force of law providing direction for meeting the regulations and may describe an acceptable solution. In order to standardize the safety reviews of license applications, NRC has issued Standard Review Plans which describe the basis of NRC technical reviews. Neither Review Plan has the force of law.

A non-regulatory source of technical criteria for nuclear plants is the consensus standards developed by industry with participation by the NRC staff. These standards are usually issued by the American National Standards Institute (ANSI) after being prepared by working groups within the framework of professional societies, such as the American Society of Mechanical Engineers (ASME), the American Nuclear Society (ANS), or the Institute of Electrical and Electronics Engineers (IEEE).

3. West German Nuclear Licensing

The central role in the nuclear licensing process is assumed by the State Supreme Licensing Authority of the individual German State. The applicant for a nuclear facility license applies to the
Supreme Licensing Authority of the State where the facility will be located (fig. 1). That State then coordinates the licensing proceeding and ultimately issues the license. The State Supreme Licensing Authorities place heavy reliance upon outside experts to conduct the technical review of license applications. The supervision of operating plants is conducted by the State Supervising Authorities.

FIG. 1. Licensing process in FRG.

The States do not, however, implement and enforce the Atomic Law completely autonomously. The Federal Government has the responsibility and authority under the Atomic Law to ensure State compliance with the law and to coordinate the activities of the various States to obtain consistency. This task was assigned to the Federal Ministry responsible for reactor safety BMU (Bundesminister für Umwelt, Naturschutz und Reaktorsicherheit), formerly to the Federal Ministry of the Interior (BMI). The BMU nuclear staff must still make use of outside technical experts for much of their technical work. In addition to outside technical companies, BMU makes continuing
use of an advisory body of scientists and engineers very similar in nature to the U.S. Advisory Committee on Reactor Safeguards (ACRS). This advisory committee, called the Reactor Safety Commission (RSK), reviews each license application as well as generic safety issues. There is also a Radiation Protection Commission (SSK) that performs a similar advisory function in radiation protection matters.

As mentioned above, both the State Supreme Licensing Authority and the BMU make use of outside technical experts during license proceedings. Each of the States has a private but nonprofit company of technical experts called a Technical Inspection Agency (TÜV). The TÜV's have numerous functions from car inspections to boiler certification, and most have a nuclear division which acts as experts for the nuclear licensing and supervising bodies.

Another nonprofit company, the Company for Reactor Safety (GRS), performs technical contracts for the various TÜVs, the Supreme Licensing Authorities, the BMU, and the RSK. Other organizations having an impact on German nuclear licensing criteria include the Länderausschuss für Atomkernenergie (State Committee for Nuclear Energy), the Nuclear Safety Standards Committee (KTA), and the German Institute for Standardization (DIN). The first organization is a coordinating body of the States and the BMU, while the latter two deal with the development of standards.

The construction permit and operating license are normally issued in a number of individual stages in the form of so-called partial licenses. Each stage covers a part of the erection of the plant or of its operation. With the first partial license for erection, the site must also have been conclusively assessed and the assessment of the safety concept as a
whole must also have been completed resulting in a positive judgement. After the construction permits or operating licenses have been granted, erection work on site of the part of the plant concerned or the operation stage concerned can commence.

The Atomic Energy Act lays down, as an essential prerequisite for granting a license that preventive measures needed to prevent damage arising from erection and operation must be taken in accordance with the state of science and technology. The Radiation Protection Ordinance (Strahlenschutzverordnung) lays down limit values and system requirements as a framework within which the handling of radioactive substances must take place.

The Federal Minister of the Interior, - new BMU -, together with the licensing authorities of the Federal States have issued "Safety Criteria for Nuclear Power Plants" /1/ to detail the necessary preventive measures. These Criteria contain protection objectives and basic requirements for the design of systems. A set of nuclear regulations has been drawn up by the Nuclear Safety Standards Committee (KTA) to specify in detail the state of science and technology in the design and realization of individual components. The "KTA Rules" take account of the available conventional technical regulations, but generally go far beyond them.

The Reactor Safety Commission (RSK) has drawn up RSK guidelines specifically for Pressurized Water Reactors. These guidelines which are far more detailed than the safety criteria serve to unify the safety requirements.
4. Applicable Codes and Guides for High Temperature Gas Cooled Reactors

In the FRG safety-technical aims and aspects for plant and environment are laid down in the "Safety Criteria for Nuclear Power Plants" /1/. These apply strictly to the Light Water Reactors and also to other reactor types such as HTGR, and indirectly when considering plant-specific systems.

The analogous application for HTGRs refers mainly to the criterion for the testability with respect to periodic inspections and tests, the criteria for shutdown systems, residual heat removal systems, reactor coolant boundary and the reactor containment. In order to avoid difficulties which arose from analogous application in the licensing procedure of the THTR-300, a draft of safety criteria was drawn up for HTGRs /2/ and the Safety Criteria for Nuclear Power Plants were redrafted /3/ to include the HTGR.

Planning specifications /4/ interpreting the Safety Criteria for Nuclear Power Plants for the THTR-300 have been especially adapted as a working basis for those involved in the licensing procedure of THTR-300. Experience has been positive with this procedure. A similar procedure is planned for the licensing of HTR-500, which is at the design stage.

Rules issued by the KTA, which apply either fully or analogously to HTGRs including rules especially developed for HTGR design, have been referred to for the detailed design of HTGR nuclear power plants, such as HTR-Module and HTR-500. Relevant items on HTGRs which are not yet included in nuclear standards are laid down in specifications agreed within the licensing procedure and which can be orientated to conventional standards such as DIN
KTA work for HTGR has been intensified within the last two years by a special HTGR-Subcommittee (HTR-Unterausschuss) in KTA. Also there are major activities on the part of the KFA Nuclear Facility at Jülich together with other companies in proposing technical rules for HTGR. They involve proposals for rules for high temperature materials and components for both electricity-generating and nuclear heat-generating HTGR power plants. Temperatures greater than 800°C are included.

Activities in developing criteria and guides for HTGRs have also been undertaken in the USA in the past, e.g.:

- a draft of "Nuclear Safety Criteria for the Design of Stationary Gas Cooled Reactor Plants" with supplements prepared by the American Nuclear Society /5/,

- an analysis showing whether the Regulatory Guides are applicable to HTGRs,

- revision of the "General Design Criteria for Nuclear Power Plants" presented in Appendix A of Part 50, Title 10, Code of Federal Regulations, for the application to HTGRs (draft) /6/,

- expanding the ASME code in respect to high temperature design.

Since the relevant codes and guides in the USA were drawn up against the background of Light Water Reactors, it is necessary there as in the FRG to make
certain interpretations for HTGRs, and the drafting of special codes and guides for HTGRs might be the objective for the future. Worthy of mentioning here are the existing supplements to the ASME code, which are also referred to in the FRG with regard to design.

On the Eighth International Conference on the HTRG in September this year we heard that it is currently planned by NRC to review the Preliminary Safety Information Document (PSID) for the "Advanced HTGR" /7/. The main purpose of the review will be to establish the licensing criteria which apply to the advanced HTGR and to make an assessment of the potential of the proposed design to meet those criteria. The licensing criteria will be established by building upon existing LWR criteria, where applicable, and by developing additional criteria, as necessary to address the unique aspects of the HTGR.

5. HTGR-Specific Features

The unique features of the HTGR must be discussed during regulatory work, important items here are:

5.1 Core Design and Systems for Control and Shutdown of the Reactor

Similarly to LWRs, two independent and diverse reactivity control systems are required. One of these shall be capable to shutdown the reactor from all operational and accident conditions for a sufficient period; the second shall be capable of maintaining cold shutdown for an unlimited time.

The following inherent safety characteristics of the HTGR

- graphite structure with a high heat capacity, thermal conductivity and phase stability
- phase stability of the coolant helium
- fuel element not sensitive to overheating
- negative temperature coefficient of reactivity

should be considered when specifying requirements for shutting down the reactor.

Therefore, a design is acceptable in which hot shutdown conditions are provided by increasing the average core temperature. This has been verified experimentally with the AVR High Temperature Reactor Power Plant in the FRG by turning off the helium circulators without moving the control rods which results in a reduction of the coolant flow and an limited increase of the core temperature /8/.

5.2 Reactor Coolant Boundary and Pressure Bearing Vessel

In the case of LWRs and small HTGRs, the reactor coolant pressure boundary including the pressure vessel consists entirely of metallic components.

According to the design of HTR-500 it is preferable to specify requirements for the metallic components which represent the enclosure of the reactor coolant and for the pressure bearing vessel separately.

In detail, the reactor coolant boundary consists of

- the leak-tight liner of the pressure bearing vessel,
- penetrations through the vessel including their closures,

- reactor coolant piping including the first isolation valves,

- pipelines penetrating the vessel and interfacing with the reactor coolant at their outer surface (e.g. heat exchangers in the primary circuit).

In addition to general requirements to design, testing, materials and leakage monitoring instrumentation, HTGR-specific features are to be considered, e.g.: the penetrations of the pressure bearing vessel must be secured against outward forces, and consequences of closure failure must be mitigated by limiting the blowdown flow area, e.g. by provision of flow restrictors.

5.3 Residual Heat Removal Systems

Reliable residual heat removal systems are required for operational and accident conditions.

The following features have to be considered when establishing requirements for the accident residual heat removal system:

- No total loss of coolant occurs in an HTGR-system that a minimum helium pressure remains in the primary circuit.

- Ingress of foreign media, e.g. water or air into the primary circuit or chemical reactions may occur.
The inherent safety characteristics should be considered, discussed earlier, e.g. the properties of graphite and the fuel element.

The proposal for the design requirements include the aspect, that if inherent characteristics can assure residual heat removal or storage after accidents so that design limits are not exceeded, hardware requirements can be softened, e.g. the requirement for meeting the single failure criterion with respect to the residual heat removal system during its maintenance can be suspended for an adequate period.

A three hour interruption of residual heat removal has been investigated for the THTR-300 power plant /9/. According to this, the structure of the core is preserved in order that the residual heat removal can be resumed by the active systems after three hours and that the reactor can be brought into a safe state without exceeding radiological limits at all times.

If development and engineering lead to a design where heat removal may be interrupted for a long time period without doing harm to the public, so-called "passive safety" is obtained. Progress in that direction is evident for the Advanced HTGR, HTR-Module and HTR-500.

5.4 Nuclear Reactor Containment

In the design requirements of the containment a high-pressure containment or a controlled vented containment are adequate. It should be stated that during external events
the containment shall remain both leak-tight and structurally intact, if it cannot be shown that the requirements of the Radiation Protection Ordinance /10/ with respect to radioactive releases is adhered to as a result of accident or operational leakage from a non-leak-tight structure, or

- the containment shall only remain structurally intact, if it can be shown that even without leak-tightness the requirements of the Radiation Protection Ordinance are met.

6. Applicability of the IAEA Standard "Design for Safety of Nuclear Power Plants, A Code of Practice" to HTGRs

On the international level, the Code of Practice "Design for Safety of Nuclear Power Plants" /11/ is published within the IAEA Safety Standards. This code can be adopted by countries interested in nuclear energy. The discussion, as to whether this is applicable to HTGRs without detailed interpretation can be restricted to five items, because the other parts of the standard cite fundamental protection objectives or contain plant non-specific requirements.

6.1 Provisions for In-Service Testing, Maintenance, Repair, Inspection and Monitoring (Section 2.9 of /11/)

In principle, here measures are required for in-service testing, maintenance, repair, inspection and monitoring of the functional capability of components. For some of the HTGR components in special
design a restricted accessibility applies. In section 2.9 of /11/ this restricted accessibility is taken into account by requiring "adequate safety precautions" to compensate for potential undiscovered failures.

6.2 Reactor Core (Section 4 of /11/)

The criteria for core and fuel design and reactor control and shutdown system layout are kept sufficiently general so that they can be applied also for HTGR power plants.

6.3 Reactor Coolant System
(Section 6 of /11/)

Section 6 of /11/ comprises design requirements for the residual heat removal systems and for the reactor coolant boundary including the pressure vessel. The requirements for the reactor coolant boundary are general so that they can be applied to HTGRs as well. However, for criteria more specific to HTGRs, it is better to differentiate between the enclosure of the coolant and the pressure bearing vessel for some HTR-designs as has been discussed above.

Section 6.6 of /11/ "Emergency Core Cooling" is LWR-specific in essential aspects so that interpretations for the application to HTGRs are necessary. Thus, section 6.6 requires an emergency core cooling system for the case of the loss-of-coolant accident in order to comply with the design value of the cladding temperature of the fuel elements, which is LWR-specific.

6.4 Containment System (Section 8 of /11/)

Within the requirements of section 8.1 of /11/ "Purpose of Containment System" it is possible to
have a vented or a hermetically sealed containment depending on other means of limiting the release of radioactive substances. These features are consistent with the requirements for an HTGR-containment within Chapter 5 of this paper. Effects of potential HTGR-specific energy sources on the containment structure, e.g. from reactions of air with graphite or the formation of combustible gases, are included within section 8.2 of /11/ "Containment Structure Strength".

6.5 Conclusion

The conclusion to be drawn is that the IAEA standard "Design for Safety of Nuclear Power Plants, A Code of Practice" is applicable to HTGRs without major difficulties.

References

/1/ Der Bundesminister des Innern, Sicherheitskriterien für Kernkraftwerke Bonn 1977

/2/ TÜV-Arbeitsgemeinschaft Kerntechnik West, Sicherheitskriterien für Anlagen zur Ernergieerzeugung mit gasgekühlten Hochtemperaturreaktoren Draft September 1980

/3/ Der Bundesminister des Innern, Sicherheitskriterien für Kernkraftwerke Draft 21 May 1984

/4/ HKG und BBC/HRB Planungsgrundlagen für die Errichtung des 300 MWe-THTR-Prototyp-Kernkraftwerks Hamm/ Uentrop as of: 22.06.1976
American Nuclear Society,
Nuclear Safety Criteria for the Design of Stationary Gas Cooled Reactor Plants
ANS-23 Subcommittee, Draft No. 9, Rev. 2, Illionois (Jan. 1974)

Code of Federal Regulations 10 CFR Part 50,
Draft Appendix A, General Design Criteria for High Temperature Gas Cooled Reactors,

King, T.L. et al.,
NRC-Review of the Advanced HTGR - A Status Update and Perspective
Eighth Annual International Conference on the HTGR, San Diego, Sept. 15 and 16, 1986

Knüfer, H.,
Abschaltvorgänge beim AVR-Hochtemperaturreaktor
Brennstof-Wärme-Kraft 26 (1974) Nr. 12, Pages 495 - 502

Kietzer, K., et al.,
Safety Analysis of the THTR-300 MW(e) - Prototype-Reactor and Future HTRs under extreme Accident Conditions
International Conference on Nuclear Power and its Fuel Cycle, Salzburg (1977)

Der Bundesminister des Innern,
Verordnung über den Schutz vor Schäden durch ionisierende Strahlen (Strahlenschutzverordn.-StrlSchV)
Bonn (1976/1982)

IAEA,
Design for Safety of Nuclear Power Plants, A code of Practice
Vienna (1978)
GAS-COOLED REACTOR APPLICATIONS
(Session E)
The high temperature reactor (HTR) is an advanced reactor concept: it is "unrestrictedly safe against catastrophes", can be used for electricity production and process heat applications and is economically attractive. It offers a broad variety of possible applications in the future energy market. The next step in the development is the systems demonstration, e.g. with AVR-extension.

1. HTR for world energy supply

Nuclear energy is established in the market for electricity production in many countries of the world. The main driving force is its economical competitiveness.

The high temperature reactor (HTR) is considered to be an advanced reactor concept. Of advance are the following two facts: 1) the HTR offers "unrestricted safety against catastrophes" and 2) it opens up a broad variety of applications in addition to electricity production: these are process heat applications. These applications will be described in the following from the development point of view. The users perspectives in electricity production and in coal refinement are described by Dr. Klusmann in this meeting, Session G, /1/.

2. Status of research, development and demonstration

The status of the work on research, development and demonstration, done since more than two decades on the HTR, can be described as follows, FIG. 1.
HTR R+D+D status

- electricity → commercialization
- process heat → semi technical pilot plant

next steps: systems demonstration (AVR Extension)

FIG. 1: Status of the Research and Development and Demonstration Work on High Temperature Reactors

For HTR electricity production the R+D+D-status has reached that of commercialization.

For HTR process heat applications the R+D+D-status has reached that of semi-technical and pilot plant demonstration of the main components. The next steps in R+D+D require the systems demonstration, e.g. via the extension of the AVR.

3. Unrestricted safety against catastrophes

The accidents in Chernobyl and Harrisburg have influenced and may further on influence the development and application of nuclear energy in the world to a large extend. The analysis indicates that a new quality in reactor safety may be necessary.

This new quality may be summarized as follows: catastrophic accidents should be impossible.

The R+D work on the HTR since Harrisburg and discussions after Chernobyl on advanced technology in nuclear energy, /2/, can be summarized as follows, FIG. 2: The HTR-system provides: by the development and the application of "inherent stabilizing qualities", e.g. the negative temperature coefficient of the reactivity and decay heat removal by thermal radiation and conduction:

"Unrestricted safety against catastrophes, even if mistakes are made by operators".
the HTR system provides:
by development and application of inherent stabilizing qualities
* unrestricted safety against catastrophes
* even if mistakes are made by operators.

FIG. 2: Fundamental Safety Qualities of the HTR-system:
unrestricted safety against catastrophes

The development and application of inherent stabilizing qualities have the potential to produce, FIG 3, 1) an enhancement of basic safety and - by the same effects - 2) a reduction of investment costs by a decrease of control systems, additional systems and auxiliary systems. This is an indication for the statement that economics and safety do not necessarily represent conflicting goals.

inherent stabilising qualities
produce
1) enhancement of basic safety
and - by the same effects -
2) reduction of investment costs

FIG. 3: Effects of Inherent Stabilizing Qualities:
economics and safety are not necessarily conflicting goals

4. Economical competitiveness of HTR

4.1 HTR electricity production

The results of recent studies by Arbeitsgemeinschaft Hochtemperaturreaktor, /3/ and /4/, can be summarized as follows, FIG. 4:
In the field of the production of electricity the HTR (medium and small size) is competitive compared to electricity on the basis of German hard coal in base load, and the HTR (mean size, single unit or multiple small units) is competitive (at about the same costs) compared to electricity from large PWRs, German conditions, and the HTR is potentially competitive compared to electricity from world market coal, base load. The fossil competitors have - of course - to be taken including additional investment for environmental reasons.

For indicating the cost situation for HTR electricity production the "cost-flow-diagram", following /3/, is given in FIG. 5. It indicates the following: from fuel elements for about 1 $/GJ high temperature helium is produced in the HTR-core at costs of about 3.3 $/GJ,
which than is converted in life steam in the steam generator to give about 4.4 $/GJ, which finally is converted in the turbomachine into electricity at about 13.6 $/GJ, respectively 5 $/kWhe ("real" cost evaluation method, 2.5 DM/$).

4.2 HTR process heat in general market

From the cost-flow-diagram for HTR-electricity-production it can be concluded, that HTR-heat in the form of high temperature helium (primary helium circuit) costs about 3.3 $/GJ, FIG. 6.

<table>
<thead>
<tr>
<th>Process Heat Cost Figures comparison</th>
</tr>
</thead>
<tbody>
<tr>
<td>HTR</td>
</tr>
<tr>
<td>High Temp</td>
</tr>
<tr>
<td>Helium (primary helium)</td>
</tr>
<tr>
<td>$/GJ</td>
</tr>
<tr>
<td>3.3</td>
</tr>
</tbody>
</table>

FIG. 6: Comparison of Cost Figures for Process Heat: HTR process heat and process heat from coal

In autothermal conversion and refinement processes the required process heat is taken from the fossil input. In most cases this heat production consists of a partial combustion of the fossil feed with oxygen. A rough calculation for coal, in particular for german hard coal respectively world market coal, FIG. 6, indicates, that process heat from these sources costs about 6.3 respectively 4.7 $/GJ, /5/.

The rough calculation of the costs for the fossil process heat include the costs of the fossil energy carrier, FIG. 7, costs for oxygen, separated from air, and costs for equipment. Right know, autumn 1986, the market prices in Germany for natural gas and oil are about the same as the world market price for coal.
Therefore, the cost evaluation for fossil process heat has about the same result, see FIG. 7.

In summary it can be concluded that the HTR-process is by a factor of about 2 competitive compared to German hard coal and by a factor of 1.4 competitive compared to world market coal as well as natural gas and oil.

An overview on cost figures of primary energy carriers is given in FIG 7. In the upper part it describes cost figures for primary energy carriers in autumn 1986 for the Federal Republic of Germany and Western Europe. In the lower part it describes cost figures for future primary energy sources as these may be of relevance for Western Europe.

The cost situation on the energy market is characterized by the sharp decrease of a price of oil: it costs now about 10 $/barrel, equivalent 1.8 $/GJ. A few years ago it has been more expensive by almost the factor of 4. For the future increases of the oil price are expected. An indication is given by the cost figures of enhanced oil recovery and even oil production from oil sands and oil shales, FIG. 7.
An overview on the broad variety of possible applications of the HTR in given in FIG. 8. The first branch is HTR-electricity production by the turbine cycle and in cogeneration as well as district heat. The second branch is called "process heat applications" and is divided into three categories:

1) HTR process heat application for recovery, conversion and refinement of fossil energy carriers,
2) HTR process heat applications via the chemical heat loop (EVA/ADAM-system) and
3) HTR process heat applications via the thermo-chemical cycle for the production of hydrogen and oxygen from water.

In the following three examples for process heat applications in combination with fossil energy carriers are given in overview. The R+D+D-work on the technology for this application is performed (and has been performed) in the projects "Prototype Plant Nuclear Process Heat (PNP)" and "Nuclear Long Distance Energy (NFE)". The results of this work are described by Dr. Klusmann, /1/.

4.3 Integration of coal, steel and nuclear energy

The energy conversion system "Integration of coal, steel and nuclear energy" is a concept for the production of methanol (energy alcohol) and pig iron on
the basis of German hard coal, iron ore (+ additional substances) and nuclear energy in the form of process heat from the HTR. These concepts combine three groups of processes, FIG. 9, in particular:

1) the steam coal gasification in partial gasification for the production of gas and coke,
2) the Klöckner-steel-gas-process for the reduction of iron ore in a converter for the production of pig iron and converter gas and
3) the joint application of the product gases from the gasifier and the converter for the production of methanol.

An economical evaluation has indicated that the system "Integration of coal, steel and nuclear energy" (in german: Verbund von Kohle, Stahl und Kernenergie) has the potential to be economically competitive compared with the market situation of 1984/85, FIG. 10. The integration can convert German hard coal at about 100 $/t (257,- DM/t) into methanol at 200 $/t and pig iron at 150,- (respectively 167,-) $/t, including (excluding) a subsidy on coke (Kokskohlenbeihilfe) given for blast furnace operation.

Methanol at 200 $/t is of interest as motor fuel and as heating fuel. As motor fuel methanol in 1984/85 seemed to be economically competitive compared to
high octan gasoline in Otto-cars. Meanwhile the oil price and therefore the gasoline price has decreased. The results of the economic evaluation can therefore be summarized as follows: economical attractiveness seems to be achievable at a price level for oil of about 25 $/barrel (and 2.5 DM/$) that means compared to oil from the North Sea, future Norwegian Shelf, /6/.

4.4 In-situ oil sand synthetic crude production

Another promising example for the application of process heat from nuclear energy, in particular HTR-process heat, is the in-situ oil sands synthetic crude production. The reference is the oil sand mining project "Syncrude" in Fort McMurray, Canada, in operation since a few years with production costs of about 19 $/barrel (1984).

For the future it is important that most of the oil sand is deeply buried. Therefore "in-situ"-production has been proposed, e.g. by steam injection.

The required injection steam can be produced by the HTR. Compared to other nuclear reactors the HTR can produce high pressure steam for almost all requirements feasible.

In addition to injection steam the HTR can also deliver heat for refinement processes and electricity.
for the production of hydrogen from water. Following these lines increasing amounts of nuclear energy can be integrated into the recovery, conversion and refinement of the overall process.

In a feasibility study, /7/, on the basis of the existing plant in Fort McMurray, but for in-situ-production by steam injection it has been shown, that the product yield could be increased from less than 70 % stepwise up to more than 130 %. At the same time the side production of CO₂ is reduced equivalently.

The fundamental process steps are described in FIG. 11. The HTR produces the injection steam, as well as process steam and electricity. The conversion consists of upgrading, flexicoking and hydrotreating. The final products are synthetic crude, methanol and purge coke.

4.5 Natural gas conversion to liquids

Natural gas is today and will be more in future an important energy carrier in the world. But today it is exported only to an extend of about 15% compared to the production. The reason is obviously the expensive transportation by pipe lines or as liquified natural gas.

Therefore the conversion of natural gas into liquids, e.g. methanol (may be together with ammonia) is a means to make it a "world trade energy carrier".
This conversion can be done by autothermal processes or by the application by HTR-process heat. The latter one would increase the product yield from about 60% to about 110%. An overview on the process for the synthesis of methanol from natural gas (together with carbon dioxide) is given FIG. 12.

An analysis on the market possibilities for remote fields of natural gas of limited content, /8/, shows the following results: for remote natural gas fields of limited content the conversion into "liquids" (methanol + ammonia) using HTR-process heat may be economically attractive, even if the production costs are higher compared to the autothermal process, because the increased yield produces a larger cash flow for the operator.

5. AVR-extension

As indicated in FIG. 1, the next step in the R+D+D-work for process heat applications is the systems demonstration of the coupling of high temperature helium from a nuclear reactor to the heat consuming components. Therefore it has been proposed by the KFA and others that the existing AVR-plant should be used for this system demonstration by an extension.

The AVR-reactor is (and has been) used for the production of steam which finally is converted into
electricity which is given to the grid. The outlet temperature of the reactor coolant helium is 950 °C.

The proposal "AVR-extension", /9/, and /10/, contains the following, FIG. 13: half of thermal power (and helium mass flow) is at 950 °C used for the demonstration of the coupling of nuclear helium to heat consuming components. These are the steam reformer (steam-methane-reforming), the steam-coal-gas-generator (steam-coal-gasification) and the intermediate heat exchanger.

**AVR for NUCLEAR PROCESS HEAT**

**Process Heat Loop 25 MW**

- Hot gas duct 950 °C
- Heat consuming components
  - Steam reformer
  - Gas generator
  - Intermediate heat exchanger
- Confinement


6. Future R+D-prospectives

Future perspectives for research and development work are opened up by the "better utilization of the temperature potential of the HTR". This potential will become applicable if R+D-work on ceramic materials, in particular on ceramic heat exchangers, will be successful.

Today the process temperature in the heat consuming processes is limited by the metallic heat transfer surfaces to about 900 °C. Both fundamental process heat reactions, the "steam-methane-reforming"-reaction and the "steam-coal-gasification"-reaction, would be posi-
tively influenced by an increase of 50 K - 100 K (or more) because of the following reasons:
1) better fitting of the HTR-heat-vector and
2) higher space velocity (in particular for coal gasification) and
3) higher reaction degree (with respect to the required products).

The specified maximum fuel temperature of the THTR fuel is 1250 °C. Between the specified process temperatures of 800 to 850 °C and 1250 °C there is a large potential for improvement of the processes, particularly with respect to economics, FIG. 14.

7. Results of this report
1) The HTR R+D+D-work has reached the state of commercialization for electricity production respectively the state of semi-technical and pilot plant demonstration of processes for process heat applications. For the latter the next step is the systems demonstration, e.g. by AVR-extension.
2) The HTR provides by the development and application of inherent stabilizing qualities "unrestricted safety against catastrophes", even if mistakes are made by operators.
3) This development offers - by the same effects - a reduction of investment costs. Economics and safety do not need to be conflicting goals.

4) The economic competetiveness of the HTR can be characterized as follows: in electricity production the HTR (medium and small size) is competetive compared to German hard coal, base load; and the HTR (medium size, single unit or multiple small units) is competetive (about the same costs) compared to large PWR, german conditions; and the HTR is potentially competetive compared to world market coal, base load.

5) From this it is concluded that in process heat applications the HTR (medium and small size) is potentially competetive compared to German hard coal and world market coal (as well as oil and natural gas). HTR process heat costs about 3.3 $/GJ, and is therefore by a factor of about 2 respectively 1.4 cheaper than fossil competitors. However, the market situation has changed due to the sharp decrease of oil prices.

6) The HTR offers a very broad variety of possible applications in electricity production and process heat applications. Three typical examples are described in the following:

7) The system "Integration of coal, steel and nuclear energy" for the production of methanol and pig iron, using HTR process heat has the potential to be economically competitive compared to the market situation of 1984/85. Economically attractiveness seems to be achievable at an oil price level of 25 $/barrel (and 2.5 DM/$) that means compared to oil from the North Sea, future Norwegian Shelf.

8) In the "in-situ oil sand syncrude production" (ref. syncrude production in Fort McMurray, Canada, in operation since a few years) the application of HTR-process heat offers the possibility of an increase of the product yield from about 70 % to (stepwise) more than 130 %.
9) Natural gas can be converted into "liquids" and thereby become a "world trade energy carrier". For remote fields of limited content the conversion using HTR process heat may be economically attractive, even if the products costs are higher compared to the autothermal process, because the increase of the product yield produces a larger cash flow.

10) The next step in the R+D+D-work for process heat applications is the systems demonstration of the coupling between high temperature helium from a reactor and heat consuming components. Therefore, the extension of the AVR has been proposed.

11) Future R+D-prospectives are open up by "the better utilization of the temperature potential of the HTR". This becomes feasible by ceramic materials for heat exchangers.

REFERENCES


A High Temperature Gas-Cooled Reactor (HTGR) can solve the problems in improving the economy and broadening the field of utilizing nuclear energy while maintaining safety, due to its specific characteristics of high temperature heat, inherent safety, and high burn-up. Considering the importance to use nuclear energy in non-electric fields in the future, this paper presents a recent study on the perspective of the HTGR in the chemical and iron & steel industries of Japan, conducted by the Japan Atomic Industrial Forum.

Although the heat generated by the HTGR will be expensive for use by the chemical and iron & steel industries in the rest of the 20th century, the chemical industry has many good processes which can apply nuclear heat when the cost is competitive compared with fossil fuels. Among them, the process for hydrogen production could be considered for major application. The iron & steel industry has a big potential for use of reducing gas produced by nuclear heat with sufficiently competitive cost.

1. TEMPERATURE RANGES AND HEAT QUANTITIES REQUIRED FOR PROCESS HEAT APPLICATIONS

The temperature ranges and the heat quantities required for typical processes in petroleum refining and petrochemistry are shown in TABLE I and TABLE II, respectively.
<table>
<thead>
<tr>
<th>Process</th>
<th>Feedstock</th>
<th>Process Temp. °C</th>
<th>Required Energy (Per Feedstock 1000 bbl)</th>
<th>Representative Plant Capacity (10⁶ bbl/SD)</th>
<th>Corresponding Thermal Output of HTR (MW) (Effectiveness 100%)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td>Fuel (10⁶ Kcal)</td>
<td>Steam (ton)</td>
<td>Electric Power (KWH)</td>
</tr>
<tr>
<td>Atomospheric Distillation</td>
<td>Crude Oil</td>
<td>250-350</td>
<td>17</td>
<td>24</td>
<td>22</td>
</tr>
<tr>
<td>Vacuum Distillation</td>
<td>Atomospheric Residue</td>
<td>350-430</td>
<td>20</td>
<td>20</td>
<td>20</td>
</tr>
<tr>
<td>Catalytic Reforming</td>
<td>Naphtha</td>
<td>470-540</td>
<td>108</td>
<td>41</td>
<td>1,530</td>
</tr>
<tr>
<td>Hydro-cracking</td>
<td>Naphtha-Vacuum Residue</td>
<td>260-450</td>
<td>40</td>
<td>4</td>
<td>10,000</td>
</tr>
<tr>
<td>Hydro-desulfurization</td>
<td>Naphtha-Lub Oil</td>
<td>260-430</td>
<td>20</td>
<td>64</td>
<td>1,100</td>
</tr>
<tr>
<td>Fluidized Catalytic Cracking</td>
<td>Light Oil-Heavy Oil</td>
<td>465-520</td>
<td>27</td>
<td>10</td>
<td>20</td>
</tr>
<tr>
<td>Visbreaking</td>
<td>Vacuum Residue</td>
<td>450-540</td>
<td>58</td>
<td>-0.2</td>
<td>1,080</td>
</tr>
<tr>
<td>Process</td>
<td>Feedstock</td>
<td>Process Temp. °C</td>
<td>Required Energy (Per Product 1 ton)</td>
<td>Representative Plant Capacity (10^6 t/y)</td>
<td>Corresponding Thermal Output of HTR (MWt) (Effectiveness 100%)</td>
</tr>
<tr>
<td>--------------------------</td>
<td>-------------</td>
<td>------------------</td>
<td>-------------------------------------</td>
<td>----------------------------------------</td>
<td>-------------------------------------------------------------</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Fuel (10^6 Kcal)</td>
<td>Steam (ton)</td>
<td>Electric Power (KWH)</td>
</tr>
<tr>
<td>Ammonia (Steam Reforming)</td>
<td>Naphtha-LNG</td>
<td>~ 780 ~ 800</td>
<td>3.6</td>
<td>-</td>
<td>14</td>
</tr>
<tr>
<td>Methanol (Steam Reforming)</td>
<td>Naphtha-LNG</td>
<td>~ 850 ~ 850</td>
<td>4.4</td>
<td>-</td>
<td>45</td>
</tr>
<tr>
<td>Ethylene (Thermal Cracking)</td>
<td>Naphtha-Ethane</td>
<td>~ 870 ~ 880</td>
<td>6.6</td>
<td>-</td>
<td>70</td>
</tr>
<tr>
<td>Propylene (Dehydrogenation)</td>
<td>C_3 LPG</td>
<td>~ 650</td>
<td>1.6</td>
<td>3.5</td>
<td>50</td>
</tr>
<tr>
<td>Butene Isobutyrin (Dehydrogenation)</td>
<td>C_4 LPG</td>
<td>~ 650</td>
<td>2.7</td>
<td>-2.1</td>
<td>-25</td>
</tr>
<tr>
<td>Vinyl Chloride Monomer (Oxychlorination)</td>
<td>Ethylene</td>
<td>~ 500</td>
<td>1.1</td>
<td>1.5</td>
<td>120</td>
</tr>
<tr>
<td>Styrene</td>
<td>Benzene</td>
<td>~ 500</td>
<td>1.5</td>
<td>0.4</td>
<td>88</td>
</tr>
<tr>
<td>Xylene Separation</td>
<td>Mixed</td>
<td>~ 550</td>
<td>5.4</td>
<td>0.2</td>
<td>270</td>
</tr>
<tr>
<td>Phenol</td>
<td>Benzene</td>
<td>~ 300</td>
<td>4.5</td>
<td>-0.4</td>
<td>305</td>
</tr>
</tbody>
</table>
The processes in petroleum refining shown in TABLE I require comparatively mild temperatures between 250°C-550°C and can use the heat from the HTGR. In the processes relating to petrochemistry shown in TABLE II, the processes requiring high temperatures (600-880°C), such as the steam reforming process for ammonia and methanol, and the dehydrogenation process for propylene and butylene can also use the heat from the HTGR.

Specifically, the thermal cracking process for producing ethylene needs a lot of energy and the process temperature is in the range of heat obtained from the HTGR. To increase the conversion ratio, however, it is necessary to increase the heat input rate by increasing the temperature of the heating medium up to about 1100°C because the reaction time is restricted for high selectivity.

Steam reforming also has a high process temperature level. The heating medium temperature of 950-1000°C is adequate because the restriction in reaction time of the reforming process is less severe than that of ethylene production by thermal cracking.

In summary, nuclear heat utilization is highly feasible for the following processes:

- Steam reforming process
- Petroleum refining process
- Process for producing propylene and butylene by dehydrogenation reaction of C₃, C₄ and LPG
- Process for producing C₁ derivatives

FIG. 1 shows an example of a coal chemistry system. The nuclear process heat in the high temperature ranges can be used for hydrogen production, ammonia synthesis, methanol synthesis and so on, as shown in the upper half of the figure. The processes in the lower half of the figure use low temperature heat.
As mentioned above, in the chemical industry, nuclear process heat in the high temperature ranges can effectively be used in the processes of production of hydrogen, reducing gas, ammonia, and methanol. On the other hand, the CIS process, a
water splitting process which uses CH₃OH, HI, and H₂SO₄ in its chemical reaction cycle, which is being developed by JAERI (the Japan Atomic Energy Research Institute) is one of the promising processes in producing hydrogen by thermochemical water splitting.

Since all these processes have a steam reforming unit as a key unit it is felt necessary to extend the development of the helium-heated steam reformer (1), (2), (3), (4) carried out as part of the National Project of the Nuclear Steelmaking (5), (6) and proceed to make a demonstration unit so that future nuclear heat utilization can be realized.

2. POSSIBLE APPLICATIONS IN CHEMICAL INDUSTRIES

If the heat exchanger type steam reformer, which is the interface between the HTGR and chemical plant, is demonstrated on a commercial scale, the production processes for ammonia, methanol and oxosynthesis gas will have a good chance of realization with the reformer.

In this case, the primary helium outlet temperature from the HTGR largely affects the reforming efficiency and the nuclear heat utilization factor. The factor decreases with the helium outlet temperature from the HTGR as shown in TABLE III. It also shows that the factor changes with the secondary helium temperature returning to the intermediate heat exchanger (IHX). As shown in the table, raising the secondary helium temperature returning to the IHX, if the design condition of the HTGR allows it, is worth taking into consideration.

Nuclear heat utilization for endothermic reaction processes in the temperature range lower than that of the steam reformer, such as the processes of producing propylene and butylene by thermal cracking of saturated hydrocarbons, is also promising, especially when using these materials as feedstock in a series of petrochemical plants for derivatives.
The following, however, are the problems that need to be solved for these plant configurations:

- A fundamental rearrangement of energy balancing is needed for introducing nuclear heat because energy-saving, effective use of heat, or improvement of consumption rate (by combustion of off-gas or purge gas in heating furnaces) is highly achieved in present chemical plant design.
- If chemical plants are to be regulated as nuclear facilities, different factors, such as periodic inspection, decrease in plant availability, safety design, and aseismic design will affect economy, making the present chemical plants uneconomical in design and operation.

3. POSSIBLE APPLICATIONS IN IRON & STEEL INDUSTRY

Annual production of steel when the National Project of Nuclear Steelmaking was started was 120 million tons, but the present figure is hovering around the 100 million-ton level. This is because the double oil shock has changed the progress of the nation's economy to stable growth. And further, because of the change in industrial background and efforts in energy saving, energy consumption in the iron & steel industry has remarkably decreased. For these reasons, although it depends on high grade coal and iron ore, iron making by the blast furnace will play a primary role for the time being.

The most feasible way of applying nuclear heat in the future iron & steel industry will be utilization of secondary energy from a secondary energy center instead of exclusive use of nuclear heat for the iron & steel industry. In this case, competitive price of energy compared with fossil fuels and a stable supply of abundant energy are the keys to realization.

4. FORMS OF SITING FOR HTGR AND PROCESS PLANTS

When using nuclear heat from the HTGR in process plants, generally, the following two forms of siting are considered:

- Adjacent Siting
- Independent Siting

In the former siting, the HTGR is located adjacent to the process plant to supply high temperature helium and steam directly by a piping system. In the latter siting, after nuclear heat from the HTGR is transformed into a secondary energy form (e.g. hydrogen energy), it is transferred or transported to remotely located process plants and used independently.
(1) ADJACENT SITING

The distinctive features of adjacent siting are as follows:

- Energy loss accompanied with energy conversion and heat transmission can be minimized and economy as a whole is improved.

- Nuclear heat can be utilized in different forms of supply (process heat, high temperature steam, electric power, etc.) in response to the user's energy demand.

- When organizing a group of complexes consisting of different processes, the utilization of nuclear heat covering a wide range of temperatures from high to low can be achieved reasonably and efficiently. Also, it becomes possible to make a concise plant layout with the HTGR located in the center of the complexes.

However, an investigation is necessary on the following points:

- The scheme of coupling the nuclear facilities and the chemical plants with a piping system on the same site requires strict verification of safety and reliability.

- As the HTGR is regarded as utility facilities for the chemical plants, load following operation that responds to chemical plant operation is required for the HTGR.

- The problem of conformity between the nuclear facilities and the chemical plants arises in the laws and regulations.

(2) INDEPENDENT SITING

The distinctive features in isolating the process plant from the nuclear facilities are as follows:

- Safe decoupling can be expected.

- The consideration of energy loss during transfer or transportation of the converted secondary energy medium is not necessary.
- Depending on the forms of the converted energy selected, a variety of combinations of chemical plants are possible.

- In addition, provision of a secondary energy center which supplies hydrogen, reducing gas, methanol, or electric power is possible. Especially, the transformation into methanol as a secondary energy is promising from the viewpoint of transportation and utilization.

5. CAPACITY OF HTGR FOR PROCESS HEAT APPLICATIONS

Considering the controllability of the HTGR, the possibility of interruption of the nuclear heat supply for regular inspections or shutdowns owing to causes from the chemical plant side, or the adaptability to the unplanned increase of production capacity in the future, the modular scheme, with an HTGR of appropriate scale, is superior to an installation with a large HTGR.

In this case, the most appropriate capacity for one HTGR module can be determined from the HTGR's own safety features as well as the operational factors, such as the production capacity of the chemical plants.

The HTGR is recognized as having many safety advantages, but for use in the chemical industries, intrinsic safety and passive safety without active components will strongly be required from the safety point-of-view. The capacity of one HTGR module is determined from the viewpoint of effectiveness in passive safety.

6. COMMENTS ON ECONOMIC EVALUATION OF NUCLEAR PROCESS HEAT APPLICATIONS

Regarding nuclear heat as a substitute for heat from a heating furnace in the chemical industries, an economic comparison by using the cost per unit heat is made as follows:

- Conventional method (Use of fossil fuel)
  Fixed cost of heating furnace + Fuel cost + Other operation cost
- Nuclear heat utilization method
  
  Fixed cost of HTGR and primary heat exchanger + HTGR operation cost + Fixed cost of process heaters + Other operation cost

  Based on the assumption that the fixed cost of process heaters using nuclear heat is equal to the fixed cost of the heating furnace, an economic comparison is made between the cost per unit heat quantity of fossil fuel and the fixed cost of the HTGR and the primary heat exchanger including the operation cost of the HTGR.

  TABLE IV shows a forecast of the fossil fuel costs. For the HTGR to be competitive, the (fixed cost of HTGR and primary heat exchanger + HTGR operation cost) per unit heat quantity must be less than the value of the fossil fuel shown in TABLE IV.

  TABLE IV  Forecast of Fossil Fuel Cost with 7% Escalation

<table>
<thead>
<tr>
<th>Classification</th>
<th>1984</th>
<th>2000</th>
<th>2020</th>
<th>2040</th>
<th>2060</th>
</tr>
</thead>
<tbody>
<tr>
<td>Imported Coal</td>
<td>1.8</td>
<td>5.3</td>
<td>20.6</td>
<td>79.6</td>
<td>308</td>
</tr>
<tr>
<td>Naphtha</td>
<td>5.4</td>
<td>15.9</td>
<td>61.7</td>
<td>239</td>
<td>924</td>
</tr>
<tr>
<td>Kerosene</td>
<td>7.7</td>
<td>22.7</td>
<td>88.0</td>
<td>340</td>
<td>1,320</td>
</tr>
<tr>
<td>Heavy Oil (A)</td>
<td>7.0</td>
<td>20.7</td>
<td>80.0</td>
<td>309</td>
<td>1,200</td>
</tr>
<tr>
<td>L P G</td>
<td>6.7</td>
<td>19.8</td>
<td>76.5</td>
<td>296</td>
<td>1,150</td>
</tr>
<tr>
<td>L N G</td>
<td>7.5</td>
<td>22.1</td>
<td>85.7</td>
<td>332</td>
<td>1,280</td>
</tr>
<tr>
<td>Electric Power</td>
<td>20.2</td>
<td>59.6</td>
<td>230.8</td>
<td>893</td>
<td>3,460</td>
</tr>
</tbody>
</table>

(Yen/1000 Kcal)
7. CONCLUSION

This paper presents a recent study on the perspective of HTGRs in the chemical and iron & steel industries of Japan. As for future applications, the following are concluded:

- A secondary energy center which supplies secondary energy such as hydrogen, reducing gas, methanol, and electric power is feasible. Especially, the transformation into methanol as a secondary energy is promising from the viewpoints of transportation and future utilization.

- When C₁ chemistry and aromatic chemistry, which are in a basic study stage at the present time, achieve the prospect of commercialization, nuclear heat utilization could extend to the whole petrochemical industry and the concept of a petrochemical center will be feasible.

- The method of producing hydrogen by decomposing water with the nuclear heat, will become promising in the near future, and it is considered that the chemical industry will develop a new configuration using hydrogen energy which will be a most important innovation.

- The most feasible way of applying nuclear heat in the iron & steel industry is utilization of secondary energy from a secondary energy center. In this case, the competitive price of this energy compared with the conventional method and the stable supply of abundant energy are the keys to realization.

REFERENCES


(2) T. MIYASUGI, S. YOSHIOKA, S. KOSAKA and A. SUZUKI, "Experimental Study of a Heat-Exchanger Type Steam Reformer with a Low Steam/Carbon Ratio. Effect of Carbon Deposition


ISRAELI PERSPECTIVE ON HTGRs

A. BARAK, A. BECK, E. GREENSPAN, J. SZABO
Atomic Energy Commission,
Tel-Aviv

L. BLUMENAU, H. BRANOVER, A. EL-BOHER,
E. SPERO, S. SUKORIANSKY
Ben-Gurion University of the Negev,
Beer-Sheva

Israel

Abstract

A preliminary assessment of the interest Israel should have in the HTGR technology, and of possibilities for improving the efficiency of HTGRs using the Liquid Metal MHD (LMMHD) energy conversion technology is undertaken. Modular HTGRs were found to offer a collection of features which make them attractive in the Israeli conditions. These features include inherent safety, resistance against acts of war, investment protection, suitability to the Israeli seismic conditions and to dry cooling, possibility for optimal development of the Israeli grid and, hopefully, for economical sea-water desalination and oil-shale utilization.

Being free of rotating machinery and being applicable over the entire temperature range of HTGR energy delivery, the LMMHD technology opens new promising possibilities for utilizing the high temperature operation - ability of HTGRs for the production of electricity. A number of novel LMMHD cycles were conceived and attractive matches to the unique HTGR heat source characteristics were identified; they enable producing electricity with efficiencies which exceed 70% of the ideal efficiencies even in the very high temperature domain. For example, the LMMHD technology might enable attaining net efficiencies of 43% and 47% in, respectively, the first and second generation of HTGRs as compared with efficiencies of 37% to 40% attainable with conventional steam energy conversion technology.
1. INTRODUCTION

Of the several advanced nuclear reactor concepts emerging in recent years, modular High Temperature Gas Cooled Reactors (HTGRs) attracted particular attention in Israel. This is due to a collection of a number of interesting features, including the following: inherent safety, small unit size, adaptability to dry cooling and potential for a variety of industrial applications in addition to the production of electricity.

Consequently, the Israel Atomic Energy Commission (IAEC) undertook a preliminary assessment of the interest Israel should have in the HTGR technology. In parallel, the IAEC encouraged the Center for Magneto-Hydro-Dynamic (MHD) Studies (CMHDS) of the Ben-Gurion University (BGU) to investigate possibilities for improving the efficiency of HTGRs for the production of electricity, capitalizing on the high temperature operation ability of both the HTGR and the Liquid Metal MHD (LMMHD) energy conversion technologies.

The present paper reports upon the preliminary findings obtained so far. It consists of two distinct parts: the first part (Sec. 2) refers to the IAEC perspective on HTGRs whereas the second part (Sec. 3) reports upon the preliminary findings from the study of the promise of the LMMHD energy conversion technology for HTGRs.

2. THE INTEREST ISRAEL MIGHT HAVE IN MODULAR HTGRs

2.1 Implications on the Development of the Israeli Electricity Generation System

The expansion of the electricity generation system in Israel is more problematic than in many other countries due to its (1) Relatively small size - the maximum peak demand is around 2600 MWe, and (2) Isolation - so far Israel can not buy electricity from its neighbours, in case of a sudden generation shortage. These constraints impose penalty on the reserve
generating capacity that need be installed in the country and, hence, on the
cost of electricity in Israel.

Modular HTGRs (MHTGR) might enable expanding the Israeli grid
with minimum over-capacity. This is due to the following reasons:
(1) Minimum overshooting of the total reserve capacity needed.

Consider, for example, the two options that were recently considered
for further expansion of the ~4000 MWe Israeli grid: adding 550 MWe coal
fired units or 950 MWe LWRs - the smallest size LWR of proven design
commercially available. As the average increase in the peak demand of
electricity in the mid-nineties is expected to be ~150 MWe per year, the
average overcapacity (beyond the reserve capacity) can be of the order of 200
MWe and 400 MWe for, respectively, the coal and nuclear power plants. If
the total installed capacity in the mid-nineties will be ~5000 MWe, an extra
400 MWe corresponds to an average capital cost penalty of up to ~8%.
(2) Reducing the installed reserve capacity needed.

The installed reserve capacity - aimed at replacing long-term outages,
depends on the largest size unit installed in the grid. Thus, if the 950 MWe
LWRs were to be installed in Israel, the installed reserve capacity would have
to be at least 950 MWe. If, on the other hand, MHTGRs will be installed, the
reserve capacity need be only ~550 MWe - the size of the largest coal plant in
the system. The extra 400 MWe required for the reserve corresponds to an
additional capital cost penalty of up to ~8%.
(3) Reducing the non-spinning reserve capacity.

This reserve capacity is to take care of short outages (of the order of a
fraction of an hour) - it consists of inexpensive units such as gas turbines.
The smaller the largest generating unit size, the smaller need be the
non-spinning reserve.
(4) Reducing the spinning reserve.

This reserve is to take care of outages for very short periods (of the
order of a few minutes) - it is obtained by operating a fraction of the units at
partial load. Partial load operation usually involves cost penalty due to reduced efficiency. The smaller the largest generating unit size, the smaller this penalty is expected to be.

(5) Improving the grid's load following capability.

The need for a spinning reserve forces at least a fraction of the generating units to operate at least a fraction of the time at below rated capacity. Due to their relatively low operation costs, nuclear units are to operate at the nominal capacity. Hence, the larger the fraction of the installed nuclear generating capacity in the grid, the larger becomes the load reduction of the fossil fuel fired power plants. Since the electricity generation efficiency decreases faster than the load reduction, the smaller the overcapacity of nuclear generating units the more efficient, on the average, the electricity generation is expected to be.

Additional economic attributes of MHTGRs include the following:

(6) Short installation period.

Obviously, the shorter the installation period the lower is the interest during construction.

(7) Reduced uncertainty in commissioning time.

Standardization, factory fabrication and shorter fabrication and construction time expected for the MHTGRs reduces the probability that the power unit will be untimely commissioned - either because of an inaccurate demand forecast or due to delays in installation. Late commissioning may impose a penalty in the form of shortage in electricity supply. Too early a commissioning will impose a capital cost penalty.

(8) Increased availability.

Generally speaking, the higher the availability the smaller need be the installed capacity. On-line refuelling, low radiation levels, system simplicity, standardization and small size components are expected to make the availability of HTGRs significantly higher than of conventional LWRs.
Taken all together, it appears that MHTGRs might enable developing the Israeli grid in an optimal way. The cumulative economic benefit resulting from the above outlined attributes can be very significant. Whether or not this benefit will make MHTGRs economically competitive with other types of power sources in Israel is an open question, as yet.

2.2 Safety implications and investment protection

The inherent safety aspect of MHTGRs was borne out by a number of safety studies [1-5] which considered "internally" initiated accidents (i.e. due to the malfunction or failure of components, or to operator errors) as well as to certain "externally" initiated accidents, such as loss of offsite power and earthquakes.

As part of the assessment of the interest Israel should have in HTGRs, we set to examine the implications, on the safety of modular HTGR's, of yet another type of "externally" initiated accidents namely, accidents resulting from acts of sabotage or acts of war. Also examined are ways to reduce the damage by such acts and their consequences. The assessment is done with reference to the MHTGR power plant proposed in Ref. 6, in which the NSSS is located underground, inside a silo.

The preliminary assessment carried-out on the standard underground design, modified to have some additional, relatively inexpensive sheltering features, could not identify any credible act of sabotage or war which could bring about excessively high (i.e. >1600°C) fuel temperature and, hence, a significant release of fission products.

Turning, next, to the issue of investment risk, it is clear that having a relatively small capacity per module, the modular design is significantly more investment risk resistant than an integrated design. Moreover, a preliminary assessment indicates that by relatively simple and inexpensive design modifications it should be possible to make the MHTGR's NSSS practically undamageable by any conceivable act of sabotage or act of war. The above
features are of great significance for the acceptance of nuclear power reactors in Israel.

2.3 Siting implications

The present population density precludes the possibility for siting conventional LWRs in the Mediterranean coastal zone in Israel. However, the combination of inherent safety, relatively small unit size and resistance to acts of sabotage may enable siting MHTGRs by or near the Mediterranean coast. If coastal sites will be accepted, MHTGRs may have a significant economic benefit (See Sec. 2.4) on conventional nuclear power plants, such as LWRs, which may not be accepted for siting in the relatively densely populated coastal zone.

Preliminary seismological studies performed on a number of possible sites in Israel indicate that nuclear power reactors will have to be designed to withstand a relatively high g-factor. Recently it was proposed that the California seismic standards be adopted for Israel. This is due to the many similarities between these two regions: (1) The California seismology is dominated by the San Andreas fault while in Israel the Dead Sea rift is the dominant factor. (2) The San Andreas fault runs about 100 km inland parallel to the Pacific shoreline throughout California while the Dead Sea rift runs along Israel parallel to the Mediterranean shoreline. (3) Many minor but potentially hazardous faults branch out from both the San Andreas and the Dead Sea rift, and (4) In both cases there is a significant amount of uncertainty related to the earthquake frequency and the damage potential associated with each minor fault line.

It has been reported [6] that MHTGRs designed to have their NSSS properly supported inside a silo located underground can meet the California seismic standards (of 0.5 g horizontal acceleration) with, essentially, no capital cost penalty. Standard LWRs, on the other hand, are designed to withstand a horizontal g-factor of 0.15 to 0.25. Upgrading their design for
withstanding a horizontal ground acceleration from 0.25 g to 0.5 g is expected to increase the LWR capital cost by as much as 6%.[7]

2.4 Adaptability to Dry-Cooling

Due to the relatively dense population along Israel's Mediterranean coast, coastal and near coastal sites are not likely to be available for nuclear power plants in Israel. The other major water bodies in Israel are located along the Dead Sea rift, and none of them can offer acceptable sites. Consequently, unless a major water transport system (such as the proposed canal to connect the Dead Sea with the Mediterranean) is constructed, the only practical way to discharge the heat from nuclear power stations in Israel appears to be via cooling towers.

Dry, wet and dry-wet cooling towers are being considered in Israel. However, with the increasing shortage of water Israel is facing, it might be in the national interest to spare the water for applications for which there is no substitute for water. Consequently, a very preliminary assessment of the adaptability, to dry cooling, of HTGRs vs. LWRs under the Israeli conditions was undertaken. The adaptability of HTGRs to wet and dry-wet cooling towers will be examined in the future.

Relative to direct, once-through cooling, dry cooling penalizes the economics of power plants in Israel in a couple of ways: (1) By reducing the attainable efficiency - as a result of increasing the steam condensation temperature, and (2) By increasing the capital and operating costs - as a consequence of the capital investment in the cooling towers and of the power consumption for air circulation.* Table I summarizes our preliminary estimates of the penalty thus imposed on LWRs and two generations of HTGRs. The two generations are characterized by a helium inlet/outlet

* Natural drought dry cooling towers appears to be impractical under Israeli conditions.
TABLE I. Preliminary Estimates of the Effect of Dry Cooling on the Efficiency and Economics of Nuclear Power Reactors in Israel.

<table>
<thead>
<tr>
<th>Characteristic</th>
<th>LWR</th>
<th>HTGR 250/700°C</th>
<th>HTGR 550/950°C</th>
</tr>
</thead>
<tbody>
<tr>
<td>Net Efficiency (%)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Coastal site; sea cooling</td>
<td>32</td>
<td>40</td>
<td>47</td>
</tr>
<tr>
<td>Inland; dry cooling</td>
<td>31.1</td>
<td>39.2</td>
<td>46.5</td>
</tr>
<tr>
<td>Relative efficiency loss</td>
<td>2.9%</td>
<td>1.9%</td>
<td>1.1%</td>
</tr>
<tr>
<td>Installed capital cost increase ($/KWe)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Due to efficiency loss</td>
<td>60</td>
<td>36</td>
<td>18</td>
</tr>
<tr>
<td>Cooling towers + energy needs</td>
<td>400</td>
<td>282</td>
<td>212</td>
</tr>
<tr>
<td>Total</td>
<td>460</td>
<td>318</td>
<td>230</td>
</tr>
<tr>
<td>Installed cost relative increase</td>
<td>23.0%</td>
<td>15.9%</td>
<td>11.5%</td>
</tr>
<tr>
<td>Cost of electricity increase (%)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Due to efficiency loss</td>
<td>2.9</td>
<td>1.9</td>
<td>1.1</td>
</tr>
<tr>
<td>Due to installed cost increase</td>
<td>16.9</td>
<td>9.6</td>
<td>6.5</td>
</tr>
<tr>
<td>COE relative increase</td>
<td>19.8%</td>
<td>11.5%</td>
<td>7.6%</td>
</tr>
</tbody>
</table>

The preliminary assessment indicates that the economic penalty to HTGRs associated with the use, in Israel, of dry cooling can be 60% to almost 40% of the penalty to LWRs. Moreover, if the second generation HTGRs will be used for process heat applications (say for synthetic fuel
production (see Sec. 2.5) in addition to electricity generation, their cost penalty is expected to be significantly less than 40% that of LWRs. This is due to the very large fraction (approaching, possibly, 90%) of this HTGR energy which can be utilized for useful purposes.

If either gas-turbine technology or LMMHD energy conversion technology (see, for example, Sec. 3.4.3) be developed to become an attractive match to HTGRs, dry-cooling will impose very little penalty, if at all, on the efficiency attainable from HTGRs. This is due to the relatively high temperature for heat-rejection from the corresponding gas cycles.

2.5 Israeli oil-shale utilization

The main organic fuel resources of significant economic potential discovered so far in Israel are oil-shales. The presently proven reserves are 10 billion tons. The organic material contents in the Israeli shales is in the vicinity of 15% - significantly lower than in rich deposits (found, for example, in Colorado), but comparable to the quality of shales found in many parts of the world (such as Australia, Morocco, China and the Eastern U.S.A.). The total amount of oil which could be extracted, using conventional methods, from the $10^{10}$ tons of Israeli shales is 600 million tons. At the present level of energy consumption in the country, this corresponds to a potential for supplying the total national energy needs for a period of approximately 75 years.

Conventional methods for extracting liquid and gaseous fuel materials from oil-shales call for burning part of the organic material of the shale in order to supply the heat required for the process. Rather than burning organic material it is possible to provide the high temperature (i.e., $\geq 800^\circ$C) heat required for the production of fuel from shales from HTGRs. A number of studies of the feasibility of so doing were performed both in the U.S.A. [8-10] and in Germany.[11] A general conclusion of these studies is that with HTGRs as the primary heat source, it might be possible to increase the
amount of fuel extractable from a given quantity of oil-shales by as much as 
\(1/3\), and to do so at a comparable (in certain cases even somewhat lower) 
cost and with significantly lower adverse environmental effects than with the 
conventional approaches.

The prospects for increasing the liquid and gaseous fuel extractable 
from the limited resources of oil-shales in Israel by \(1/3\) is, of course, very 
appealing. A crucial question is whether this could be done, in the Israeli 
conditions, with no, or with relatively small economic penalty. This question 
will be addressed in the future.

The largest and first to be exploited oil shale field - the Rotem field, is 
located in the dry planes of the northern Negev district. Siting of HTGRs in 
these planes appears to be quite attractive - from the point of view of distance 
from population centers, and quite acceptable - from the point of view of 
seismic (relative high g-factors) and dry-cooling design requirements. As 
discussed in Secs. 2.3 and 2.4 the economic penalty these design 
requirements impose on HTGRs is relatively small.

2.6 Adaptability to Sea-Water Desalination

Israel suffers today from a severe shortage of water; the average annual 
consumption exceeds the natural annual supply by a few hundred m.cu.m. 
(millions of cubic meters). This water overdraft has accumulated to about 
1500 m.cu.m. - the consumption equivalent of about 10 months, causing 
overpumping from the aquifers and the Sea of Galilee. This overpumping 
brought the water table down by a few meters, to a dangerous level. 
Desalination of sea-water is one of the major possible solutions to the water 
shortage problems.

The two seawater desalination processes that seem most promising to 
the professional community in Israel are the Multi-Effect Distillation (MED) 
combined with power generation and the Reverse Osmosis (RO).
2.6.1 Multi-Effect Distillation

The MED process has been developed by the Israel Desalination Engineering (I.D.E.) company in the last 19 years [12]. The driving force for this evaporation process is low grade heat; the upper practical temperature for heat supply - about 75°C, is dictated by material compatibility considerations. The most economical source for such heat in Israel is the exhaust steam from power-plants.

A preliminary assessment of the promise of HTGRs for sea-water desalination revealed a number of potential advantages over LWRs as the heat source for MED:

1) Direct coupling to the MED plant.

As the pressure in the primary (or reactor) cooling system of a PWR is higher than the pressure in the secondary cooling (or turbine) system, any leak in the steam generators is bound to cause water from the reactor system to leak into the turbine system. Since the reactor system water may be radioactive a direct coupling of the turbine system steam with the MED plant is unacceptable. Instead, the heat from the turbine cycle is transferred to the desalination plant via a pressurized sea-water loop which interfaces with the turbine loop through a conventional condenser.

A different situation exists in HTGRs. As the primary helium system pressure is significantly lower than the secondary coolant (i.e., steam) pressure, a leak in the steam generators is not likely to contaminate the turbine steam. Consequently, it might be possible to directly couple the turbine exhaust steam with the MED plant. Being free of the condenser, huge circulation pump and flashing unit, such a direct interfacing is expected to be simpler and cheaper than in the case for LWRs, and to save the pumping energy.

2) Higher efficiency.

The higher the driving steam temperature, the lower is the energy consumption and cost of desalination in MED plants. Conventional turbines
can operate with a backpressure as high as 5.5" Hg, corresponding to a steam exit temperature of 58.5°C. This is the steam temperature HTGRs can provide to the MED plant. If LWR is the power source, 5.5" Hg condensation pressure is equivalent, due to the indirect coupling to the MED plant, to a ~54°C driving steam, resulting in a reduction of almost 20% in the water production per unit of the turbine exhaust steam.

(3) Using gas-turbine with high heat rejection temperature.

HTGRs may attain an additional significant efficiency gain by the use of gas turbines instead of steam turbines. HTGR-GT reject their unused heat at a relatively high temperature. This reject heat could provide ~75°C steam to the MED plant with no or very small penalty on the electricity generation efficiency. The 75°C steam can produce twice as much desalinated water as produceable by the same quantity of LWR steam.

(4) Higher availability.

The economics of desalination favors high availability and high capacity factor. As discussed in Sec. 2.1, MHTGRs are expected to have higher capacity factors than LWRs.

(5) Coastal siting

As discussed in Sec. 2.3, it is much more likely that MHTGRs could be sited by the Mediterranean coast than LWRs. Coastal or near coastal location has a significant economic attribute for sea-water desalination.

2.6.2 Reverse Osmosis

The reverse-osmosis desalination process is energized entirely by electricity. It can produce 130 m.cu.m. per year per 100 MWe. Thus 700 MWe can provide about 50% of the present water consumption of Israel. As the RO units are suitable for load shedding, the electricity production capacity destined for driving the RO plants could serve as the spinning as well as non-spinning reserve capacity for the Israeli electricity grid, and possibly even as part of the installed reserve capacity (see Sec. 2.1). Such a multi-purpose
scheme is likely to improve the overall economics of electricity generation and water desalination in Israel. Whether or not this scheme could make water desalination economical in Israel is an open question, as yet.

2.7 Conclusions

The preliminary assessment indicates that modular HTGRs offer a collection of features which might be very attractive for the Israeli conditions. Among the potential attributes of MHTGRs identified are the following:

1. Very high level of public safety.
2. Resistance against acts of war and sabotage with negligible economic penalty.
3. Better investment protection ability than conventional nuclear reactors.
4. Possibility of meeting the demand for electricity with minimum overcapacity and minimum reserve capacity.
5. Better suitability than conventional LWRs to the seismic conditions in Israel.
6. Better suitability than LWRs to dry-cooling in Israel.
7. Higher likelihood for coastal siting than conventional nuclear reactors.
8. Higher efficiency than conventional LWRs as an energy source for sea-water desalination.
9. Potential for significant improvement in the utilization of the Israeli oil-shale resources.

All combined, these attributes might significantly improve the economics of MHTGRs in Israel. Whether or not this improvement will make MHTGRs economically competitive with other power sources is, however, not clear as yet. What is clear is that the inherent safety feature of MHTGRs, combined with their resistance to acts of sabotage and war, small unit size and adaptability to siting in the sparcely populated parts of the arid Negev district...
give an MHTGR type reactor the best chance of being accepted in Israel in the post Chernobyl era.

3. POSSIBILITIES AND IMPLICATIONS OF LIQUID METAL MHD ENERGY CONVERSION IN HTGRs

3.1 Introduction

Unique features of HTGRs, as far as their energy conversion is concerned, is that (a) their helium coolant can be provided at very high temperatures - possibly exceeding 950°C, and (b) they deliver their energy over a relatively very large temperature range - up to 650°C in the AVR.[13] The latter feature is a consequence of two constraints: (1) Relatively low volumetric heat capacity of a gaseous coolant and, in particular in the German HTGR design approach,[13] (2) The need to maintain, with the primary coolant, the metal structural components of the reactor at below ~300°C.

HTGRs destined for power production are usually designed to have a conventional superheated steam Rankine cycle for their Power Conversion System (PCS), yielding a net efficiency approaching 40%.[13] The cycle upper steam temperature of ~550°C, set by materials strength limitations, is well below the helium outlet temperature attainable. Thus, the conventional Rankine cycle does not make a good thermodynamic match to HTGRs. Moreover, the probability for water ingress from the secondary water system into the primary system is a safety hazard.

Gas Turbines (GT) were proposed [13] for utilizing the above 550°C operation ability of HTGRs for power production. The predicted efficiency of HTGR-GT was 40%-41%, pertaining to a helium coolant outlet temperature of 850°C imposed by material strength limitations. A significantly higher efficiency of ~47% has been calculated for HTGR-GT operating with a binary cycle consisting of a gas-turbine (i.e., Brayton) topping cycle and an organic coolant Rankine bottoming cycle.[13]
Despite the inherent simplicity of the direct-cycle HTGR-GT and to the high efficiency expected from HTGR-GT having a binary cycle, the development of HTGR-GT has been abandoned both in Europe and in the U.S.A. due to technological and safety difficulties encountered, including: [13] turbomachine integrity; lubrication oil ingress; seal design; primary system acoustic level; materials and maintenance considerations. All of these difficulties are associated with the rotating nature of the GT technology.

Being free of rotating machinery and being applicable over the entire temperature range of HTGR energy delivery, the LMMHD technology opens new possibilities for utilizing the high temperature operation-ability of HTGRs for the production of electricity. The purpose of this section is to report upon preliminary results from the search for possibilities of improving the efficiency of HTGRs using the LMMHD energy conversion technology.

Starting with a brief description of the principles of LMMHD energy conversion (Sec. 3.2) we describe a number of novel LMMHD components and cycles (Sec. 3.3). Section 3.4 describes a number of possibilities for matching PCS, based on these and other LMMHD cycles, with HTGRs and estimates the attainable efficiencies. Other implications of coupling the LMMHD energy conversion technology with HTGRs are discussed in Sec. 3.5.

### 3.2 Principles of LMMHD energy conversion

The principle of LMMHD energy conversion is illustrated in Fig. 1 which shows an MHD generator of DC-Faraday type. This generator consists of a channel through which an electrically conducting fluid flows perpendicular to a magnetic field, creating an induced electric field. By connecting the electrodes to a load, a closed path is created through which current can flow. The power produced by the MHD generator is the product of the load voltage and the current. As gases have very low conductivities, they must be heated to about 2000°K and seeded with materials having low ionization energies to achieve adequate conductivities for use in an MHD...
generator. For example, helium seeded with 0.45 atom % of cesium will have a conductivity of roughly $10(\Omega m)^{-1}$ at 2000°K and 1 atm. By contrast, liquid metals have conductivities of the order of $10^6 - 10^7 (\Omega m)^{-1}$ at low temperature. This implies that LMMHD generators use significantly lower flux density and cost less than their plasma MHD counterparts. Moreover, LMMHD generators can be driven by HTGRs whereas plasma MHD generators cannot.

### 3.3 Novel LMMHD cycles and components

A description of the LMMHD approach to energy conversion and a review of the evolution of this technology since its inception in the late 1950s, can be found in Ref. 14. Reference 15 provides a review of novel LMMHD components and cycles recently conceived at the BGU CMHDS. The following is a brief description of selected cycles of relevance to HTGRs.

#### 3.3.1 Wet-Vapor Cycle

A simplified LMMHD cycle operating in the "wet-vapor" regime is illustrated in Fig. 2. The working fluid can be any liquid metal having proper vapor pressure at the temperature range of interest. Heat is transferred, in an isobaric process, from the source to the working fluid bringing it to its
saturation point (Process 4-1, Fig. 2). The fluid is partially flashed (i.e., a small fraction of it boils) isenthalpically before entering the two-phase MHD generator (point 1). As the gas-liquid mixture expands isentropically through the divergent MHD duct (1-2), work is extracted. As the pressure drops, more liquid vaporizes and the temperature falls. After expansion, the low temperature low pressure saturated fluid is cooled (2-3) and the vapor condenses. The liquid is pumped back (3-4) to the system high pressure before entering the heat source.

Unique features of the wet-vapor cycle include the following:

(1) It can use a single working fluid; its liquid phase serves as the electrodynamic (i.e. high conductivity) fluid whereas its vapor phase serves as the thermodynamic fluid. In contrast, all other LMMHD cycles proposed use different materials for their electrodynamic and thermodynamic fluids.

(2) It is free of mixers and separators and, hence, of the kinetic energy losses associated with the operation of these components.

(3) It is free of the need for regenerative heat exchangers.

(4) It can make a perfect match to heat sources which deliver their energy over the entire temperature range of the cycle.
3.3.2 LMMHD compressors and Ericsson cycles

By operating a two-phase LMMHD generator in the reverse (i.e., analogously to the operation of a LM electromagnetic pump), it is possible to compress the gas or vapor being in direct contact with the LM. The resulting compressor has two unique properties:

1. It performs the compression nearly isothermally.
2. It is free of rotating or reciprocal machinery and is free of the need for sealing shafts (i.e., it can be hermetically sealed and nearly maintenance free).

The LMMHD compressor, when combined with a LMMHD generator, enables the design of gas cycles characterized by nearly isothermal expansion and compression. Fig. 3 shows a schematic layout of the resulting Ericsson cycle.

Fig. 3  Ericsson cycle schematic with T-S diagram
cycle with the corresponding T-S diagram of the thermodynamic working fluid (i.e. gas, in this cycle). The isothermal expansion and compression make the theoretical efficiency of this cycle close to the Carnot efficiency. In contrast, conventional gas cycles based on turbomachinery are of the Brayton type; they are characterized by adiabatic expansion and compression and are less efficient than Ericsson cycles operating between the same temperature limits.

Ericsson cycles can be designed to have a variable fraction of the heat source energy delivered to the electrodynamic (i.e., LM) and thermodynamic (i.e., gas) fluids. In the following we shall assume that all the HTGR power is transferred to the gas, i.e., that $e_2 = 1$. After being heated by the HTGR (point 9), the fraction $e_1$ of the PCS helium is mixed with the other fraction $(1-e_1)$ which comes out of the regenerative heat-exchanger (point 8). The combined high-temperature high pressure helium is mixed with the LM to provide a two-phase mixture with a proper inlet void fraction. Nearly isothermal expansion through the MHD generator channel to the maximum void fraction follows (1-2), with subsequent separation of the liquid and gas. The low pressure gas is then cooled in the regenerators in two steps: first against the fraction $e_1$ of the high pressure gas (2-3) and then against the entire flow of the high pressure low temperature gas (3-4). The reverse of the expansion process, isothermal compression, occurs in the LMMHD compressor (5-6). The two phases are again separated and the high pressure gas returns to the high temperature mixer (point 10) after being heated in the regenerative heat exchangers and in the HTGR. By varying the fraction $e_1$ of the helium that goes through the HTGR, it is possible to adjust the helium temperature difference across the reactor to be consistent with the reactor design requirements.
3.4 LMMHD PCS for HTGR

3.4.1 Mercury Wet-Vapor Topping + Rankine Steam Bottoming PCS

Consider an HTGR the helium inlet/outlet temperature of which is 300°C/950°C. Fig. 4a shows the temperature dependence of the heat delivered from the primary (helium) to the secondary (steam) coolants when conventional Rankine steam cycle having one internal reheat stage is used for the PCS. The shaded area represents exergy (and, hence, efficiency) loss.

One way to reduce the exergy loss is to interpose an LMMHD wet-vapor cycle in between the helium heat source and the Rankine steam cycle. The temperature-heat diagram of the resulting binary cycle is shown in Fig. 4b. Mercury is assumed to be the working fluid of the topping cycle due to its very favorable vapor pressure in the temperature range considered. To limit the number of secondary coolants, mercury exchanges heat with the
helium throughout the heat source temperature range. In other words, the mercury serves also as a heat transport medium between the helium and water. The efficiency of the mercury topping cycle is calculated to be 8.5%, when 50°C temperature difference between the helium and mercury is assumed. This makes the binary cycle efficiency 47%, as compared with 42% of the conventional Rankine PCS (of Fig. 4a).

Notice that the Rankine steam cycle assumed for the binary cycle of Fig. 4b is of the LMMHD type. [15] This cycle is characterized [15] by an isothermal expansion and by the use of supercritical steam.

3.4.2 LMMHD Ericsson PCS

The "pure" Ericsson cycle described in Sec. 3.3.2 is not particularly attractive, as it is, for conventional HTGRs because the cycle efficiency drops significantly as the temperature range for energy delivery increases.

There are, however, a number of possibilities for better matching the HTGR energy source characteristics with the LMMHD gas cycle PCS. One possibility is to superimpose a number of classical Ericsson cycles, each driven by a relatively small helium temperature drop (say, of the order of 100°C).

A combination of an Ericsson bottoming cycle and a wet-vapor homogenous topping cycle can make a good match to HTGRs, the inlet helium temperature of which is in the vicinity of 550°C. The efficiency of the resulting binary system is estimated at 47%.

The LMMHD Ericsson PCS can also make a very attractive match to special purpose reactors, such as the graphite-moderated, heat-pipe cooled reactor proposed in Ref. 16. Such a reactor delivers its heat at almost a constant temperature and provides a heat source perfectly suitable for heating the LM of the PCS. With a heat source temperature of 750°C (950°C), an efficiency of 46% (53%) is expected.
3.4.3 A dual LMMHD - GT PCS

This cycle, shown in Fig. 5, is similar to the dual cycle LMMHD system proposed for ship propulsion in Ref. 17, with one exception: it carries out part of the regenerative heat exchange between the gas expansion in the MHD generator and the turbine, whereas in Ref. 17 the gas turbine immediately follows the MHD generator. Our arrangement eases the design of the GT, as it limits the helium admission temperature to ~750°C. The relatively low gas admission temperature also reduces the LM vapor content in the helium entering the turbine. The pressure drop across the turbine is adjusted so as to provide the power requirement of the turbocompressor.

The helium temperature drop across the gas turbine makes this dual gas cycle more compatible, thermodynamically, with HTGRs than the ideal Ericsson cycle. The need of turbomachinery can, however, be a disadvantage for certain applications.

![Fig. 5 A schematic layout of a dual LMMHD-GT modified Ericsson cycle and the associated T-S diagram](image)

3.4.4 Attainable Efficiencies

Table II illustrates the efficiency attainable from selected combinations of HTGRs and LMMHD PCSs. These efficiencies pertain to the following assumptions: efficiency of the MHD generator and maximum outlet void
### Table II. Efficiencies of Selected LMMHD PCS - HTGR Combinations

<table>
<thead>
<tr>
<th>HTGR Inlet/Outlet Temp (°C)</th>
<th>PCS Efficiency %</th>
<th>PCS Efficiency %</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Absolute</td>
<td>% of ideal</td>
</tr>
<tr>
<td><strong>Reference PCSs</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1. 250/750</td>
<td>Conventional Rankine steam cycle</td>
<td>42</td>
</tr>
<tr>
<td>2. 500/850</td>
<td>Gas turbine Brayton cycle (HTGR-GT)</td>
<td>42*</td>
</tr>
<tr>
<td>3. 500/850</td>
<td>Binary cycle: Brayton topping + organic Rankine bottoming</td>
<td>47*</td>
</tr>
<tr>
<td><strong>LMMHD PCSs</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>4. 300/750</td>
<td>LMMHD Rankine Superheated steam cycle</td>
<td>45</td>
</tr>
<tr>
<td>5. 300/950</td>
<td>Binary: Hg Wet-Vapor + &quot;4&quot; bottoming</td>
<td>47</td>
</tr>
<tr>
<td>6. 550/950</td>
<td>Binary cycle, as of &quot;5&quot;</td>
<td>49</td>
</tr>
<tr>
<td>7. 550/950</td>
<td>Binary: Hg Wet-Vapor + Ericsson</td>
<td>47</td>
</tr>
<tr>
<td>8. 550/950</td>
<td>Dual: LMMHD Ericsson cycle with GT below 750°C</td>
<td>47</td>
</tr>
<tr>
<td><strong>LMMHD PCS - Special Application</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>9. 700/750</td>
<td>Heat-pipe reactor with Ericsson</td>
<td>46</td>
</tr>
</tbody>
</table>

* This is a net plant efficiency; it accounts for the HTGR coolant circulation power requirement. This power requirement corresponds to 1% to 2% (absolute) of the PCS efficiency.

The efficiencies are calculated considering the following assumptions: the helium fraction are 0.85; gas turbine and compressor efficiencies are 88%; effectiveness of regenerative heat exchangers is 97%; friction losses in PCS are taken into account, but auxiliary power needs of the HTGR are not; the electricity generated by the LMMHD PCS is DC.

### 3.5 Discussion

All the LMMHD PCSs considered in this work offer, in addition to improved efficiency, a safety attribute - either elimination of the water from the HTGR plant altogether or, at least, separation of the water from the primary helium (and, therefore, graphite) system by a liquid metal. Thus, the use of the LMMHD energy conversion technology can eliminate, or significantly reduce, the water ingress accident probability.
On the other hand, the handling of the LM of the LMMHD PCSs may pose maintenance and environmental difficulties. For example, being toxic under certain conditions, mercury must be handled with special care. Alkaline materials, on the other hand, are destructive to graphite, so that the LMMHD - PCS should be designed to have no possibility for liquid-metal ingress accident.

Two other design difficulties are associated with the LMMHD PCS - compatibility with structural materials at elevated temperatures, and the need to bring the LM to the liquid state before starting the system.

### 3.6 Conclusions

The recently conceived LMMHD components and cycles open new interesting possibilities for the conversion of HTGR energy into electricity, including:

1. Designing conventional HTGRs (i.e., HTGRs designed to have a helium outlet temperature of 750°C) to have a PCS efficiency of about 45%. This corresponds to a net HTGR efficiency of ~43% versus ≤40% with conventional energy conversion technology.

2. Making efficient use of the high temperature operation-ability of HTGRs for generating electricity; it might be possible to design HTGRs providing net efficiencies approaching 47%.

3. Attaining efficiencies as high as, and even somewhat exceeding, 70% of the ideal efficiency pertaining to the HTGR heat source characteristics. This is due to the versatility of the LMMHD PCSs and their flexibility in matching the HTGR heat source characteristic and due to the relatively high efficiency of the LMMHD PCS components.

4. Designing HTGRs to be free of rotating machinery, promising high reliability and low maintenance systems suitable for unattended applications.

5. Reducing the risk for water ingress accidents.
The incorporation of a liquid metal system in an HTGR power plant introduces, however, a number of safety and maintenance difficulties. Thus, an assessment of the promise of the LMMHD energy conversion technology requires a comprehensive techno-economical feasibility study. In view of the potential gain in efficiency offered by the LMMHD energy conversion technology, it is recommended that such a thorough feasibility study be undertaken.

4. REFERENCES


GAS-COOLED REACTOR TECHNOLOGY
(Session F)
OVERVIEW OF US MHTGR BASE TECHNOLOGY DEVELOPMENT PROGRAM*

J.E. JONES, Jr., P.R. KASTEN
Oak Ridge National Laboratory,
Oak Ridge, Tennessee,
United States of America

Abstract

The Base Technology Development Program for the Modular High Temperature Gas Cooled Reactor (MHTGR) provides basic information on fuel performance, fission product retention capabilities, and material properties; the program plan for obtaining such information is summarized, considering the specific needs of the MHTGR concept. Continued international cooperation in HTGR base technology development will contribute to the planned program.

Introduction

At this time, existing technology is sufficient to show the feasibility of the MHTGR concept and to develop the conceptual design. The base technology development program has been established in direct response to specific design data needs and licensing requirements for the MHTGR concept.

The base technology program is planned to develop the basic correlations concerning the behavior of physical and chemical phenomena; they also provide experimental validation of mathematical models predicting system behavior necessary for licensing (e.g., retention of fission products). The specific development areas discussed here are (1) fuels performance and fission product transport, (2) graphite behavior, and (3) metals behavior.

Component development requirements have not yet been specifically defined. These development requirements are not included in the base technology program.

* This paper was written by a contractor of the US Government under Contract No. DE-AC05-84OR21400.
Fission product retention within the fuel coatings is the key ingredient to achieving passive safety in the MHTGR, and this area will be emphasized. Figure 1 illustrates a Triso-coated fuel particle used in the MHTGR. The fuel kernel is a mixture of UO₂ and UC₂, termed uranium oxycarbide; it is surrounded by a porous layer (or buffer layer) of pyrolytic carbon which provides expansion volume. Surrounding the buffer volume are successive layers of pyrolytic carbon, silicon carbide, and pyrolytic carbon, which together contain the fission products. These coated fuel particles have very high fission product retention capability, even at high temperatures. It is now well established that high-quality coated fuel particles retain essentially all fission products under anticipated reactor temperature conditions, as indicated in Figure 2; this strongly supports the feasibility of the MHTGR concept.

FIG. 1. TRISO coated fuel particles retain fission products at high temperatures.

FIG. 2. Coated fuel particles maintain integrity at high temperatures.
At the same time, during the coating process some very small fraction of defective coated fuel particles may be produced. Process development is planned to limit defective coatings in fuel to a very low value either by removal of defective coated particles per se or by reducing the number of defective particles produced in the process itself. Preliminary indications are that process development can effectively reduce the number of defective particles. However, defective particles will not be completely eliminated in practice, and the technology development program will address the performance of coated fuel particles having defective coatings to determine fission product release from such particles as well as to guide the requirements associated with process development. To obtain such information, purposely defective coated fuel particles are planned to be irradiated in capsule experiments in the High Flux Isotope Reactor (HFIR) at the Oak Ridge National Laboratory (ORNL), with on-line monitoring of fission gas release and with subsequent postirradiation examinations. The latter examinations determine the retention of fission products within the coated particles.

Experiments are also ongoing relative to determining the influence of water ingress on hydrolysis of fuel particles having defective coatings. Figure 3 shows an irradiation capsule used to test MHTGR coated particles and fuel rods. These irradiations are carried out in the reflector region of the HFIR at ORNL, as illustrated in Figure 4. As shown, the fuel is located within the graphite cylinder portion of the capsule.

Coated fuel particles have to be fabricated into fuel rods, as illustrated in Figure 5. This step involves fabrication processes which can lead to some breakage of fuel particle coatings. The fuel technology development program concentrates on (1) developing economic processes which maintain the integrity of the fuel particle coatings, and (2) performance testing of fuel rods under normal reactor conditions as well as under postulated accident conditions.

Fuel coating process development involves fabricating coated particles which have a high degree of sphericity and producing coating layers having uniform properties. Process development associated with fabrication of fuel rods involves development of processes which minimize the likelihood of breakage of the particle coatings. The rod fabrication step involves injecting pitch into a bed of coated particles plus matrix graphite. It is
3 Fuel Rods per Body

Graphite Fuel Body
(8 Bodies per Capsule)

FIG. 3. Irradiation capsule HRB-21 (USDOE).

FIG. 4. Irradiation will be conducted in the HFIR removable beryllium (RB) facility.
planned that the performance of the fabricated fuel rods will be obtained by irradiation testing in capsules in the HFIR at ORNL; these capsules were illustrated previously in Figure 3. The fuel rods will be irradiated under nominal reactor operating conditions; the fuel capsules will be instrumented with thermocouples, and a sweep gas will be used to determine fission gas release during irradiation. Following irradiation, it is planned that the fuel rods undergo postirradiation examination in the High Radiation Level Evaluation Laboratory (HRLEL) (Figure 6) to determine the integrity of the coatings relative to retention of fission products. In addition, fuel will be heated to high temperatures simulating postulated accident conditions, and the performance of such fuels for retaining fission products will be obtained using the irradiated microsphere gamma analyzer (IMGA) and postirradiation gas analyzer (PGA) equipment at the Oak Ridge National Laboratory. Such equipment examines coated particles and determines the fission product inventory still remaining within particles; a stable fission product is used as a reference to give quantitative information on fission product retention of fuel coatings. Large numbers of particles are examined to give statistically meaningful information. Figure 7 illustrates the IMGA system. As shown, a singularizer picks up individual particles and a sample changer indexes the particles to a detector where the gamma ray spectrum is measured. Depending upon the results of gamma analysis, the examined particle is placed in a particular compartment. Various classification compartments can be utilized, with particle separation based on predetermined criteria.
AUTOMATIC PARTICLE HANDLING MECHANISM FOR THE IRRADIATED-MICROSphere GAMMA ANALYZER (IMGA) SYSTEM. (A) SINGULARIZER PICKS UP INDIVIDUAL PARTICLE, (B) SAMPLE CHANGER INDEXES PARTICLE TO Ge(Li) DETECTOR (NOT SHOWN); (C) CLASSIFICATION COMPARTMENTS WHERE PARTICLE IS PLACED IN ONE OF TWENTY COMPARTMENTS DEPENDING ON PREPROGRAMMED CRITERIA.

FIG. 7. Irradiated microsphere gamma analyzer for MHTGR.

Figure 8 shows the PGA equipment which can be operated either at room temperature or up to 2000°C. The PGA measures (by a mass spectrometer) the gases released after particle breakage.

The heavy metal contamination of the fabricated fuel rod is specified to be ≤1x10⁻⁵ at a 50% confidence level, and ≤2x10⁻⁵ at a 95% confidence level. The fraction of defective particles is specified to be ≤1.5x10⁻⁴ at 50% confidence level, and ≤6x10⁻⁴ at a 95% confidence level.

A small amount of uranium contamination will be present in the fuel rod matrix material as a result of the coating process. Further, a small fraction of coating defects are expected to occur during the coating process. As a result, fission product behavior within the reactor system must be quantified in order to assure meeting the user requirements of "no-evacuation planning." Fission products released from the fuel rod can be
transported to the heat exchanger system where "plateout" tends to occur on the cooler surfaces. Figure 9 shows the relative location of the steam generator and the reactor core. Over a period of time, fission products will "build up" on the steam generator surfaces. During rapid depressurization, some of these fission products could be released to the atmosphere. Present knowledge indicates there is sufficient retention of fission products during rapid depressurization, but more quantitative knowledge is needed. A definitive program has been developed for demonstrating specific behavior and for validating the mathematical models used to predict fission product transport behavior. More specifically, it is planned to measure the diffusion coefficients of fission products in coatings and in graphite moderator under various reactor conditions. Also, the sorptivity of fission products on materials such as graphite moderator and metallic surfaces will be determined for various conditions. In addition, studies of fission-product "plateout" (i.e., without dust and without water vapor present, and with dust and with water vapor present) are planned along with fission-product "liftoff" experiments under reactor-depressurization conditions.
conditions. Both basic data for improving the modeling of fission product transport as well as integral data for validation of system models will be obtained. Nuclides to be studied in various tests are iodine, cesium, strontium, and silver.

**Graphite Technology**

Graphite is an important material in the reactor and constitutes a major volume fraction of the core region. Figure 9 shows the general location of the fueled core, while Figure 10 illustrates more clearly the annular characteristics of the core and the inner and outer reflector graphite regions. Great Lakes Research Corporation grade H451 graphite has been chosen for the fuel elements and the replaceable reflector elements. Stackpole Corporation grade 2020 graphite has been chosen for the permanent side reflectors and the core support structures. There is much information
on graphite behavior in gas-cooled reactors which generally supports the feasibility of the MHTGR concept. At the same time, additional physical and chemical property data is needed to support the detailed design of graphite components for the MHTGR under normal operation conditions and under postulated accidents. The graphite technology development program will obtain the needed data base and involves evaluating variations in graphite strength, determining irradiation creep behavior (an important mechanism for relieving stresses), and reducing oxidation characteristics. Most of the graphite technology program effort is related to obtaining property data required for reliable component design and reducing associated investment risk.

Most of the planned graphite property studies involve irradiation-creep experiments, fracture-mechanics studies, and corrosion/oxidation behavior. Irradiation-induced creep and strain data are particularly needed since creep strongly influences stresses and also affects graphite properties. Graphite fracture-mechanics behavior tends to deteriorate at high neutron fluences, and experimental creep data will determine graphite reflector lifetimes. The fracture-mechanic studies relate to the initiation, and growth of cracks and provide a basis for understanding the phenomena of fracture and for developing graphite failure criteria. Relative to steam oxidation of graphite, both surface and internal oxidation behavior are planned to be determined.
Since they are not pure materials, graphites by their very nature have existing cracks or flaws which are induced by the raw materials used and by the manufacturing process. Because of the scatter in measured graphite property data, and because of insufficient design margin associated with the lower values of the measured properties, it is planned to provide graphite property behavior and material failure criteria in statistically meaningful ways. This requires large data bases.

**Metals Technology**

Metals technology is sufficiently known to assure the feasibility of metal components in the MHTGR. Additional information needs relate primarily to property data bases required for reliable component design, licensing, and reducing associated investment risk. A metals technology development program has been identified which will provide the required material property correlations. The pertinent components are included in Figure 11 and consist of the reactor vessel (constructed of SA 533B steel,
which is a Light-Water Reactor pressure-vessel steel); the duct which connects the two vessels (made of Alloy 800H); the core lateral restraint system (includes the core barrel) surrounding the graphite core (made of 2.25Cr-1Mo and Alloy 800H); and the steam generator (made of Alloy 800H and 2.25Cr-1Mo steel). Long-term testing is needed to complete the materials property data base; key measurements involve fracture mechanics, mechanical properties, radiation effects, environmental and corrosion effects, and weld properties. These measurements on material properties are planned in order to provide assurance that components will perform satisfactorily over their design lifetime. Further, materials are to be qualified under an ASME Code Case specifying data base requirements. Sufficient data will be obtained so that rational extrapolations of materials performance can be made into areas where data is limited; that is, data trends will be established.

Overall, information will also be obtained which provides material property data important to the design of components. For example, thermal aging can lead to changes in material ductility and creep strength and this has a direct influence on design requirements. Additional information on mechanical properties as a function of environment and aging are to be obtained for both 2.25Cr-1Mo and for Alloy 800H. In particular, emphasis will be placed on understanding the influence of aging effects on material properties since such information reduces investment risk associated with component performance.

A significant portion of the Metals Technology Development Plan is associated with high temperature fracture mechanics studies. Fracture mechanics involves basic measurements associated with the initiation, propagation, and arrest of cracks and in particular, investigates conditions associated with crack growth and crack arrest. Such studies are believed to be a key to understanding why materials fail; this understanding is particularly important in high temperature applications of materials. Such studies are complicated by the influence of factors such as aging and environment on fracture mechanics behavior. It is planned that significantly more work in the above areas be carried out on 2.25Cr-1Mo and Alloy 800H, including decarburization studies. Irradiation testing of Alloy 800H and SA 533B steels is included in the planned program. Also, the creep and rupture behavior of SA 533B steel at the temperatures associated with postulated accident conditions will be studied. Other areas of R&D being planned in the Metals Technology
FIG. 12. Creep testing equipment and specimens for MHTGR metals.
Development Program are irradiation behavior of materials, and the effects of HTGR-He environment and thermal aging on materials properties.

Some of the equipment being used in Metals Technology Development are illustrated in Figures 12 and 13. Figure 12 shows a creep test assembly being used to test 2.25Cr-1Mo and Alloy 800H, along with creep test specimens. Figure 13 shows some lever-arm test units for material creep tests at high temperatures in an HTGR-He environment.

FIG. 13. Lever-arm creep test equipment for MHTGR metals at high temperature.

International Cooperation and Base Technology

The United States and the Federal Republic of Germany presently carry out a cooperation under an Umbrella Agreement covering MHTGR technology development. Also, the United States and Japan are cooperating in MHTGR research and development. The U.S./FRG cooperation involves (1) a subprogram in metals development which addresses eight specific studies and (2) a subprogram in fuels, fission products, and graphite which covers
sixteen specific investigations. The U.S./Japan cooperation involves testing of MHTGR fuels including their performance at high temperatures; further cooperation is planned in the graphite area and is being proposed in the metals area. These cooperative agreements contribute to expand base technology information pertinent to the MHTGR. Continuing and expanding such cooperation can have a beneficial effect on increasing the amount of property-data information and limiting development costs.

Conclusions

Base Technology Development for the Modular High-Temperature Gas-Cooled Reactor (MHTGR) provides basic information on fuel performance, fission product retention capabilities, and material properties. Present information is sufficient to show the feasibility of the MHTGR concept, and to develop the conceptual design. The base technology development program is based on the specific design data needs of the MHTGR concept; these programs will develop the data bases and validate the system response models required for detailed design and for licensing. The planned base technology program is relatively modest and is well focused on the needs of the MHTGR concept. Continued emphasis on international cooperation in base technology development will increase the amount of property behavior information as well as limit development costs.
STATUS AND DEVELOPMENT OF THE GERMAN MATERIALS PROGRAMME FOR THE HTGR

H. NICKEL, F. SCHUBERT, H. SCHUSTER
Institute for Reactor Materials,
Kernforschungsanlage Jülich GmbH,
Jülich, Federal Republic of Germany

Abstract

During the last decade the research and development of materials for the HTGR in Germany have mainly been directed towards the qualification of high temperature structural alloys for heat transferring components of advanced nuclear process heat plants. The availability of Fe/Ni-base and Ni-base alloys for long term service loading has in principle been established. Furthermore, materials and optimization have indicated that improvements in the corrosion properties are also achievable.

The status of the material evaluation and optimization are demonstrated with NiCr 22 Co 12 Mo (Alloy 617) and Thermon 4972 for heat exchanging components, with the steel DIN Werkstoff-Nr. 1.4981 (X 8 CrNiMoNb 16 16) for absorber and control rods and carbon fibre reinforced carbon (CFC) for hot ducting.

The continuing HTR structural materials programme is aimed at improving the quality and data base of the materials for components of advanced HTGR's, both for nuclear heat application and for the steam cycle. Emphasis is given

- to the determination of long term design values for different semi-finished products including weldments;
- to improve service life estimation rules;
- to improve constitutive equations for inelastic analysis and to develop design rules;
- to develop fracture mechanics criteria for leak - before - break argumentations.

The general aim of this work is to assure the long term integrity of class I components. Using typical examples, this paper shows the status of the theoretical and experimental work related to these objectives.

1. Introduction

It is intended at this meeting to provide a review of the current status and the recent progress made in the technology of gas-cooled nuclear reactors. This report shows the status achieved in the solution of the materials problems.

Extensive effort is devoted to the qualification of materials, concentrated on the two German advanced HTGR concepts, the HTGR steam cycle and HTGR for nuclear process heat.
The materials for the helium-cooled HTGR of the pebble bed type have been defined and they are qualified, the manufacturing procedures for the required product forms have been proved /1, 2, 3, 4/ and the semi-finished products and the fabrication methods are available. This statement stands without any reservation for the components for a steam cycle HTGR with working temperatures up to 750 °C as is demonstrated by the commissioning of the THTR. The spherical fuel elements and the structural graphite have been developed and qualified, although long-time irradiation experiments and tests of simulations of emergency conditions are still going on /1, 5, 6, 7/. Some restrictions, however, must be made for components in a nuclear process heat plant with the highest working temperature (up to 1000 °C) concerning the permitted service life. The ongoing work for structural materials is concentrated to the improvement of the long term reliability of metallic components for nuclear process heat application /3, 7, 8/.

In this paper we will concentrate on metallic materials for structural components of the steam cycle and process heat HTGR's.

2. Comments on selection of candidate materials

For the selection and qualification of metallic materials the following properties have been taken in account:

- strength and long term ductility for components which can be designed with time independent materials data;
- creep resistance and microstructural stability for those components which must be designed using time-dependent design values;
- corrosion resistance in the working atmospheres;
- hot- and cold-formability and weldability;
- possibility of non-destructive inspection during manufacture and in the plant;
- sensitivity to neutron irradiation induced embrittlement (this only for those components which are in the core-region).

For the design and construction of THTR components steels are used, for which experience from fossil-fired plants is available. Those steels are:

- the ferritic heat resistant steel 15 Mo 3 for the fuel loading tube system;
- the steel X 20 CrMoV 12 1 for the external components of the steam circuit;
- the creep resistant high temperature alloy X 10 NiCrAlTi 32 20 (Alloy 800) for the steam generator and live steam header and tubes enclosed in the prestrained concrete pressure vessel.
- the high temperature stainless steel X 8 CrNiMoNb 16 16 for control and absorber rods. For this steel there exists already experience concerning the effect of neutron irradiation from experiments done for the fast breeder reactor projects.
In the case of components for nuclear process heat plants, creep resistant alloys are needed. From the possible alloys used in conventional petrochemical plants or in gas turbines, the alloy (FIG. 1)

- NiCr22 Co12 Mo (Alloy 617)

was selected for the highest working temperatures. For component parts with lower temperatures of about 850 °C the alloys

- NiCr22 Fe18 Mo (HASTELLOY X)

and

- X 10 NiCrAlTi 32 20 (Alloy 800 H)

have been qualified.

![Graph showing the creep rupture strength of various steels and alloys.](image)

Fig. 1: Mean 100,000 h creep rupture strength of some typical high temperature steels and alloys

As an alternative for Alloy 617 a new Fe-Ni-Cr alloy with about 12 % W has been developed by Thyssen-Edelstahlwerke AG, Krefeld, FRG. This alloy, Thermon 4972, is now also under qualification for nuclear components /9/.

The steel X 20 CrMoV 12 1, one of the well known materials for modern German coal-fired plants, has been improved concerning the level of trace elements for the application in the THTR steam circuit /10, 11/.
To minimize the sensitivity of Alloy 800 for loss of ductility due to ageing at temperatures between 550 and 650 °C the contents of C and Al+Ti is limited. Meanwhile it has been proved that long-term properties of this THTR-version reach the same level of creep rupture strength as the solution heat treated Alloy 800 H with higher level in C and Al+Ti.

A comparison of the $10^5$ h creep rupture strengths /12/ of the candidate materials shows that Alloy 617 has the potential for application at the highest service temperatures.

In the design of heat exchanging components, creep rupture strength was given priority for the selection of the materials. It is, however, important to take into consideration the influence of corrosion by the operational environments on the creep behaviour.

Protection against corrosion in the circuits of an HTGR is achieved in the following way:

Steam corrosion in the secondary circuit is controlled by conventional means, i.e. appropriate conditioning of the feed water.

The heat transferring gas in the primary circuit, helium, is slightly contaminated by $H_2O$, $CO$, $CH_4$, $H_2$ and $N_2$ in the $\mu$bar range. Chromium is the metal element governing the corrosion reactions with the Fe- and Ni-base alloys. No deleterious hot gas corrosion effects have been found up to temperatures of 750 °C. Compositions of the impurities with excessively high or very low carbon activities may cause carburization or decarburization, respectively (FIG. 2). By keeping the CO partial pressure of the cooling gas helium in a well defined range, unacceptable carburization/decarburization reactions

![Fig. 2: Corrosion areas in the stability diagram for chromium](image)

390
can be avoided. This range may already be obtained by the impurity composition which usually is formed without conditioning of the gas /13/.

3. Components for the steam cycle of power generating HTGR

For the structures of a steam cycle HTGR the required steels are well specified either in the German standards /14/ or are given in ASME Boiler Code, Code Case N47 /15/. Nevertheless, in order to take advantage of the materials capacity, refined methods of life time prediction and insurance of long-term integrity of the components are required. The following tasks are handled in ongoing research work:

- determine the influence of preageing and cold deformation on long-term properties;
- establishing life-time fraction rules;
- establishing fracture mechanical criteria in the creep-fatigue region.

For the THTR components these criteria have been fulfilled in a conservative manner.

The ongoing work for the above mentioned areas is not only being carried out in Germany, but it is part of the international cooperation with the United States, Japan and Switzerland /2/.

During the THTR materials program for the external steam cycle tubes, the postulation of "leak before break" has been proved for the steel X 20 CrMoV 12 1 /11/. Calculations using the fracture mechanics approach predicted satisfactorily that for this steel the "leak before break" criterion is fulfilled in the component.

4. Material for control rods

The control and absorber rods are essential parts for the safety of an HTGR plant. These metallic components are exposed to irradiation by thermal and fast neutrons. The effect of fast neutrons on material properties is limited to the temperature regime up to about 350 °C. At higher irradiation temperatures, however, the helium atoms, produced by an n,α-reaction with boron and nickel, decrease the ductility of steels and Ni-base alloys. This high temperature embrittlement increases with increasing neutron fluences and test temperature, as well as with the duration of the irradiation at high temperature. Emergency conditions in advanced HTGR's may lead to short-term temperature excursions up to about 850 °C in the control rods.

For the advanced HTGR, therefore, the selection of material for the control rod tubes is of crucial importance.

In a screening program a number of high temperature steels and Ni-base alloys have been irradiated at 400, 500 and 600 °C and were post-examined in tensile and short-term creep tests. The comparison of post-irradiation tensile rupture elongation (FIG. 3) lead to an optimized version (KA) of
Fig. 3: Creep rupture strain of various metallic high temperature alloys before and after neutron irradiation ($\phi_{th} = 3 \times 10^{25} \text{ m}^{-2}$) and short fracture life-time (< 100 hrs) unirradiated irradiated

1.4981 (X 8 CrNiMoNb 16 16) in comparison with Alloy 800 and HASTELLOY X. By the alloy optimization work /16/ a satisfactory level of post-irradiation rupture strain both in tensile and creep testing could be preserved (FIG. 4). The irradiation tests are continuing in order to accumulate a fluence of $10^{22} \text{ cm}^{-2}$, the expected maximum fluence of control rods in HTGR plants. The results confirmed the use of this optimized thermal mechanically treated 1.4981 for future control rods.

5. Materials for intermediate heat exchanger and methane reforming tubes in a nuclear process heat plant

The qualification of Alloy 617 has reached a status which allows to realize a design time for He/He-IHX of 100,000 hours at a materials temperature up to 950 °C and 140,000 hours for steam reformer tubes at working temperatures of up to 900 °C.

All the basic work concerning
- creep rupture testing up to 30,000 hours (FIG. 5);
- high cycle fatigue testing;
- low cycle fatigue (without and with hold times, FIG. 6);
and
- influence of ageing on short-term properties (FIG. 7)

has been done by the joint effort of the German partners in the HTGR materials working group. So we obtained design curves both for creep and fatigue (FIG. 8), and confirmed the
Fig. 4: Rupture strain of stainless steel 1.4981 (X8 CrNiMoNb 16 16) after neutron irradiation (Irradiation temperature: 400 °C; Test temperature: 850 °C)

Fig. 5: Creep properties of INCONEL 617
Fig. 6: LCF results for INCONEL 617

Fig. 7: Influence of ageing at 900 °C on fracture elongation of NiCr 22 Mo 10 Co (Alloy 617)

application of the linear life time fraction rule for creep and fatigue

\[ D_c = \sum \frac{t_i}{t_{Ri}} \]

where

- \( D_c \) = exhaustion due to creep
- \( t_c \) = actual duration at \( \Delta \varepsilon_i \) and temperature \( T_i \)
- \( t_{Ri} \) = allowable time at \( \Delta \varepsilon_i \) and temperature \( T_i \)

and

\[ D_f = \sum \frac{N_i}{N_{fi}} \]

where

- \( D_f \) = exhaustion due to fatigue
- \( N_i \) = actual number of cycles at \( \Delta \varepsilon_i \) and \( T_i \)
- \( N_{fi} \) = allowable number of cycles at \( \Delta \varepsilon_i \) and \( T_i \)
The analysis of component life for the high temperature PNP-components cannot be treated by elastic analysis; inelastic analysis is necessary to confirm a realistic service life. The strain range for fatigue is calculated to be less than 0.03 % which can be neglected for life time calculations /17/.

For steam reformer tubes, in the operating time of 140,000 hours 160 start-up and shut-down loading cycles are assumed, which gives a running time of 875 hours between shut downs.

Taking into consideration the relaxation of stresses caused by temperature gradients through the tube wall, the life-time calculation gives the following result (FIG. 9) /18/: Assuming that each cycle produces the same amount of consumption of life-time as does the first cycle, the life-time exhaustion factor $D = 1$ allows 455 start-up and shut-down procedures, more than twice the number postulated for the operation time.

The life of heat exchanger tubes is mainly determined by the primary stress level; thermally induced secondary stress may be treated as unimportant due to the small wall thickness. The estimated primary stress in the tube wall is less than 3 N/mm², so that a safe operation time of 100,000 hours can be achieved. An extension of this operation time needs a very carefully analysis of the tubes, which does not seem feasible.

The question of the importance of creep ratcheting effects on the He/He-heat exchanger components cannot be handled by using any calculation given in the ASME-CC N47. An inelastic
analysis of finite element programs leads to very expensive and time consuming computer times. Therefore, the degree of ratcheting due to hot streaks in the helium has been experimentally investigated. It has been found that /19/:

- creep ratcheting may occur if the primary stress is low (FIG. 10);
- but the accumulated ratcheting strain remains less than the permitted strain limits.

For the hot header of a heat exchanger, the postulated emergency condition determines the dimensioning. The header must be safe against creep buckling. In the case of a helium/helium heat exchanger for the postulated emergency condition, loss of pressure in the secondary circuit exposes the components to a pressure of 40 bar at 950 degrees. This exposure may cause a loss of geometrical stability. The judgement of the component behaviour must be done by inelastic analysis, in which the deformation with increasing time must be evaluated.
A parameter study of creep buckling for the hot header of a He/He-heat exchanger has been performed /20/: dimensions were 437.5 mm/125 mm; external pressure 42 bar, $T = 957$ °C. FIGURE 11 shows the increase of ovality as a function of time. According to this calculation, a total exposure of $t = 20$ hours for the starting ovality of $\varepsilon_0 = 1\%$ may be tolerated. In reality, the temperature will decrease immediately after the emergency event. If the cooling rate is $5$ K/h, failure due to creep buckling is not expected (FIG. 12).

**Fig. 10:** Creep curve under axial primary stress (▲) and superimposed alternating secondary stress (i.e. ratcheting, ●).

**Fig. 11:** Time dependence of tube ovality.
7. Materials for hot ducts

For steam cycle plants, the gas duct can be made from sheet of those materials which have already been mentioned, such as HASTELLOY X. But for a nuclear process heat plant, with higher temperatures, a hot duct of high temperature capability is necessary. In the German materials programme, CFC-components are under development (FIG. 13). The figure shows a test of a ring-section. The high temperature evaluation of this material is now in progress.

8. Future necessary effort

For steam cycle plants, design codes for concrete vessels and graphite components are to be formulated. In respect to codes for metallic components, some additional effort is necessary.

For nuclear process heat components, especially for the heat exchanging components, there will be a demand for inelastic analysis or simplified inelastic analysis for the structural design. The problems with inelastic analysis which remain to be solved are:

8.1 Constitutive equations

Constitutive equations should describe the three-dimensional stress/strain behaviour. In the work for the HTR design rules, beside the analytical approach given in /12, 21, 22/, Norton's creep law in the three-dimensional formulation is used. Further the applicability of the ORNL-model as well as
the Interaot-model for HTGR components is under consideration.

The main problem with all constitutive equations remains the scatter of the materials data which have to be used.

Fig. 13: CFC-ring test from a tube with 900 mm Ø.
8.2 Influence of anisotropy of mechanical properties in the semi-finished products

A comparison of the creep behaviour of bar material and axially loaded tube specimens (FIG. 14) shows the importance of the manufacturing process. In our case the tubes were subjected to a final cold deformation in order to fulfil the tolerances of straightness. The prior cold work lead to a significant change in the stress-time relationship /23/.

![Creep curves for axial loaded specimens at 950 °C](image)

**Fig. 14: Creep curves for axial loaded specimens at 950 °C (ε = 30 Nmm⁻²)**

8.3 Weldments

The strain in the areas of a weld seam should be restricted to 50 % of that allowed for the parent metal according ASME CC N-47. The weld itself is very difficult to treat with inelastic analysis due to different stress/strain behaviour of its sections (FIG. 15). The figure demonstrates the remaining creep deformation across a weldment after creep exposure at 850 °C of 500 and 1610 hours /24/. The above mentioned limitation of the strain in the weldment may be too restrictive.

8.4 Flaws in the components

At ambient temperature the fracture mechanics calculation is used if detectable flaws are present or a postulated flaw in the component is expected. The analysis should result in the proof that within the next inspection period or the total service life the crack growth will not lead to spontaneous catastrophic fracture. In high temperature fracture mechanics both fatigue and creep crack growth must be examined, these are just now under intensive examination. So far in steam reformer tubes of Alloy 800 H, the fatigue crack growth may be described by threshold and a Paris type of law (FIG. 16) /25/. Creep crack growth evaluation is underway.
**Fig. 15:** Axial strain distribution along a circumferential welded tube after axial creep exposure

**Fig. 16:** Comparison of fatigue crack growth data for 1"CT-specimens, 1/2"CT-specimens and steam reformer tubes with circumferential defects (X 10 NiCrAlTi 32 20, 850 °C)
8.5 Uncertainties in the knowledge of expected loading history

In addition to the uncertainties and scatter concerning materials and product properties, the uncertainties in the loading history hinder the exact calculation and prediction of the real life-time of a component.

9. Final comments

The problems mentioned above are the main areas of future efforts.

The material programmes executed have demonstrated that the main materials problems are solved. Continuing efforts are necessary to ensure the long term reliability and safety of the components.

ACKNOWLEDGEMENT

The authors thank to all the members of the working group "Arbeitskreis 8 of the EG-HTR 'Materials'" and the "Fachkreis Auslegungskriterien" for the fruitful cooperation.

The work is sponsored by Bundesminister für Forschung und Technologie (BMFT), Bundesminister des Innern (BMI), and Ministerium für Wirtschaft, Mittelstand und Technologie (MWMT) des Landes Nordrhein-Westfalen.

10. References


/4/ "HTR-Komponenten"
Editor: Der Minister für Wirtschaft, Mittelstand und Verkehr des Landes Nordrhein-Westfalens, Band 16-1, 16-2, Schriftenreihe "Energiepolitik in Nordrhein-Westfalen", 1984


FRACTURE BEHAVIOUR AND NONDESTRUCTIVE TESTING, 8. MPA-Seminar, Stuttgart, 1982, Staatliche Materialprüfungsanstalt Universität Stuttgart


/14/ VDTÜV-Blätter (VDTÜV-Essen), TRD-Vorschriften und AD-Merkblätter (Carl Heymanns Verlag, Köln / Berlin), DIN-Vorschriften (Beuth Verlag Berlin)

/15/ ASME-Code, Case N 47-17, "Class I Component in Elevated Temperature Service, Division 1", ASME (1979)


404
/22/ SCHUBERT, F.; SCHUSTER, H.; DIEHL, H.; HONEFF, H.: "Zeitstandverhalten und Gefüge des Werkstoffs Nr. 2.4663"
Presentation at MWMV-VDM-Seminar, Werdohl, 23.01.1985

SMIRT 8-Post Conf., Post-Seminar No. 6, Paris, August 1985

KFA-JüL.-2092, Oct. 1986

/25/ R. J. Kwasny: "Bruchmechanische Untersuchungen an hochwarmfesten Legierungen im Temperaturbereich von Raumtemperatur bis 900 °C"
Dissert. RWTH Aachen, 1986
KVK AND STATUS OF THE HIGH TEMPERATURE COMPONENT DEVELOPMENT

W. JANSING, H. BREITLING, R. CANDELI, H. TEUBNER
Internationale Atomreaktorbau GmbH, Bergisch Gladbach, Federal Republic of Germany

Abstract

The High Temperature Helium Test Facility (KVK*) which started operation in August 1982 has reached an operating time of 11000 hrs.

It serves in testing components of high-temperature gas-cooled reactors for direct heat application and for the steam cycle HTR-Module.

Tests of a 10 MW U-tube heat exchanger, the cylindrical hot header of the He-heat exchangers, and a secondary hot gas duct have been successfully completed.

Operation will be continued with tests of a 10 MW helical-tube heat exchanger, a 10 MW steam generator and primary and secondary hot gas duct test objects like bends and a coaxial duct. The availability of the test facility is better than 80%.

1 The Component Test Facility (KVK)

The KVK was built at the Interatom company's site in Bergisch Gladbach in 1980 - 1982. It serves in testing the components of a high-temperature gas-cooled reactor for direct heat application as well as for the steam cycle HTR-Module.

* The KVK and the PNP test components are sponsored by the Minister for Economics, Small Business and Technology of the State of North Rhine/Westphalia in the Federal Republic of Germany.
The main operating conditions are (Fig. 1):

Thermal power: 10 MW (max. 12.8 MW)
Heat transfer medium: Helium
Temperature in the Primary Loop: 950 °C (max. 1000 °C)
Temperature in the Secondary Loop: 900 °C (max. 950 °C)
System pressure: 40 bar (max. 46 bar)
Coolant flow: 3 kg/s (max. 4.3 kg/s)
Max. temperature transient: ± 200 K/min
Max. pressure transient: 5 bar/s

Operating data:

<table>
<thead>
<tr>
<th></th>
<th>Primary System</th>
<th>Secondary System</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal power</td>
<td>10 MW (max. 12.8 MW)</td>
<td></td>
</tr>
<tr>
<td>Temperature</td>
<td>950 °C (max. 1000 °C)</td>
<td>900 °C (max. 950 °C)</td>
</tr>
<tr>
<td>Pressure</td>
<td>40 bar (max. 46 bar)</td>
<td>40 bar (max. 46 bar)</td>
</tr>
<tr>
<td>Flow rate</td>
<td>3 kg/s</td>
<td>3 kg/s</td>
</tr>
<tr>
<td>Helium velocity</td>
<td>60 m/s</td>
<td>60 m/s</td>
</tr>
</tbody>
</table>

FIG. 1. COMPONENT TEST FACILITY (KVK)

The test facility consists of a primary and a secondary helium loop (Fig. 2). The heat generating sources are a steam-heated preheater, a natural gas fired and a electrical heater. Heat is transferred from the primary loop to the secondary loop via the He/He heat exchanger. The heat sink is a steam generator. Helium is circulated by radial blowers.

This two-loop test facility allows the installation of reactor test components under their respective operating conditions.
2 Test Components (KVK)

The following main reactor components have been or will be tested under simulated reactor operating conditions (Fig. 3).

- He/He heat exchanger of U-tube design (tests completed)
- He/He heat exchanger of helical design
- Hot header of an He/He heat exchanger (tests completed)
- Hot helium duct for the primary loop, including a duct bend and an expansion bellow
- Hot helium duct for the secondary loop, including a duct bend and an expansion bellow (tests partially completed)
- Hot gas valves
- Steam generator
3 Operating Experience and Test Results in KVK

3.1 10 MW U-tube Heat Exchanger

Fig. 4 shows the test unit. It has been designed by Interatom and Balcke-Dürr and manufactured by Balcke-Dürr. The heat exchanger tubes are bent to a U-shape. Primary helium flows along the tubes outside in a countercurrent arrangement, secondary helium is flowing inside the tubes. The hot header is positioned in the upper part of the heat exchanger, resulting in a rather short central tube. The cold gas header is separated from the support plate and is suspended by springs.

The test results are described below.

- Steady-state operation was performed at 40 - 100 % with 950 °C on the primary side and 900 °C on the secondary side:

A comparison of the most important design and test data for the U-tube heat exchanger is
FIG. 4. 10 MW HE/HE U-TUBE HEAT EXCHANGER

given in Fig. 5. In the upper part of the figure, the temperature profile over the heat exchanger tubes is presented for the 100% load case. There is good agreement, particularly in the hot region of the U-tube bundle. Due to a helium bypass flow the design and test values differ in the cold region. This also has an influence on the primary outlet.
FIG. 5. COMPARISON OF DESIGN AND TEST DATA OF THE U-TUBE HELIUM HEAT EXCHANGER

<table>
<thead>
<tr>
<th>LOAD 100%</th>
<th>PRIMARY SIDE</th>
<th>SECONDARY SIDE</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>DESIGN</td>
<td>TEST</td>
</tr>
<tr>
<td>TEMPERATURE [°C]</td>
<td></td>
<td></td>
</tr>
<tr>
<td>• INLET</td>
<td>950</td>
<td>950</td>
</tr>
<tr>
<td>• OUTLET</td>
<td>293</td>
<td>306</td>
</tr>
<tr>
<td>PRESSURE [bar]</td>
<td>39,9</td>
<td>40,0</td>
</tr>
<tr>
<td>PRESSURE DROP [bar]</td>
<td>0,50</td>
<td>0,40</td>
</tr>
<tr>
<td>MASS FLOW RATE [kg/s]</td>
<td>3,0</td>
<td>2,97</td>
</tr>
<tr>
<td>HELIUM VELOCITY [m/s]</td>
<td>17,6</td>
<td>17,5</td>
</tr>
<tr>
<td>OVERALL HEAT TRANSFER COEFFICIENT [W/m² K]</td>
<td>DESIGN</td>
<td>446</td>
</tr>
<tr>
<td>HEAT TRANSFER [MW]</td>
<td>10,24</td>
<td>9,53</td>
</tr>
</tbody>
</table>

temperature (306 instead of 293 °C), the secondary mass flow (2.71 instead of 2.9 kg/s) and the transferred thermal output (9.53 instead of 10.24 MW).

The inspection of the U-tube heat exchanger showed the essential components, such as tubes, hot header and insulation to be in good condition. Only the sheet metal enclosure of the tube sections exhibited signs of leakage.
close to the bend region. But that enclosure is only an expedient for testing. It will not exist in the full-scale heat exchanger.

- Non-steady state tests were performed such as:
  . Tenfold startup and shutdown of the heat exchanger at a transient level of $\pm 1$ K/min over a temperature range of 750 °C.
  . Cutoff of flow through the IHX at 25 % load over 1.5 h. No measurable convection resulted. Temperature profiles remained stable at a uniformly falling rate.
  . Fifty thermal cycles at 7 K/min and a temperature rise of 300 °C.
  . Simulation: disturbed primary inlet temperature between $+10$ K/min and $-50$ K/min for a max. temperature difference of 200 °C.

- The bearing forces for the load transfer system of the cold header and the bundle are within the range of the calculated values.

- The measured vibrations did not result in any noteworthy loading of the heat exchanger tubes.

- Successful tests showed leaktightness between the primary and secondary side of the IHX.

- Ultrasonic inspection of the IHX tubes was performed after a test time of more than 4700 hrs.

The following photo (Fig. 6) shows the 10 MW U-tube Intermediate Heat Exchanger at the fabricator's site.

Six tube sections are installed. The bends can be seen in detail.
FIG 6 10 MW HEAT EXCHANGER (U-TUBE TYPE)
PREPARATION FOR HE-LEAKAGE TEST

The photo was taken, while the heat exchanger was prepared for the helium leakage test.

3.2 10 MW Helical-tube Heat Exchanger

Figure 7 shows the heat exchanger manufactured by Steinmüller/Sulzer.

Primary helium enters the inner shell space from the bottom at a temperature of 950 °C and is cooled down to 293 °C by countercurrent-cross flow at the helical heat exchanger tubes. It then flows back to the primary outlet through the outer annular space, while cooling the pressure shell. Secondary helium enters the helical tubes at a temperature of 220 °C via the cold gas header, and exits from the heat exchanger at a temperature of 900 °C via the hot header and the central tube.
The next photo (Fig. 8) shows the heat exchanger bundle with 117 helical tubes in the final stage of mounting. This test unit is now installed in the Component Test Facility. Beginning of the test runs is scheduled for October '86.
FIG. 8. 10 MW HEAT EXCHANGER (HELICAL TYPE)
TEST BUNDLE WITH 117 HELICAL TUBES

Instrument Penetration
Water Cooling
Strain Gauge Insertion
Test Vessel
Insulation
Liner
Hot Header
864 Tube Nippel
Helium-Inlet

Operational Data
Temperature: 950°C - 1000°C
Diff Pressure: 44 bar

Dimensions
- Tube Nippel: 22 x 2
- Hot Header: ø 920 x 100 Wd
- Test Vessel: ø 2300 x 44.90
- Liner

Material
- 24663 (Nicrofer 5520 Co)
- 24663 (Nicrofer 5520 Co)
- 15-54 (T5Mo3)
- 14-876 (Incoloy 800 H)

FIG. 9. HOT HEADER CREEP BUCKLING TEST
3.3 Hot Header

Fig. 9 shows a cross section of the hot header as provided for large helium heat exchangers. Helium is flowing around the outside. The inside of the header had been equipped with a measuring insert for the creep buckling test. The 864 nozzles are closed by caps.

In order to perform the fatigue tests, the test arrangement had been altered (Fig. 10). The caps of the nozzles have been cut off. The helium is now flowing through the nozzles.

**FIG. 10. HOT HEADER FATIGUE TEST**
The test programme comprised three test series (Fig. 11).

1. The creep-buckling tests on the non-fatigued material of the hot header consisted of two tests with starting temperatures of 936 °C and 992 °C. The temperature drop was 250 °C in each case, whereby the differential pressure of 43 bar and the temperature transient of 0.42 K/min were identical.

2. In order to fatigue the material of the hot header, 656 cycles were run from 950 °C to 715 °C and then back up to 950 °C. The temperature transient amounted to ± 40 K/min.

<table>
<thead>
<tr>
<th></th>
<th>T [°C]</th>
<th>Δp [bar]</th>
<th>$\dot{T}$ [K/min]</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>CREEP BUCKLING TEST</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>936</td>
<td>43</td>
<td>$-0.42$</td>
</tr>
<tr>
<td></td>
<td>686</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>992</td>
<td>43</td>
<td>$-0.42$</td>
</tr>
<tr>
<td></td>
<td>742</td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>FATIGUE TEST</strong> (656 Cycles)</td>
<td>950</td>
<td>±40</td>
<td></td>
</tr>
<tr>
<td></td>
<td>715</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>950</td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>CREEP BUCKLING TEST</strong></td>
<td>970</td>
<td>43</td>
<td>$0$</td>
</tr>
<tr>
<td></td>
<td>903</td>
<td></td>
<td>$-0.21$</td>
</tr>
<tr>
<td></td>
<td>970</td>
<td>40</td>
<td></td>
</tr>
</tbody>
</table>

**FIG. 11. CYLINDRICAL HOT HEADER-MAIN TEST PARAMETERS**
3. Creep-buckling testing of the fatigued material of the hot header consisted of one test with a temperature drop from 950 °C to 903 °C with a transient of 0.21 K/min and a pressure difference of 43 bar, as well as continuous loading for a period of 455 h at a constant temperature of 970 °C and a differential pressure of 40 bar.

Fig. 12 shows a photo of the hot header. You can see the tube nozzles and some of the ther-
mocouples to measure the temperature distribution. The surface shows a grey protective oxide layer.

3.4 Primary Hot Gas Duct

Fig. 13 shows the test section of the primary hot gas duct with fibre insulation. The gas liner consists of graphite and of CFC. Adjustable supporting elements fix the gas liner radially and axially.

Tests on the primary hot gas duct under normal operating conditions have been completed. Fig. 14 gives a comparison of test results and design data. It is significant that the heat loss from the hot to the cold side is substantially lower than calculated (12 kW/m as compared to 36 kW/m).
<table>
<thead>
<tr>
<th></th>
<th>DESIGN</th>
<th>TEST</th>
</tr>
</thead>
<tbody>
<tr>
<td>TEMPERATURE [°C]</td>
<td></td>
<td></td>
</tr>
<tr>
<td>• HOT GAS</td>
<td>950</td>
<td>961</td>
</tr>
<tr>
<td>• COLD GAS</td>
<td>293</td>
<td>327</td>
</tr>
<tr>
<td>MASS FLOW RATE [kg/s]</td>
<td>3,0</td>
<td>2,9</td>
</tr>
<tr>
<td>PRESSURE [bar]</td>
<td>40,0</td>
<td>38,0</td>
</tr>
<tr>
<td>HELIUM VELOCITY [m/s]</td>
<td></td>
<td></td>
</tr>
<tr>
<td>• HOT GAS</td>
<td>19,1</td>
<td>19,6</td>
</tr>
<tr>
<td>• COLD GAS</td>
<td>9,1</td>
<td>9,5</td>
</tr>
<tr>
<td>HEAT EXCHANGE PER METER [Kw/m]</td>
<td>36,0</td>
<td>12,0</td>
</tr>
<tr>
<td>TEMPERATURE DROP PER METER [°C/m]</td>
<td>3,0</td>
<td>1,0</td>
</tr>
</tbody>
</table>

FIG. 14. PRIMARY HOT GAS DUCT WITH GRAPHITE LINER AND FIBRE INSULATION
Typical temperature profiles of the cold support tube and pressure tube and at the hot gas liner are shown on the left-hand side of the figure. The bandwidth of the temperature distribution is included. While the max. bandwidth is 16 °C at the gas liner, it drops to values of down to 5 °C at the pressure and support tube.

Fig. 15 shows the primary hot gas duct as installed in the KVK. In the foreground you see one of the two header of the cold helium.
3.5 Steam Generator

A cross section of the steam generator is shown in Fig. 16. The following operational experiences and test results have been gathered:

- Fault-free operation for 11000 hrs at high thermal loads and temperatures (up to 900 °C).
- No instability on the water steam side in all load ranges.

---

Operating Data

<table>
<thead>
<tr>
<th>He</th>
<th>Steam</th>
</tr>
</thead>
<tbody>
<tr>
<td>Flow rate</td>
<td>2.9 kg/s</td>
</tr>
<tr>
<td>Inlet Temperature</td>
<td>900 °C/950 °C</td>
</tr>
<tr>
<td>Outlet Temperature</td>
<td>210 °C</td>
</tr>
<tr>
<td>Inlet Pressure</td>
<td>410 bar</td>
</tr>
<tr>
<td>Outlet Pressure</td>
<td>405 bar</td>
</tr>
<tr>
<td>Power</td>
<td>0.5 MW</td>
</tr>
</tbody>
</table>

---

Dimensions and Materials

| Number of Tubes | 12 |
| Tube Length | 71 m |
| Heating Surface | 66.5 m² |
| Tube Dimensions | 25 x 4/25 x 2.5 |
| Tube Material | Incoloy 800 H/15 Mo 3 |
| Weld Material | W 81 E 36 |

---

FIG. 16. 10 MW STEAM GENERATOR
- Max. deviation from the average outlet steam temperature + 20 °C.
- Good agreement between measured and calculated tube-wall temperatures.
- Good agreement of the overall coefficients of heat transfer (< 2 %) between calculation and test.
- Total pressure drop 12 bar compared to 15 bar calculated.
- No inadmissible vibrations of tubes and other structures.
- Alkaline mode of operation in compliance with the VGB Guidelines.
- Perfect condition of the steam generator: bundle, insulation and pressure shell as shown during inspection performed in 1986.

Steam Reformer in EVA-II Plant

A test module of the steam reformer for the PNP-Project with a power of 5 MW was designed by Interatom and Steinmüller and manufactured by Steinmüller.

A longitudinal section through the test steam reformer as well as through a reformer tube is given in Fig. 17.

Primary helium enters the steam reformer at 950 °C. It flows upwards on the shell side of the reformer tubes and is cooled down to 700 °C. Below the insulated tube sheet the primary helium is conveyed to the steam generator. Having cooled to 300 °C, the helium is returned through the concentric annulus back to the reactor.
The process gas enters the steam reformer via the process gas chamber. It cools the support plate before being preheated in a small recuperator which is integrated in each reformer tube. Hereafter it enters the catalyst region, is heated up to about 810 °C by the hot helium and reformed to a synthesis gas with a high content of hydrogen. It leaves the steam reformer at 460 °C.

The test bundle consists of 18 reformer tubes and one inspection tube. The material of 13 tubes is Nicrofer 5520 (2.4663), the remaining 5 tubes are made of Incoloy 800 H (1.4676).
The test component (Fig.18) is under testing by KFA, Jülich in the EVA-II facility plant.

Four phases of steam reformer tests have been scheduled, the first three ones have been completed in April '86 with a testing time of 3800 hours.

- The steam reformer performance under normal operating conditions was demonstrated in test phases 1 and 2:
  - Helium temperatures between 900 and 950 °C
  - Helium pressures between 30 and 40 bar
  - Helium flow between 2 and 3.85 kg/s
  - Methane flow between 0.3 and 0.5 kg/s
  - temperature transients between 0.1 and 25 K/min
The high transients apply to disturbed operation.

- Phases 3 and 4 include non-stationary operation, e.g.
  . startup and shutdown
  . reduced system pressure
  . low steam/methane-ratio
  . simulation: tube failure
  . emergency shutdown behavior
  . leave standing in hot state
  . blocking of one reformer tube

5 Experience with the Materials in the KVK

In connection with the operation of KVK several tests have been carried out with of the material Nicrofer 5520 (2.4663).

The effect of impurities in the helium atmosphere on the corrosion behaviour has been measured by inserting material specimens into the KVK loop. Fig. 19 shows surface microstructure and change in carbon content of monitor samples removed after about 4000 hrs. The metallographic examination reveals only slight internal oxidation; the change in carbon content lies within the scatterband of laboratory test results.

Internal pressure creep rupture tests on circumferentially welded tubes (20 mm Ø x 2 mm) of the URKO and HELIX-heat exchangers have demonstrated the superior strength of the weldment in comparison to the base material. Fig. 20 shows that the tubes failed by developing cracks outside the weld in the base metal region.
FIG. 19. SURFACE MICROSTRUCTURE AND CHANGE IN CARBON CONTENT FOR KVK MONITOR SPECIMENS

FIG. 20. INTERNAL PRESSURE CREEP TESTS ON WELDED HELIX-HEAT EXCHANGER TUBES
HTR FUEL DEVELOPMENT AND QUALIFICATION — TREATMENT OF SPENT FUEL

G. KAISER
Kernforschungsanlage Jülich GmbH, Jülich

K. HACKSTEIN
Nuklear-Chemie und -Metallurgie GmbH, Hanau

Federal Republic of Germany

Abstract

The paper gives a description of design and performance requirements for spherical HTGR fuel elements together with the fabrication technology and the irradiation qualification. Different options for spent fuel treatment are presented and discussed.

1. Introduction

Mixed uranium/thorium oxide of 93 % uranium enrichment (High Enriched Uranium/Thorium fuel) had been the reference fuel for pebble-bed High Temperature Reactors in the Federal Republic of Germany until 1979. Thereafter, non-proliferation aspects and difficulties envisaged with the longterm supply of high enriched uranium led to a change into an uranium fuel with an initial enrichment of around 10 % (Low Enriched Uranium Fuel) 1/. As a consequence, emphasis in research and development work on fuel fabrication and qualification is now placed on this type of fuel which will be used for THTR follow-on reactors.

Recovery of valuable fuel materials by reprocessing was the prime target in spent fuel treatment as long as the use of high enriched uranium/thorium fuel had been under consideration. Neither economical reasons nor aspects of fuel conservation necessitate this treatment, however, when HTRs are operated with low enriched uranium in a high burnup modus. Therefore, interim storage in engineered surface storage facilities, followed by final disposal in a repository is the reference management concept for spent HTR fuel in the near future.
2. Fuel Element Development & Qualification

2.1 Design and Performance Requirements

The overall objective of the German HTR fuel development program was and still is to qualify an element which minimizes fission product release under normal, transient and accident conditions and which can be used for all types of HTR applications /2/.

The design of the FRG fuel element is shown in Figure 1. About $10^4$ coated fuel particles are dispersed in a graphite matrix. The fueled zone is surrounded by a fuel-free shell composed of the same graphite material. The overall diameter of the element is 6 cm with a 0.5 cm thick fuel-free shell. The coated particles are of Triso-type and contain a 500 µm diameter oxide fuel kernel. Triso refers to a four-layer coating with a silicon carbide (SiC) layer sandwiched between two high-density pyrocarbon (PyC) layers.

Table 1 summarizes the design data of the present FRG reference fuel and the operating requirements. To keep already developed fabrication processes as constant as possible, the particle design parameters of the LEU-fuel have been chosen to be identical the design data of the former HEU fuel.

![Diagram of HTR fuel element](image-url)
The Triso coating was introduced to fulfil all requirements defined by the different reactor projects, especially the stringent requirements for gas-turbine and process-heat applications. Because of the elevated operating temperatures of these advanced HTRs, the silicon carbide layer only provides sufficient retention of the more volatile solid fission products cesium and silver.

To guarantee a clean primary circuit for easy maintenance, the following specifications have been used as a development goal:

- Defect particle fraction due to fabrication $60 \times 10^{-6}$
- Defect particle fraction due to operation $200 \times 10^{-6}$

This tolerates out of a total of 1,000,000 coated particles not more than 60 individuals to have fabrication induced defects and, in addition, not more than 200 additional particles to fail during reactor operation.

### 2.2 Fabrication Technology

The fabrication process for spherical HTR fuel elements consists of the following steps /3-7/:

- Kernel formation by gel precipitation
- Chemical vapor deposition of the coating layers
- Preparation of resinated graphitic matrix powder
- Cold molding of the fuel element spheres
Kernel formation is achieved by a gel precipitation process as outlined in Figure 2. After formation of an uranium nitrate broth, spherical droplets are produced by a vibrational dropping technique. The droplets are aged to improve the internal structure and then washed to remove the nitrate. After drying and calcining, a special UO$_2$ related step - reduction of the calcined kernels to UO$_{2.6}$ - is applied. Kernel fabrication is completed by sintering to produce highly dense material.

Coating of the kernels (Fig. 3) is performed in fluidized beds by pyrolysis of hydrocarbons for PyC and of silane derivates for SiC. Because of criticality restrictions, the maximum coating batch size is limited to 10 kg of heavy metal oxide. A coater with this capacity, having a coating tube of 400-mm inner diameter, has been developed and sucessfully tested.

In the manufacturing process (Figure 4) phenol and hexamethylene-tetramine are first mixed at elevated temperatures with graphite powder. During the warm mixing process, resin binder is formed by chemical reaction of the components. The resulting material is then ground to a fine powder. A portion of the resinated powder is used to overcoat the coated particles; another portion is premolded together with these overcoated particles to form the fueled zone of the fuel elements. The remaining material is taken for the final molding of the fuel element giving the element its fuel-free shell.
Final steps in the process are: lathing the elements to obtain the appropriate diameter and heat treatment for binder coating and removal of impurities.

The crucial point in fuel fabrication is to reduce particle defects to the lowest possible level, that means at least below the design value of $60 \times 10^{-6}$. To reach that goal, two mechanisms which might cause particle defects have to be overcome:

- during the molding process, adjacent particles can mechanically interact, inducing local peak stresses that
might crack the coating layers. This effect can be avoided by overcoating the particles with layers of resinated graphite powder which act as spacers. The best in overcoating qualities have been obtained by classifying the overcoated particles using vibrating tables.

- extremely odd shaped particles - whether they are overcoated or not - cannot withstand the pressure applied during molding. To exclude this, the coated particles have also to be classified by vibration separation before overcoating.

Figure 5 shows how the product quality is influenced by the classification procedures. In 1977/1978 statistically no fuel element as fabricated was fully free of defect particles, and many elements (≈30 %) had three or more particles with a cracked coating. Once the technique of precise classification of overcoated particles was established, the defect rates reduced significantly. On a developmental as well as on a production scale, more than 50 % of the fuel elements were found free of particle defects, and only a very low percentage (≈ 5 %) showed three or more particle defects per fuel element. Introduction of the additional classification procedure for coated particles resulted in a significant further improvement of the product quality: more than 97 % of the fuel elements have been free of any failed particles, and in the remaining 2 to 3 % one particle defect only per element could be found. To date, the defect particle fraction is $1.5 \times 10^{-6}$, which is equivalent to 1 defective particle in a total of 40 spheres. This positive

![Figure 5. Percentage of SiC defect numbers in fuel elements.](image-url)
difference between design and actual measured values, the latter of which correspond to pilot-scale manufacturing, is a good basis of guarantee that the design goal can be met in a large-scale production too /8/.

2.3 Irradiation Qualification

Irradiation testing of fuel samples and single fuel spheres was and is performed in Material Test Reactors, mass testing of thousands of fuel elements in the 15 MJ/s experimental HTR power plant AVR.

Important performance related parameters applied in the simulation of normal operating conditions have been:
- irradiation temperature: 800°C-1200°C
- burnup: 8 - 14 % fima
- accumulated fluence of fast neutron: \(1-8 \times 10^{25} \text{n/m}^2\)

The operating behavior of LEU fuel has been investigated in a comprehensive basic irradiation program, including tests with the following main objectives:
- Reference test: irradiation of reference fuel elements under conditions enveloping the demands of the different HTR projects on temperature, fast neutron fluence, and burnup
- Determination of particle defects: testing of fuel under conditions exceeding the demands of the HTR projects with reference to fast fluence and burnup to investigate design limits
- Investigation of burnup influence: irradiation of fuel in thermal test reactors with low fast neutron fluxes to separate burnup-controlled effects from neutron-induced effects
- Investigation of solid fission product release: generation of input data for release calculations

The results of these experiments, summarized in Table 2, demonstrate the excellent irradiation performance. Neither the in-pile measured R/B values nor the results of post irradiation examination work indicate any particle breakage.

For mass testing, more than 40,000 LEU fuel elements have been inserted into the AVR in the meantime and irradiated up to
TABLE 2. FISSION GAS RELEASE RATE OF LEU UO₂ FUEL

<table>
<thead>
<tr>
<th>Experiment</th>
<th>Temperature (°C)</th>
<th>Fast Fluence (10²¹ cm⁻²)</th>
<th>Burnup (% fima)</th>
<th>R/B* (Kr)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reference test</td>
<td>1200</td>
<td>4.0</td>
<td>8.5</td>
<td>3 x 10⁻⁷</td>
</tr>
<tr>
<td></td>
<td>1000</td>
<td>6.0</td>
<td>10.1</td>
<td>2 x 10⁻⁷</td>
</tr>
<tr>
<td>Determination of particle defect rate function (35 µm SiC)</td>
<td>1200</td>
<td>8.0</td>
<td>11.6</td>
<td>9 x 10⁻⁸</td>
</tr>
<tr>
<td></td>
<td>1000</td>
<td>8.0</td>
<td>11.6</td>
<td>8 x 10⁻⁸</td>
</tr>
<tr>
<td>Determination of particle defect rate function (50 µm SiC)</td>
<td>1000</td>
<td>8.0</td>
<td>11.6</td>
<td>8 x 10⁻⁸</td>
</tr>
<tr>
<td>Investigation of burnup influences</td>
<td>1200</td>
<td>&lt;1</td>
<td>12.0</td>
<td>2 x 10⁻⁸</td>
</tr>
<tr>
<td>Investigation of solid fission product release</td>
<td>1200</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>1000</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>800</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

*End-of-life.

burnups of 10 % fima. Behavior under reactor operating conditions is also excellent; to date no fuel element damage was found.

2.4 Core Heatup Accident Simulation

In small modular HTRs the maximum fuel temperatures will not exceed 1600°C even under the severe accident conditions of core heatup and simultaneous depressurization /9,10/.

In the HTR 500 the design basis accident is defined by some loss of coolant while forced convection is still maintained by at least one loop removing the afterheat. Maximum fuel temperature is kept below 1250°C. The hypothetical accident sequence is given by an unrestrained core heatup in the depressurized reactor in combination with a failure of all active cooling systems. This sequence leads to maximum temperatures of the fuel in the pebble bed core of 2350°C /11,12/.

At these high temperatures, fission products can leave the fuel elements. Two failure mechanisms are prominent:

- thermal decomposition of the SiC layer, and
- SiC corrosion resulting from the interaction of fission products, mainly Pd, with the silicon carbide layer

In addition, diffusion processes in the coating layers can increasingly contribute to fission product release.

Core heat-up simulation tests with irradiated fuel elements have demonstrated the excellent performance of HTR fuel up to 1800 °C: The release of key nuclides is negligible for up to
hundreds of hours at 1600°C and small for dozens of hours at 1800°C (Fig. 6). Ceramographic sections through coated particles show no damage to fuel coatings.

SiC thermal decomposition is setting in at temperatures well beyond 2000°C. This leads to near 100% cesium release in temperature ramps to 2500°C.

FIG. 6. Accumulated release of fission products at 1800 and 1600°C from spherical fuel elements with LEU TRISO particles. *

*The contributions from contamination in the fuel element and from release during irradiation have been subtracted from measurement data.

3. Spent Fuel Treatment

3.1 Spent Fuel Treatment Options

For the management of spent HTR fuel 2 principal options exist:

- reprocessing or
- final disposal

In the Federal Republic of Germany the maximum amount of spent fuel to be expected by the year 2000 from the presently
existing HTR plants AVR and THTR is estimated to approach 3.5 million fuel spheres or 25 metric tons of uranium/thorium oxide. For this limited amount, final disposal is definitely planned. For the follow-up plants, which will be operated with low enriched uranium in a high burnup modus, technical and economic arguments call for the same treatment.

3.2 Interim Storage

To store fuel discharged from the AVR and to prepare, on a R&D basis, for the interim storage of spent HTR fuel, KFA was and still is operating two engineered storage systems of different designs: a vault-type storage facility and transportation/storage cask prototypes. The vault-type storage facility /15/, consists of a stainless steel rack with 36 storage positions. Each position can be loaded with 2 storage canisters containing about 1,000 fuel elements each. This facility has been at full capacity since April 1984. The total activity of stored fuel amounts to about 600,000 Curies, including about 10,000 Curies of Krypton-85 and about 2,000 Curies of Tritium. No release of radioactivity has been monitored to date and the maximum surface temperature of the sealed canisters has been only 10°C above the ambient air inlet temperature /16/.

Cast iron casks have also been used in Germany for some years as transportation/storage systems for spent nuclear fuel. An experimental program performed by KFA with prototypes of HTR casks (Fig. 7) demonstrated, impressively, the safety features of interim storage of spent HTR fuel in those systems (Table 3).

3.3 Final Storage

Spent HTR fuel has been included by the Physikalisch Technische Bundesanstalt, the designated operator of the future Federal repository, in the group of waste with medium heat production which will be disposed of in 300 meter deep boreholes. The development of the general storage techniques and the testing of necessary below-ground components like transport system, charging machine, borehole plug etc. is performed in
FIG. 7. HTR transportation/storage casks prototypes.

TABLE 3. CHARACTERISTICS AND OPERATIONAL DATA OF HTR-TRANSPORTATION/STORAGE CASKS PROTOTYPES

<table>
<thead>
<tr>
<th>Characteristics</th>
<th>Cast Iron</th>
</tr>
</thead>
<tbody>
<tr>
<td>Material</td>
<td>Cast Iron</td>
</tr>
<tr>
<td>Dimensions</td>
<td>2.6 m x 1.3 m Dia</td>
</tr>
<tr>
<td>Wall thickness</td>
<td>~0.3 m</td>
</tr>
<tr>
<td>Cask weight</td>
<td>~20 Mt</td>
</tr>
<tr>
<td>Capacity</td>
<td>1 THTR spent fuel canister or 2 AVR spent fuel canisters (2000 FE)</td>
</tr>
<tr>
<td>Cask loading for DEMO-test</td>
<td>2 AVR spent fuel canisters</td>
</tr>
<tr>
<td>Data of loaded material decay time</td>
<td>~1 YR</td>
</tr>
<tr>
<td>total activity</td>
<td>~42,000 Ci</td>
</tr>
<tr>
<td>Kr-85 activity</td>
<td>~700 Ci</td>
</tr>
<tr>
<td>H-3 activity</td>
<td>~95 Ci</td>
</tr>
<tr>
<td>decay heat</td>
<td>~180 Watts</td>
</tr>
</tbody>
</table>

| Operational Data                        |          |
| Cask leakage                            | ~10E-7 mbar 1/s |
| Canister leakage                        | ~10E-4 mbar 1/s |
| Cask surface temperature                | ~1°C above ambient temperature |
| max γ-dose rate (contact)               | ~12 mR/hr |
| max n-dose rate                         | ~4 n/cm E-2 s |
| Activity release over 8 months          |          |
| inside cask                             | Kr-85 ~5 x 10E-7 Ci |
|                                         | H-3 ~5 x 10E-8 Ci |
| outside cask                            | 'zero-release' |
connection with the R&D work related to the disposal of specific heat generating wastes resulting from LWR fuel reprocessing. Therefore, the emphasis of HTR work has been and still is on the following 4 major items:

- experimental evaluation of HTR fuel performance under normal and upset operational conditions of the repository
- development and testing of packaging concepts
- safety analysis work on the long-term safety of HTR fuel disposal in salt formations
- in situ demonstration of ultimate HTR fuel disposal in an experimental salt mine facility.

Available results on the release of gaseous fission products /23/ and on the leachability of solid fission products by salt brines /24/ indicate that the multiple barrier concept of the coated particles, developed for fission product retention in reactor operation, also ideally suits HTR fuel for final disposal. Not only are gaseous and solid fission products safely retained, but chemical attack from the outside is prevented too. The full demonstration program is, however, time consuming and it will take some additional years of R&D work until the HTR fuel will finally be qualified as a product which can be safely stored in a salt mine forever. The results of a planned in situ pilot experiment in the ASSE salt mine, in the frame of which a limited amount of AVR fuels will be stored for 5 years in monitored boreholes (Fig. 8), will be of considerable importance in this qualification process /19/.

FIG. 8. Schematic view of the MAW test drift EV (retrievable disposal test).
4. Summary and Conclusions

The state-of-art in HTR fuel cycle work in the Federal Republic of Germany can be summarized as follows:

- The development of processes and equipment for fuel fabrication has progressed to an advanced stage. Basic irradiation experiments with low enriched uranium fuel have been successfully completed; final tests simulating the operating conditions of commercial HTRs are in preparation. High temperature heating tests with irradiated material indicate an excellent behavior of the fuel also under core heatup accident conditions up to 1600/1800°C.

- The technology of interim surface storage of spent HTR fuel is developed. Taking advantage of the experience with the storage facilities in operation and using additional data resulting from ongoing R&D work, interim storage facilities for upcoming HTRs can be designed properly. The development of techniques for final storage of spent HTR fuel in salt domes is in progress and will be demonstrated until end of 1992.

References

1. D.F. Leushacke and G. Kaiser, "HTR fuel cycle program in the FRG" Trans. ENC 31 (1979), 190


3. M. Kadner and J. Baier, "Ober die Herstellung von Brennstoffkernen für Hochtemperaturreaktor-Brennelemente" Kerntechnik 18 (1976), 413


17. R. Duwe, U. Brinkmann, Freisetzung gasförmiger Nuklide aus lagernden HTR-Brennelementen", JTK 1985, Tagungsbericht, ISSN 0720-9207


CHARACTERISTICS OF HTGR SPHERICAL FUEL ELEMENTS

A.S. CHERNIKOV, L.N. PERMYAKOV, L.I. MIKHAILICHENKO, S.D. KURBAKOV
State Committee on the Utilization of Atomic Energy,
Moscow, Union of Soviet Socialist Republics

Abstract

The paper presents the investigation results for the basic characteristics of spherical fuel element components: fuel microspheres (dimension, shape, density, structure, strength), pyrocarbon and silicon carbide coatings (thickness, density, macro- and microstructure, crystalline structure, thermal stability, gaseous fission product release, radiation stability etc) and matrix graphite (density, effective thermal conductivity, linear thermal expansion coefficient, elasticity modulus, bending strength). In manufacturing fuel microspheres sol-gel process and slip-casting methods were used; coating deposition was accomplished in fluidized bed, high-density pyrocarbon coatings being deposited from CH₄-Ar or C₃H₆-Ar; SiC coatings - from CH₄-SiCl₄-H₂-Ar or CH₄SiCl₃ - H₂ - Ar mixtures; matrix graphite was prepared on the basis of bulk graphite fillers, 30 PG and MPG-6.

1. Introduction

The concept of the pebble-bed core of high-temperature helium-cooled reactors (HTGR) with spherical fuel elements is being developed in the USSR for VGR-50 and VG-400 plants /1/.

The current efforts are directed to improvement and stabilization of the main characteristics of the spherical fuel element (FE) components: fuel microspheres (FM), coated particles (CP) and matrix graphite (MG).
FM (500 μm) are produced from uranium dioxide; fission products (FP) are retained within CP by the multi-layer barrier coating of pyrocarbon (PyC) and silicon carbide (SiC); CP dispersed over the graphite matrix constitute a spherical fuel element of HTGR.

The realized processes of FM, CP, FE manufacturing are based on utilization of various versions.

FM - sol-gel process and slip casting method;
CP - deposition of so-called high-temperature and low-temperature PyC coatings (HTC and LTC), as well as SiC coatings from SiCl₄·CH₄·H₂·Ar and CH₃SiCl₃·H₂·Ar mixtures;
FE - matrix graphite based on bulk carbon graphite materials, of various types (e.g. 30 PG and MPG-6 type graphites).

Comparison of the main characteristics of FE components (FM, CP and MG) realized by the above versions is the aim of the present work.

2. Fuel microspheres
2.1. Main requirements

In selection of FM spheroidization process various physical and chemical methods were considered and tested, including:
- various versions of so-called dry agglomeration from finely dispersed powders;
- slip casting methods;
- sol-gel method.

Comparison of the main geometrical and structural characteristics of FM manufactured using these methods as well as analysis of various spheroidization versions showed that the most acceptable of them are the sol-gel and dross spheroidization methods /2/. Both methods allow FM of the required quality, meeting increasing requirements to HTGR fuel to be obtained.
As shown by the literature analysis and investigations FM quality is determined by the following main characteristics:
- density (apparent and pycnometric);
- average dimension;
- non-sphericity factor;
- crushing force;
- structural characteristics (grain and pore sizes, pore distribution over FM, particle surface relief, etc);
- phase and chemical composition.

Among the specific requirements to radiation performance of HTGR spherical fuel elements the most important one is that FP release in them would not exceed $10^{-5}$ (R/B), which is equivalent to one damaged CP per $10^4$ particles. These requirements can be only met when the starting product, i.e. FM, is highly homogeneous. Therefore, along with ensuring specified average values of the FM basic characteristics a high degree of their homogeneity is needed both within one batch and between FM batches.

2.2. Spheroidization methods

The sol-gel method of FM spheroidization from uranium dioxide is based on the sol-gel process similar to the "H-process" of KFA reported in [3]. The working solution contains uranium, urea and urotropine in relation 1:2:1.5 and has a density of 1.65 g/cm$^3$. Spherical particles produced by internal gelation in petroleum jelly at 90-95 °C are washed in CCl$_4$ and ammonia solution, then subjected to azeotropic drying in CCl$_4$ and calcinated in argon - 4.0% hydrogen mixture atmosphere.

Another method of FM production is based on spheroidization of plastic slip which is prepared of finely dispersed uranium dioxide powder and binder. At 70-75 °C the slip mass is highly fluid, it is injected drop-by-drop into glycerine, where it solidifies with formation of homogeneous spherical particles.
After spheroidization particles are washed from glycerine and then the binder is distilled out of them.

Particles produced using both methods are sintered in argon at 1500-1700 °C, sieved till the desired fraction is obtained and classified by their shape.

2.3. FM characteristics

Figs. 1 and 2 show the curves for FM distribution by their sizes, and figs. 3 and 4 present the curves for FM distribution by non-sphericity factor ($D_{max}/D_{min}$) for particles manufactured by the slip spheroidization and sol-gel methods, respectively.

---

![Graph showing FM characteristics](image)

**Fig. 1.** Distribution of UO$_2$ fuel microspheres manufactured slip casting method by sizes:
- a - batch 9789;
- b - batch 9827;
- c - batch 9858
Fig. 2. Distribution of UO₂ fuel microspheres manufactured by sol-gel method by sizes:
- a - batch 9792;
- b - batch 9726;
- c - batch 9857

Fig. 3. Distribution of UO₂ fuel microspheres manufactured by slip casting method by sizes:
- a - batch 9858;
- b - batch 9827;
- c - batch 9789
Fig. 4. Distribution of UO₂ fuel microspheres (sol-gel method) by non-sphericity factor:
  a - batch 9726;
  b - batch 9857;
  c - batch 9792

Fig. 5. Cumulative distribution of UO₂ fuel microspheres by porosity:
  a - batch of sol-gel-manufactured microspheres;
  b - batch of slip casting
Spread in porosity, measured by the quantitative metallography method using the image analyzer, is characterized by the curves of FM distribution by porosity presented in fig 5.

It follows from comparison of the curves that the geometrical characteristics of spherical particles manufactured by the two methods are good from the homogeneity viewpoint. FM manufactured by the sol-gel method have somewhat better homogeneity in the geometrical characteristics and FM produced by slip-spheronidization method are more homogeneous in porosity. The typical microstructure of FM prepared by both methods is shown in fig 6.

Fig. 6. Microstructure of UO₂ fuel microspheres manufactured by slip casting (left) and sol-gel (right) methods: a, b X100; c, d - X240
3. Coated particles

3.1. Manufacturing methods

The CP design has been developed on the basis of calculation taking into account operation conditions /4/ and implies, as mentioned above, use of highly dense PyC coatings of two types: HTC or LTC, as well as SiC layer. The requirements to the density and geometrical characteristics of protection layers of coatings, we reported earlier in /5, 6/, are nearly identical for application of any PyC types. In deposition of high-density LTC layer on buffer PyC layer, the medium density (1.4-1.6 g/cm³) PyC layer in the CP design is absent.

Deposition of protection coatings on CP is accomplished in the apparatus of pseudofluidized bed with a conic gas-distributing device.

Table 1 presents protection coating deposition conditions and lists the reagents used.

---

**TABLE 1. PROTECTION COATING DEPOSITION CONDITIONS**

<table>
<thead>
<tr>
<th>Coating layer</th>
<th>Gas mixture</th>
<th>Reagent concentration, kPa</th>
<th>Temperature, °C</th>
</tr>
</thead>
<tbody>
<tr>
<td>Buffer (PyC-1)</td>
<td>C₂H₂ - Ar</td>
<td>40-60</td>
<td>1500-1550</td>
</tr>
<tr>
<td>Dense isotropic layers:</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>- HTC</td>
<td>CH₄ - Ar</td>
<td>10-20</td>
<td>1900-2000</td>
</tr>
<tr>
<td>- LTC</td>
<td>C₃H₆ - Ar</td>
<td>15-30</td>
<td>1250-1400</td>
</tr>
<tr>
<td></td>
<td>SiCl₄</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>SiCl₄ - CH₄ - H₂ - Ar</td>
<td>1.0±2.5</td>
<td>1450-1550</td>
</tr>
<tr>
<td></td>
<td>CH₄</td>
<td></td>
<td></td>
</tr>
<tr>
<td>SiC</td>
<td>1.0±2.0</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>(Si/C)=1.0±2.0</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>CH₃SiCl₃ - H₂ - Ar</td>
<td>2.5±3.5</td>
<td>1500-1600</td>
</tr>
</tbody>
</table>

452
Rejection of CP with damaged coatings implies their treatment in the acid and subsequent distribution by their densities. CP of the specified shape and size are then separated from the batch using appropriate classifier.

3.2. CP coating characteristics

The typical macrostructure of HTC and LTC as well as with SiC layer is shown in fig 7. The microstructure of high-density HTC and LTC, found by oxidation in the glow-discharge plasma /7/.

Fig. 7. Macrostructure of HTC (a) and LTC (b) coated particle sections, X70
and that of SiC layer after electrochemical pickling has a fine-crystalline structure with equiaxial grain (fig. 8).

The PyC crystalline structure was determined in all cases as graphite-type one hexagonal but with no long range order). The results of X-ray and neutron diffraction investigations indicate that PyC grains consist of several crystallites (0.002–0.03 μm in size), whose number and orientation determine the grain shape and size. SiC coatings obtained using the above mentioned mixtures had a cubic modification with the lattice parameter a=0.4360±0.001 nm. We have not observed any essential effect of the type of gas mixture on the level of other properties of high-density (≥ 3.20 g/cm³) SiC coatings.

Fig. 8. Microstructure of high-density coatings HTC (a) X2000; LTC (b) X2000 and SiC (c) X1000
The difference between LTC and HTC is accounted for by differences in their structure dispersity. The sizes of LTC crystallites we have obtained are essentially (by 5-10 times) smaller than those of HTC, while interlayer distances in LTC are longer than those of HTC.

3.3. Results of CP tests

The characteristics of typical CP batches with HTC and HTC as well as those with SiC coatings are listed in table 2.

<p>| TABLE 2. CHARACTERISTICS OF COATED PARTICLES WITH LOW-TEMPERATURE (LTC) AND HIGH-TEMPERATURE (HTC) COATINGS |
|-------------------------------------------------|-------------------------------------------------|-------------------------------|</p>
<table>
<thead>
<tr>
<th>Parameters</th>
<th>Coated particles with LTC</th>
<th>Coated particles with HTC</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel microsphere :</td>
<td></td>
<td></td>
</tr>
<tr>
<td>diameter, μm</td>
<td>490</td>
<td>500</td>
</tr>
<tr>
<td>sphericity factor</td>
<td>1.03</td>
<td>1.02</td>
</tr>
<tr>
<td>density, g/cm³</td>
<td>10.8</td>
<td>10.5</td>
</tr>
<tr>
<td>oxygen coefficient, O/U</td>
<td>2.0003</td>
<td>2.0020</td>
</tr>
<tr>
<td>Coated particle coating thickness, μm</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pyc-1</td>
<td>95</td>
<td>90</td>
</tr>
<tr>
<td>Pyc-2</td>
<td>67</td>
<td>35</td>
</tr>
<tr>
<td>yC-3</td>
<td>-</td>
<td>40</td>
</tr>
<tr>
<td>SiC</td>
<td>50</td>
<td>53</td>
</tr>
<tr>
<td>Pyc-5</td>
<td>60</td>
<td>50</td>
</tr>
<tr>
<td>coating density, g/cm³</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pyc-1</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td>Pyc-2</td>
<td>1.85</td>
<td>1.55</td>
</tr>
<tr>
<td>Pyc-3</td>
<td>-</td>
<td>1.60</td>
</tr>
<tr>
<td>SiC</td>
<td>3.2</td>
<td>3.2</td>
</tr>
<tr>
<td>Pyc-5</td>
<td>1.82</td>
<td>1.8</td>
</tr>
<tr>
<td>Layer contamination:</td>
<td></td>
<td></td>
</tr>
<tr>
<td>surface, x10¹³ Cl/cm³</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td>bulk, g²³⁵/g coating, x10⁵</td>
<td>1.0</td>
<td>3</td>
</tr>
<tr>
<td>crushing force CP, kg</td>
<td></td>
<td></td>
</tr>
<tr>
<td>mean</td>
<td>7.7</td>
<td>6.6</td>
</tr>
<tr>
<td>max/min</td>
<td>10.0/5.2</td>
<td>8.0/4.2</td>
</tr>
<tr>
<td>Leakage at the stage of &quot;weak&quot; irradiation, X 10⁶</td>
<td>1.6</td>
<td>2.0</td>
</tr>
</tbody>
</table>

*The uranium dioxide FM manufactured by slip casting methods (CP with LTC) and sol-gel process (CP with HTC) were used.
The mass-spectrometric measurements of the gas volume, within CP carried out earlier /8/ showed that at O/U not exceeding 2.001 in FM after coating deposition at temperatures lower than 1400 °C $P_{CO}$ in the beginning of the exploitation will be 20-40 ata.

CP thermogradients tests made it possible to establish /2/ that at a given CP operation temperature of 1250 °C and fuel burnup to 15% fima CP migration will not exceed 3-5 μm.

Long-term heating of CP under isothermal conditions at 1800 °C gives evidence of the integrity of protection coating layers. Short-term tests of CP of various temperatures indicate that damages in individual CP appear beginning from 2200 °C, which is in agreement with the data of work /9/.

Thermocycling (temperature range 350 °C - 1500 °C at a rate up to 8 K/s with the total number of cycles up to 2000) did not result in violation of the integrity and initial leaktightness of CP.

In different types versions of coat designs for four and five-layer CP and PyC structure composition (LTC and HTC), mentioned above, CP irradiation performance was somewhat different. The difference manifests itself mainly at the initial irradiation stage when fission gas release at 1000-1200 °C from coatings of both versions corresponds to $R/B < 10^{-7}$, but rather when critical burnups corresponding to the beginning of coat destruction have been reached. For example, at fully identical thicknesses of protection layers, in CP with LTC depressurization begins at higher (by 20-40% rel) burnups then in CP with HTC. This is because radiation shrinkage and, therefore, cracking of protection layer in LTC is less than in HTC.

The results of metallographic investigations permit to make a conclusion that CP performance under irradiation does not practically differ from that described in the literature /10/.

On example of CP irradiated at about 1200 °C and at 15% fima...
burnup the following results can be pointed out:

- porosity structure is fully rearranged, pore size increased, distribution by sizes became wider and their void content increased;
- metallic FP represent light bright inclusions a part of which is connected with pores;
- etching in boiling nitric acid revealed grey inclusions locating both in the body and on the grain boundaries, which seem to be oxides of metallic FP not solved in the fuel matrix;
- after irradiation microhardness of the fuel matrix increased from 640±60 to 950±150 kg/mm².

4. Spherical fuel elements

The currently adopted technological process of PE manufacturing is based on powder metallurgy methods, such as compaction with subsequent thermal treatment of blanks /11/.

In Ref. 12 the main results of technological and experimental investigations on substantiation of PE design for the VGR-50 and VG-400 reactors which are under development in the USSR are presented.

Because of low volume filling of CP in the two-component PE (<8 vol.%)/12/, its thermophysical and mechanical characteristics are determined essentially by the matrix graphite properties. The latter, in turn, can be varied on account of initial materials (granulometric and component composition of moulding powders). In this section of the paper emphasis is given to the influence of one of the basic factors predetermining the PE operation characteristics, namely, the graphite filler type.

The matrix composition fillers were prepared from artificial bulk graphites based on petroleum cokes: calcinated (30 PG) and non-calcinated (MPG-6). The moulding powder of the two-component granulometric composition contains 0.2*0.4 and 0.05 mm graphite fractions and an appropriate amount of coal-tar pitch binder.
Initial graphites differ essentially both in main physical (thermal conductivity, strength etc) and structure (perfection degree, porosity, crystallite size) characteristics which, to some extent, are also MG and FE characteristics.

Table 3 lists the results of investigation of the main MG and FE characteristics, determining the operability level of the latter.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>30PG-based matrix graphite</th>
<th>MPG-6-based matrix graphite</th>
</tr>
</thead>
<tbody>
<tr>
<td>Density, g/cm³</td>
<td>1.89</td>
<td>1.92</td>
</tr>
<tr>
<td>Effective thermal conductivity at t = 250°C, W.m⁻¹K⁻¹</td>
<td>74</td>
<td>78</td>
</tr>
<tr>
<td>Specific electrical resistance at t = 20°C</td>
<td>1580*</td>
<td>1240*</td>
</tr>
<tr>
<td>Coefficient of linear expansion (CLE) in the temperature range 20-1000°C, X10⁶ degree⁻¹</td>
<td>5.6*</td>
<td>6.5*</td>
</tr>
<tr>
<td>Elasticity modulus, X10⁻⁴ MPa</td>
<td>0.99*</td>
<td>1.07*</td>
</tr>
<tr>
<td>Bending strength (t = 20°C), MPa</td>
<td>24*</td>
<td>28*</td>
</tr>
<tr>
<td>Breaking force of FE at t = 20°C, kN</td>
<td>27</td>
<td>39</td>
</tr>
<tr>
<td>FE wear, g</td>
<td>9.8</td>
<td>4.2</td>
</tr>
</tbody>
</table>

*The data refer to specimens cut in direction parallel (numerator) and perpendicular (denominator) to the moulding axis.

The matrix graphites investigated are high-density carbon materials with a reasonable combination of thermophysical and physical mechanical characteristics comparable with artificial bulk graphites. At the same time presence of some amount of the second phase of non-graphitized carbon (pitch coke) results in a noticeable change in such properties as the electrical and
thermal conductivities, CLE, without the strength characteristics being affected. Comparison of both MG types shows that use of high-strength non-calcinated coke-based graphite (MPG-6 type) as a filler makes it possible to increase FE strength and wear resistance by about 1.5-2 times. Insignificant texture of such a filler predetermines lower anisotropy of the MG properties (e.g., by CLE) than that of 30 PG-based MG, properties. At the same time the 30 PG based MG possessing a more perfect structure than MPG-6 has lower CLE and elasticity modulus, which is favourable for the FE radiation stability. As far as the radiation stability of the MPG-6 material is concerned it is expected that it should be higher due to high strength, low anisotropy of this material and low level of radiation dimensional changes in the graphite-filler itself; currently this material is in the stage of reactor tests.

5. Conclusion

The complex of investigations performed permitted the main characteristics needed for meeting the operation requirements to HTGR spherical fuel elements to be selected.

References


The paper contains results of studies for optimizing fuel utilization in HTGRs. Different U235 enrichments, core power densities and heavy metal concepts have been analyzed and results are presented and discussed.

1. Introduction

Possibly one of the best thermal reactor system for the utilisation of thorium is the pebble bed kind of high temperature gas cooled graphite reactor. In this system, the fuel is in the form of coated particles, which are then formed into balls. The reactor core vessel is filled with such balls and they are cooled by helium passing through it. The specific features which make this system attractive for thorium U233 cycles are the relatively hard spectrum, the continuous refuelling facility, and the absence of structural material inside the reactor core. All these tend to raise the conversion ratio in the core.

The HTGR core designs that one meets with in literature come mostly from the UK, the USA, and the Federal Republic of Germany. The nuclear design in these is generally optimised for power generation cost, and acts to the great detriment of fuel utilisation characteristics. The conversion ratios quoted in these designs are generally of the order of 0.85 or less. In India we have made some studies
in optimising on fuel utilisation. The results of these studies are presented in this paper.

2. The System

The reactor system that has been considered is very similar to the THTR reactor. The fuel is assumed to be in the form of tiny particles of \((\text{Th} + \text{U}^{233})_2\) mixed oxide surrounded by soft graphite and coated with pyrocarbon. Large numbers of such coated particles are embedded in a graphite matrix and formed into spheres of 60 mm diameter. The core is assumed to be cylindrical in shape, with a diameter of 560 cms and a height of 600 cms. A conical shaped lower end leading into a fuel element discharge pipe permits spent fuel to be continuously discharged, while fresh fuel is continuously fed in from the top.

The hard spectrum in such a core results in low cross-sections and large mean free path leading to high leakage. In the geometry described above, the leakage of neutrons is about 15%. In order to improve the fuel utilisation, it is necessary to capture these neutrons. To do this, the core is surrounded by a thorium blanket which absorbs these neutrons and raises the conversion ratio.

3. The Calculational Formalism

This study is intended to be a survey kind of study, which necessarily entails calculations on a large number of cases. It was therefore decided to use a few group treatment for the computations, even though a rigorous calculation would require multigroup treatment.
Four groups of neutrons were used for the calculations. The group structure was as follows: Group 1 is above 821 kev. Group 2 is 5.5 kev - 821 kev. Group 3 is 0.625 ev to 5.5 kev and Group 4 lies below 0.625 ev. This group structure was assumed to be adequately accurate for a survey type calculation. The line of reasoning was as follows:

In group 1, the spectrum would correspond to the fission spectrum. In thermal reactors, the internal spectrum of group 1 is likely to stay constant from system to system and so it is permissible to treat it as one group.

In group 2, the primary effect is slowing down and so the intra group spectrum could well depend very strongly on the moderator. However, in this group, all cross-sections are quite low and fairly constant, so that even if the spectrum within the group is not very well calculated, the error caused by any inaccuracies in the calculated group 2 cross-section for any thermal reactor will be small.

In group 3, the overwhelming effect is that of resonances. In the high temperature graphite reactor, with its hardened spectrum, the effect is likely to be felt in the third group. The internal spectrum in group 3 could be different from what it is in other thermal reactors. It is possible that the spectrum is more peaked for the HTGR towards the lower edge of the group. The third group spectrum will thus peak towards the lower group boundary. Since most of the larger resonances are near the lower end of the group, this will cause the resonance absorption to increase when compared to the normal thermal reactors.
It was decided that some increase in the resonance integral of thorium in the third group is thus in order. To estimate the magnitude of this increase, this method of calculation was applied to the THTR core. Reactor physics data was obtained from the Nuclear Reactors Directory. The date of the information is 1972. Although a lot of literature on THTR has been received of late, we were unable to locate an adequate amount of reactor physics data in them. The 1972 date gave the core description as well as the burnup, which was quoted as 113,000 Mwd/T. The maximum excess reactivity was quoted as 111 mk. On the basis of these available numbers, it was decided to increase the resonance integral in the third group by about 15%, in which case both the burnup and the reactivity are fairly well-predicted.

In the thermal group, there are only three nuclides which influence the group internal spectrum. They are U233, thorium, and carbon. Of these, thorium and carbon are 1/v absorbers and treated as such. The effect of U233 on the spectrum is accounted for by considering it a 1/v material. Since the U233 cross-section does not have very sharp variations, it is assumed that this approximation is adequate for survey calculations which we have in mind.

4. The results of the Study

Calculations were made for different U233 enrichments in thorium, different values of heavy metal content in the fuel balls, and different power densities. In order to evaluate the merit of each case, two quantities were calculated each time. One was the discharge burn-up. The other
is fissile inventory ratio, which is defined as the ratio between the fissile content in the discharged fuel, and that in the fresh fuel.

Fig. 1 shows both discharge burn-up and fissile inventory ratio plotted as a function of the U233 content in the thorium. As is to be expected, the burn-up increases with enrichment, while the fissile inventory ratio decreases.

FIG. 1 HTGR CHARACTERISTICS AS FUNCTION OF ENRICHMENT
Fig. 2 shows a plot of the same two quantities, namely discharge burn-up and fissile inventory ratio, as a function of the heavy metal content of the ball. This is expressed in terms of the weight percent of the ball formed by the heavy element. As the heavy metal content increases, the spectrum becomes harder and so the conversion ratio increases. At the same time, because of the decline in fission reactions of the thermal neutrons, the burn-up decreases. It can be seen that at lower heavy metal concentrations, the discharge burn-up increases dramatically. The increase in burn-up is almost of an order of magnitude. In the light of fig. 2, it appears quite understandable how one could be tempted to give
up the prospect of better fuel utilisation and prefer to take advantage of the extremely attractive burn-up which this reactor system can give.

Fig. 3 shows the effect of power density on the discharge irradiation and fissile inventory ratio. The discharge irradiation comes down very sharply with increased power density. This figure does not immediately show any effect on conversion ratio. That effect will however, come as a two step process. With a decrease in discharge burn-up,
an effort will be made to increase the discharge burn-up by increasing the enrichment, and this will cause the conversion ratio to come down.

5. **Conclusions**

This study seems to point that conversion ratios in excess of 1.0 are possible with the HTGR. Two or three cases which look promising have been selected for a more rigorous analysis using multigroup codes. These cases are the following:

<table>
<thead>
<tr>
<th>Enrichment % U_{233}</th>
<th>Wt. % of HM in the ball</th>
<th>Power density (kw/litre)</th>
</tr>
</thead>
<tbody>
<tr>
<td>6.9</td>
<td>25.0</td>
<td>3.0</td>
</tr>
<tr>
<td>7.0</td>
<td>25.0</td>
<td>3.0</td>
</tr>
<tr>
<td>6.9</td>
<td>24.0</td>
<td>3.0</td>
</tr>
<tr>
<td>6.9</td>
<td>25.0</td>
<td>4.0</td>
</tr>
</tbody>
</table>

From the point of view of fuel utilisation, the HTGR is a very promising system when thorium is used as fertile material. The disadvantage is the rather high specific inventory that is required if conversion ratios exceeding 1.0 have to be attained. The other disadvantage is the problem of initiating the self-sustaining cycle. In Indian conditions, where enrichment facilities are not available, the route to SSET lies via self-generated plutonium. But plutonium is a disadvantaged fuel in thermal reactors. This disadvantage is accentuated in the HTGR spectrum which is harder than other thermal reactors, but not hard enough to
reach those energies at which the $\alpha$ of plutonium starts decreasing again. One solution will be to produce the first charge of $\text{U}^{233}$ in some other system like a heavy water reactor, or a fast reactor blanket.

6. **References**

1) Various publications on THTR-300.
USER’S PERSPECTIVES ON GAS-COOLED REACTORS
(Session G)
FUTURE APPLICATIONS OF THE HIGH TEMPERATURE REACTOR

A. KLUSMANN, M. STELZER
Ruhrkohle Öl und Gas GmbH,
Bottrop, Federal Republic of Germany

Abstract

The advanced reactor systems, the fast breeder and the high temperature reactor (HTR), have reached maturity for commercial use. At the same time nuclear energy is in a critical situation. In addition to reliability, safety, and competitiveness new requirements have come up for future projects in many countries. More safety, an easy fuel cycle, a wide variety of process applications, public acceptance and low environmental impacts are demanded by the public. The HTR can fulfil most of these criteria as is proven by the German prototype plants 15 MW AVR-Jülich and 300 MW THTR Schmehausen. The Federal Republic of Germany can contribute to the future use of nuclear energy a new safe reactor system with application possibilities beyond the electric power production.

1. INTRODUCTION

The development of the pebble bed reactor has been pursued for 30 years in the Federal Republic of Germany. The main goals have been reached: The 15 MW AVR Jülich operates since 1967 and at temperature levels of up to 950 °C since 1974 and the 300 MW THTR Schmehausen is on full power. The steel vessel concept for small reactors of 80-100 MW (AVR-type) and the prestressed concrete construction for 500 - 600 MW (THTR-type) plants are ready for commercial use.

For the future application a great variety of technical options exists. Besides the normal electricity production the use of nuclear process heat of 1.000 °C allows projects from coal gasification to water fission. The development of new processes, materials, components, and their adaption to the high temperature reactor is therefore subject of a program in the Federal Republic of Germany.

Until the end of 1985 appr. 7 billion DM were spent on the HTR development. This sum includes the construction projects 15 MW AVR Jülich and 300 MW THTR Schmehausen and the expenses for nuclear process heat development work of appr. 1,4 billion DM. The Minister of Economy, Small Business, and Technology of North Rhine Westphalia, the Federal Minister of Research and Technology, and the German industry have funded the program.

2. ENERGY SITUATION

The energy reserves of the world are not abundant and secure for ever. The present picture of a cheap energy surplus does not correspond to long-term realities. Petroleum and natural gas account for more than half of the energy consumption worldwide.
Large proportions have to be imported from politically unstable regions into those countries with no major own sources. The oil prices are arbitrary and have great impacts on the economy of all countries.

The search for alternative energy supply systems concentrates on coal and uranium conversion to electric power and heat or gaseous and liquid products. This is a comprehensive challenge to science and technology. But these new ways are required since in the next two decades the energy consumption in the world will increase with the population. Therefore the full benefit from the two raw materials coal and uranium with the biggest resources should be reached.

A long time is needed to prepare for the industrial use of new technologies for the next generation, in particular if nuclear application is envisaged beyond generating of electricity. The optimum system that delivers different energies such as electric power, gas and liquids by interlocked energy conversion has to be found. The key component HTR is available, but the technologies to be applied are not yet developed for large-scale industrial use.

3. APPLICATION SURVEY

The basic idea to use high temperature nuclear heat is to break complex natural materials such as coal or heavy hydrocarbons into their basic components. These products are reformed by various synthesis ways into new materials with the required specific characteristics. The natural resources of coal, petroleum, and natural gas can be stretched by synthetic fuels, such as hydrogen or carbon monoxide, generated in nuclear/coal complexes. These very clean substances can be produced and used with considerably reduced environmental impact.

First steps down this road have been taken in the Federal Republic of Germany in the last 20 years. The prototype HTR's are in operation. Subsequent plants of 500 MW or 100 MW are to follow. With the next reactors the heat range of 150 - 530 °C can be served. More than half of the heat market requires this temperature level. Even for temperatures up to 750°C the materials and components are available today. For temperatures around 1,000 °C, however, more long-term experience and special processes are necessary to take full advantage. But also for this high temperature level very promising materials and components qualifications have been achieved.

Only mass production of products justifies the use of large amounts of nuclear heat. Nuclear coal gasification is therefore a major research area. The gas mixtures obtained with the heat from the HTR by coal conversion can be fed into the gas networks and used as fuel gas for home-heating or for industrial processes. Also synthesis gas for the production of synthetic materials, fertilizers, and liquid fuels can be produced. But for both purposes industrial plants are not yet under consideration.

Further proposals concern the steel industry, such as nuclear smelting of iron ore. A switch from blast furnaces and coke to shaft furnaces is envisaged in these proposals. - Also the cold
transport of heat from the HTR may become a new area. The conversion of methane into hydrogen and carbon monoxide and then in the place of consumption the conversion of the fission gases back to methane are proposed. - Also the noble form of energy, hydrogen, can be produced by coal or heavy hydrocarbon gasification or later by direct water splitting.

Thus the pebble bed reactor application to the heat market make the industrial countries more independent from energy imports. The HTR can contribute almost all energy qualities, but there is a long technical way until these routes will be competitively established. This applies also for the use of a helium gasturbine in direct cycle.

4. PROJECTS

The 300 MW THTR Schmehausen represents the basis for the commercialization of the HTR. It is the modern German reference plant for all subsequent projects in the power range between 100 and 600 MW. The extensive design reserves can be reduced with growing operating experience on the way to competitiveness. By a twin design with one secondary part a power range up to 1,200 MW can be covered. The next step of commercialization is a 500 - 600 MW steam cycle plant for electric power generation with the possibility of process steam and district heat extractions.

Industrial reactors designed for 100 MW power output and as multiple stations for 200 and more MW make use of the proven principle of the 15 MW AVR Jülich steel vessel design as well as the THTR-technology. In the Federal Republic of Germany two reactor suppliers are competing in this area with the pebble bed design. In the United States a bloc fuel reactor of slightly larger size is under development. The lay-out difference concerns the integrated and non-integrated arrangement of the steam generator. These proposals are suitable for the supply of energy to industrial complexes and to small grids as well as for countries with less developed industrial structures.

The R+D-program for the conventional lay-out of the HTR-plants is retrogressive concerning the public funds. The planning of new HTR power stations has to be financed by the private organizations in the Federal Republic of Germany. However, the development work for the use of high temperature heat is a subsidized programme known under the name "Prototypenanlage Nukleare Prozeßwärme" (PNP-project) with the participation of several German organizations.

To meet the long-term objectives of the high-temperature process heat use the absolute prerequisite is the realization of an intermediate HTR for cogeneration of electricity and steam. The point has been reached by studies that the present designs are feasible, licensable and almost competitive. For further technical progress a follow project is needed to demonstrate serial production, convoy construction, and other ways to reduce costs by a simple, but fully safe plant lay-out.
5. PROCESSES UNDER DEVELOPMENT

The potential for energy systems with high temperatures is in several industrial large-scale processes, such as hard coal and lignite gasification, iron or hydrogen production.

5.1. Steam Gasification of Hard Coal

To take full advantage of the HTR heat a new gasification process had to be developed. For this purpose from 1976 to 1984 a semi-technical fluidized bed gas generator was in operation at the Bergbau-Forschung GmbH in Essen. The gas generator with a coal capacity of 0.2 t/h operated at 40 bar and 700 to 800 °C. The test runs had the objectives to demonstrate that the heat demand for hard coal gasification can be supplied from outside. Improvement of the conditions in the fluidized bed, the heat transfer by an immersion heater, the tests of refractory materials, dust separators, heat recovery systems, and high temperature fittings were the test targets. Within a total operation time of 26,000 h, all objectives were achieved. About 1,700 t of anthracite and semi-anthracite coal were gasified. Finally, it was possible to gasify also baking high volatile bituminous coal (700 t) after a jet injection system was installed.

After the test runs an engineering assessment will be available by the end of 1986. The evaluations indicate so far the technical feasibility of the steam gasification process for industrial application, whereas the competitiveness is not yet given. In the reviewed concept a helium cycle with a capacity of 340 MW(th) at 900 °C is adjoined to a gas generator. A horizontal type gas generator with the immersion heater concept would have a length of 33 m and a diameter of 7.2 m at a capacity of 28 t/h and a gasification rate of 93%. Boosting the gasification reactions by adding a catalyst could increase the capacity to 69 t/h.

Also a preliminary design of an industrial hydrogen production plant is performed. The concept includes three He-cycles and gas generators connected to a 1,000 MW(th) HTR. To increase hydrogen production the installation of a helium heated steamreformer is planned. A demonstration plant would have a coal throughput of 0.6 mill. t/a to produce 2.0 billion m³ hydrogen. There is further development potential if high temperature materials can be proven for catalytic gasification or new gasification concepts are considered.

5.2. Hydrogasification of Lignite

A hydrogasification process with a vertical fluidized bed reactor is under development for lignite. Hydrogasification is an exothermal reaction. However, HTR heat is necessary to reform the methane produced in the gasifier. The reformer is heated by helium gas of 950°C. The produced synthesis gas is processed to hydrogen which is fed back to the hydrogasifier.

The successful test runs of a semi-technical plant at Rheinische Braunkohlenwerke AG in Köln-Wesseling permitted the scale-up to a 10 t/h-pilot plant which is in operation since May 1983. The experimental test runs were finished at the end of September 1986. In that time the plant operated for 14,723 h including 7,777 h on coal. More than 38,100 t of lignite were
gasified to 19 million m$^3$ of methane. The methane concentration of the raw gas ranged between 22 and 36% at a gasifier efficiency of 50 to 60%. Until April 1987 a detailed documentation will be concluded.

The gasification process for lignite is due to the pilot plant operation far ahead of the gasification of hard coal. However, no actual plan to scale-up the process is presently made for economic reasons.

5.3. Materials Program

From the beginning the development of the materials and components was a crucial issue. New alloys suitable to process coal at high temperatures in an eroding and corroding atmosphere were necessary. Until today material scientists are still testing new resistant qualities especially for the catalytic process conditions. A new field is the heat exchanger technology at 800 to 1,000 °C. The program for high temperature materials and new components such as the He/He-heat exchanger, the high temperature helium loop, the reformer, the hot gas tubes, and the fittings requires up to 15 years.

Component modules have been tested in the high temperature helium loop facility at Interatom GmbH in Bensberg. Since August 1982 test conditions at temperatures up to 1,000°C and pressures up to 46 bar in steady or dynamic mode have been available. Until September 1986 the test facility operated for 11,000 h including 3,300 h at temperatures higher than 900°C. Test items are the hot-gas pipes and fittings and other crucial new designs. The efforts are concentrated also on carbon compounds and ceramic high temperature insulation materials.

Many important results have been achieved. The He/He-heat exchanger and steamreformer materials are qualified for 60,000 to 100,000 h. For high operation temperatures "Inconel 617" and for lower temperatures "Hastelloy X" meet the requirements, but for catalytic gasification conditions the material problems are not yet solved. The feasibility to produce largesized forgings has been demonstrated by fabrication of a 11 t hot collector of the He/He-heat exchanger. - For hot gas lines at temperatures higher than 600°C new carbon compound tubes with a diameter of 900 mm are available. - The future development activities are directed towards industrial application of the materials and components tested. It is expected that the material development period will last until the late 1980's or even longer if ceramic materials are also considered.

5.4. Iron Production

The HTR is intended to be used to supply heat and steam to gasify hard coal up to 50% to synthesis gas. The residual coke is fed back to an iron bath for reduction of ore. With the Klöckner steel-gas-process coke, oxygen, and iron ore are processed in an iron bath. The synthesis gas generated can be converted to methanol. The products of the integrated plant are electrical power, methanol, and pig iron. Until now there are no actual plans for an industrial plant. Before industrial use the technical feasibility of all production steps has to be demonstrated.
5.5. Hydrogen Production

Hydrogen production by using electrical power and heat is a further HTR-application, which could become very important for the future energy market. Conventional electrolysis processes utilize nearly 80% of the electric power. The efficiency of this process with light water reactors is approx. 26%, whereas the HTR may reach 31%. Efficiencies of more than 45%, however, can be expected by melt or high temperature electrolysis, because a part of the energy can be introduced as heat. Electrolytic-chemothermal hybrid processes with sulphuric acid separation at temperature levels of 400-500°C and 800-900°C are further possibilities.

6. MARKETS

6.1. Electricity/Steam Cogeneration

The primary use of the HTR is electricity production and steam supply with units up to 600 MW. The competitiveness in this market against light water reactors has to be proven, but is a realistic assumption. The electricity market will be open for the HTR when economic proposals are made by the reactor industry.

In areas with medium-sized grids small HTR's can be used for electric power generation, steam production, and community district heating. Also for those countries with no nuclear program yet the small HTR may become a very safe option. In countries without industrial infrastructure small reactors can also serve certain island regions or deliver high-pressure steam up to 530°C to the industry. Energy supply for tertiary oil extraction, processing of heavy oil and tar-sand or for desalination, and coal gasification are the fields for application.

Recently the IAEA identified in detail the potential markets for small nuclear power plants up to 300 MW. The advantages quoted are: Lower absolute capital cost causes a smaller financial burden and reduces the economic risk. The early introduction of small nuclear power plants reduces the environmental pollution. Finally small plants are suitable for low capacity grids and applications for selected industrial processes. The IAEA expects that some 25 additional countries have grids with a capacity of 2,000 - 6,000 MW in the period of 1990 to 2000 with a market ready for a number of small nuclear plants.

In countries with large interconnected grids, high technical infrastructure and energy consumption the market for the HTR comprises the whole electric power and heat supply. Nuclear power plants in those countries have sizes between 500-1,300 MW(e) for economic reasons. Smaller reactors would not be competitive against large nuclear plants. In those areas with a well developed technical infrastructure and high industrial production a wide field of new applications for nuclear steam is open. Low temperature heat is required in a number of industrial branches, such as the chemical, paper, cellulose, sugar, textile industry or for district heating. But in almost all cases the required amount of heat from a nuclear reactor is limited in each site. Steam supply therefore has to be coupled with electricity production or small HTR-plants of some 100 MW are to be built.
6.2. Situation in the Federal Republic of Germany

In a time of moderate growth rates of energy consumption new sales can only be made at the expense of competing energies. The prospects for the introduction of HTR in the Federal Republic of Germany can be regarded as an example of a typical industrial country.

In the Federal Republic of Germany nearly 95 GW power station capacity is installed, which produces 400 TWh. A forecast of 1984 predicted for the year 2000 nearly 110-125 GW capacity to produce 480-560 TWh. These values may be lower considering the power plant structure. But from 1995 several GW per year additional capacity is required. Some HTR with 500 MW could be accommodated from the market point of view.

Another example is the German market for synthesis gas. The chemical industry needs about 10 billion m$^3$/a. The conversion of 0.6 Mio tce (maf) hard coal delivers up to 2 billion m$^3$ of hydrogen and carbon monoxide using nuclear heat from the HTR. The total annual demand of synthesis gas can be produced from 3 Mio tce of hard coal with heat supplied by two large HTR's. This market could expand considerably in case the methanol sales raise for fuel supply. This market section will be determined by competitiveness with mineral oil products. Three percent methanol in the usual fuel means an additional sale of methanol from 0.7 Mio t/a. A complete substitution of the gasoline market by methanol would require the additional amount of methanol from 35 Mio t per year and is unrealistic.

The existing heat market in the Federal Republic of Germany is divided into the low temperature range and the process steam supply for industry and small consumers. Forecasts assume 180 Mio tce in the year 2000. Oil saving potentials and increasing demand represent approx. 45 Mio tce. Sufficient potential exists therefore for the introduction of nuclear heat.

Today the proportion of district heating is approx. 6 Mio tce rising to 12 Mio tce in 2000. A detailed study about the application of district heat concludes that in nearly 190 cities existing heat demand could be served by district heating. The main demand is in urban centers to be connected with power stations. Assuming a grid enlargement nuclear power stations could substitute approx. 6 Mio tce per year competitively if a base load of 7.300 MJ/s and 6.500 h of operation were presumed. At the same time the nuclear plants could produce between 5 an 10% of the additional electricity demand until the year 2000. The introduction of nuclear district heat depends on the availability of large grids. At the moment the introduction of nuclear power plants for this purpose is a matter of single cases. A maximum of four power plants for district heating can be considered until the year 2000.

The demand of industrial process steam energy will reach 65 Mio tce in 2000. This includes a modest growth of 0.4% per year. The main part of the process energy is distributed amongst the following industries with two crucial temperature peaks: Up to 400 °C: chemicals 25%, paper/cellulose 10%, textile 4% and sugar 2%; at a level between 900 and 1700 °C: steel industry 56%, cement/brick 29%, glass/ceramics 8%, chemicals 6% and aluminium 1%.
With its special advantages to produce electricity and process steam simultaneously the HTR can serve the chemical industry first. The energy intensive processes for producing chemical feed stocks are large-scale operations with a high and constant operation rate over the whole year. The industrial application of high temperature steam in the industry is also a matter of a few single cases. Nevertheless, because of the existing infrastructure this market segment is very important as a first application.

Although long-term development programmes are underway it is still too early for a prediction of the future market for HTR high temperature heat for coal gasification, steel production or water splitting.

7. COST AND SCHEDULE

Whereas the HTR use for electricity and steam production is a commercial matter, the introduction of technologies around the HTR is determined by the requirements of the future markets for gaseous and liquid products. Also the availability of special processes, and the political and industrial determination for new projects play a major role. Furthermore the introduction of advanced systems and products depends on their development costs and the prices for other energies.

As a first guideline for the economy of the new processes coal gasification costs can be used. Synthesis gas of modern autothermal coal-dust gasification plants costs around 0.22-0.24 DM/m³ with feed coal prices of about 200 DM/tce. Economically less favourable is the production of substitute natural gas estimated to cost 0.75 DM/m³, more than twice as much as imported natural gas.

For the nuclear coal gasification production cost estimates had been calculated in the early 1980's. At that time appr. 70% higher costs were assumed than the natural gas import price of 0.30 DM/m³. Recent calculations show that since then no significant improvements towards better economy have been achieved. Moreover, more development time is required than originally anticipated, which means, that the present low oil and gas prices might have increased again by the time the technology will become available.

The options for the advanced HTR-concepts can be maintained only if the 300 MW THTR Schmelhausen operates with full capacity for several years and if the follow-on projects HTR 500, HTR 100 (BBC), or the modular HTR (KWU) are successful. Many long-term experimental steps with public funding are necessary. The project schedule requires time until the next century, because the chances to connect a new application route to a HTR reactor have to be regarded as open or very long-term. Also for example for the hard coal gasification with nuclear process heat very high development costs will arise. First estimates for a nuclear test of the new gasification process arrive at approx. 500 Mio DM, although a successful programme of 1.4 billion DM has already been performed in the past years. Therefore for the future programme the PNP-project might encounter some delay.
8. SUMMARY

The HTR for electric power and steam (530 °C) production is feasible and almost competitive for plants up to 500 - 600 MW. Smaller reactors are under development. Concerning the advanced applications the transfer of heat from an HTR at a helium outlet temperature of 950 °C for processes such as coal gasification is feasible. The work on the nuclear island for future plants is underway. The new industrial concepts take into consideration the option for the future use of the high temperature range.

The tests of the heat transfer components and the selected materials for temperatures of 950 - 1,000 °C are successful, but continuation for some more years is required. Hydrogasification of lignite is more advanced when compared with the steam gasification process of hard coal. The future for both concepts is open. Follow-on plants are not yet decided. By the end of 1986 an engineering assessment will be completed providing the basis to discuss the next project stage. All other technologies are considerations for the far future.

The decisions ahead will be difficult to make. At a time of scarce funds for development, very low oil prices, and a general distrust in nuclear energy application the long-range goals of the HTR-application have to be reassessed. Since the present energy situation does not justify the necessary expenditure for private companies only further public support is able to maintain the incentives for further advancing the described projects.

A more general view on the world energy situation is advisable in this situation. It is assumed that increasing world population and demand in developing countries will lead to more energy consumption. Representatives of the oil industry regard the present low oil price level as temporary and assume that the world oil production and demand will be in balance again latest in the 1990's at a higher price level. Furthermore new political problems cannot be excluded. Another reason to develop new technologies is the ratio of fossil energy reserves and consumption. About 75 % of the fossil world energy reserves are coal and 25 % oil and natural gas. The consumption data are almost the reverse: oil and gas 60 %, coal 30 %. Therefore clean technologies for coal upgrading are needed and the use of nuclear heat could well be the solution.

The contribution of nuclear energy to the worldwide energy supply is 5 % only although 350 nuclear power stations are in operation. There is a certain hesitation to proceed with new plants after the accidents in the United States and in Russia and the cost increase for nuclear power in the last years. The HTR has safety advantages, application possibilities for new processes and stands in competition with the established light water reactors. In this situation a new general discussion how to proceed long-term may arise but requires time. In any case a next HTR-project is needed soon to avoid a scattering of the capable engineering and research teams built-up with billions. A planning order for the HTR 500 is the most advanced next step and also the studies for smaller HTR plants could contribute.
The chance to choose an alternative nuclear system is given, since the 300 MW THTR Schmehausen has reached full power. The use of the HTR besides power generation for raw materials production might determine its market introduction. For the frequently announced high application potential the HTR community needs an international cooperation of various branches of the energy industry. Coal and nuclear energy are the future resources, and their combined use is a common long-term task. The continuity of the development should not be subject to ever changing price cycles of the energy market, but rather of a consequent energy policy.

9. REFERENCES


(15) Arbeitsgemeinschaft Fernwärme e.V. (AGFW), Hauptbericht Fernwärme 1982.
FINANCING MODELS FOR HTR PLANTS
Co-financing, counter trade, joint ventures

J. BOGEN, D. STÖLZL
Brown, Boveri und Cie AG,
Mannheim, Federal Republic of Germany

Abstract

Structure and volume of investment cost for HTR nuclear power plants are different in comparison to other types of nuclear power plants. Even if the share of local participation is in comparable order of magnitude to other nuclear power plants, the required technical infrastructure for HTR plants is more suitable for existing and still practised technologies in countries which are in development processes.

These HTR specific features offer special possibilities in HTR project financing. Various models are discussed in respect of the special HTR situation. Even if it is not possible to point out in a general manner the best solution - due to national, local and time dependant situations - this paper discusses the HTR specific impacts to buyer's credit financing, supplier's credit financing, barter trades or joint ventures and combined financing.

1. GENERAL

In general financing of nuclear power plants is characterized by high capital requirements in form of investment cost and low cost for operation, maintenance and the fuel cycle. Nations which desire to use nuclear energy usually have to

- ratify some international agreements
- start energy demand analyses in accordance to population and economic growth rates
- start economic analyses on different energy generation alternatives (water, oil, gas, coal, nuclear)
- start technical know-how improvements
- start actions for improvement of the general infrastructure basis
- evaluate the environmental situation.
Only if such an (independent) assessment or feasibility study comes to the result that a nuclear power plant is

- viable
- desirable

for the foreign country, it makes sense to discuss financing models. - In this paper it is not the task to create new financing models for HTR-plants but to show that HTR-plants are more convenient for financing than nuclear power plants (NPP) in the range of 1300 MWe.

2. Financing nuclear power

Nuclear power projects are characterized by a 10 to 5 years' period of planning prior to the start of construction. The construction period of the NPP takes about 6 years. After handover the operation period of the plant is in the order of 30 years. - No one can say with the confidence required for the assessment of financing risks what will happen to the political stability or economic strength of a country (under development) over such a time period of 40 - 50 years.

Therefore, one of the first prerequisites for financing a nuclear power project is to shorten the construction time of the plant and to minimize the pay-back period for credits and financing cost by sales of energy to the energy consumers such as industry, trade, private consumers and eventually by export of electric energy.

This requires

- a proven technology of the NPP technology decided for
- an experienced supplier's management during construction
- a well trained operation personnel
- guaranteed earnings from sales of energy
- economic and financial conditions to convert earnings from energy sales from national into international currency.

The HTR-line with spherical fuel elements of graphite and the inert gas helium as primary coolant represents a proven technology with high safety standards and good-minded operation characteristics.

This had been demonstrated by the AVR-15 MWe experimental power plant, which is in operation in Jülich, (F.R. Germany) since nearly 20 years. An experienced supplier's management
could be created during construction and commissioning of THTR-300 MWe prototyp power plant in Hamm-Uentrop, (F.R. Germany). By implementation of this project also the required infrastructure for licensing and sub-supplier's supplies and services was built-up. The THTR-300 MWe plant is the first nuclear power plant in a technical scale worldwide on basis of the pebble-bed principle. On Sept. 23, 1986, the THTR-300 generated 100 % thermal capacity. That means that a prototype plant of a pebble-bed reactor on a technical scale is available as reference plant for this reactor line.

2.1 BUYERS CREDIT BASIS FOR EXPORTS FINANCING

In this case the buyer of the plant asks a bank in the exporter's country for credit to pay the plant supplier according to the supply contract and project progress on a cash basis (fig. 1).
For follow-on projects BBC/HRB concentrated its interest to plants of medium and small capacities, such as HTR-500, HTR-100 and HTR-2 x 100 as well as GHR-10 (th) [2]. Due to the fact that these plants are in the lower or medium capacity scale they meet the requirements of small and medium sized networks and the absolute amount of the investment cost (in Deutschmark (DM)) only is in the order of

\[
17 \times 10^6 \text{ [DM]} \text{ for GHR-10 (th)} \\
630 \times 10^6 \text{ [DM]} \text{ for HTR-100} \\
900 \times 10^6 \text{ [DM]} \text{ for HTR-2 x 100} \\
1690 \times 10^6 \text{ [DM]} \text{ for HTR-500}
\]

This is demonstrated in fig. 2. The figures are only for demonstration in the order of magnitude for investments of the mentioned HTR plants. In case of sales/contracts they have to be adopted to the actual date and scope of the plant supplier.

![Fig. 2: HTR-Power Plants Investment Cost](image-url)
HTR-500

The absolute amount necessary as credit for financing of an HTR-500 [3] is only about 50 % of the amount for a large 1300 MWe plant. This reduction of the absolute investments is not by the cost of the economy of the generated nuclear power. Due to specific characteristics of HTR plants such as high total efficiency and high burn-up of the nuclear fuel (110,000 MWD/t HM compared to 35,000 MWD/t HM for PWR's) the electricity generating cost of an HTR-500 are comparable to those of a large 1300 MWe plant.

This makes financing of an HTR-500 easier, assuming that normally

\[ 20\% \text{ (} = 340 \times 10^6 \text{ DM) } \]

are financed by local sources/credits and the remaining

\[ 80\% \text{ (} = 1350 \times 10^6 \text{ DM) } \]

can be granted by export-credit agencies to a value of 75 % (\( = 1268 \times 10^6 \text{ DM) }, \) according to present conditions of the OECD-Consensus as agreed upon in August 1984. The remaining 5 % (\( = 85 \times 10^6 \text{ DM) } \) then must be financed by commercial loans.

As any commercial bank is likely to limit its participation for one project to about \( 50 \times 10^6 \text{ DM or less, there are still two commercial banks necessary for financing the remaining share. In the case where also the local share of 20 \% (340 \times 10^6 \text{ DM) has to be financed about 10 commercial banks have to be linked together for such a project. This is only half the number of banks necessary for financing of a large 1300 MWe plant. This helps to reduce financing cost for HTR projects. The main difference between commercial loans and export credits is the grace period, that is the period before repayment of the debt starts. For commercial loans this grace period is in the order of 4 - 5 years whereas for export-credits this period is in the order of 7 - 10 years. This means that for commercially financed shares repayment of the debt starts before the plant delivers earnings by energy generation. The investment cost of HTR-plants for heat and power generation can be roughly characterized, as follows

- 40 % belong to the nuclear heat supply system (NHSS) and turbogenerator
and

- 60% belong to sub-suppliers for the rest of the plant.

This is demonstrated in fig. 3 for an HTR-500 plant.

![Diagram](image)

Fig. 3: HTR-500 Shares for Suppliers

This gives an opportunity to participate local industries during the construction of the plant. The share which local industry can take depends on the existing industrial infrastructure. The fact, that

- an HTR-500 has a reactor pressure vessel of pre-stressed concrete which has to be constructed, isolated and tested at the site

- the components such as steam generators, valves, piping (including supports/hangers) isolations are of smaller dimensions than for large 1300 MWe plants

- the turbogenerator of a conventional type as required for fossil-fired power plants with comparable capacity

indicates that the share for local industries is even for the first HTR-500 plant about 40% or more. This is a greater number than normally is achieved for large 1300 MWe plants.
For the HTR-100 and HTR-2 x 100 the advantage of low total investment cost (see fig. 2) leads to an increase of the specific investment cost (see fig. 4).

Fig. 4: Specific Investment Cost for HTR-Power Plants

This influences economic figures, so that the competitiveness for power and heat generated by an HTR-100 (HTR 2 x 100) only exists in comparison to fossil fired plants and no longer for large nuclear power units. But HTR-100 plants are of high interest in cases to avoid

- dependencies on imported fossil fuels (oil, gas) for power generation
- investments for an additional infrastructure, e. g. for transportation of coal
- investments for stack-gas cleaning systems (desulphurisation, denitrification).
The share which local industry can take during implementation of an HTR-100 plant is comparable to that one of the HTR-500 and depends strongly on the technical infrastructure of the country in which the HTR-100 plant will be erected (see fig. 5).

The HTR-100 normally has a pressure vessel of steel which is pre-manufactured, tested and pre-installed with the reactor internals at the workshops of the plant supplier. This helps to reduce the construction period to 4 years, commissioning included. This is a very important factor for the financing point of view and contributes to low financing cost. For an HTR-100 the 20% share of local credits – as assumed before – amounts to 126 x 10^6 DM. The 75% share financed by export-credit agencies amounts to 472 x 10^6 DM and the remaining 5% share which needs to be financed on a commercial credit basis amounts to 32 x 10^6 DM. Even if the local share needs financing by commercial loans, there are only three banks necessary to be linked together, each one with a participation of 50 x 10^6 DM.

**GHR-10 (th)**

As far as the gas cooled heating reactor (GHR-10 (th)) is concerned, the total investments are only in the order of 17 x 10^6 DM. The nuclear part of this plant is completely premanufactured by the supplier. This means that the local industry only can participate in the civil works and the installation of the district heating network which the GHR-10 (th) serves. The construction period for one plant is reduced to only two years in total, commissioning included.
Financing of about $17 \times 10^6$ DM of investment cost is not a general problem. If the local share is assumed to be 20% this corresponds to $3 \times 10^6$ DM. The 75% share financed by export-credit corresponds to about $13 \times 10^6$ DM. Even if the local share requires for credits on a commercial basis, this corresponds together with the remaining 5% to about $4 \times 10^6$ DM. Due to the short construction period these amounts can be easily financed by one single bank.

The only problem is that no licensed sites for such plants are available and often the infrastructure for a (expensive) district heating network has to be erected before.

Above we discussed the advantages of the various HTR plants on basis of a buyer's credit and cash money for the plant-supplier and sub-suppliers. Today this will be something like an ideal situation, more convenient to demonstrate the advantages of HTR-plants from the financing point of view than for realistic project financing.

It is not only the amount of money which hinders the financing of nuclear power plants but also the credit worthiness of countries as it is seen by international lender organisations. For the assessment of the credit worthiness

- the debt situation of the buyer's country
- the comparison of the level of debt and debt service against the cross national product
- the supply of foreign exchange through export of goods

are the main indicators. There are countries which have rescheduled the reimbursement of their debts and the interest due are generally paid step by step.

In this situation there is no buyer's credit granted for nuclear power plants. Then other financing models are possible which have their justification in the interest of the plant supplier and the economic policy of the plant supplier's government to promote the export of power plants [4].

2.2 **SUPPLIERS CREDIT FOR EXPORT FINANCING**

In this case the plant supplier asks for a credit on basis of the signed contract (see fig. 6).
For minimization of the credit amount necessary there is a certain interest to increase the participation of the buyer (high local participation) or to split financing between a large number of sub-suppliers. Each party then is responsible for financing of his scope (multi-national financing approach). In this case the foreign buyer remains responsible for the plant operation and pays back the credits to the supplier. The risk of conversion of earnings in local money to internationally convertable currency is in this case with the buyer.

2.3 BARTER BASIS FOR EXPORT FINANCING

An additional risk of the buyer's country is the conversion of raw-materials like crude oil or metallic ores on the world market into internationally convertable currency. So the buyer's country is interested to shift this
risk to the plant supplier. He can use the earnings of this barter deal for the reimbursement of the supplier's credit. The risk of unstable monetary values of the raw-materials according to variations in the world market trading prices then lies with the supplier.

The foreign buyer remains responsible for the plant operation after handover and the management of energy supply and earnings (fig. 7).

Fig. 7: Export Financing on Barter Basis

2.4 JOINT VENTURE BASIS FOR EXPORT FINANCING

One of the most severe problems in the success of a NPP project is the management of commercial operation of the plant. This requires a well-trained and experienced operation staff. For this risk the foreign
buyer is interested to use the experience of the supplier during the period of commercial operation. This can be arranged by foundation of a joint-venture utility, responsible for power plant operation and energy management. The risk of conversion of the earnings from local currency to an internationally convertible currency is then with this utility. From there the money flows back to the plant supplier for reimbursement of the supplier's credit (fig. 8). Also a mixture of joint venture and barter deal is possible (fig. 9).

---

Fig. 8: Export Financing on Joint Venture Basis
2.5 **MIXED SOURCES FOR EXPORT FINANCING**

In special situations a mixture of various financing sources can reduce the credit amount from the bank which has to be granted to the supplier (or buyer).

Such a situation occurs when money, for example, from development aid can be granted by the exporter's country with subject to the implementation of a special project such as a nuclear power station.
3. CONCLUSIONS

Financing of NPP projects becomes easier with HTR plants due to

- low total investment cost
- low construction periods
- good-minded operation characteristics
- high safety standards and inherent safety
- high local participation by proven conventional technologies and smaller dimensions of the components used.

It is not possible to discuss the best solution for financing in a general manner but there are a lot of suggestions indicating that export financing is possible even when the creditworthiness of the buyer or the buyer's country does not exist.

All these financing models show a tendency to shift financial risks to the supplier's side. This increases the complexity of the project during implementation. In case of barter financing risks of international trade additionally are shifted to the supplier's side.

In the case of joint venture's additionally the risk of operation of the plant is shifted partially to the supplier's side.

From the supplier's point of view these increases in financial risks must be compensated by guarantees on the political and financial level of the buyer's country and additionally by export credit insurances of the supplier's country for successful implementation of an HTR nuclear power project.

References:

1 | IAEA-TECDOC 378
Costs and Financing of Nuclear Power Programmes in developing countries
Proceedings of a Seminar 09-12 Sept. 1985

2 | Brandes, Kohl, Schmitt
Small Nuclear Power Plants: 10 MW GHR Gas-Cooled Heating Reactor and 100 MWe Industrial Nuclear Power Plant
List of Abbreviations

<table>
<thead>
<tr>
<th>Abbreviation</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>BBC</td>
<td>Brown, Boveri &amp; Cie AG Mannheim</td>
</tr>
<tr>
<td>HM</td>
<td>Heavy Metal</td>
</tr>
<tr>
<td>HRB</td>
<td>Hochtemperatur Reaktorbau GmbH Mannheim/Düsseldorf</td>
</tr>
<tr>
<td>HTR</td>
<td>High Temperature Reactor</td>
</tr>
<tr>
<td>PWR</td>
<td>Pressurised Water Reactor</td>
</tr>
<tr>
<td>DM</td>
<td>Deutschmark</td>
</tr>
<tr>
<td>MWd/t</td>
<td>Megawatt-Days per ton (of HM)</td>
</tr>
<tr>
<td>NHSS</td>
<td>Nuclear Heat Supply System</td>
</tr>
<tr>
<td>NPP</td>
<td>Nuclear Power Plant</td>
</tr>
</tbody>
</table>
PERSPECTIVES ON THE HTGR FROM UTILITIES IN THE USA

H.L. BREY
Public Service Company of Colorado, Denver, Colorado

L.D. MEARS
Gas-Cooled Reactors Associates, San Diego, California

United States of America

Abstract

The Gas Cooled Reactor Associates (GCRA) represents utility/user interests within the United States in establishing and achieving HTGR program objectives. GCRA conducts Demonstration Project development activities, plus provides overall program coordination/integration support services to the U. S. Department of Energy. This paper summarizes three of the activities currently being undertaken by GCRA. These include:

° The 1986 Summary of the United States Utility/User Questionnaires concerning new electrical generating capacity, the future of nuclear power, and the potential of the HTGR,

° The utility/user requirements for the modular HTGR plant, and

° An update on the modular HTGR Demonstration Project development activities.

Introduction

This paper is a review of three significant programs currently being undertaken within the U.S.A. by the Gas-Cooled Reactor Associates (GCRA). These programs include: a summary of the responses received from U.S. electric utility and process heat users to questionnaires concerning new generating capacity, the future of nuclear power, and the potential of the HTGR; a review of the utility/user requirements for the Modular High Temperature Gas-Cooled Reactor (MHTGR) and; an update on the Modular Gas Reactor (MGR) Demonstration Project.

SUMMARY REPORT ON THE UTILITY/USER QUESTIONNAIRE

The Gas Cooled Reactor Associates recently compiled the results of a questionnaire responded to by 80 utilities and 13 industrial companies as of January 31, 1986. The utility response represents approximately
59% of the installed generating capacity and over 72% of the installed nuclear capacity within the United States. The ranking on the associated graphs on a scale of from one to five corresponds to the following criteria:

<table>
<thead>
<tr>
<th>Value</th>
<th>Interest, Significance, or Importance</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>No</td>
</tr>
<tr>
<td>2</td>
<td>Low</td>
</tr>
<tr>
<td>3</td>
<td>Moderate</td>
</tr>
<tr>
<td>4</td>
<td>High</td>
</tr>
<tr>
<td>5</td>
<td>Extremely High</td>
</tr>
</tbody>
</table>

The significant highlights of the 1985 GCRA utility questionnaire are provided below (Ref. 1):

- Although new power plant orders are scarce in today's environment, the number of utilities planning one or more capacity additions in the late 1990s and beyond is significant.

<table>
<thead>
<tr>
<th>Planned Inservice Date</th>
<th>Number of Utilities Planning Additions</th>
</tr>
</thead>
<tbody>
<tr>
<td>Before 1995</td>
<td>9</td>
</tr>
<tr>
<td>1995 to 1999</td>
<td>38</td>
</tr>
<tr>
<td>2000 to 2004</td>
<td>37</td>
</tr>
<tr>
<td>2005 to 2010</td>
<td>29</td>
</tr>
</tbody>
</table>

- The preferred size of new capacity additions or replacements was divided between the 200 to 400 and the 400 to 700 MWe ranges (Figure 1). These results are consistent with the 1983 GCRA survey results on size preference, and supports both the reference plant size of 550 MWe (4 modules) and the 278 MWe (2 module) variant.

![Figure 1 Nuclear Plant Size Preference](image-url)
- 200 to 400 MWe favored by small and non-nuclear
- 400 to 700 MWe favored by larger nuclear utilities

° On the average, recommendations on allocating utility R&D resources were as follows:

- 40% for maintaining and improving existing plants and
- 36% for advanced generation alternatives, including
- 11% for advanced nuclear options, including
- 3% for the HTGR

° Obstacles rated highest for new orders for large LWRs (Figure 2):

- Financing difficulties
- Investment protection concerns, and
- PUC and NRC regulatory risks

![Figure 2: Q4-Obstacles to Large LWR Orders (Averaged)](image)

° The major obstacles identified that impede the commercialization of the HTGR were (Figure 3):

- Difficulties in establishing a combined utility/government/vendor development and demonstration program and
- Lack of a strong, established HTGR vendor/supplier infrastructure.

° Advanced generation options with the highest interest (Figure 4):

- Advanced coal technologies
- Conservation and load management

° Advanced nuclear generation alternatives received, on average, low to moderate interest with the improved LWR and the HTGR indicating the highest relative interest (Figure 5).
Except for load management and conservation, company interest in advanced generation alternatives ranked below the perceived industry interest.
The preferred institutional arrangement to ensure the success of an HTGR Demonstration Project favored the balanced utility, government and industry control and support slightly over a total DOE Project and a total utility-led Project, respectively. GCRA utilities showed a definite preference for a Project controlled by utilities with complementary support from the government and the vendor/supplier industry.

The demonstration project objectives receiving the highest ratings were demonstration and confirmation of:

- Safety characteristics to support design certification,
- Overall plant performance,
- System and component reliability,
- Investment protection characteristics, and
- Vendor/supplier capability

In selecting between coal, LWR, and MHTGR baseload options that were assumed to be available with equivalent commercial infrastructures, the following results were obtained:

- Utilities favored coal over nuclear even though the coal was presumed to have a higher busbar cost by 10-20%;
- Coal options, on average, received moderate to high interest and nuclear received low to no interest (Figure 6):
- Non-nuclear utilities clearly favored the 300 MWe coal plant and showed no interest in the 600 and 1200 MWe LWR, no to low interest in the 560 MWe HTGR, and low interest in the 280 MWe MHTGR plant; and
Nuclear utilities showed moderate to high interest in the coal options, and relatively low interest in the MHTGR and LWR options, although their interest in the nuclear options was much greater than for non-nuclear utilities. GCRA utilities clearly preferred HTGR options over any other nuclear alternative.

Utility preferences for the advanced reactor systems was highest for the improved LWR with the HTGR a close second. The HTGR was ranked higher than either the advanced LWRs (e.g. PIUS) or the LMRs from both the company and industry perspectives.

The significant highlight of the 1985 GCRA industrial questionnaire responses is provided below:

- Process heat users are looking to utility cogeneration and coal for future energy supply with the least interest in nuclear or externally generated electric energy sources (Figure 7).

**UTILITY/USER REQUIREMENTS FOR THE MODULAR HTGR PLANT**

A major role of GCRA is to establish utility/user requirements for the MHTGR design. A summary (Figure 8) of these requirements for a MHTGR electric generating plant with 4 reactors and 2 turbine generators with an electrical output of 550 MWe net includes:

**AVAILABILITY**

The equivalent availability average over the lifetime of the plant shall be at least 80% for electrical generation when modeled using the equipment mean time to failure, mean time to repair data for like type, or similar systems and/or components.
Figure 7  Process Heat User - Q1 - Advanced Energy Preference (Averaged)

<table>
<thead>
<tr>
<th>CRITERIA</th>
<th>UTILITY/USER REQUIREMENT</th>
</tr>
</thead>
<tbody>
<tr>
<td>EQUIVALENT ANNUAL AVAILABILITY</td>
<td></td>
</tr>
<tr>
<td>- TOTAL OUTAGE</td>
<td>20% MAXIMUM OVER LIFETIME</td>
</tr>
<tr>
<td>- SCHEDULED OUTAGE</td>
<td>10% MAXIMUM OVER LIFETIME</td>
</tr>
<tr>
<td>PLANT INVESTMENT PROTECTION</td>
<td></td>
</tr>
<tr>
<td>- UNSCHEDULED OUTAGE</td>
<td>10% MAXIMUM OVER LIFETIME</td>
</tr>
<tr>
<td>- OUTAGES &gt; 6 MONTHS</td>
<td>10% MAXIMUM OF UNSCHEDULED OUTAGES</td>
</tr>
<tr>
<td>- EXPECTED VALUE OF LOSS</td>
<td>&lt; ANNUAL INSURANCE PREMIUM</td>
</tr>
<tr>
<td>- PROBABILITY OF EXCEEDING</td>
<td>&lt; 10^-5/PLANT-YR</td>
</tr>
<tr>
<td>SAFETY RELATED DESIGN LIMIT</td>
<td></td>
</tr>
<tr>
<td>SAFETY AND LICENSING CRITERIA</td>
<td></td>
</tr>
<tr>
<td>- OVERALL CRITERIA</td>
<td>EXISTING NRC/EPA DOSE AND RISK CRITERIA</td>
</tr>
<tr>
<td>- EMERGENCY PLANNING CRITERIA</td>
<td>NO SHELTERING OR EVACUATION REQUIRED</td>
</tr>
<tr>
<td>SITING PARAMETERS</td>
<td></td>
</tr>
<tr>
<td>- EXCLUSION AREA BOUNDARY RADIUS</td>
<td>425 METERS</td>
</tr>
<tr>
<td>- SEISMIC (GROUND ACCELERATION)</td>
<td>.3 g SSE/.15 g OBE</td>
</tr>
<tr>
<td>FUEL CYCLE</td>
<td></td>
</tr>
<tr>
<td>- ENRICHMENT LEVEL</td>
<td>LOW, &lt; 20%</td>
</tr>
<tr>
<td>- SPENT FUEL MANAGEMENT</td>
<td>ONCE-THROUGH THROWAWAY</td>
</tr>
<tr>
<td>ECONOMIC GOALS</td>
<td></td>
</tr>
<tr>
<td>- BUSBAR POWER COST</td>
<td>AT LEAST 10% ADVANTAGE OVER COMPARABLY SIZED ADVANCED COAL PLANTS</td>
</tr>
<tr>
<td>- INSTALLED CAPITAL COST</td>
<td>&lt; 2000$/KW (1986 DOLLARS)</td>
</tr>
</tbody>
</table>

Figure 8  Summary of Utility/User Requirements for MHTGR
The equivalent unavailability averaged over the lifetime of the plant because of scheduled outages shall not exceed 10%. (This is an average of 876 hrs. per year).

PLANT INVESTMENT PROTECTION
It is a design goal that the unavailability averaged over the lifetime of the plant because of forced outages shall not exceed 10%.

Outages which last 6 months or more shall not contribute more than 10% of the total unavailability from forced outages.

The calculated financial annual value for risk to plant equipment or property when averaged over the plant lifetime is not to exceed the annual property damage insurance premium used in economic assessments. The current assumed portion of the annual premium for property damage is 4.5 million dollars U.S.

The likelihood of exceeding a design limit associated with safety-related design conditions for a reactor shall be less than \(10^{-5}\) per plant year.

SAFETY AND LICENSING
Design and licensing documentation will be developed with the goal of obtaining design certification of the standard Nuclear Island by the NRC. This includes meeting existing NRC/EPA dose and risk criteria.

This plant shall be designed to meet regulatory requirements without taking credit for the need of sheltering or evacuation of the public beyond the plant's exclusion area boundary.

SITING PARAMETERS
The radius of the exclusion area boundary shall be 425 meters.

The horizontal ground acceleration seismic criteria will be .3G for the safe shutdown earthquake and .15G for the operating basis earthquake.

FUEL CYCLE
This modular reactor plant will have a fuel cycle based upon the use of once-through low enriched uranium. However, other fuel cycles such as the use of high enriched fuel or recycled fuel which can be utilized without requiring additional modification to the basic core/fuel design, or to the physical plant, or to refueling components will be identified by the completion of this conceptual design effort.

ECONOMIC GOALS
A design goal of this plant is to have an economic advantage of at least 10% in the 30 year busbar cost of electricity relative to a comparable sized state-of-the-art coal plant. Based on current coal plant estimates, this results in a target of approximately 45 mils per KW total busbar cost and a total capital cost of approximately $2000 per installed KW electric including contingencies, owners' costs, and AFUDC (Ref. 2).

MODULAR HTGR DEMONSTRATION OBJECTIVES
As indicated previously, GCRA, under the direction of its members, has undertaken development activities for the eventual construction of
a modular HTGR which would have as its purpose to demonstrate the following objectives (Figure 9):

° Demonstrate licensing process and support design certification
  Through the Reference Plant development effort, a disciplined requirements-based approach to licensing is being developed that should fully capitalize on the MGR's unique safety characteristics. This process will be put to practice and demonstrated through such a Project.

° Demonstrate Plant Performance
  Performance of a full-scale reactor module and its interaction within the Nuclear Island and the overall plant will be demonstrated. In addition to the normal steady state power operation and anticipated transients demonstrations, design basis events may be demonstrated to verify inherent safety and investment protection features of the plant.

° Demonstrate Maintenance and Reliability
  The Project will provide for a demonstration of normal operation and maintenance (O&M) activities as well as identify and demonstrate O&M which might be required in the unlikely event of major equipment failures. In addition, a reliability improvement program will be established as an integral part of the Project.

° Establish basis for commercial plant cost and schedule
  The project will provide first-of-a-kind experience with adapting modular fabrication techniques and separation of the Nuclear Island

- DEMONSTRATE THE LICENSING PROCESS USING THE CRITERIA AND METHODOLOGY ESTABLISHED FOR THE MHTGR PROGRAM AND SUPPORT DESIGN CERTIFICATION EFFORT, AS REQUIRED, FOR THE STANDARD MHTGR

- DEMONSTRATE PLANT PERFORMANCE CHARACTERISTICS

- DEMONSTRATE CRITICAL MAINTENANCE ACTIVITIES AND SYSTEM/COMPONENT RELIABILITY

- ESTABLISH BASIS FOR COMMERCIAL PLANT COST AND SCHEDULE PLUS FOSTER THE DEVELOPMENT OF A VENDOR/SUPPLIER INFRASTRUCTURE

- ESTABLISH UTILITY/USER/INVESTOR CONFIDENCE TO BUY COMMERCIAL PLANTS

Figure 9  MHTGR Demonstration Project Objectives
and Turbine Island portions of the plant. The cost and schedule experience in the overall design, licensing, fabrication and construction effort will serve as invaluable demonstration experience for prospective vendors and customers.

- Develop Vendor/Supplier infrastructure
  The Project will foster the development of one or more vendor/supplier entities and related infrastructure for offering subsequent commercial plants.

- Establish Utility/Investor confidence
  Once the objectives above have been met, the utilities and the financial community would have sufficient confidence to proceed with commercial units (Reference 3).

As an initial requirement of this modular HTGR demonstration project, a definition study was undertaken beginning in August, 1985. The overall objective of this study was to provide an assessment for determining the feasibility of a M6R Demonstration Project. The tasks associated with this study include the development of the demonstration plant scope and layout, establishment of the project licensing approach, evaluation of utility siting options, development of a preliminary testing program, definition of the supporting technology requirements, an estimate of total program costs, and a defined program schedule (Figure 10). Figure 11 provides the estimated time frame for the phase-in of the modular demonstration plant.

- SCOPE & LAYOUT
- LICENSING APPROACH
- SITING OPTIONS
- TESTING PROGRAM
- TECHNOLOGY REQUIREMENTS
- PROGRAM COST
- PROGRAM SCHEDULE

Figure 10 Project Definition Study Objectives

Summary

Utilities representing approximately one-third of the installed generation capacity in the United States form a major part of the constituency of GCRA. Our utility membership has a strong interest in seeing the development of the HTGR as a future major electrical generation option in the United States. We are committed to the continuing development of nuclear power with specific emphasis on the HTGR. The significant attributes provided by this advanced concept will play an important role as an energy source of the future.
PHASE I - PROJECT DEFINITION (THRU 1988)

PHASE IIA - PRELIMINARY DESIGN AND LICENSING
(1989 - 1990)


PHASE IV - COMMERCIALIZATION

Figure 11 MGR Demonstration Project Phases

REFERENCES


MARKET PROSPECTS OF HTRs IN NEWLY INDUSTRIALIZED AND DEVELOPING COUNTRIES

S. GARRIBA
Politecnico di Milano,
Milan, Italy

C. VIVANTE
Commission of the European Communities,
Brussels

Abstract

Some unique characteristics seem to make HTR a nuclear system suitable to cope with future power and energy supply needs of developing world. Aspects which deserve consideration are economics, technological flexibility, safety and proliferation issues.

As compared with other nuclear reactor concepts, HTRs can be constructed into relatively small sizes according to a standardized approach. This characteristic renders the reactor suitable to be introduced in power grids of limited extension. The small size allows to increase the installed power generating capacity by following the variable trend of demand, all within a short construction period. The high degree of inherent safety makes the reactor system especially resistant to equipment failures and mismanagement. Safety and high-temperature features would allow the use of the reactor for supplying steam and process heat a newly industrializing country may need. In principle, non-proliferation records of HTRs depend upon the type of nuclear fuel and nuclear fuel cycle. If low-enriched or medium-enriched uranium is used, reprocessing would be not necessary. The exhausted fuel can be easily accommodated and stored before ultimate disposal. Finally, as for its relation with the local context, HTR entails flexible arrangements, which can be adapted to participation and seem suitable to technology transfer. Given these opportunities, it can be suggested that forms of co-operation with developing countries are looked for to monitor technology progress and prepare appropriate solutions.

1. PREMISE

The newly industrialized and developing countries represent a wide spectrum of states and their interests towards nuclear commitment vary substantially. Some nations are economically and industrially better placed. Few ones are exporters of energy, while most of them are in a shortage of indigenous energy resources and technologies. A widespread reaction to the basic issues of energy is that the international community has shown very little care or interest in understanding and helping resolve the acute energy problems of the Third World, although these problems are paralyzing national economies and threatening their survival. In the recent past, the rising costs of oil and the growing uncertainty in its supply have undermined
growth perspectives and created balance-of-payment difficulties. Although the price of oil has now fallen in real terms, countries in the Third World remain concerned about their energy supplies and consider nuclear power to be an economic and viable alternative for meeting their power needs.

Instead of relying on an early advent of soft or exotic technologies for quickly rescuing from energy shortages, quite a few developing countries continue to believe that serious attention should be paid toward making nuclear power economically and technically more accessible for extended use, under appropriate safety and safeguards regimes.

Two main general difficulties are limiting an effective development of the nuclear option, however. First, are the delays accumulated by nuclear programs in the major industrial countries. Second, are the serious problems encountered in nuclear technology transfer and adaptation.

Needless to say that nuclear programs of the major industrial countries are at a standstill. This is partly because of the slowing down of national energy demand growth projections, and partly because of strong opposition and pressures from certain groups that express fears about the environmental impact of nuclear power. Therefore some observers pretend that if the technologically more advanced nations are curtailing their nuclear power programs, there is perhaps even less justification for developing countries to turn to nuclear energy. This argument does not allow for the fact that the sustained increase in power supply is a prerequisite for economic and industrial development on one side. On the other side, use of nuclear energy would ease the problem of proper management of the natural resource base. Such a policy goal is especially important in developing countries, because these countries can least afford the opportunity costs of irreversible loss of their renewable natural resources, or efforts to remedy environmental damage.

The worry that the spread of nuclear technology may accelerate the proliferation of nuclear weapons adds further concerns. It has been sometimes argued that newly industrialized countries have sought nuclear reactors, not out of any serious interest in commercial power generation, but rather to enhance national prestige, to gain access to nuclear technology and training, to get on the technological trajectory of nuclear learning. Some countries indeed seem to have based their decisions to start up their nuclear programs on such considerations. In this process, companies from industrialized economies were sometimes led to exaggerate the promise of nuclear power and to understate its risk. Developing countries were advised, for example, to prepare themselves by installing large power reactors and to take advantage from scale factors. As a result, if nuclear reactors do prove to be a major technical or financial burden (and this likelihood cannot be ruled out), a frustrated developing country has no means of justifying the substantial investment in civilian reactors and human resources.

As for this special difficulty related with lack of an adequate industrial and technological base in developing countries, there is a case for designing and constructing standardized, small or medium-sized power reactors that could be operated in needy regions, and technically serviced through
international arrangements to ensure proper and efficient operation [1,2]. A new path should be found towards the development of uses of nuclear energy in the Third World. HTRs basing on small sizes and highly standardized units, like the ones proposed by some major nuclear vendors, seem to offer a viable option [3-5]. In this frame, four main problem areas seem worthy of evaluation and will be considered in the following, economics, role of local participation, safety, and proliferation characteristics.

2. THE ECONOMICS OF HTRs

The external cost of constructing and operating nuclear power plants and their facilities in developing countries is a key and still unresolved problem. This fact is not surprising, since few developing countries possess the infrastructure which is required for extensive participation in the construction of nuclear power plants ordered from foreign vendors. Usually, developing economies have negotiated turn-key packages that include arrangements for the financing of the nuclear projects by foreign vendors. While as far as plant operation is concerned, almost all the commercial nuclear power stations in the developing countries have come on-line recently, and there is insufficient experience in the running of the installations. Thus real costs over time cannot be assessed with any degree of accuracy.

The question of the costs of nuclear power is, of course, not peculiar to developing countries. Industrialized economies confront themselves with the same issue, although with a different set of constraints and a different framework of comparative choices. What is special to developing countries is their need to consider additional decision issues that, by definition, appear to be settled for advanced countries. Among these special issues are:

(i) The structure of the national grid, its size and the portion that a nuclear power plant would contribute to it;

(ii) The search for and the access to financial means from indigenous and external sources to support the project;

(iii) The degree of industrial and technological maturity of the country.

Let us examine these three issues one by one. Regarding unit size, it is apparent that present-generation commercial reactors, like LWRs, only operate economically above a certain minimum threshold of installed capacity. Such a break-even is in practice dictated in advance by the fact that, with present economies of scale, it is only those reactors which are working in large numbers already which can compete with other sources of power. These reactors are currently located in the industrialized countries and their size exercises a determining influence on the reactor market through the world.

Conversely, in almost any developing country, even a single reactor of the minimum unit size now readily available in international trade would represent a large proportion of the total electricity supply system, with obvious implications for the vulnerability of the same system to the withdrawal of a single generating unit from service. Technical considerations dictate that only a limited percentage of the total electricity
output of a country should be generated by one individual power station (regardless of how the unit is powered). Would this limit be exceeded, the danger exists that any unplanned outage in the station could create problems throughout the system, and maybe even cause it to break down totally. This also applies where there is no national transmission system, and where a single power station is responsible for supplying a particular economic or demographic area: in such a case the risk exists of a black out.

In principle, HTRs can be designed and constructed to overcome these problems. Adoption of small sizes would allow to follow an approach of standardized design and construction, and to avoid overcosts connected with the diminishing scale of the plant. A considerable part of the plant could be shop-constructed, rather than erected in the field.

Concerning the issue of financial resources, two key limiting factors are obvious: first, the level of real interest rates; and second, the availability of external capital. There are other important factors too, chiefly the allocation policy of scarce national investments. It can be remarked indeed that the real cost of nuclear electricity, of creating and supporting a nuclear generating capacity, with the necessary industrial and regulatory infrastructure, may preempt a larger share of the capital available for investment in energy supply systems than does a generating systems designed for fossil fuels. Again small standardized HTRs would ease these constraints. By allowing a gradual approach to nuclear power development it would become possible to plan a balanced distribution of funds which may become accessible. Moreover, HTRs could be adopted to produce electricity, or process heat, or both. This means that the same reactor can cope with the variety of electrical and other energy supply requirements in the country, so that possible uses of nuclear technology appear extended [6].

3. ENERGY SECURITY AND INDUSTRIAL DEVELOPMENT

The issue of the degree of industrial and technological maturity which is needed to embark on a nuclear power program, deserves careful examination. In a developing or newly industrialized country, the introduction of nuclear power may help to diversify supplies of energy in general and electricity in particular. The dependence on any one source of supply would be diminished and the dependence on imported energy sources would be reduced correspondingly. However, nuclear electricity demands a rather elaborate apparatus, in the shape of a reactor and its safety features, and a considerable series of industrial processes to transform uranium ore into the fuel for such a reactor. The most difficult questions about energy security in connection with nuclear power arise, not from comparing dependence on uranium to that on fossil fuels, but from considering what other forms of external supply a developing country must rely upon in launching a nuclear power program.

A common dilemma has been that technology imported from industrialized countries is often difficult to adapt to local socio-economic conditions. Such technology transfer can sometimes even hamper scientific innovations and entrepreneurial initiatives for indigenous development, leading to repeated
technology imports. Yet many developing countries do not have a cadre of scientific and technical personnel, who are capable of advising their national authorities on a strategy for the introduction of a sophisticated technology, such as nuclear energy, which would be suitable to local conditions. Most countries do not have adequate resources for establishing their own R&D efforts. The few developing countries that have the financial resources are, by and large, short of manpower and without adequate educational infrastructure. Another constraint which all developing countries have to face is the absence of a diversified industrial base.

Much the same is true of dependence on external nuclear fuel-cycle services. As in the case of reactor manufacture, for energy security, there is an obvious reason for seeking to reduce national dependence on foreign fuel-cycle services. A country choosing HTRs for its nuclear program needs access, not only to natural uranium, but also to all the services and processes involved in enriching uranium and fabricating suitable fuel elements. There is a question of economical, as well as technological, prudence in relation to indigenous fuel cycle development, however. Especially, careful judgment is required with regard to the establishment of enrichment or reprocessing capacity in a single country.

These limitations would not prevent that benefits of a well-planned and successfully executed nuclear program are industrial and technical, as well as energy-related, because the program can result in the progressive accumulation of knowledge and formation of construction and maintenance aptitudes. It is admitted indeed that HTRs entail a flexible technology, which can be adapted to a variety of local situations and prerequisites. Due to the design characteristics of the reactor system, opportunities for local participation in building HTRs could and should be exploited in ways that promote a generally higher level of technical and industrial expertise, embodied in a skilled work force and deployed in the form of manufacturing capability. The consequences would be a lasting benefit to the economy concerned.

These prospects must, therefore, be in the planners' minds when they assess the impact of embarking on the nuclear program. On their part, nuclear vendors ought to consider the problem of technology transfer as a strategic goal in the development of the HTR concept. Access to advanced technology and industrial skills needed in a nuclear program may be seen, in a wider context, as a way of raising the level of scientific and technical development, just as electrification relying on nuclear power may be seen as an optimal path to economic development based on industrialization. Clearly, for technology transfer can succeed, further investments would be required to create industrial capacity, only a fraction of which might be anything like fully employed once an initial reactor program has been completed.

4. SAFETY AND ENVIRONMENTAL ISSUES

Basing on both experimental and theoretical evidence, small and medium-size HTRs exhibit favourable characteristics relative to safety and environmental concerns. Inherent and
passive safety features make these reactors a low risk to the public and to the operating personnel.

A difficult matter for the planner to consider is that of the public's acceptance of nuclear power. It may seem strange to quote this an issue for developing countries, because so far it is believed to be in advanced industrial countries alone that serious problems of public opposition to nuclear power have arisen. But aftermaths of Chernobyl disaster have made apparent once more, that an accident which happen in whatsoever nuclear plant has a negative feedback on the overall nuclear market. In this view, the high degree of inherent safety of HTRs makes these reactors particularly resistant to equipment failures and mismanagement.

Inherent safety features of most HTRs relate primarily to fundamental laws of nature. These features are the first lines of defence in maintaining the integrity of the barriers against the release of radioactivity. An additional characteristic of critical importance is that the consequences of accidents develop slowly, allowing time for the operators to take deliberate and planned actions [7].

All aspects of inherent safety features, engineered safety equipment, barriers to fission products have been discussed at length by other authors and there is no need to add further remarks. These aspects seem also more valuable for countries where the lack of adequate skills and resources might lead to operate nuclear reactors along sub-optimal conditions and procedures.

Particularly, well known passive safety characteristics of the HTR are:

(i) A negative temperature coefficient of the core;
(ii) Coated-particle ceramic fuel that releases fission products slowly, even under extreme temperatures;
(iii) A graphite moderator/reflecter that responds smoothly to transient temperature, while maintaining structural integrity at very high temperatures;
(iv) Helium coolant which remains a single-phase gas under all conditions, and is neutronically transparent and chemically inert.

Should accidents be considered, engineered safety equipments that contribute to their low probability and low consequences, include:

(i) The triply redundant core auxiliary cooling system (CACS), which is used when main loop cooling is unavailable;
(ii) The reserve shut-down system, which can bring the reactor to cool shut-down in the absence of control rod insertion;
(iii) The plant protection system, which initiates various safety actions, such as reactor trip, steam generator isolation and dump, and CACS start-up.

The usefulness of probabilistic risk assessment has been demonstrated especially with LWRs. It would be appropriate to extend this type of analysis to HTRs, so as to embody external events and operating conditions which could be encountered in developing nations.

Lastly, as for the environmental concerns, it can be recalled the high efficiency of HTRs in the conversion of thermal energy into electrical energy even with dry cooling. This fact means less stringent site constraints because of the reduced low-quality thermal waste which must be discharged.
Another possibility which has its own merits and does not reserve further comments is cogeneration to cope with industrial uses of energy.

5. THE SAFEGUARDS PROBLEM

Developing and newly industrialized countries are conscious of the fact that prevailing fears about proliferation of nuclear weapons are an obstacle to their acquiring nuclear technology for peaceful purposes. Nuclear proliferation is basically a political problem. Hans Blix emphasizes that "the first and most important obstacle to the proliferation of nuclear weapons is a matter of political judgment and determination, as it emerges from assessments of political and security conditions, of benefits possibly accruing from NRT affiliation, and of drawbacks possibly connected with retention of the nuclear-weapons options" [8].

The reasons for the proliferation of sensitive nuclear technology seem indeed linked with perceptions of security needs and national interests. However, should a developing country decide to acquire sensitive technology while following a program of nuclear power development, it would inevitably put an almost unbearable economic burden and cause serious social and political problems within its own borders. Any unbiased analysis would indicate that this would be too high a price to pay for some kind of nuclear military capability of doubtful effectiveness.

In any case, it may be meaningful to analyze the important problem of non-proliferation in an objective manner with reference to HTR. Proliferation concerns vary widely with the type of fresh fuel utilized. All proposed HTR versions and fuel cycle options rely upon different grades of enriched uranium. For the low-enriched uranium and medium-enriched uranium fuel cycles the uranium in the fresh fuel itself would not be weapons usable without further enrichment. It should be also noted that once the fuel elements are fabricated, the fuel is present in a highly diluted form, due to the coatings and the associated graphite moderator material. Once exhausted, fuel can be easily accommodated and stored before ultimate disposal. And if reprocessing should be maintained as a possibility for the future, technology would still need further study and experimentation.

The different HTRs require the implementation of safeguards procedures according to their special design features, mainly off-load refuelling for prismatic elements and on-load refuelling for the pebble-bed version. The accessibility of an individual fuel element is quite different, whether the elements are in fixed positions, or move randomly through the core. The safeguards systems take these differences into consideration. Despite the recent high visibility of terrorism, an HTR would not greatly augment terrorist's ability to threaten damage, even with the help of a co-operative plant insider intent on causing core damage or some other radiological incident. Compared with other reactor systems, too much emphasis has not to be placed on physical deterrent measures which might not be as effective against insider threats from one side, and which might impair emergency operability from the other.
6. CONCLUSIVE REMARKS

Most industrialized countries believe that nuclear energy offers a viable solution to their electric power problems. The construction and operation of nuclear reactors, with the concomitant savings in raw materials and fossil fuels along with the protection of the environment, is essential to continuing economical growth and industrial productivity. There are strong incentives for developing and newly industrialized countries to turn to a technology that is already proven and available, and that can be put to use for the generation of electricity and possibly heat production at competitive costs. This is the motivation for integrating HTRs in Third World economies, as a flexible solution to yield industrial, technological and energy-related benefits.

While expanding their capability for using nuclear energy, Third World countries cannot go it alone, however. They need meaningful co-operation from the industrialized countries and an understanding of the problems. They require sharing and transfer of technology and dependable supplies of equipment, materials, and services. They ask for help in acquiring appropriate management skills, know-how, and financing.

Our assumption is that small-size standardized HTRs have several characteristics that make this reactor concept suitable to cope with the special requirements set forth by developing countries. In the recognition of this evidence, it seems appropriate that potential nuclear suppliers and possible host countries take a number of positive measures aimed at building a market for HTRs, owing to the fact that this reactor concept and system are close to commercial deployment in the more industrialized economies. Forms of inter-country co-operation can be perhaps devised to collect, evaluate and disseminate a wide range of information about HTRs and their applications, including such questions as nuclear power (and heat) economics, safety and environmental considerations, plant performance and operation, all with particular application to the Third World. Formal associations or interest groups might also be created, where potential nuclear suppliers and recipient countries work together to finalize HTR design, construction and operation. Several actions can be proposed regarding adaptation of HTRs to special environments, support to licensing procedures and safety analysis, collection of data concerning the operation of existing HTRs, organization of local industry and operator training, establishment of joint ventures. Finally, the scope of the work could be broadened to look at nuclear power option in the context of total energy needs and resources.

Every available opportunity should be used for conducting a serious dialog between the Third World and the more advanced nuclear countries to comprehend each other's perceptions and to narrow down existing differences.

REFERENCES


CHAIRMEN AND SECRETARIAT

GENERAL CHAIRMAN, VICE-CHAIRMAN AND CHAIRMEN OF SESSIONS

General Chairman:  Mr. R. Schulten  Federal Republic of Germany
Vice-Chairman:  Mr. E. Baithesen  Federal Republic of Germany

Chairmen of Sessions

Session A  Mr. E. Baithesen  Federal Republic of Germany
Session B  Mr. A. Pexton  United Kingdom of Great Britain and Northern Ireland
Session C  Mr. A. Millunzi  United States of America
Session D  Mr. C. Vivante  Commission of the European Communities (CEC)
Session E  Mr. V.N. Grebennik  Union of Soviet Socialist Republics
Session F  Mr. T. Hayashi  Japan
Session G  Mr. R. Schulten  Federal Republic of Germany

SECRETARIAT

Scientific Secretary:  J. Kupitz  Division of Nuclear Power, IAEA
LIST OF PARTICIPANTS

BELGIUM

Mr. J. PLANQUART

Director - Nuclear Program
C.E.N./S.C.K. Mol
Boeretang 200
B-2400 Mol
Tel. 0032/14/316871/311801
Telex 31922 ATOMOL B

CHINA

Mr. Gong YUNFENG

Deputy Chief Engineer of Beijing Institute of Nuclear Engineering
P.O. Box 840
Beijing

FRANCE

Mr. Y.M.F. BERTHION

CEA/CEN de Saclay
DEDR/DEMT
B.P. 2
F-91191 Gif-sur-Yvette
Tel. 908.24.04
Telex 690 641F ENERGAT-SACLAY

GERMAN DEMOCRATIC REPUBLIC

Mr. H.F. BRINCKMANN

Zentralinstitut für Kernforschung
Rossendorf, PSF 19
DDR-8051 Dresden
Telex zfk rossendorf 02 167

FEDERAL REPUBLIC OF GERMANY

Mr. M. ANDLER

Interatom GmbH.
Dep. PF113
Friedrich-Ebert-Straße
D-5060 Bergisch Gladbach 1
Tel. 02204/84-0
Telex 8878 457 iagl (d)
<table>
<thead>
<tr>
<th>Name</th>
<th>Address</th>
</tr>
</thead>
<tbody>
<tr>
<td>Mr. G. ALTES</td>
<td>Kernforschungsanlage Jülich GmbH. IRE</td>
</tr>
<tr>
<td></td>
<td>Postfach 1913</td>
</tr>
<tr>
<td></td>
<td>D-5170 Jülich 1</td>
</tr>
<tr>
<td></td>
<td>Tel. 06-02461.61-0</td>
</tr>
<tr>
<td></td>
<td>Telex 833556 KFA D</td>
</tr>
<tr>
<td>Mr. E. ARNDT</td>
<td>Hochtemperatur-Reaktorbau GmbH. Gottlieb-Daimler-Straße 8</td>
</tr>
<tr>
<td></td>
<td>D-6800 Mannheim 1</td>
</tr>
<tr>
<td></td>
<td>Tel. 0621/4511</td>
</tr>
<tr>
<td></td>
<td>Telex 462041; Telefax 0621451599</td>
</tr>
<tr>
<td>Mr. L.T. AUMLLELLER</td>
<td>NUKEM</td>
</tr>
<tr>
<td></td>
<td>Rodenbacher Chaussee 6</td>
</tr>
<tr>
<td></td>
<td>Postfach 110080</td>
</tr>
<tr>
<td></td>
<td>D-6450 Hanau 11</td>
</tr>
<tr>
<td></td>
<td>Telex 4184 780 nuk d</td>
</tr>
<tr>
<td>Mr. K.P. BACHUS</td>
<td>Bundesminister für Umwelt, Naturschutz und Reaktorsicherheit</td>
</tr>
<tr>
<td></td>
<td>Husarenstraße 30</td>
</tr>
<tr>
<td></td>
<td>D-5300 Bonn</td>
</tr>
<tr>
<td>Mr. G. BALLENSEVEN</td>
<td>Kernforschungsanlage Jülich GmbH. IRE</td>
</tr>
<tr>
<td></td>
<td>Postfach 1913</td>
</tr>
<tr>
<td></td>
<td>D-5170 Jülich 1</td>
</tr>
<tr>
<td></td>
<td>Tel. 06-02461.61-0</td>
</tr>
<tr>
<td></td>
<td>Telex 833556 KFA D</td>
</tr>
<tr>
<td>Mr. E. BALTHESEN</td>
<td>Kernforschungsanlage Jülich GmbH. Projektträger HTR</td>
</tr>
<tr>
<td></td>
<td>Postfach 1913</td>
</tr>
<tr>
<td></td>
<td>D-5170 Jülich 1</td>
</tr>
<tr>
<td></td>
<td>Tel. 06-02461.61-554</td>
</tr>
<tr>
<td></td>
<td>Telex 833556 KFA D</td>
</tr>
<tr>
<td>Mr. H. BARNERT</td>
<td>Kernforschungsanlage Jülich GmbH. IRE</td>
</tr>
<tr>
<td></td>
<td>Postfach 1913</td>
</tr>
<tr>
<td></td>
<td>D-5170 Jülich</td>
</tr>
<tr>
<td></td>
<td>Tel. 06-02461.61-0</td>
</tr>
<tr>
<td></td>
<td>Telex 833556 KFA D</td>
</tr>
<tr>
<td>Mr. E. BAUST</td>
<td>Hochtemperatur-Reaktorbau GmbH. Gottlieb-Daimler-Straße 8</td>
</tr>
<tr>
<td></td>
<td>D-6800 Mannheim 1</td>
</tr>
<tr>
<td></td>
<td>Tel. 0621/4511</td>
</tr>
<tr>
<td></td>
<td>Telex 462041; Telefax 0621451599</td>
</tr>
</tbody>
</table>
Mr. B. BEINE                   Siempelkamp Gießerei
                                 GmbH + Co.
                                 Abteilung Reaktortechnik
                                 Siempelkampstraße 45
                                 Postfach 2570
                                 D-4150 Krefeld-1
                                 Telex 2151338

Mr. R. BERNARD                  Badenwerk AG
                                 Badenwerkstraße 2
                                 Postfach 1680
                                 D-7500 Karlsruhe 1
                                 Telex 7825749

Mr. J. BOGEN                    Brown, Boveri & Cie AG
                                 Nuclear Energy Dept.
                                 Kallstadter Straße 1
                                 Postfach 351
                                 D-6800 Mannheim 31
                                 Telex 462 411 107

Mr. J. BOOSTEN                  Kraftwerk Union AG
                                 Department V 11/RK 1
                                 Hammerbacherstraße 14
                                 D-8520 Erlangen
                                 Tel. 09131-18-4141
                                 Telex 62929-0

Mr. T. BORNSCHEIN              Kernforschungsanlage Jülich GmbH.
                                 HBK
                                 Postfach 1913
                                 D-5170 Jülich 1
                                 Tel. 06-02461.61-0
                                 Telex 833556 KFA D

Mr. S. BRANDES                  Hochtemperatur-Reaktorbau GmbH.
                                 Gottlieb-Daimler-Straße 8
                                 D-6800 Mannheim 1
                                 Tel. 0621/4511
                                 Telex 462041; Telefax 0621451599

Mr. D. BRINKMANN               Kraftwerk Union AG
                                 Dept. V 49
                                 Hammerbachstraße 12 + 14
                                 D-8520 Erlangen
                                 Tel. 09131-18-0
                                 Telex 62929-0

Mr. H. BRUNNER                  NUKEM GmbH.
                                 Rodenbacher Chaussee 6
                                 Postfach 110080
                                 D-6450 Hanau 11
                                 Telex 4184 780 nuk d
Mr. R. CANDELI
Interatom GmbH.
Dep. PF113
Friedrich-Ebert-Straße
D-5060 Bergisch Gladbach 1
Tel. 02204/84-0
Telex 8878 457 iagl (d)

Mr. H.-P. CLASSEN
Interatom GmbH.
Dep. PF114
Friedrich-Ebert-Straße
D-5060 Bergisch Gladbach 1
Tel. 02204/84-0
Telex 8878 457 iagl (d)

Mr. Cl.-B. von der DECKEN
Kernforschungsanlage Jülich GmbH.
Institut für Reaktorbauelemente
Postfach 1913
D-5170 Jülich 1
Tel. 06-02461.61-0
Telex 833556 KFA D

Mr. K. DUMM
Interatom GmbH.
Dep. PF22
Friedrich-Ebert-Straße
D-5060 Bergisch Gladbach 1
Tel. 02204/84-0
Telex 8878 457 iagl (d)

Mr. C. ELTER
Hochtemperatur-Reaktorbau GmbH.
Gottlieb-Daimler-Straße 8
D-6800 Mannheim 1
Tel. 0621/4511
Telex 462041; Telefax 0621451599

Mr. Ch. EPPING
University of Duisburg
Energy Technology
Lotharstraße
Duisburg

Ms. R. EXNER
Balcke-Dürr AG
Homburger Straße 2
D-4030 Ratingen
Telex 8585113

Mr. J.A. FAßBENDER
Kernforschungsanlage Jülich GmbH.
PTH
P.O. Box 1913
D-5170 Jülich
Tel. 06-02461.61-0
Telex 462041
<table>
<thead>
<tr>
<th>Name</th>
<th>Address</th>
</tr>
</thead>
<tbody>
<tr>
<td>Mr. H. HUEBEL</td>
<td>Interatom GmbH. Dep. T2</td>
</tr>
<tr>
<td></td>
<td>Friedrich-Ebert-Straße D-5060 Bergisch Gladbach 1</td>
</tr>
<tr>
<td></td>
<td>Tel. 02204/84-0 Telex 8878 457 iagl (d)</td>
</tr>
<tr>
<td>Mr. X. HUANG</td>
<td>at present: Kernforschungsanlage Jülich GmbH.</td>
</tr>
<tr>
<td></td>
<td>Postfach 1913 D-5170 Jülich 1 Tel. 06-02461.61-0</td>
</tr>
<tr>
<td></td>
<td>Telex 833556 KFA D</td>
</tr>
<tr>
<td>Mr. G.P. IVENS</td>
<td>Arbeitsgemeinschaft Versuchsreaktor AVR GmbH.</td>
</tr>
<tr>
<td></td>
<td>Postfach 1411 D-4000 Düsseldorf 1 Telex 211 4285</td>
</tr>
<tr>
<td>Mr. W. JAHNS</td>
<td>Interatom GmbH. Dep. PF12</td>
</tr>
<tr>
<td></td>
<td>Friedrich-Ebert-Straße D-5060 Bergisch Gladbach 1</td>
</tr>
<tr>
<td></td>
<td>Tel. 02204/84-0 Telex 8878 457 iagl (d)</td>
</tr>
<tr>
<td>Mr. W. JANSING</td>
<td>Interatom GmbH. Dep. T31</td>
</tr>
<tr>
<td></td>
<td>Friedrich-Ebert-Straße D-5060 Bergisch Gladbach 1</td>
</tr>
<tr>
<td></td>
<td>Tel. 02204/84-0 Telex 8878 457 iagl (d)</td>
</tr>
<tr>
<td>Mr. T. JOBSKY</td>
<td>INNOTEC KG Huyssenallee 70-72</td>
</tr>
<tr>
<td></td>
<td>D-4300 Essen 1 Facsimile 201472 INNOTEC</td>
</tr>
<tr>
<td>Mr. G.G. KAISER</td>
<td>Kernforschungsanlage Jülich GmbH. HBK-Projekt</td>
</tr>
<tr>
<td></td>
<td>Postfach 1913 D-5170 Jülich 1 Tel. 06-02461.61-0</td>
</tr>
<tr>
<td></td>
<td>Telex 833556 KFA D</td>
</tr>
<tr>
<td>Mr. A. KASPER</td>
<td>Hochtemperatur-Reaktorbau GmbH. Gottlieb-Daimler-Straße 8</td>
</tr>
<tr>
<td></td>
<td>D-6800 Mannheim 1 Tel. 0621/4511 Telex 462041; Telefax 0621451599</td>
</tr>
</tbody>
</table>
Mr. W. KROEGER
Kernforschungsanlage Jülich GmbH.
ISF, P.O. Box 1913
D-5170 Jülich
Tel. 06-02461.61-0
Telex 462041

Mr. K. KRUEGER
Arbeitsgemeinschaft Versuchsreaktor AVR GmbH.
Hambacher Forst
D-5170 Jülich
Telex 833598

Mr. K. KUGELER
University of Duisburg
Energy Technology
Lotharstraße
Duisburg

Mr. W. KUEHNLEIN
Kernforschungsanlage Jülich GmbH.
KFA-HZ
Postfach 1913
D-5170 Jülich
Tel. 06-02461.61-0
Telex 833556 KFA D

Mr. W.F. von LENSA
Kernforschungsanlage Jülich GmbH.
Projektträger HTR
Postfach 1913
D-5170 Jülich 1
Tel. 06-02461.61-0
Telex 833556 KFA D

Mr. G. LANGE
Hochtemperatur-Reaktorbau GmbH.
Gottlieb-Daimler-Straße 8
D-6800 Mannheim 1
Tel. 0621/45111
Telex 462041; Telefax 0621451599

Mr. G. LOHNERT
INTERATOM GMBH.
Dep. PF211
Friedrich-Ebert-Straße
D-5060 Bergisch Gladbach 1
Tel. 02204/84-0
Telex 8878457 IAGL (D)

Mr. V. Maly
Kernforschungsanlage Jülich GmbH.
HBK
Postfach 1913
D-5170 Jülich 1
Tel. 06-02461.61-0
Telex 833556 KFA D
Mr. R. MAUERSBERGER
Hochtemperatur-Reaktorbau GmbH.
Gottlieb-Daimler-Straße 8
D-6800 Mannheim 1
Tel. 0621/4511
Telex 462041; Telefax 0621451599

Mr. Kh. MARCH
Kraftwerk Union AG
Dept. V 49
Hammerbacherstr. 14
D-8520 Erlangen
Tel. 09131-18-3811
Telex 62929-0

Mr. M. MARK-MARKOWITSCH
Kernforschungsanlage Jülich GmbH.
P.O. Box 1913
D-5170 Jülich
Tel. 06-02461.61-0
Telex 833556 KFA D

Mr. C. MARNERT
Member of the Board
Arbeitsgemeinschaft Versuchsreaktor
Luisenstraße 105
D-4000 Düsseldorf

Mr. A.W. MEHNER
NUKEM GmbH.
Postfach 110 080
D-6450 Hanau 11

Mr. G. MEISTER
Kernforschungsanlage Jülich GmbH.
ISF
Postfach 1913
D-5170 Jülich 1
Tel. 06-02461.61-0
Telex 833556 KFA D

Mr. T. MONSAU
Ministerium für Wirtschaft, Mittelstand und Technologie des Landes Nordrhein-Westfalen
Haroldstraße 4
D-4000 Düsseldorf

Mr. R. MOORMANN
Kernforschungsanlage Jülich GmbH.
ISF
Postfach 1913
D-5170 Jülich 1
Tel. 06-02461.61-0
Telex 833556 KFA D

Mr. W.D. MUELLER
Lurgi GmbH.
Gervinusstr. 17
D-6 Frankfurt/Main 11
Telex 41236-0 g d
Mr. H. NABIELEK
Kernforschungsanlage Jülich GmbH.
HBK
Postfach 1913
D-5170 Jülich 1
Tel. 06-02461.61-0
Telex 833556 KFA D

Mr. A. NAOMIDIS
Kernforschungsanlage Jülich GmbH.
P.O. Box 1913
D-5170 Jülich
Tel. 06-02461.61-0
Telex 833556 KFA D

Mr. H. NICKEL
Kernforschungsanlage Jülich GmbH.
Institute for Reactor Materials
P.O. Box 1913
D-5170 Jülich
Tel. 06-02461.61-0
Telex 833 556 KFA D

Mr. W.K. PANKNIN
L. & C. Steinmueller GmbH.
Fabrikstraße 1
D-5270 Gummersbach
Telex 884551-0 SG D

Mr. P. PHLIPPEM
University of Duisburg
Energy Technology
Lotharstraße
Duisburg

Mr. M. PODHORSKY
Balcke-Dürr AG
Homberger Straße 2
D-4030 Ratingen
Telex 8585113

Mr. H. POHL
Bundesministerium für Forschung und Technologie
Ref. 312,
Postfach 1913
D-5300 Bonn 1

Mr. H. PUETZ
Interatom GmbH.
Dep. PF211
Friedrich-Ebert-Straße
D-5060 Bergisch Gladbach 1
Tel. 02204/84-0
Telex 8878 457 iagl (d)

Mr. U. QUAST
Gesellschaft für Reaktorsicherheit mbH.
Schwertnergasse 1
D-5000 Köln 1
Mr. J. RAUTENBERG
Hochtemperatur-Reaktorbau GmbH.
Gottlieb-Daimler-Straße 8
D-6800 Mannheim 1
Tel. 0621/4511
Telex 462041; Telefax 0621451599

Mr. W. REHM
Kernforschungsanlage Jülich GmbH.
ISF
Postfach 1913
D-5170 Jülich 1
Tel. 06-02461.61-0
Telex 833556 KFA D

Mr. R. REIMERT
Lurgi GmbH.
Gervinusstr. 17
D-6 Frankfurt/Main 11
Telex 41236-0 lg d

Mr. H. REUTLER
Interatom GmbH.
Dep. PF21
Friedrich-Ebert-Straße
D-5060 Bergisch Gladbach 1
Tel. 02204/84-0
Telex 8878 457 iagl (d)

Mr. E. ROEHLER
Hochtemperatur-Reaktorbau GmbH.
Gottlieb-Daimler-Straße 8
D-6800 Mannheim 1
Tel. 0621/4511
Telex 462041; Telefax 0621451599

Mr. M. ROETH-KAMAT
Kernforschungsanlage Jülich GmbH.
IRE
Postfach 1913
D-5170 Jülich 1
Tel. 06-02461.61-0
Telex 833556 KFA D

Mr. R. ROTTERDAM
Hochtemperatur-Reaktorbau GmbH.
Gottlieb-Daimler-Straße 8
D-6800 Mannheim 1
Tel. 0621/4511
Telex 462041; Telefax 0621451599

Mr. H.J. RUETTEN
Kernforschungsanlage Jülich GmbH.
IRE
Postfach 1913
D-5170 Jülich 1
Tel. 06-02461.61-0
Telex 833556 KFA D
Mr. M.K. SCHAD
Lurgi GmbH.
Gerwinusstraße 17
D-6 Frankfurt/Main 11
Telex 41236-0 lg d

Mr. U. SCHARFER
INNOTEC KG
Huyssenallee 70-72
D-4300 Essen 1
Facsimile 201472 INNOTEC

Mr. W. SCHENK
Kernforschungsanlage Jülich GmbH.
HZ/IRW
Postfach 1913
D-5170 Jülich
Tel. 06-02461.61-0
Telex 833556 KFA D

Mr. E. SCHERER
Kernforschungsanlage Jülich GmbH.
IRE
Postfach 1913
D-5170 Jülich
Tel. 06-02461.61-0
Telex 833556 KFA D

Mr. K. SCHIMMELPFENNIG
Zerna, Schnellenbach und Partner
Gemeinschaft Beratender Ingenieure GmbH.
Viktoriastraße 47
D-4630 Bochum 1
Tel. (0234) 6890712
Telex 825424

Mr. H.-J. SCHLESINGER
Interatom GmbH.
Dep. T233
Friedrich-Ebert-Straße
D-5060 Bergisch Gladbach 1
Tel. 02204/84-0
Telex 8878 457 iagl (d)

Mr. P. SCHMIDTLEIN
University of Duisburg
Energy Technology
Duisburg
Lotharstraße

Mr. F. SCHMIEDEL
Hochtemperatur-Reaktorbau GmbH.
Gottlieb-Daimler-Straße 8
D-6800 Mannheim 1
Tel. 0621/4511
Telex 462041; Telefax 0621451599
Mr. R. SCHMIDT
Fichtner Beratende Ingenieure
Sarweystraße 3
D-7000 Stuttgart

Mr. H. SCHMITT
Hochtemperatur-Reaktorbau GmbH.
Gottlieb-Daimler-Straße 8
D-6800 Mannheim
Tel. 0621/4511
Telex 462041; Telefax 0621451599

Mr. G. SCHNELLENBACH
Zerna, Schnellenbach und Partner
Gemeinschaft Beratender Ingenieure GmbH.
Viktorikastraße 47
D-4630 Bochum
Tel. (0234) 1 50 22
Telex 825 424

Mr. J. SCHOENING
Hochtemperatur-Reaktorbau GmbH.
Gottlieb-Daimler-Straße 8
D-6800 Mannheim
Tel. 0621/4511
Telex 462041; Telefax 0621451599

Mr. B. SCHROEDER
Kernforschungsanlage Jülich GmbH.
IRE
Postfach 1913
D-5170 Jülich
Tel. 06-02461.61-0
Telex 833556 KFA D

Mr. G. SCHROEDER
Steag AG
Bismarckstr. 54
D-43 Essen

Mr. H.J. SCHROETER
Bergbau-Forschung GmbH
Abt. Kohlechemie Grundprozesse
Franz-Fischer-Weg 61
Postfach 13 01 40
D-4300 Essen 13
Telex 857 830 BERG D

Mr. F. SCHUBERT
Kernforschungsanlage Jülich GmbH.
Institute for Reactor Materials
P.O. Box 1913
D-5170 Jülich
Tel. 06-02461.61-0
Telex 833556 KFA D
Ms. M.L. SCHUELLER  
Kernforschungsanlage Jülich GmbH.  
Projektträger HTR  
Postfach 1913  
D-5170 Jülich 1  
Tel. 06-02461.61-554  
Telex 833556 KFA D

Mr. H. SCHUG  
Bundesministerium für Forschung und Technologie  
Postfach 200706  
D-5300 Bonn 2

Mr. R. SCHULTEN  
Director IRE  
Kernforschungsanlage Jülich GmbH.  
Postfach 1913  
D-5170 Jülich 1  
Tel. 06-02461.61-554  
Telex 833556 KFA D

Mr. H. SCHUSTER  
Kernforschungsanlage Jülich GmbH.  
Institute for Reactor Materials  
P.O. Box 1913  
D-5170 Jülich  
Tel. 06-02461.61-554  
Telex 833556 KFA D

Mr. D. SCHWARZ  
Vereinigte Elektrizitätswerke Westfalen AG  
Theinlanddamm 24  
D-4600 Dortmund

Mr. W. SEELE  
Brown, Boveri & Cie AG  
Dept. Switchgear  
Postfach 351  
Kallstadter Straße 1  
D-6800 Mannheim  
Telex 462 411 202

Mr. J. SINGH  
Kernforschungsanlage Jülich GmbH. IRE  
Postfach 1913  
D-5170 Jülich 1  
Tel. 06-02461.61-0  
Telex 833556 KFA D

Mr. B. STAUCH  
Kernforschungsanlage Jülich GmbH. IRE  
Postfach 1913  
D-5170 Jülich 1  
Tel.: 06-02461.61-0  
Telex 833556 KFA D
Mr. H. STEHLE
Interatom GmbH
Dep. T311
Friedrich-Ebert-Straße
D-5060 Bergisch Gladbach 1
Tel. 02204/84-0
Telex 8878 457 iagl (d)

Mr. W. STEINWARZ
Interatom GmbH
Dep. PF113
Friedrich-Ebert-Straße
D-5060 Bergisch Gladbach 1
Tel. 02204/84-0
Telex 8878 457 iagl (d)

Mr. M. STELZER
Ruhrkohle Oel und Gas GmbH.
Gleiwitzer Platz 3
D-4250 Bottrop

Mr. D. STOELZL
Brown, Boveri & Cie AG
Mannheim, Germany
Nuclear Energy Dept.
Postfach 351
Kallstadter Straße 1
D-6800 Mannheim 31
Telex 462 411 107

Mr. H.W. STRAUß
Rheinische Braunkohlenwerke AG
Stüttgenweg 2
D-5000 Köln 41
Telex 8883011 a rbw d

Mr. H. TEUBNER
Interatom GmbH.
Dep. T311
Friedrich-Ebert-Straße
D-5060 Bergisch Gladbach 1
Tel. 02204/84-0
Telex 8878 457 iagl (d)

Mr. E. TEUCHERT
Kernforschungsanlage Jülich GmbH.
IRE
Postfach 1913
D-5170 Jülich 1
Tel. 06-02461.61-0
Telex 833556 KFA D

Mr. W. THEYMANN
Hochtemperatur-Reaktorbau GmbH.
Gottlieb-Daimler-Straße 8
D-6800 Mannheim 1
Tel. 0621/4511
Telex 462041; Telefax 0621451599
Mr. M. VALSAN  
Kernforschungsanlage Jülich GmbH.  
IRW  
Postfach 1913  
D-5170 Jülich  
Tel. 06-02461.61-0  
Telex 833556 KFA D

Mr. K. VERFONDERN  
Kernforschungsanlage Jülich GmbH.  
P.O. Box 1913  
D-5170 Jülich  
Tel. 06-02461.61-0  
Telex 833556 KFA D

Mr. W. WACHHOLZ  
Hochtemperatur-Reaktorbau GmbH.  
Gottlieb-Daimler-Straße 8  
D-6800 Mannheim 1  
Tel. 0621/4511  
Telex 462041; Telefax 0621451599

Mr. J. WALDMANN  
Kraftwerk Union AG  
Department R 74  
Hammerbacherstraße 14  
D-8520 Erlangen  
Tel. 09131-18-0  
Telex 62929-0

Mr. I. WEISBRODT  
Interatom GmbH.  
Dept. PF1  
Friedrich-Ebert-Straße  
D-5060 Bergisch Gladbach 1  
Tel. 02204/84-0  
Telex 8878 457 iagl (d)

Mr. M. WILL  
Interatom GmbH.  
Dep. PF114  
Friedrich-Ebert-Straße  
D-5060 Bergisch Gladbach 1  
Tel. 02204/84-0  
Telex 8878 457 iagl (d)

Mr. L. WOLF  
Kernforschungsanlage Jülich GmbH.  
IRE  
P.O. Box 1913  
D-5170 Jülich  
Tel. 06-02461.61-0  
Telex 833556 KFA D

Mr. J. WOHLER  
Hochtemperatur-Kernkraftwerk GmbH.  
Siebenbeckstr. 10  
D-4701 Hamm-Uentrop  

539
Mr. E. ZIERMANN
Arbeitsgemeinschaft Versuchsreaktor AVR GmbH.
Hambacher Forst
D-5170 Jülich
Telex 833598

Mr. I.L. ZIERMANN
AVR - GmbH
c/o Kernforschungsanlage Jülich
Postfach 1913
D-51700 Jülich
Telex 833556 KFA D

INDIA
Ms. K. BALAKRISHNAN
Bhabha Atomic Research Centre
Reactor Engineering Division
Trombay
Bombay-400 085
Tel. 523321
Telex 2817 Ab Code Barc

INDONESIA
Mr. M. DJOKOLELONO
National Atomic Energy Agency
Jl. K.H. Abdul Rochim
Kuningan Barat
Mampang Prapatan
Jakarta Selatan
Telex 46354 BATAN JKT

Mr. I.R. SUBKI
National Atomic Energy Agency
Jl. K.H. Abdul Rochim
Kuningan Barat
Mampang Prapatan
Jakarta Selatan
Telex 46354 BATAN JKT

IRAQ
Mr. D.K. AL-DABAGH
Atomic Energy Commission
Tuwaitha-Baghdad
P.O. Box 765
Baghdad, Iraq

ISRAEL
Mr. J. ALTER
Israel Atomic Energy Commission
P.O. Box 7061
Tel-Aviv 61070
Telex 33450 ST PF IL
Mr. W.S. DUKHAN  
NRCN-IAEC  
P.O. Box 9001  
Beer-Sheva 84190  
Telex 5363 RTKMG IL

Mr. E. GREENSPAN  
Israel Atomic Energy Commission  
P.O. Box 7061  
Tel-Aviv 61070  
Telex 33450 ST PF IL

Mr. A. SEROUSSI  
Israel Atomic Energy Commission  
P.O. Box 7061  
Tel Aviv 61070  
Telex 33450 ST PF IL

ITALY

Mr. S. GARRIBA  
Politechnical University  
Milano

Mr. O. MODONESI  
ENEA, CRE  
Cassaccia, Postal Box 2400  
I-00100 Rome  
Tel. 040 6 6948  
Telex 613296

JAPAN

Mr. Ryuichi ARAKI  
Chiyoda Chemical Engineering & Construction Co., Ltd.  
1-12 Tsurumichuo 2-chome  
Tsurumi-ku  
Yokohama 230  
Telex 47726 chiyo J

Mr. Osamu BABA  
Japan Atomic Energy Research Institute (JAERI)  
Tokai Research Establishment  
Dept. of Power Reactor Projects  
Tokai-mura, Naka-gun  
Ibaraki-ken, 319-11  
Tel. 02928-2-5764  
Telex 3632339 JAERI F J

Mr. Toshikazu HAYASHI  
Japan Atomic Energy Research Institute (JAERI)  
Tokai Research Establishment  
Dept. of Power Reactor Projects  
Tokai-mura, Naka-gun  
Ibaraki-ken, 319-11  
Tel. 02928-2-5764  
Telex 3632339 JAERI F J
MEXICO

Mr. J.B. MORALES
Institut de Investigaciones Eléctricas
Apartado Postal 475
Centro-Cuernavaca, Mor.
62000 Mexico, D.F.
Tel. 514-8004
514-8259
Telex 17776352 IIEEMME

THE NETHERLANDS

Mr. J.M. van den BRINK
NUCON Engineering & Contracting B.V.
P.O. Box 4026
1009 AA Amsterdam
Telex 11996

Mr. J.B.M. de HAAS
Netherlands Energy Research Foundation
Westerduinweg 3,
P.O. Box 1
1755 ZG Petten
Telex 57211 REACP NL

Mr. G.C. van UITERT
Ministry of Economic Affairs
P.O. Box 20101
2500 EC The Hague
Telex 31099 B ECZA NL

POLAND

Mr. Edward OBRYK
Institute of Nuclear Physics
Ul. Radzikowskiego 152
31-342 Krakow
Tel. 3370222 ext. 280
Telex 0322461 ifj PL

PORTUGAL

Ms. I.M. CANHAO RORIZ
Gabinete de Protecção e Segurança Nuclear
Ministerio da Industria
Av. Republica 45 – 6º
1000 Lisbon
Telex 14344 Nucseg P

Mr. F. CARDEIRA
Laboratorio de Physica en Engenharia
Estrada Nationale 10
Sacavam
Telex 12727 NUCLAB P
SPAIN

Ms. Consuelo PERES DEL MORAL
Representante Técnico
c/o Consejo de Seguridad Nuclear (CSN)
International Affairs
Sor Angela de la Cruz, 3
28020 Madrid, Spain
Tel. 4561812
Telex 45869 CSNM E

Mr. C. FERNANDEZ PALOMERO
Central Nuclear de Vandellos
Director de Vandellos I
Hospital del Infante
Tarragona
Tel. 823050
Telex 56430

ROMANIA

Mr. I. NEAMU
Institute of Physics and Nuclear Engineering
B.P. 35
Bucarest
Tel. 00400-807040
Telex 11397 CSEN R

SWEDEN

Mr. C.U. RUNFORS
Studsvik Energiteknik AB
S-611 82 Nyköping
Tel. 0155/210 00
Telex 64013 STUDS S
Telefax 0155-63044

SWITZERLAND

Mr. P.O. BURGSMUELLER
Gebrüder Sulzer AG
Abt. 0350
CH-8401 Winterthur
Telex 896 060 30

Mr. H. HOLM
General Atomic Europe
Dorfstraße 4
CH-8126 Zumikon-Zurich
Tel. 01-9180505
Telex 53499
Mr. G. SARLOS
Schweizerische Interessen-
gemeinschaft zur Wahrnehmung
gemeinsamer Interessen an der
Entwicklung nuklearer
Technologien (IGNT)
CH-5303 Würenlingen
Telex 53714 EIR CH
Tel. 056/992742

USSR

Mr. V.N. GREBENNIK
I.V. Kurchatov Institute
of Atomic Energy
42 Ulitsa Kurchatova
P.O. Box 3402
123182 Moscow
Tel. (095) 1965158
(State Committee on Utilization
of Atomic Energy
Tel. 2331718, Telex 411888 MEZON SU)

Mr. I.V. CHROMOV
USSR State Atomic Energy Committee
Staromonetny, 26
Moscow
109180 USSR

Mr. I.M. IONOV
State Committee on the
Utilization of Atomic Energy
Staromonetny peureulok, 26
Moscow Zh-180
Tel. 2331718
Telex 411888

Mr. L.N. PERMERKOV
State Committee on the
Utilization of Atomic Energy
Staromonetny peureulok, 26
Moscow Zh-180
Tel. 2331718
Telex 411888

UNITED KINGDOM

Mr. J.R. ASKEW
UK Atomic Energy Agency
E511, Risley
Warrington, Cheshire WA3 6AT
Tel. 092531244
Telex 629301 atomry g

Mr. B. KEEN
National Nuclear Corporation Limited
Booths Hall, Chelford Road
Kuntsford, Cheshire WA16 8QZ
Telex 666000
Mr. A. PEXTON
SSEB
Cathcart House
Spean Street
Glasgow, G44 4BE
Tel. 041-637-7177
Telex 777703

Mr. P.T. SAWBRIDGE
Central Electricity Generating Board
Berkeley Nuclear Laboratories
Berkeley, Gloucestershire, GL13 9PB
Telex 43227

UNITED STATES OF AMERICA

Mr. L. BREY
Manager of Nuclear Engineering
Public Service of Colorado
2420 West 26th Avenue
Denver, CO 80211

Mr. J.L. HELM
President
Proto-Power Cooperation
Groton, Connecticut 06340
Telefax 203/4468292

Mr. J. JONES
Oak Ridge National Laboratory
P.O. Box X
Oak Ridge, Tennessee 37831
Tel. (615) 574-0377, Telefax 624-0377

Mr. A. MILLUNZI
Division of HTGR
Office of Nuclear Energy
DOE Washington
Mail Stop-B-107/GTN
Washington, D.C. 20545
Tel. 900-1-301353-2947
Telex 710-828 0475
Telefax 9001-301-353-3870/3888

Mr. A.J. NEYLAN
GA-Technologies Inc.
P.O. Box 85608
San Diego, CA 92138
Tel. (619) 455
Telex 695065

YUGOSLAVIA

Mr. Z. VARAZDINEC
Institut za Elektroprivredu
Proleterskih brigada 37
41000 Zagreb
Tel. 041/530-604 or 041/513-822/338
Telex 22154 YU INSTEP
INTERNATIONAL ORGANIZATIONS

Commission of the European Communities (CEC)

Mr. C. ViVANTE

200, rue de la Loi
B-1049 Bruxelles
Tel. 235 11 11
Telex COMEU B 21877

Organization for Economic Co-operation and Development (OECD)

Mr. S. HORIUCHI

Deputy Director for Nuclear Science and Development
OECD Nuclear Energy Agency
38, boulevard Suchet
F-75016 Paris
Tel. 524 96 50
Telex 630 668 AEN/NEA

International Atomic Energy Agency (IAEA)

Mr. J. KUPITZ

Division of Nuclear Power
Wagramerstraße 5
P.O. Box 100
A-1400 Vienna
Tel. 2360-ext.2814 through-dialing
Telex 1-12645
Telefax 222-230184

INTERNATIONAL PROGRAMME COMMITTEE

AUSTRIA

Mr. G.F. REITSAMER
OFZS – Seibersdorf

FEDERAL REPUBLIC OF GERMANY

Mr. E. BALTHESEN
PTH – Jülich

Mr. H. POHL
BMFT – Bonn

Mr. R. SCHULTEN
KFA – Jülich
UNITED STATES OF AMERICA

Mr. A. MILLUNZI DOE - Washington

COMMISSION OF THE EUROPEAN COMMUNITIES

Mr. C. VIVANTE

INTERNATIONAL ATOMIC ENERGY AGENCY

Mr. J. KUPITZ