FOREWORD

The Fifth Meeting of the IAEA International Working Group on Gas-Cooled Reactors was held in Heysham, UK, 5–8 June 1984.

The Summary Report (Part I) contains the Minutes of the Meeting.

The Summary Report (Part II) contains the national programmes in the field of Gas-Cooled Reactors and other presentations at the Meeting.
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HTR RESEARCH AND DEVELOPMENT IN AUSTRIA
1982–1984 PROGRESS REPORT

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1. General Status

In Austria there was never a definitive plan to construct or build a gas-cooled reactor. Furthermore, since 1979 it is prohibited by law to operate nuclear reactors for the purpose of electricity production, and there is no indication that this unsatisfactory situation will change in the near future. Therefore, there has never been and there is still no national programme in the field of GCRs.

Nevertheless, traceable back to Austria’s participation into the late OECD Dragon project, a considerable amount of R & D work has been performed and is still going on in several selected areas related to high-temperature gas cooled reactors (HTRs). These areas are:

- High temperature reactor fuels,
- High temperature structural materials,
- High temperature helium test rig/prestressed concrete pressure vessel.

During the period covered by this report the following institutions have been active in this field:

- Österreichisches Forschungszentrum Seibersdorf GmbH (ÖFZS)
- Vereinigte Edelstahlwerke AG (VEW)
- Reaktorbau, Forschungs- und Baugesellschaft mbH (RFB) (to a limited extent only since the beginning of 1983).

The total extent of the R & D work performed showed a slight downward trend as compared to the period covered by the previous status report (1).

2. High Temperature Reactor Fuels

R & D in the field of HTR fuels is performed exclusively at ÖFZS. This work is devoted to some selected areas where a high degree of qualification and of international competitiveness has been achieved. Contract work for customers outside Austria is continuing at a constant rate; additional self-supported R & D work is proceeding as well in order to maintain and to improve further the level of qualification.

Development of particle production methods.
No activities are to be reported except the publication of some previous work (2).

Equipment for physical quality control
A new particle size analyser of compact size was constructed and is now under test. That redesigned equipment half in size compared with the old one provides diameter measurements as well as shape factor measurements (3).

Comparative measurements of PyC and SiC layers with the X-ray technique against the standard methods and studies about types of pyrocarbon density profiles were made.

Post-irradiation examination
Work under contract to customers in the FRG continued during the period reported. It included the desintegration of various fuel
bodies, examinations and measurements related to the coated par-
ticle, graphite and electrolyte samples obtained (e.g. by gamma
and mass spectrometry, optical inspection, and other methods), as
well as hot chlorine gas leaching of total fuel elements, post-
irradiation annealing, and other methods. A survey about the
methods available at Seibersdorf is given in (4).

On the basis of a re-evaluation of all the CO measurements perfor-
med previously, a novel correlation was postulated between CO
production in UO$_2$ kerneled coated particles, and irradiation para-

Progress has been made also in further improving the methods
applied, especially in mass-spectrometric isotope-dilution tech-
niques. Special devices have been constructed and built for remote
preselection of large numbers of coated particles in a hot cell,
either via a visual inspection (microscope-coupled TV-camera) or
via gamma spectrometry (4). A remotely controlled thermogravi-
metric apparatus has been developed and built. It will be made
available commercially in the near future.

Budget
Amounts of AS 8,5 mio each have been spent at ÖFZS in the above
mentioned activities during 1982 and 1983.

3. High-Temperature Structural Materials
The investigation on the testing and development of high tempera-
ture metal alloys was continued. Special emphasis in these acti-
vities were given to the long term behaviour of austenitic steels
and Nickel-based alloys, i.e. Hastelloy X, Inconel 617 and
Nimonic 86.

Tasks of the projects in this field are

- Creep rupture tests
  In five test rigs at ÖFZS a large number (up to 240) of
  specimens are continuously tested under condition related to
  solid gasification, i.e. high temperatures (750 to 950°C) and a
  special atmosphere containing H$_2$, CH$_4$, CO$_2$, CO and H$_2$O vapour.
  The maximum predicted life time in these tests is 30,000 h. At
  the beginning of 1984 the maximum time, which a specimen was ex-
  posed to the test condition was about 12,000 h. The results of the
  specimens ruptured already showed the very high temperature
  resistivity of the alloys and especially of Hastelloy X (6).
  In addition tests with short life times (1000 h) were carried
  out at the same test conditions to study the creep properties of
  these alloys.

- Investigation of influence of variations in composition and
  microstructure on creep properties
  The alloy Incoloy 800H is widely qualified for high temperature
  application. Nevertheless the influence of small variations in
  composition and the related changes in microstructure on the
  creep properties of this alloy is studied to enable a carefully
  directed further improvement of the alloy's properties. Beside
  the creep tests, mechanical properties tests (tensile test, im-
  pact test etc) are carried out. The microstructure is investiga-
  ted by scanning electron microscopy combined with electron
  microprobe analysis. The structure analysis is mainly directed
to the precipitation of carbides and carbonitrides.

Both activities are carried out at ÖFZS in cooperation with the
Austrian steel industry (VEW).
Budget
In 1982 and 1983 total amounts of AS 6 mio and AS 7 mio, resp., have been spent on these programmes, which were partly sponsored by the "Forschungsförderungsfonds der gewerblichen Wirtschaft".

4. High Temperature Helium Test Rig/Prestressed Concrete Pressure Vessel

The concept of a prestressed concrete pressure vessel with elastic hot liner, developed at ÖFZS in cooperation with Austrian industries (7, 8) has been verified in an extensive experimental program, covering the following activities (9):

- concrete and liner technology
- measuring techniques in concrete at elevated temperatures
- erection and testing of an 1 m experimental ring
- design and erection of a 4 m dia, 12 m high prototype vessel
- testing the prototype vessel up to 300°C, 100 bar.

As reported in the last Progress Report (1), the prototype vessel was subjected to its design conditions for the first time in 1982. Testing has been continued since by executing two long-term cycles, each of six months duration. Thus it was possible to examine the behaviour of the vessel under prolonged stress conditions. In a thorough inspection of the vessel, in particular by ultrasonic testing of the liner, the bolts and the welds, no failure could be identified, proving the advantages of the concept.

The accompanying materials development and testing program was continued as well and will yield data for further extrapolations. The detailed analysis of measured data concerning vessel behaviour was started, indicating good agreement of measurements with design predictions.

Budget
The partners (ÖFZS and RFB) spent in 1982 AS 11 mio, in 1983 AS 9 mio for this project.

5. References
(2) E.BONEK, K.KNOTIK, P.LEICHTER, G.MAGERL, L.ROHRECKER, Microwave-assisted Solidification of free-falling Radioactive Drop­lets. Archiv Elektron. Übertragungstechn. 37, 222-228 (1983)
(3) E.M.HÖRL, K.WALLISCH, Dichte- und Dichteprofilemessungen an Hüllschichten von HTR-Brennstoffteilchen, Atomkernenergie-Kerntechnik (in press)
GAS-COOLED REACTOR PROGRAMME IN BELGIUM
PROGRESS REPORT 1982–1984

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A. GENERAL

1. NUCLEAR ENERGY IN BELGIUM

In BELGIUM, the electricity from nuclear origin is produced by Pressurized Water Reactors.

Five PWR power stations are in operation, two others are due to start operation in 1985; this represents about fifty percent of the total installed capacity. One small plant, mostly dedicated for fuel assemblies tests, (BR3 - 11 MW(e)), is operated by the research center CEN/SCK.

Belgium is engaged, together with the Federal Republic of Germany and the Netherlands, in Research and Development of sodium-cooled fast reactors, in the construction and future operation of the SNR-300. The DEBENE partnership is also part of a larger cooperation with France, Italy and UK on LMFBR's.

Because the already large effort involved in the above mentioned actions, it is not possible to develop on its own new reactor types. For HTR and/or GCFR, for which Belgian authorities and/or electricity producers have indicated interest, research and development must be carried on, on a large international collaboration basis.

B. HIGH TEMPERATURE REACTOR

2. IRRADIATIONS IN THE BR2 REACTOR

In the past, in the scope of contracts with the European Communities, with Research Centers in Germany, in the United Kingdom and in France, irradiation programmes have been conducted in the material research reactor (BR2) at MOL; for instance : irradiation of materials (graphite) for MAGNOX, AGR and HTR reactors, fuel for HTR reactors.

A large experience has been gained in this field and in CO₂ and helium technology.
3. HTR FUEL

In the past, extensive research has been made by CEN/SCK and industry (BELGONUCLEAIRE) on the coated particle concept and fuel element design for HTR reactor, mainly in the scope of the OECD-DRAGON project.

4. HELIUM TECHNOLOGY

Work carried out at CEN/SCK - MOL.

The helium test facility is designed for out-of-pile testing of materials and components for helium cooled reactors and for process heat applications.

5. EURO HKG

The electricity producers follow the evaluation work carried on in the scope of the associates EURO HKG.

C. GAS BREEDER REACTOR

6. VENTED FUEL ASSEMBLIES IRRADIATION

The "GSB" Experiment.

Work carried out at CEN/SCK - MOL.

The aim of the GSB experiment, which is carried out in the frame of a joint KFA-Jülich - CEN/SCK-Mol programme, is to investigate the irradiation behaviour of fuel bundles, pins and components, including the fuel pin venting system.

Irradiation in the BR2 reactor of the UO$_2$-PuO$_2$ fuel element (HELM 3) was terminated in December 1981 and the fuel element was unloaded in March 1982; the burn-up was 70,000 MWd/t.

From March to June 1982, the fuel element HELM 2A, with a defect fuel pin, was irradiated in BR2.

The in-pile section is located in the peripheral H4 channel of the BR2 reactor. The 96 mm OD in-pile section is surrounded by a cadmium screen which is an integral part of the BR2 type 210 driver fuel element. The test fuel element has 12 fuel pins of 8 mm outer diameter, with a fissile length of 600 mm and a total length of 1218 mm.

The main irradiation conditions of the Helm 2A irradiation experiment were:

- Loop pressure: 60 bar
- Fuel element power: 284 to 290 kW
- Main helium flow rate: 225 to 237 g.s$^{-1}$
- Linear pin power in the maximum flux: 450 W.cm$^{-1}$
- Maximum clad surface temperature: 800 °C
- Gas temperatures:
  - Fuel element inlet: 250 to 270 °C
  - Fuel element outlet: 500 to 505 °C
- Main helium gas impurities:
  - Hydrogen content: 90 vpm
  - Water content: 30 vpm

During the whole irradiation period extensive measurements have been carried out to determine the activity release of the HELM 2A fuel element.

The main conclusions are as follows:

1) Noble gas activity release to the fission gas adsorption system (spgas).

From the results it can be seen that the activity released from the fuel element via the venting system is considerably higher for the HELM 2A irradiation than for the HELM 3 irradiation, except for $^{133}$Xe (which has a much longer half-life than the other isotopes).

One should mention the influence of the reactor power reduction (reverse) on the activity release: after the power reduction the activity release remained considerably lower than before.

2) Noble gas activity in the main loop.

The total activity of the main loop gas at the end of the HELM 2A campaign is slightly higher than for the previous HELM 3 irradiation campaign.

The measured V/B values for $^{133}$Xe, $^{135}$Xe and $^{85}$Kr are, however, lower than for the Helm 3 irradiation but this is due to higher background activity coming from short living isotopes vented to the fission gas adsorption system.

The influence of the reactor power reduction in the activity in the main loop is also very remarkable (as for the activity released to the fission gas adsorption system).
The activities in 1983 were limited to the conservation of the complete loop and of the in-pile section, which is still in the BR2 reactor. Evaluation of the lifetime of the in-pile section has proceeded, due to the availability of results of the steel irradiated in BR2 from samples originating from the first in-pile section. As to the post-irradiation of the two test fuel elements, the HELM 2A test element is stored in the BR2 reactor pool, while the HELM 3 test element is under examination in KfA-Juicich. This test element will be reassembled with mainly irradiated pins. The reassembled test element will be returned to Mol, for eventual execution of a new irradiation campaign to a higher burn-up of the order of 100 MWd/kg. Funds for this campaign are not yet available, but are investigated in an European and International context.

7. CANNING MATERIALS

Work carried out at CEN/SCK - MOL.

The work performed at the Belgian Nuclear Research Center on canning materials for Fast Breeder Reactors (dispersion-strengthened ferritic steels) is going on.

The main effort is presently concentrated on a demonstration project of a bundle with this canning material. The results are given in CEN/SCK Semi-annual progress reports.

Recently, and in the scope of the R & D programme coordinated at the European Communities level (Task Force on GCFR), the CEN/SCK has been involved in the evaluation of compatibility studies between canning material(s) and CO₂ or He.

8. GBRA - BRUSSELS

The European Association for the Gas-Cooled Breeder Reactor (G.B.R.A.) has produced, in April 1982, a memorandum to the Commission of the European Communities, entitled "DESIRABILITY OF THE CONTINUATION OF THE GCFR R & D EFFORT IN THE EUROPEAN COMMUNITY"; this document has been sent to the members of the IAEA-IWGCCR.

A new memorandum has been issued in June 1984 as a contribution to the GCFR - task force; it covers the following items: cost of 12 year GBR demonstration plant programme, time required to develop, design and build the first GCFR demo plant, revised GCFR development philosophy.

GAS-COOLED REACTORS IN FRANCE IN 1983

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Abstracts

France operates eight reactors cooled by carbon dioxide and has acquired a 25% interest in a reactor of the same type built by it in Spain (Vandellos). From their commissioning, to 31 December 1983, these power plants have together generated 185 million MWh net. Moreover, based on its participation in the Dragon experiment, France focused research activities on high-temperature helium-cooled reactors, and it also observes the international efforts aimed to develop gas-cooled breeder reactors.

1 CARBON DIOXIDE-COOLED REACTORS

Since the shutdown of the G2 reactor at Marcoule in February 1980, the total installed capacity of carbon dioxide-cooled reactors is 2030 MWe, not including the Vandellos 480 MWe power plant. Six reactors are moderated by graphite (natural uranium/graphite/gas type: Marcoule G3, Chinon A2 and A3, Saint Laurent A1 and A2, Bugey 1), and the seventh by heavy water (heavy water/gas type: Monts d'Arré).
1.1 REACTOR OPERATION IN 1983

<table>
<thead>
<tr>
<th>Reactor</th>
<th>Power Rating (MWe)</th>
<th>Went Critical</th>
<th>Net Output 1983 (GWh)</th>
<th>Load Factor 1983 (%)</th>
<th>Cumulative Load Factor (%)</th>
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<td>Marcoule C3</td>
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<td>4/1960</td>
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<tr>
<td>Chinon A2</td>
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<td>1341</td>
<td>85</td>
<td>72</td>
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<tr>
<td>Chinon A3</td>
<td>360</td>
<td>10/1967</td>
<td>1456</td>
<td>46</td>
<td>47</td>
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<tr>
<td>Saint Laurent A1</td>
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<td>1359</td>
<td>40</td>
<td>65</td>
</tr>
<tr>
<td>Saint Laurent A2</td>
<td>450</td>
<td>8/1971</td>
<td>1152</td>
<td>29</td>
<td>56</td>
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<tr>
<td>Bugey 1</td>
<td>540</td>
<td>4/1972</td>
<td>2290</td>
<td>48</td>
<td>62</td>
</tr>
<tr>
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<td>480</td>
<td>5/1972</td>
<td>3030</td>
<td>72</td>
<td>74</td>
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NATURAL URANIUM/GRAPHITE/GAS

<table>
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<th>Reactor</th>
<th>Power Rating (MWe)</th>
<th>Went Critical</th>
<th>Net Output 1983 (GWh)</th>
<th>Load Factor 1983 (%)</th>
<th>Cumulative Load Factor (%)</th>
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<tr>
<td>Marcoule C3</td>
<td>40</td>
<td>4/1960</td>
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<td>74</td>
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HEAVY WATER

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<th>Power Rating (MWe)</th>
<th>Went Critical</th>
<th>Net Output 1983 (GWh)</th>
<th>Load Factor 1983 (%)</th>
<th>Cumulative Load Factor (%)</th>
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<td>Monts d'Arrée</td>
<td>70</td>
<td>7/1967</td>
<td>217</td>
<td>35</td>
<td>56</td>
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</tbody>
</table>

MARCOULE G3 (40 MWe)

The reactor generated 203 million kWh, with a load factor of 58%. This value reflects a modest operation of the installation, which underwent a long shutdown this year, during which the fuel channels were rebored to correct misalignments of the stack bricks.

CHINON A2 (180 MWe)

With a load factor of 85% for the year, the operation of this reactor was highly satisfactory. The annual maintenance shutdown, which took place from 10 May to 17 June, revealed nothing abnormal. Despite its excellent operation, this reactor is to be shut down in July 1985.

CHINON A3 (360 MWe)

Among the natural uranium/graphite/gas reactor range, Chinon A3 is the most severely affected by the problem of steel corrosion by hot CO₂, due to the types of material used and their assembly by non-gastight welds. Hence, while awaiting the completion of a major operation, called Isis, scheduled for 1984 and intended to confine and consolidate the metal structures situated in the upper part of the core, the reactor was subjected to special operating conditions:

- limitation of the CO₂ temperature at the outlet of the hottest channel to 385 °C,
- periodic inspection of the structures after four months of operation,
- no in-service refuelling.

This led to two shutdowns in 1983 (from 2 April to 16 August and 18 November to 15 December) and output limited to 345 MWe. The arrangements made at the end of the second shutdown allowed the resumption of in-service refuelling, and inspection failed to reveal any acceleration of corrosion damage.

Since the Safety Authorities issued a favorable decision concerning the methods to be implemented for the planned repairs, the latter have been scheduled for the second quarter of 1984. This corresponds to a preliminary set of works expected to last six months. They will be executed using a robot for which the necessary motions were simulated on a full-scale mock-up.

Given these long shutdowns, the reactor only produced 1456 GWh in 1983, corresponding to a load factor of 46%. Apart from shutdowns, the reactor operated successfully.
SAINT LAURENT A1 (390 MWe)

Major maintenance operations on a turbine generator and on two turboblowers caused a prolonged shutdown of the installation from 29 April to 9 September. Furthermore, a malfunction of a turboblower thrust bearing led to another shutdown from 15 to 31 December. Hence annual output was only 1359 GWh, representing a load factor of 40%.

Erosion/corrosion affecting the steam generators of this type of reactor was minor during the year 1983, in which only one leak occurred.

SAINT LAURENT A2 (450 MWe)

Following the melting of two irradiated fuel elements on 13 March 1980, partial filtration was installed to recover the uranium particles, and the reactor was restarted in May 1982. Since then, its operation has been affected by this filtration system, for which pressure drop (filter clogging) and activity limits must be observed. The latter condition the blower rate, and hence the power and time of operation.

From 25 March to 7 July 1983, the reactor was shut down for periodic maintenance and to extend the filtered zone which proved inadequate. The reactor was again shut down from 9 September to 4 October 1983 to repair a number of filter cartridges. Following this shutdown, the reactor reached its rating for the first time since the incident, but, in view of the foregoing, the load factor for 1983 was only 29%.

Two steam generator leaks were detected in 1983.

BUCEY 1 (540 MWe)

Following a first half during which the reactor displayed very stable operation, it was shut down on 25 June 1983 for maintenance after 180 days of operation without scram. The operations performed on a generator and three turboblowers kept the reactor shut down until 6 November 1983.

Simultaneous with these operations, the shock absorber cans were also replaced to deal with the jamming of these components by corrosion by hot CO₂. To reduce this corrosion, the gas outlet temperature was lowered, cutting the reactor's capacity to 480 MWe (annual load factor 48%).

Moreover, the injection of methane into the coolant to inhibit graphite corrosion of the stack creates hydrogenated deposits, causing a loss of reactivity. To offset this effect, a decision was taken to refuel the reactor with very slightly enriched fuel (0.76% U 235) effective mid-1984.

VANDELLOS (480 MWe) (joint venture with Spain)

As at Saint Laurent A, the erosion/corrosion process affecting the Vandelloes steam generators remains the most preoccupying problem of this reactor. In fact, whereas the number of leaks was limited to 20 at Saint Laurent A1, and 14 at Saint Laurent A2, 62 leaks occurred at Vandelloes, where one of the two half-exchangers is partly affected. Apart from this factor, operation is satisfactory (203 days of operation without scram) and the maintenance performed from 25 April to 13 May revealed nothing abnormal. Annual output is 3030 GWh, with a load factor of 72%, which is to be compared with the maximum possible of 85% corresponding to a power limitation to preserve the steam generators.
MONTS D'ARREE (70 MWt)

The reactor was shut down for maintenance from 30 September to 30 November. Shutdowns caused by problems on a turboblower impeller and a turbine generator cut output for the year to 217 GWh. This corresponds to a load factor of 35% only. This reactor, which is severely disadvantaged by its small size from the cost standpoint, will be shut down in July 1985.

1.2 BEHAVIOR OF NATURAL URANIUM/GRAPHITE/GAS FUEL

As of 31 December 1983, out of 462,100 'graphite core' fuel elements loaded in the Chinon, Saint Laurent and Vandellos reactors, only seven clad failures occurred that could not be attributed to operating faults (like the incident at Saint Laurent A1 on 13 February 1980). The failure rate of $1.4 \times 10^{-7}$ confirms the outstanding reliability of this type of fuel.

As for the annular fuel element of Bugey 1, three clad failures occurred out of 77,200 elements loaded. This corresponds to a failure rate of $3.7 \times 10^{-5}$, which is also very satisfactory.

1.3 RESEARCH AND DEVELOPMENT

The abandonment of the natural uranium/graphite/gas reactor by France in 1969 stopped research and development on this type of reactor. With the ageing of the installations, research has been resumed in recent years, with the aim of demonstrating the reliability of certain repairs and maintenance operations, and to build appropriate tools. This includes operations performed on Chinon A3 to try to repair the upper structures affected by steel corrosion by CO$_2$.

2 HIGH-TEMPERATURE REACTORS

In the year 1983, a test program was prepared on heat insulation panels in normal and incidental conditions (up to 1260 °C) on behalf of General Atomic Company. The Carmen 2 test facility (helium at high pressure and high temperature) was subjected to maintenance work, while awaiting a testing schedule of hot duct and steam generator models which were installed in 1980.

Similarly, the Comédie in-core test loop was kept in condition for subsequent tests. The very limited budget allocated by the CEA to high-temperature reactors in 1983 only allowed active standby, with attendance at the principal meetings.

3 GAS-COOLED FAST REACTORS

The CEA and EDF have observed design work on a helium-cooled 1300 MWt reactor conducted within the framework of Institut National des Techniques Nucléaires at Saclay. They feel that the Commission of European Communities could justifiably fund a two-year research and development program aimed to clarify certain technical aspects of the gas-cooled fast reactor, whose potential advantages concerning the breeder rate and investment cost need clarification.

4 PUBLISHED REPORTS

D. Bastien

Concept, evolution, and reliability of the fuel element for French CO$_2$ cooled reactors.

IAEA Specialists' Meeting (18-21 October 1983, Moscow)
PROGRAM STATUS
OF THE HIGH TEMPERATURE REACTOR DEVELOPMENT
IN THE FEDERAL REPUBLIC OF GERMANY*

Prepared by:
Projektträger für die Entwicklung von Hochtemperaturreaktoren, Kernforschungsanlage Jülich GmbH

on behalf of
Bundesminister für Forschung und Technologie
Federal Republic of Germany

Acknowledgements

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Interatom GmbH, Bergisch-Gladbach
Kernforschungsanlage Juelich GmbH, Juelich
Bergbau-Forschung GmbH, Essen
Rheinische Braunkohlenwerke AG, Köln

* Presented by E. Balthesen

I. INTRODUCTION

The status of the HTR development program in the Federal Republic of Germany in 1984 is characterized by the beginning of a transition phase from a national program to a commercial program. In the last 20 years the HTR technology program was strongly, nearly completely supported by the Federal Government and the State Government of North-Rhine-Westfalia. Funding of the program up to now exceeded 5 billion DM.

Within this framework it was possible to establish competent-reactor-system companies, to enable industries to supply HTR-specific components including fuel elements and nuclear graphites, to maintain the strong engagement of the national centre KFA Juelich in general R&D-activities, to build and operate the AVR-plant for more than 16 years, to erect the demonstration plant THTR-300 now approaching completion and to build and operate many efficient test facilities. Thereby the HTR technology development achieved a stage of maturity which is not only considered to be most advanced, but is also ready now for commercial deployment.

The transition phase towards commercialization was launched more than a year ago by a general program and technology reassessment performed by the Federal Government in close connection with the State Government of North-Rhine-Westfalia. This was initiated mainly by the high incremental costs for the completion of the THTR-plant. But ultimately it was not only an effort to consolidate the budget for the THTR-costs, it was also a clarification - which was overdue - of the positions of the users, the manufacturers and the government with regard to the technology deployment, the responsibilities for future activities and the necessary program redirection. It may be considered as a readjustment of the program emphasis to the general economic and energy situation in the Federal Republic.
The assessment report which comprised both the fast breeder and the HTR development included all major impacts, such as history, status, prospects, benefits, industrial aspects and international developments of the technology. One of the conclusions was: "The future of the advanced reactor systems can no longer be considered with the optimism of the early 70-ies, when construction of the prototype plants was decided, it also should not be assessed under the impression of the political circumstances of the late 70-ies which led to the problems and set-backs of nuclear energy in general and the advanced systems in particular. Today realistic expectations for the future indicate an increasing importance of nuclear energy and a significant potential for the advanced systems, maybe, however, later than originally envisaged."

The report did not question and again acknowledged the well known benefits of the technology. It was conceded that especially the prototype projects considerably suffered from the political uncertainties in the late 70-ies.

Ultimately, the completion of the construction of THTR-300 was recommended on the basis of a compromise between the governments, the utilities and manufacturers to cover the incremental costs with a relatively increased share to come from industry.

The main arguments were:
- the advanced status of erection of the plant
- the willingness of the industry to provide additional financial contributions
- the statement of the industry that the upper cost level of 4 billion DM for completion of the plant will not be exceeded
- the statement of the reactor industries that after THTR completion they will be able to offer commercially competitive power plants up to the size of about 500 MW(electric)

- Only the completion of the THTR and the continuation of the development program will keep the option open for a later supply of nuclear process heat.

The results of this procedure reestablished the political support for the continuation of the HTR-program, provided industry will take over the initiative and lead and assume responsibility for the further technology realization in the near future. The requirement for industrial responsibility is also applied to a later lead project or prototype plant for nuclear process heat application.

These guidelines had, of course, some influence on the rationale of the program and changed emphasis on some program elements. The program description is facilitated by distinguishing the five major program elements:

- AVR
- THTR-300
- THTR follow-up plant
- Nuclear process heat program
- Fuel cycle activities.

The main activities and milestones are shown in Fig. 1.

The AVR-plant, the small 15 MW(electric) pebble-bed reactor will be operated until the end of '86 when the main objectives and goals for the current operation will be achieved. The successful operation at very high temperatures and the good general condition of the plant motivated KFA Juelich to investigate a further utilization by modification of the reactor for a connexion and testing of a steam reformer bundle. This system would fit as a first step into plans of the KFA, Juelich, for a pilot plant scale technology centre for conversion technologies of dirty fossil fuel into clean, easy-to-use energy carriers and the production of liquid hydrocarbons, e.g. methanol, without environmental load.
<table>
<thead>
<tr>
<th>FRG-HTR-PROGRAM-ELEMENTS</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
</tr>
<tr>
<td><strong>AVR</strong></td>
</tr>
<tr>
<td><strong>1983</strong></td>
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<tr>
<td>FUEL TESTING</td>
</tr>
<tr>
<td>COMPONENT RELIABILITY</td>
</tr>
<tr>
<td>HIGH TEMPERATURE DEMONSTRATION</td>
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<tr>
<td>PLANT MODIFICATION FOR NPH COMPONENT TESTING?</td>
</tr>
<tr>
<td><strong>1984</strong></td>
</tr>
<tr>
<td><strong>1985</strong></td>
</tr>
<tr>
<td><strong>1986</strong></td>
</tr>
<tr>
<td>FURTHER OPERATION?</td>
</tr>
<tr>
<td>SHUT DOWN?</td>
</tr>
</tbody>
</table>

| **THTR**                  |
| **1983**                  |
| COMPLETION OF CONSTRUCTION |
| COMPONENT TESTING          |
| BUDGET CONSOLIDATION       |
| Δ 1. OPERATION LICENSE     |
| Δ CORE LOAD, SUBCRIT. EXP. |
| Δ 2. OPERATION LICENSE     |
| Δ POWER TESTING            |
| Δ PLANT HAND OVER TO UTILITY |

| **THTR-FOLLOW UP PLANTS** |
| USERS (AHU)-PROGRAM      |
| CONCEPTUAL DESIGN STUDIES |
| COMPACT REACTOR SYSTEM (CRS; HTR-500) |
| MODULAR REACTOR SYSTEM (MRS) |
| LEAD UTILITIES FORMATION |
| PROJECT STRATEGY PLAN     |
| PLANNING CONTRACT         |
| SITE DEFINITION           |
| PLANNING PHASE            |
| LICENSING PROCEDURE       |

| **NPH**                   |
| NHSS ADAPTATION           |
| R+D-PROGRAM CONTINUATION  |
| COMPONENT DEVELOPMENT PROGRAM |
| NFE-PILOT PLANT OPERATION |
| HKV-PILOT PLANT OPERATION |
| WKV-SEMITECHN. PLANT OPERATION |
| NUCLEAR COAL GASIFICATION REASSESSMENT |
| INTRODUCTION STRATEGY     |
| LEAD PLANT DEFINITION     |

| **FUEL CYCLE**            |
| LEU FUEL MANUFECT. PROCESS DEVELOPMENT |
| FUEL IRRADIATION TESTING   |
| SPENT FUEL TREATMENT R+D  |

*FIG. 1*
THTR-300, the demonstration power plant in Hamm/Schmehausen is now approaching completion. The budget is consolidated, construction work will be completed end of this year, the primary system is ready, zero-energy testing has been successfully performed, further obstacles for the operation license cannot be expected, start of power testing is scheduled for the beginning of '85 and surrender to the utility HKG end of '85. A trouble free commissioning phase and a successful operation at full load are now indispensable preconditions for a stronger utility engagement for a successor plant.

For the THTR follow up plants conceptual designs have been elaborated by the reactor industry. In accordance with the general program guidelines this work has already been performed as a private initiative under contracts from the user group AHR without government involvement. The AHR-Associates are listed in Fig. 2. This group is continuously pursuing the HTR development activities in order to analyse application possibilities of the HTR within their companies. Due to the present status of technology and to the present market conditions successor projects can only be steam cycle plants for the supply of electricity and/or process steam.

The BBC/HRB-group has presented a design for a THTR-type reactor of the size of 500 MW(e), called HTR-500, which adopts proven THTR component and design features as much as possible when appropriate. This will allow advantages to be taken of the THTR licensing experience. The companies claim to achieve at least the same specific electricity generating costs as big standardized PWR.

Moreover, the same companies have developed a conceptual design for a reactor of the size of 100 MW(e). It is an enlarged AVR-type reactor with a particularly high safety potential. It is also suitable for a double arrangement.
The KWU/Interatom-group has presented a system of small reactors of a size of about 200 MW(th) each in a modular arrangement (MRS). This concept, which aims more at the application for process steam and district heating or cogeneration and therefore at other users and markets, also utilizes specific safety features of small, low power density reactor cores.

The reactor industry has just handed over design reports and cost assessments for the HTR-500 and the MRS to the AHR group for analysis and evaluation. Therefore this report only contains short descriptions of the reactor designs.

Responses and further actions of the user group are expected within this year. Utility commitments, lead utility formations, time schedules, project financial arrangements as well as orders to the manufacturers for a planning phase are now needed, also to justify the continuation of the public funded R&D-program.

With regard to the nuclear process heat program it has been recognized and acknowledged that the technical development, tasks and financial expenses as well as the present market conditions and demands will postpone the commercial applicability of this technology further ahead than expected some years ago. However, the program is being continued with unreduced efforts. Due to the advanced status of reactor design work — there is no doubt that nuclear process heat up to 950°C can be supplied by HTR's — the program emphasis has been shifted from NHS design work to the development and testing of heat exchanging components and the qualification of respective materials as well as to the development of application processes. In particular with regard to the nuclear coal gasification processes a reassessment study of the process and results achieved up to now and on the applicability prospects has been commissioned by the Federal Government together with the State Government of North-Rhine-Westfalia. It will be performed by competent engineering and consultant companies which up to now have not been involved in the program. This study will include the basic engineering and cost assessment for a steam gasification pilot plant as well as comparisons with conventional, non nuclear conversion processes.

The nuclear process heat activities now consist of program elements of limited interdependence which will, by about '86, enable the parties involved to decide on an introduction strategy and to define a lead project plan if and when appropriate.

Fuel cycle activities are being continued for fuel supply and spent fuel treatment.

AVR and THTR-300 are operated with the high enriched uranium/thorium (HEU) fuel cycle. Fuel manufacturing process development and irradiation testing of the fuel elements have been completed. After intermediate storage the spent fuel is planned to be disposed of in a final storage, if possible without any further treatment in suitable casks. This is being prepared by various R&D-efforts.

For THTR follow up plants the low enriched uranium — without thorium — fuel cycle (LEU) will be applied. Fuel manufacturing processes will have been developed up to production maturity by the end of '85. Within the frame of the irradiation testing program the HTR fuel cycle flexibility has been confirmed by the operation of AVR with a mixture of HEU and LEU fuel elements. For the spent fuel treatment beside final storage the option for reprocessing is kept open by further head-end process development.
The direction of efforts, the time-schedules and milestones, the industrial activities and initiatives now satisfy the guidelines and the logic of the program.

At about the beginning of '86 we shall know about
- the feasibility and perhaps the approval for the AVR-modification
- THTR commissioning and operation experience
- the utilities' decisions and commitments for a THTR successor plant
- the decisions for the reactor concept and the reference application process for the long range nuclear process heat program as a basis for a lead project definition.

This cognizance will clarify the prospects for a commercial deployment of the technology in the Federal Republic and will be an excellent decision basis for the governments for a further support of the HTR-program.

HTR-Budget

1. Expenditures for the HTR program in the Federal Republic up to the end of 1983

<table>
<thead>
<tr>
<th>Description</th>
<th>mill. DM</th>
</tr>
</thead>
<tbody>
<tr>
<td>General HTR technology development</td>
<td>2,332</td>
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<tr>
<td>(reactor concepts, processes, fuel cycle)</td>
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</tr>
<tr>
<td>THTR-300, design and construction</td>
<td>2,745</td>
</tr>
<tr>
<td>AVR, construction and operation</td>
<td>243</td>
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</tbody>
</table>

2. Budget for 1984 (preliminary)

<table>
<thead>
<tr>
<th>Description</th>
<th>Amount</th>
</tr>
</thead>
<tbody>
<tr>
<td>HTR - technology</td>
<td></td>
</tr>
<tr>
<td>- NHS, components, materials</td>
<td>159</td>
</tr>
<tr>
<td>- coal gasification, long distance energy</td>
<td>46</td>
</tr>
<tr>
<td>- fuel cycle</td>
<td>31</td>
</tr>
<tr>
<td>AVR operation</td>
<td>16</td>
</tr>
<tr>
<td>THTR construction</td>
<td>648</td>
</tr>
</tbody>
</table>

II. AVR-PROJECT

1. Description of the power plant

The power plant has an electrical power of 15 MW and a thermal power of 46 MW. The reactor core has a height of 2.47 m and a diameter of 3 m. It consists of 94,000 spherical fuel elements of 60 mm diameter each. The core is surrounded by a graphite reflector. Two blowers transport the helium coolant through the core, where it is heated up to 950°C, and through the steam generator. The coolant pressure is 10.9 bars. Fuel charge and discharge is performed continuously under operation. Therefore no shut-down breaks are necessary for fuel element change. 600 fuel elements per day are removed from the core by means of the fuel handling system, and individually investigated on their burn-up. About 60 burnt fuel elements are separated thereby. The remaining fuel elements, about 540, along with 60 fresh elements are fed into the core again.
The power of the reactor is controlled by the throughput of the coolant gas. This is possible because of the negative temperature coefficient of the reactivity. The core temperature is determined by the position of the neutron absorber rods.

The first fuel element charges contained 1 g of U-235 (93 % enriched) and 5 g of Th-232 in particles with a pyrocarbon layer (BISO). Since the middle of 1982 low enriched fuel in elements, which have an additional silicon carbide coating (TRISO) has been used.

2. History of operation

The power plant has been operated since 1967. In 1974 the gas temperature was raised from 850 °C to 950 °C. Main emphasis during the first years has been placed on the demonstration of safety and reliability of the system. In addition, during the following years, experiments were performed to study the inherent safety of the pebble-bed reactor, and measurements were made which were of fundamental importance to the layout of future HTRs. Here, in particular, the testing of various fuel element types in production charges of at least some thousands each must be mentioned. Moreover, measurements were performed in test loops constructed after start-up to study the release of solid fission products from fuel elements and its deposition behaviour on different materials. The chemistry of impurities of the primary circuit and their interactions with ceramic structures are subjects for intensive investigations, too.

The availability achieved for the system was injured by shut-downs which have to be performed in the frame of the test program. A leakage in one of the steam generators in 1978 caused a long period of disuse which did not result from repairing the steam generator, but from removing the water from the primary system, as no technical equipment had been foreseen for water removal and this had to be installed first. The best availability for a one-year period was 92 %. The overall availability till the end of 1983 is 68 %.

3. Valuation of the concept

The hitherto operation has shown that it is possible to run a coolant outlet temperature of 950 °C with an HTR pebble-bed reactor. This demonstrates that the HTR can also be used as a source of process heat. Beyond that tests have proved the inherent safety of the AVR reactor.

The inherent safety characteristics of HTR have often been demonstrated with AVR by simulating disturbances. For that purpose the coolant flow was interrupted at full power operation by turning off the blowers and closing the gas valves, situated between blowers and core. Hereafter the reactor shuts itself down immediately. Since not shut-down rods were introduced during the experiment, the reactor became critical again after about 24 hours and reached a power of about 1 % of full load. This power corresponds to the amount of heat removed through the ceramic structures surrounding the reactor core.

The moderator temperature increased locally during this experiment; however, the fuel temperatures did not exceed the operation values.

The continuous load of the core comprises the advantage of the possibility of controlling the core reactivity very
flexibly according to the actual operational requirements by increasing or decreasing the speed of loading. It is not necessary to load large amounts of surplus reactivity and to compensate with control poisons - in contrary to rod and block reactors. Because of this and because of the exclusive application of ceramic materials in the core, the parasitical neutron absorptions are lower and the conversion rate is higher. An additional advantage of continuous loading is the possibility of changing over to another fuel type or without power decrease. The only necessity is to adapt the speed of loading to the varied content of heavy metal in the fuel balls. At present the core of the AVR reactor is being partially changed from the original load of high-enriched uranium (93 %) and thorium to low-enriched uranium.

4. Operation experience

Several different types of fuel elements of different fuel and various coatings have been tested up to now in AVR in big charges.

Summarizing evaluation of the results:

- Pressed fuel elements with highly enriched (\text{UTh})C_2 with BISO coating show up a good retention capability for fission products up to hot-gas temperatures of 900°C. No particle damages arise up to highest burn-ups. At higher temperatures mainly strontium is released which not only impedes repair works on the primary system, but also lowers corrosion resistance of fuel element graphite.

- Pressed fuel elements with highly enriched (\text{ThU})O_2 and BISO coating show up an excellent behaviour without particle damage, even at hot-gas temperatures of 950°C.

In particular all 15,000 fuel elements introduced since 1974 from the production for THTR-300 confirm all results as expected.

- Special attention is paid to fuel elements with LEU-TRISO particles that have been tested since 1982. Their excellent retention behaviour at hot-gas temperatures up to 950°C is confirmed by the presently especially low coolant-gas activity. Because of the low burn-up of these elements it is still too early to make conclusive statements.

5. Experience on radiation protection

It has often been reported on the quite good retention of coated particles with regard to solid and gaseous fission products up to high burn-ups. Thanks to these circumstances good experience has been made with radiation protection. Thus, during the first seven years of operation, it was quite possible to change components of the fuel handling system without auxiliary equipment and without shielding, since the contamination in the primary circuit was extremely low. Only after the burn-up of the carbide fuel element charges reached very high values and the above mentioned strontium release occurred, a stronger contamination was noticed in the primary circuit. This contamination consists essentially of strontium-90. So the use of respirators for dismounting and decontaminating primary circuit components was necessary.

The user holds the opinion that the extent and composition of this contamination, because of the fuel test character of the power plant, is an AVR specific problem and not a typical HTR characteristic.
The annual average radiation dose per person from 1968 until 1983 has been 415 mrem. The radiation dose during the first 3 years of operation has been relatively high since many entries to the containment vessel were necessary under radiation exposure.

It was possible to considerably reduce the number of entries by changing the equipment. Since 1978, however, the radiation dose is again comparably a little higher, for during that time the above mentioned strontium contamination turned out to be of disadvantage for repair-works. The value indicated above includes the natural radiation dose.

6. Further utilization of AVR

The operation of AVR is now approved until the end of 1986. The plant provided experience and confidence in the pebble bed reactor concept with regard to the continuous reload fueling scheme, high availability and efficiency, high component reliability, high safety potential, low environmental load and particularly the high temperature potential. The experience has given many impulses for the design studies of small reactor concepts, especially regarding safety features.

The main objectives and targets for the current operation will be achieved before the end of '86. They comprise component reliability demonstration, investigations of coolant chemistry, plate out and dust behaviour as well as fuel testing. About half of the core will be replaced by the new reference low enriched - without thorium - fuel elements.

After '86 the pebble bed HTR demonstration purpose will and has to be taken over by the THTR-300 for process heat applications.

For the further utilization of AVR, KFA Jülich has proposed in '83 a moderate modification of the plant for high temperature process heat systems demonstration.

The objectives of AVR-modification are:

1. systems demonstration for high temperature process heat applications

2. nuclear demonstration of refinement of fossile fuels, e.g. conversion of coal into synthesis gas or further more into energy alcohols

3. contribution to licensing procedures

4. investigations for novel energy-systems with approach to zero emission.

The technical concept for this first HTR-process heat application is shown in Fig. 3. It is intended, to split the primary coolant mass flow of the AVR-reactor: One half of the mass flow is no longer nourishing the reactor internal steam generator, it is extracted and feeds the external process heat plant via a hot gas duct. After intensive discussions it was decided to install the process heat plant in a separate plant building (16), to be erected close to the AVR-reactor-building at the plant site. The process heat loop will be fed by a hot gas duct (11) with half of the reactor coolant mass flow; after cooling down within the heat exchanging plant components (13, 14), the primary coolant will be fed back into the reactor circuit by a cold gas duct (12).
The modification is based on the following design principles:

- minimum changes of the configuration of the AVR plant
- free lay out for the high temperature process heat loop, i.e. the design is unaffected by the nuclear process heat source
- maximum accessibility to the process heat loop
- far-reaching decoupling of the loop from AVR with respect to safety.

Striving after maximum accessibility is one of the essentials of design: The AVR-reactor shall be usable as a versatile source of nuclear heat, that means steam reforming as the key process for hydrogasification as well as steam gasification of hard coal or other techniques of novel energy-systems. During the first phase of operation after modification the AVR-reactor would be coupled with a steam reformer in a direct cycle; a second step consists of the operation and demonstration of the maturity of an intermediate heat exchanger with the option of steam-gasification of hard coal for example.

In 1984 a feasibility study is being carried out. The considerations of modification shall be detailed to such an extent, that a solid-based decision basis for a realization can be provided. Therefore the emphasis of work is put on safety and licensing conditions, on the evaluation of a suitable concept and finally on a compilation of the investments. The work so far shows reasonable ways to fulfill the licensing criteria.
III. THTR-300-PROJECT

1. State of construction

Primary System

On August 30, 1983 nuclear commissioning of the reactor was initiated according to schedule by loading of the fuel elements and start of the zero energy tests. On September 13, 1983 the THTR-300 reached a self-sustained nuclear chain reaction for the first time (first criticality).

In mid-November 1983 the assembly of the steam/feedwater lines in the annular space for the steam generators was completed. On December 3 and 4, 1983 the gas pressure test of the steam generators and the steam/feedwater lines in the annular space for the steam generators was performed. Present activities on the construction site are concentrated on the further assembly of the steam/feedwater circuit in the turbine building, the instrumentation and control system, the ventilation system as well as the gas circuits. Start-up of the operation of the helium circulators has been initiated.

Secondary System

All major components of the steam/feedwater circuit are completed and installed. The condensate purification plant has been installed and subjected to a proof test. In the reactor hall the assembly of the high-pressure pipes has almost been completed; at the present time the small piping is being installed and the insulation of the steam/feedwater circuit is being mounted. In the turbine building the high-pressure and low-pressure piping is being installed.

The emergency diesels are installed and the relevant service systems are being mounted.

The switchgear systems and the instrumentation and control cubicles are being installed in the electrical equipment building including the cabling. Currently the equipment is being installed in the structurally completed control room of the power plant.

2. Nuclear licensing procedure

All the nuclear construction licenses for the THTR-300 have been granted with the exception of a few supplements. The nuclear operating license was split up, in agreement with the licensing authorities and experts into two partial operating licenses covering the "zero-energy-tests" and the "power operation tests including full power operation".

The first partial operating license for performing the zero-energy tests was granted on July 19, 1983. The zero-energy tests include loading of the core up to first criticality in air atmosphere, complete loading of the core in air atmosphere, drying of the primary circuit and cold testing of the components of the primary system in nitrogen atmosphere as well as reactor physics tests for verification of the core design and shutdown safety of the reactor in air and nitrogen atmospheres.

The second partial operation license is expected to be granted in the second half of 1984. The power plant is scheduled for handover to the owner in October 1985.
Start of nuclear commissioning

On August 30, 1983 nuclear commissioning of the THTR-300 was started by loading the reactor with spherical elements. On September 13, 1983 the reactor reached a self-sustained nuclear chain reaction for the first time (first criticality) after 198,180 spherical elements had been loaded. The subsequent complete loading of the reactor core with 674,200 spherical elements was completed on October 11, 1983.

The zero-energy tests were performed during and after the loading procedure for experimental verification of the licensed design of the reactor core and for verification of the function and effectiveness of the control and shutdown rods. The successful completion of the zero-energy tests performed in the cold reactor on November 27, 1983 demonstrated the satisfactory performance of the components required for initial loading and the reliability of the reactor physics calculation methods applied.

3. Structure of the reactor core

The THTR-300 reactor core consists of a loose bed of about 675,000 spherical elements of 6 cm diameter. One fuel element contains 0.96 g U-235, 93% enriched, and 10.2 g Th-232, in coated particles with mixed oxide kernels. With the exception of the initial core, the reactor contains fuel elements exclusively. These are continuously added and discharged passing through the core six times on an average with a mean run time of six months per passage. Thus in its equilibrium state the reactor core is composed of a mixture of fuel elements of different burn-up states.

For obtaining in the initial core a state similar to that of the equilibrium core, three types of spherical elements (SE) are loaded into the core: fuel elements (FE), graphite elements (GE), and absorber elements (AE).

As in the case of the equilibrium core, the initial core is designed as a two-zone core in the radial direction (Fig. 4). The outer zone of the core (D = 80 cm) contains more fissile material than the inner zone of the core in order to obtain a uniform radial power distribution.

Table 1 shows the composition of the initial core:

<table>
<thead>
<tr>
<th>Type of spherical elements (SE)</th>
<th>Number of elements</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel elements (FE)</td>
<td>358,200 (~53 %)</td>
</tr>
<tr>
<td>Graphite elements (GE)</td>
<td>272,500 (~40 %)</td>
</tr>
<tr>
<td>Absorber elements (AE)</td>
<td>43,500 (~7 %)</td>
</tr>
<tr>
<td>Spherical elements (SE)</td>
<td>674,200 (~100 %)</td>
</tr>
</tbody>
</table>

Table 1:

Mixing ratio:

Inner zone of core (approx. 40 % of SE):

Outer core (approx. 60 % of SE)
AE : GE : FE ~ 1 : 6 : 12

4. Facility for initial loading

To build up the specific configuration of the initial core, a special loading facility was developed for the initial loading of the reactor core, which does not correspond to the fuel circulating system designed for continuous refuelling. Using this initial loading facility, the spherical elements were loaded into the reactor core from the 48 m level platform by gravity (Fig. 5).
The loading station was equipped with five facilities each combined of a cask and a singulizer unit, which were mounted on support structures and coupled to the refuelling pipe system. Two loading units were provided for fuel elements, two for graphite elements, and one for absorber elements.

5. First criticality

In the first phase of the nuclear commissioning program the reactor core was filled with spherical elements to such a level as to reach criticality with no absorber rods inserted.

5.1 Performance of initial loading

The following actions preceded initial loading (Fig. 6):

- insertion of a guide tube into the fuel element discharge pipe to accommodate the auxiliary start-up neutron source
- filling of the fuel element discharge pipe and the conical reflector bottom with graphite spheres
- erection of a working platform for installing the initial load equipment
- installation of the central loading pipe, the turnable switch and the 15 loading pipes
- removal of the working platform and discharge of the graphite elements down to the upper end of the fuel element discharge pipe.

On August 30, 1983 loading of the reactor core was started, using the initial loading facility. The inner and the outer core zones were separated by separating sheets up to a cylindrical height of 2 m. Loading of the spherical elements was effected in so-called loading packages. One loading package at a time was introduced into a loading pipe, each loading consisting of about 500 spherical elements prepared in the mixing ratio designed for the respective core zone.

5.2 Reaching first criticality

In the course of loading the reactor, it had to be ensured at any time that the reactor does not reach uncontrolled criticality or supercriticality. This was ensured in the THTR-300 using the method of the inverse counting rate.

This method makes use of the fact in a subcritical assembly \( k_{\text{eff}} < 1 \) with a neutron source in the ideal case the inverse counting rate of a detector is proportional to \( (1-k_{\text{eff}}) \).

When approaching the critical state \( k_{\text{eff}} = 1 \) the inverse counting rate \( 1/Z \) approaches the value zero. By linear extrapolation of the inverse counting rate it was possible at any time of the loading procedure to determine the expected critical state.

A Cf-252 source having a source strength of approximately 230,000 neutrons/s was inserted as an auxiliary start-up neutron source in a central position at the core bottom. The neutron flux density induced by this neutron source and more or less increased by the pebble bed was measured and controlled by three high-sensitivity \( \text{BF}_3 \) detectors located in the pebble bed.
Fig. 7 shows the measured inverse counting rates of the three detectors as a function of the number of spherical elements loaded. All three curves demonstrate the expected convergent shape showing the same critical number of spherical elements in approaching criticality.

During the loading up to first criticality the incore rods were inserted with their tips contacting the surface of the pebble bed. Prior to reaching first criticality, a group of six reflector rods was partially inserted for controlling the reactivity. As schematically shown in Fig. 8, the neutron flux rises in the subcritical reactor, as the absorber rods are withdrawn and stabilizes at a higher level after termination of the rod movement. If, however, the reactor is slightly supercritical, the neutron flux continues to rise after termination of the last rod movement. This procedure permitted an exact determination of the moment of first criticality of the THTR-300.

The first self-sustained nuclear chain reaction was reached on September 13, 1983 with the loading of 198,180 spherical elements. The power generated was in the range of several watts and the radioactivity released was so low that already a few hours after shutdown of the first criticality, access of the operating personnel to the reactor core was possible again without any hazard.
By treading on the pebble bed and leveling out the piles, a filling factor (volume fraction of spheres) of the reactor core of 0.643 was achieved. For this filling factor a critical number of 204,000 spherical elements had been calculated in advance. Thus the difference is about 5,800 elements, which corresponds to a reactivity equivalent of no more than 0.4 % $\Delta k/k$. This close agreement of calculation and experiment proves the reliability of the data sets and calculation models used.

Fig. 9 shows the exponential rise of the counting rate in the slightly supercritical reactor. In this test the rise of the neutron flux was inhibited after about 700 seconds by counteraction of the reflector rods resulting in balancing out of the effective multiplication factor at a value of 1.

5.3 Zero-energy tests, complete loading

After having reached first criticality, the effectiveness of the reflector rod groups was measured. For this purpose the critical state was first established by means of the incore rods with the reflector rods completely withdrawn.

**Fig. 8** Schematic diagram of bringing the reactor to criticality. The arrows indicate addition of reactivity

**Fig. 9** Reactivity and counting rate during first criticality of THTR-300
The reactivity changes were calculated by the inverse-kinetic method from the changes of the counting rates during rapid insertion of the reflector rods. The reactivity values of the reflector rods determined by this method were in agreement with the calculated values within the accuracy of measurement.

On September 20, 1983 the further loading of the reactor core with spherical elements was initiated (Fig. 10). On October 11, 1983 the reactor core was completely loaded (Fig. 11). Thus the loading procedure took approximately six weeks, as had been envisaged.

Subsequently the following measurements and tests were performed:

- Successful testing of withdrawal and insertion of individual incore rods using a long-stroke piston drive for insertion.

- Establishing the critical state in the completely loaded initial core; the rod positions calculated for this state coincided with the experimentally determined values except for a small difference of approximately 0.5 % \( \Delta k/k \).

With these experiments the zero-energy tests in the cold reactor were completed. Power test operation will start in the second half of 1984.
IV. THTR follow-up plants

The commercial deployment of HTR's in the Federal Republic will be started with THTR follow-up plants.

By orders of the user group AHR the reactor company groups BBC/HRB and KWU/Interatom have completed conceptual designs for HTR's of medium and small size for electricity and/or process steam generation:

- BBC/HRB: HTR-500
- KWU/Interatom: Modular reactor system
- Moreover BBC/HRB has developed a small reactor design called HTR-100.

This design work has been performed by private efforts only without government involvement.

The technical design reports and cost figures from the companies as well as licensability assessment by a safety expert group which advises the Federal Ministry for the Interior have been surrendered to the utilities for evaluation and as a decision basis.

As this procedure is now going on, only short design descriptions are presented in this report.

IV.1 HTR-500 (BBC/HRB-Design)

Technical concept of the HTR-500

The structural elements or component concepts of the THTR, which had been developed and licensed according to the latest state of science and technology, are being adopted for the HTR-500 in their greater part. The experience gained during the construction of the THTR is being fully utilized and partly results in simplifications and optimizations. The plant and safety concepts of the HTR-500 are based on a low plant risk which was established by comprehensive accident analysis as well as in the THTR licensing procedure. The experience and the know-how gained in this process are used to improve the economics, maintaining a high safety standard at the same time. Thus an HTR-specific safety design is achieved.

Due to the system-specific slow accident evolution of the HTR, manual measures to be taken by the operating personnel are included in the safety concept and taken into consideration in the reliability analysis.

Also repair work on components and systems, which can be done because of the long period available, are taken into consideration in the reliability analysis for accident control.

Fig. 12 shows an overall site plan of the HTR-500 power plant showing the arrangement of the different buildings. In the center of the power station you see the reactor containment building which safely encloses the PCRV with the primary circuit, shutdown facilities, parts of the decay heat removal system, and safety-related components of the reactor auxiliary systems against external impacts. The components carrying activity are located in the reactor containment building.
Fig. 12 shows a longitudinal section of the overall plant. The engine building is connected to the reactor containment building at its front face. The reactor service building is located directly adjacent to the reactor containment building.

Fig. 14 and 15 show sections of the prestressed concrete reactor vessel with internals. The helium flows downward through the reactor core so that the in-core rods and reflector rods as well as the metal internals are cooled by the cold gas flow. The fuel elements are loaded with low-enriched uranium and pass through the reactor core only once (OTTO fuelling). They are unloaded via four discharge pipes. The increase in power as compared to the THTR 300 is achieved by increasing the number of primary loops from 6 to 8 according to a modular system and by a small increase in power per loop. The use of standardized structural elements results in a simplification of the nuclear law licensing procedure and in a reduction of the design, construction, and commissioning efforts.

The following structural elements have been standardized:
- components of the prestressed concrete reactor vessel
- steam generators
- circulators
- shut-down facilities
- ceramic blocks of the core structure
- fuel elements.

As in the THTR-300, the reactor pressure vessel is designed as a prestressed concrete reactor vessel with one large cavity accommodating the complete primary system.

The integrated design is now as before the optimum design, from safety as well as operational aspects. The use of a prestressed concrete reactor vessel saves extensive shielding measures and in-service inspections which are required
FIG. 13. Longitudinal Section of Overall plant
for steel pressure vessels. Thus this design contributes to high availability and economics.

The eight counter-current steam generators - in helicoil design as in the THTR-300 - are located in the annular space between the thermal shield and the inner wall of the prestressed concrete reactor vessel. In view of a simple design and a favourable operating and accident behaviour the steam generators were designed without re-heating.

Each steam generator has its own circulator. The design principle of the circulators corresponds to that of the THTR circulator, they are, however, mounted on active magnetic bearings and in the vertical position.

The helium flows downward through the reactor core so that the in-core and reflector rods as well as the metal internals are cooled by the cold gas flow. The fuel elements are loaded with low-enriched uranium and pass through the reactor core only once (OTTO fuelling). The main design data of the HTR-500 are summarized in Table 2.

<table>
<thead>
<tr>
<th>Thermal core power</th>
<th>1 250 MW</th>
</tr>
</thead>
<tbody>
<tr>
<td>Net electrical output</td>
<td>500 MW</td>
</tr>
<tr>
<td>Efficiency</td>
<td>40 %</td>
</tr>
<tr>
<td>Average load factor</td>
<td>80 %</td>
</tr>
<tr>
<td>Mean core power density</td>
<td>6.0 MW m⁻³</td>
</tr>
<tr>
<td>Primary gas pressure</td>
<td>50 bar</td>
</tr>
<tr>
<td>Hot gas temperature</td>
<td>700 °C</td>
</tr>
<tr>
<td>Cold gas temperature</td>
<td>260 °C</td>
</tr>
<tr>
<td>Main steam pressure, turbine inlet</td>
<td>180 bar</td>
</tr>
<tr>
<td>Mean steam temperature, turbine inlet</td>
<td>525 °C</td>
</tr>
<tr>
<td>Feed water temperature</td>
<td>190 °C</td>
</tr>
<tr>
<td>Condenser pressure</td>
<td>0.06 bar</td>
</tr>
</tbody>
</table>

Table 2: HTR-500 Main Design Data

It is possible to apply the basic concept of the HTR-500 also for lower power outputs. A unit with 300 MW(el) for instance needs only 6 primary loops as the THTR-300 reactor. The smaller core needs one fuel element discharge pipe only.

Safety design

In contrast to the THTR, the HTR-500 is equipped with a separate two-loop decay heat removal system so that the main cooling loops do not have a safety function. For this reason the main cooling loops are designed in the same way as for a conventional power plant, which is very important from the economic point of view.

In case of failure of the main cooling loops or in the event of accidents which would result in excessive impacts on their components, the decay heat removal system comes into action. For all accidents one decay heat removal system is sufficient for removing the decay heat from the core.

Total failure of the decay heat removal system is permitted over a period of 5 to 10 hours without resulting in safety-relevant damage. Subsequent restart of the decay heat removal system is possible within this time.

During this relatively long permitted failure period, the required measures can be taken to restore the decay heat removal by manual control. In this way simultaneous occurrence of single failure and repair case can be controlled by the 2 x 100 % decay heat removal systems.

An analysis on the total failure of the decay heat removal system shows that the consequences of such a failure do not represent a dominating contribution to the overall risk of the plant.
The shutdown concept is designed analogically to the THTR concept, consisting of reflector and incore rods.

Reactor scram is effected automatically by the reflector rods.

For long-term shutdown the incore rods are inserted by hand to their full depth. This is possible because, for reactivity reasons, insertion of the incore rods is necessary only after about 10 hours after shutdown by reflector rods.

The activity release concept is as follows:

As in the THTR - the primary system and its components are located in a burst-safe prestressed concrete reactor vessel.

The prestressed concrete reactor vessel and the components carrying radioactive gas are integrated in a reactor containment which is designed against external impacts only, e.g. airplane crash. The reactor containment does not contain a liner.

Operational and accidental primary gas leakages up to a leak size of about 2 cm$^2$ are controlled by filter systems and exhausted to the environment over a vent stack.

The occurrence of major leakages is extremely improbable. The maximum leak cross section is limited by structural measures to 33 cm$^2$ - as in the THTR. Thus only a slow escape of the coolant is possible; the time until pressure equilibration is about 1 1/2 hours. In such a case the coolant gas is directed immediately into the vent stack at the beginning. When the leak rate becomes smaller the pressure relief system of the reactor containment is closed again and the gas is discharged via the exhaust filter system.

Due to the low content of activity in the coolant gas the environmental load remains below the limits of the Radiation Protection Act, even in the event of such an accident, because of the short-term and only low overpressure in the reactor containment during the depressurization procedure and the low coolant gas activity, only minor activity amounts are released near to ground level.

The safety concept of the HTR-500 is in agreement with the draft of HTR-specific rules and guidelines established by the TÜV-Arbeitsgemeinschaft Kerntechnik West in November 1980.

Based on these results, a report on the technical concept has been submitted to an advisory board of the Federal Ministry of the Interior, which has been invoked to perform a first evaluation of the HTR-500 safety concept and licensability. This board had already been involved in the HTR-900, the PNP and the HHT projects and is composed of members of the reactor safety committee, the TÜV, and the licensing authorities. A positive evaluation of the licensability of the HTR-500 concept has been given.

Costs and Economics

The consequent use of HTR-specific safety characteristics as well as an optimization of the design of the components and circuits leads to the result that the HTR-500, in spite of its lower power level as compared to the 1230 MW(e1) convoy pressurized water reactor of the present design, has only slightly higher specific plant costs and nearly the same electricity generating costs in concurrence with substantially lower capital costs.
When applying the HTR-500 for electricity and process steam generation, in contrast to the PWR the HTR can also supply high-tension steam. Furthermore the HTR-500 presents clear cost advantages as compared to hard-coal fired power plants.

IV.2 HTR-100 (BBC/HRB-Design)

The concept of the small HTR reactor is based on the concept of the AVR 15 MW(el) experimental power plant.

The concept of the HTR-100 is characterized by the integrated arrangement of all primary system components in a steel reactor pressure vessel, upward helium flow through the core to the steam generator located above the core, and control and shutdown of the core by reflector rods and small absorber spherules in channels of the reflector buttresses (Fig. 16). Table 4 gives the main design data of the nuclear heat generating system.

During normal operation the thermal power is transferred to the secondary loop by three loops of steam generators and circulators. Decay heat removal is also ensured by these three loops at a high safety level. In the hypothetical event of a total failure of the primary and secondary heat sinks, decay heat removal is mainly effected by radiation and conduction of the heat from the reactor vessel to an external concrete cooling system located within the safety containment.

The reactor core, the shutdown system, the steam generators and the circulators are integrated in the reactor pressure vessel.

The reactor core is operating at a mean power density of 4.2 MW(th)/m³. Continuous reshuffling of the spheres guarantees a high and uniform burn-up.

The core is vertically arranged on a star-shaped support structure resting on the core barrel periphery. The side reflector is laterally supported by the thermal shield.

The reflector rods are used for control and reactor scram. For long-term shutdown small absorber spherules are additionally filled in vertical holes in the buttresses.

One steam generator with three sections is supported at the core barrel periphery of the upper pressure vessel ring. The circulators are located outside the reactor vessel and are flanged at the top of the pressure vessel.

The reactor core consists of a loose bed of approx. 330,000 spherical elements contained in a graphite cylinder approx. 3.5 m in diameter and about 8.7 m in height. Only 50% of the spherical elements are fuel elements, the remainder are graphite elements. Four buttresses (1 x 0.4 m) of graphite contain vertical holes for the shutdown system. The bottom reflector is conically shaped with an inclination of 30° and ends in four discharge pipes each of 500 mm diameter. Puelling and reshuffling of the elements is effected continuously. The elements are introduced through 5 pipes allowing the build-up of a two-zone reactor core. After passage through the discharge pipes they are pneumatically transported to the measuring device. Based on the results from burn-up measurement a data processing system decides upon the further residence of each sphere. The sphere is either recirculated into the reactor core or, in case of sufficient burn-up, forwarded to the discharge facility.
The maximum fuel temperature of a fuel element is approx. 1200 °C.

The excess reactivity of the equilibrium core allows load-following between 50 and 100% load in the equilibrium cycle.

The cavity holding the spherical fuel elements is formed by the ceramic internals, the bottom reflector, the side reflector and the top reflector. The reflector acts as reflector and shielding.

The bottom reflector is composed of individual horizontally separated graphite columns arranged on a metal support structure. The conical bottom surfaces of the core open up into four element discharge pipes.

The side reflector, whose design is adopted from the TMTR, is composed of an inner and an outer cylindrical wall with the graphite blocks arranged in block columns and ring layers. At the outside the side reflector is supported by the thermal side shield. In the inner section of the cylindrical wall vertical holes are provided for the reflector rods. Four buttresses composed of reflector blocks, arranged at 90° angles, are equipped with channels for small absorber spherules and protrude into the pebble bed.

The three circulators arranged above the reactor pressure vessel, convey the coolant into the cold-gas plenum. They are driven by three-phase asynchronous motors. Each circulator is equipped with a shut-off valve on its intake side. The rotor, the drive motor with its cooler and the shut-off valve form a unit.

The heat generated in the reactor core is transferred to the secondary circuit by one steam generator divided in three sections connected in parallel. The helical-type steam generator located above the core is designed for downflow boiling.

The feed water inlets and main steam outlets are located laterally at the upper cold end of the steam generator. The lines penetrate the reactor pressure vessel horizontally immediately above the steam generator.

In normal operating conditions the reactor core is cooled by an upward coolant flow entering the bottom part at a temperature of ca. 260°C and a pressure of 70 bar.

Core dimensions, power density, and coolant flow density are chosen so as to avoid lifting of the spherical elements due to excessive core pressure drop. The hot gas leaves the core at a temperature of about 730 to 750°C, passes through the top reflector and mixes with the different bypass flows to a steam generator inlet temperature of 700°C. After transferring the heat to the three secondary loops, the coolant gas enters the circulators at a temperature of 250°C.

In normal operating conditions, three sections of the steam generator, each being connected directly to a circulator, transfer roughly 260 MW to the secondary steam cycle.

In case of a reactor shutdown the residual heat of the core and the decay heat are transferred to the steam cycle in the same way as in normal operating conditions. With the reactor pressurized, one steam generator/circulator unit is sufficient for cooling the system, i.e. there is a 3 x 100% redundancy of the decay heat removal system, whereas for the depressurized reactor two units are required, i.e. 3 x 50%. 
Since the steam generator serves as a decay heat removal system, minimum flow conditions have to be observed in order to cool down the steam generator smoothly. These procedures have been well examined for the THTR prototype reactor ensuring that the plant is not exposed to undue transients.

The 3 x 100% capacity of the decay heat removal systems ensure that in all circumstances the primary circuit as well as the steel vessel are cooled with a high reliability, yielding a negligible risk of total failure of the decay heat removal systems.

Even in the hypothetical case of total failure of the decay heat removal system the decay heat could be removed solely by radiation and convection to the external concrete cooling system, which operates inherently safe by natural convection.

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**Table 4: HTR-100 Main Design Data**

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal power of the reactor</td>
<td>256 MW</td>
</tr>
<tr>
<td>Thermal efficiency</td>
<td>38 to 39 %</td>
</tr>
<tr>
<td>Mean core power density</td>
<td>4.2 MW/m³</td>
</tr>
<tr>
<td>Uranium content per fuel element</td>
<td>16 g</td>
</tr>
<tr>
<td>Mean initial enrichment</td>
<td>10.2 %</td>
</tr>
<tr>
<td>Mean burn-up</td>
<td>95,000 MWd/t</td>
</tr>
<tr>
<td>Helium pressure</td>
<td>70 bar</td>
</tr>
<tr>
<td>Helium temperatures at steam generator</td>
<td>700/253 °C</td>
</tr>
<tr>
<td>Steam conditions at SG outlet</td>
<td>190/530 bar/°C</td>
</tr>
</tbody>
</table>

**FIG. 16**

Reactor Pressure Vessel with Internals
Design features

The main boundary condition for the development of the HTR-Module concept was the demand for minimizing the technical and commercial risks in planning, construction and operation of an HTR-plant. Therefore the design features are based upon the well established technologies and the operational experience of LWRs and of the AVR with special attention to a simple safety technology, wide flexibility for application and high availability.

The basis of the HTR-Module concept is the combination of small standardized plant units, the so-called modules, to form plants of a wide range of thermal ratings.

Characteristically, a module has a power rating of 200 MJ/s (Table 5).

For reasons of economics and demand, overall plant capacities appear to be limited at a maximum of 1600 MJ/s.

An HTR-Module consists of the following system elements:

- the pebble-bed reactor with a thermal output of 200 MJ/s at an average power density of 3 MW/m³. Its control and shutdown elements move vertically within the side reflector and it is equipped with facilities for fuel element handling. Helium flows downwards through the core,
- the steel pressure vessel containing the reactor,
- the steel pressure vessel containing the steam generator as heat transfer component,
- the coaxial gas ducts between reactor pressure vessel and the steel pressure vessel of the heat transfer component,
- the primary circuit helium blower which is attached to the steel pressure vessel of the heat transfer component,
- the primary cells, two adjacent concrete cavities in the reactor building, separated from each other by a concrete wall. They house the reactor and the heat transfer system and act as confinement,
- the cavity cooling system to control the wall temperature of the reactor cavity. It also serves as an emergency cooling system.

Fig. 17 shows the section of the primary system with steam generator as heat transfer component:

The arrangement of the reactor and heat-transfer components in separate steel vessels facilitates their accessibility for inspection, repair and disassembly, leading both to a higher availability and a higher degree of safety.

The principle of coaxial gas ducting, by which cold gas at higher pressure encloses all sections filled with hot gas, ensures that the reactor pressure vessel will never be subjected to any impermissible temperature loads. The pressure vessels are designed and manufactured in accordance with LWR criteria. Cavity coolers are arranged around the reactor pressure vessel to maintain the concrete walls of the reactor cell at design temperatures during power operation. They are of triple redundancy and connected to the emergency power supply system.
The required diversity of the shut-down system is achieved by absorber rods and absorber spherules, which are positioned above the side reflector. They can be entered under gravity into vertical bores, when necessary. There are no core-rods; the high, slim geometry of the core renders them superfluous. The fuel elements are circulated repeatedly through the core until they have reached a target burn-up of approximately 80,000 MWd/t HM.

2. The operational concept

The separate HTR-Modules of a multi-module plant may be operated almost independently both from each other and from the user. Each module can be shut down and started up individually and - having reached operating conditions - be connected again to the overall plant system. Thus, even in the case of single-module outage, operation of the plant may be continued at a reduced power level.

In normal operation the thermal power of each module of the nuclear plant automatically follows the load on the user side within the range of 100% - 50% - 100%, without any limitations with regard to load cycle frequency. A multi-module plant has a wider part-load range, as entire modules can be taken out of operation.

For hot shut-down, the reflector rods are used to maintain the reactor in a cold, sub-critical condition, the absorber spherules of the secondary shut-down system are inserted into the respective bores in the side reflector.
3. The safety concept

The HTR-Module is a system which combines the general HTR-specific safety properties with the particular safety advantages of small reactors. Its limited size, the low power density and the slim core design yield an extraordinary degree of passive safety.

Attesting the licensability of the module concept, an advisory committee of the German Licensing Authorities stated in an expertise that owing to its safety characteristics a simplified and streamlined licensing procedure may be anticipated.

Safety analysis has been carried out on a great number of postulated reactivity accidents, failures of the main transfer system, primary and secondary pipe ruptures and on the radiological consequences of severe accidents.

In response even to fast changes of the primary circuit heat balance, the temperature-transients of the HTR are extremely slow owing to the high heat-capacity of core and other graphite structures. Thus, even on the assumption of extreme hypothetical accidents, ample time remains for engineered safeguards to react or for taking emergency measures to prevent or limit any potential damage to reactor components.

A fuel temperature of 1600°C was not exceeded in any of the accident combinations analysed (Fig. 18). Above this temperature fission product release due to fuel particle failure can no longer be excluded. Even for such an extremely hypothetical situation as the depressurization of the primary system in combination with a large water ingress and subsequent heating-up of the core, it could be demonstrated under conservative assumptions that the dose rates in the environment of the reactor remain distinctly below the accident dose rate limits specified by the German Radiation Protection Regulations.

As a consequence, the shut-down and decay heat removal systems as well as the confinement have to meet much less stringent safety requirements. This leads to considerable simplifications in manufacturing and installation, in in-service inspection and in the proofs to be submitted in the Nuclear Licensing Procedure.

![FIG. 18 TIME-DEPENDENT MAXIMAL FUEL ELEMENT TEMPERATURE AT ACCIDENT CONDITIONS](image-url)
Under normal shut-down conditions decay heat is removed by the main heat transfer systems. However, only the operational aspects need be considered in their design, since even in case of their failure, decay heat removal from the core to the environment will take place simply by natural convection, radiation and heat-conduction through the steel pressure vessel and the cavity coolers.

Another marked simplification is the use of only one standard shut-down procedure for any type of disturbance, to bring the reactor into a hot sub-critical condition. Also the concept of pressuretight containment may be abandoned in favour of the simpler confinement without any risks to the environment.

It could be demonstrated for a multiple-module plant that no safety problems will arise in connection with start-up or shut-down of individual modules. The reactor building is designed to withstand external events, in accordance with the usual licensing regulations.

4. Applications for the HTR-Module with steam generator

One very important field of application of HTR-Modules is the supply of process steam to medium or large industrial complexes. The HTR is the only type of nuclear reactor system, with the aid of which it is possible to generate all steam conditions usually needed in the process industries. The possibility of the combined generation of process steam and electrical power contributes towards diminishing the dependence upon the external grid. Together with the flexible power-rating, high availability and environmental advantages of the HTR-Module this will certainly make a decision easier to replace fossil fuelled steam generators by nuclear heated ones.

The flexibility in the choice of power-ratings and the excellent safety characteristics enable urban utilities to build and operate their own nuclear plants for district heating and electrical power supply even in the vicinity of towns. Not only can they choose the exact plant capacity adequate for an existing demand, but they also have the possibility of adjusting it to an increased demand at a later stage, by adding the necessary number of new modules.

The very high thermodynamic efficiency of such dual purpose plants makes for high economic viability.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal power</td>
<td>200 MW</td>
</tr>
<tr>
<td>Core diameter</td>
<td>3.0 m</td>
</tr>
<tr>
<td>Core height</td>
<td>9.6 m</td>
</tr>
<tr>
<td>Mean power density</td>
<td>3.0 MW/m³</td>
</tr>
<tr>
<td>Helium temperatures (inlet/outlet)</td>
<td>250/700 °C</td>
</tr>
<tr>
<td>System pressure</td>
<td>60 bar</td>
</tr>
<tr>
<td>Number of control rods</td>
<td>6</td>
</tr>
<tr>
<td>Number of absorber ball systems</td>
<td>18</td>
</tr>
<tr>
<td>Loading scheme</td>
<td>Reshuffling/15 times</td>
</tr>
<tr>
<td>Heavy metal loading of fuel element</td>
<td>7 g</td>
</tr>
<tr>
<td>Enrichment</td>
<td>7.8 %</td>
</tr>
<tr>
<td>Burn-up</td>
<td>80,000 MWd/t</td>
</tr>
<tr>
<td>Fuel incore time</td>
<td>1,020 days</td>
</tr>
</tbody>
</table>

Table 5: Main data for the modular HTR core
V. Nuclear process heat program

Introduction

The Nuclear Process Heat Program aims from the beginning in 1972 at the direct utilization of nuclear heat up to 950°C for coal gasification and steam reforming processes.

Fig. 19 shows the main fields of possible applications in which the HTR may enter into the market in the future. It is evident that the HTR permits optimum adaptation to coal gasification plants, because these processes will play a key role in future energy systems.

Beside these applications a non polluting/emission free energy supply concept on the basis of "chemically bound energy transport", more popular known as "ADAM/EVA"-concept has been worked out.

All these processes will play their part in the attempt to lower the environmental impact of the use of "dirty energy carriers" and to improve the overall efficiency as well as economics in the use of energy.

By taking the already described HTR-concepts (CRS, MRS) for electricity and process steam generation as prerunning basic systems, the main efforts to realize the above mentioned goals, must be undertaken in the fields of:

- adaptation of the NHSS-concepts to the higher gas outlet temperatures and operation behaviour of chemical plants
- material development and qualification
- component development and testing
- R&D on specific safety questions (such as process gas explosions)
- process development
  - hydrogasification (HKV)
  - steam gasification (WKV)
  - reforming/methanation process (NFE).

These topics form a system of program elements that will be worked upon in the next two or three years. In this period a redirection and concentration of the program will be performed according to the perspectives of the market, users engagement and possibilities of technical feasibility.

This strategy aims at a near-commercial introduction of nuclear process heat plants on a profound technical basis and reduced risks.

A brief description of the technical status of the above mentioned program elements will be given in the following.
possible Applications of the High-Temperature Reactor (direct process heat)

V.1 NHSS concept adaptation

V.1.1 Compact reactor system

The design of nuclear process heat (NPH) plants on the basis of THTR-300 and HTR-500 proposed by BBC/HRB is characterized by the following main features:

- prestressed concrete reactor pressure vessel (PCRV) with a single large cavity
- intermediate loops for all gasification processes
- intermediate helium/helium heat exchangers (IHX) in tandem type design
- concrete reactor containment without liner.

Fig. 20 shows a cross section of a 1000 MW(th) NPH-plant with its internals. The reactor pressure vessel is designed as prestressed concrete reactor vessel with one large cavity. A smaller design with 500 MW(th) power rating follows the same design principles.

All primary system components are integrated in the PCRV. The helium/helium heat exchanger cavities are covered by closures designed as prestressed concrete plugs. Prestressing of the PCRV in the vertical direction is effected by individual tendons, the circumferential prestressing by a wire winding system.

As a consequence of the compact design, and in order to reduce the problems in the high-temperature range, the IHX is subdivided into a high-temperature heat exchanger and a low-temperature heat exchanger (tandem design). Thus separate removal of the high-temperature heat exchanger is possible. The separation temperature was selected so that in
the low-temperature section wall temperatures not higher than 700°C will arise. Thus this section remains within the temperature range of conventional high-temperature alloys.

It is intended to use the same basic concept with intermediate heat exchangers for all application purposes such as steam gasification, hydrogasification and steam reforming processes. This leads to the following advantages:

- separate and independent design and construction of nuclear heat supply system and gas generating plant
- conventional testing, construction and operation of the steam reformer-/gasification processes
- no ingress of process gas or water into the primary system
- facilitation for measures to reduce tritium contamination of the product gas
- adaptation to the number of optimized process lines by combining or dividing the secondary helium flow.

For this type of NPH-reactor the IHX represents the main new component to be developed. The other components and systems such as reactor internals, shut-down systems, reactor auxiliary and service systems can be identically adopted from the HTR-500. Thus the development effort can be limited on the high-temperature metal components and the gasification processes as already pointed out. By using the experience and concept simplifications gained in the development of the HTR-500, the design and plant costs can be minimized.
V.1.2 Modular reactor systems (MRS)

The HTR-Module was designed by the KWU-group as a universal high temperature heat source. Its performance and design are, to a great extent, independent of its use for the generation of electrical power and process steam or for the production of process heat and coal gasification respectively. Of course, this does not exclude the fact that process-related requirements, which for example inevitably arise in the case of a plant for the production of process heat, must be taken into consideration when designing the MRS.

It is therefore evident that a reactor core, which is designed for a gas outlet temperature of 950°C, operates at higher fuel element temperatures than an identical reactor, which only requires a gas outlet temperature of 700°C. Nevertheless, when designing a modular HTR core, the fundamental rule is to limit the local, maximum fuel element accident temperature to 1600°C for all conceivable accidents. The core power has therefore been reduced to prevent this maximum permissible fuel element temperature from being exceeded in the event of an increase in the gas outlet temperature. The reactor power is therefore specified as being 200 MW for a outlet temperature of 700°C, while it is reduced to 170 MW for a gas outlet temperature of 950°C.

The pressure in the primary loop is also dependent on the process-related application of the heat source. Although high primary loop pressures have a favourable effect on the operating and accident behaviour of the reactor, it is, nevertheless, desirable to have relatively low pressure in the gasifier, for example in a process heat plant for steam gasification. As the primary, secondary and tertiary pressures should be more or less the same to meet the demand for low component loads, a compromise is made between these opposing requirements, by choosing a primary pressure level of 40-50 bar.

As the material exploitation in the steam reformer and He/He intermediate heat exchanger is already very high at reactor outlet temperatures of 950°C, additional loads resulting from hot or cold gas strands must be excluded wherever possible in the case of these components. Larger radial temperature differences are no longer acceptable here. For this reason, it is necessary to change over to two-zone refuelling of the core, as this permits a more even radial temperature distribution.

According to the respective application, specific boundary conditions result for the arrangement of the components. Thus, due to the demand for routine exchange of catalyst, the blower is located at the bottom of the pressure vessel in the case of a plant for the hydrogasification of coal using a steam reformer.

The different plant behaviour, e.g. in the case of decay heat removal, is largely compensated for by a suitable constructive design of the heat exchanging components.

Fig. 21 shows the primary loop arrangement and the gas duct for the HTR-Module with steam reformer and integrated steam generator, Fig. 22 for the module with He/He intermediate heat exchanger optionally in U-pipe-compact or in helical construction.
FIG. 21
HTR-MODULE WITH STEAM REFORMER AND STEAM GENERATOR

1 Pebble-Bed
2 Wall-Cooler
3 Steam Reformer
4 Steam Generator
5 Blower

FIG. 22
HTR-MODULE WITH HE/HE-INTERMEDIATE HEAT EXCHANGER

1 Pebble-Bed
2 Wall-Cooler
3 Tube-Bundle
4 Blower
For requested plant sizes with net outputs between 100 and 200 MW BBC/HRB have concepts under development on the basis of the proven AVR-Technique. Fig. 23 shows a cross section for such a reactor concept that might be used for process heat applications. The basic reactor system is nearly the same as already described under Chapt. IV.2 for cogeneration purposes.

According to the chosen processes, for example steam reforming and hydro- or steam gasification of coal, such a basic system could be coupled with the adequate heat transfer components on top of the reactor. These components are largely identical to the IHX and steam reformer bundle under development.

In contrary to the module design, this concept permits natural convection in the operational flow direction for the decay heat removal, as far as the reactor system is under pressure. In the hypothetical "loss of coolant and cooling" case the decay heat would also be transferred through the surface of the reactor vessel without surmounting a maximum temperature of about 1600°C in the fuel and thus limiting the radiological impact to the environment to a tolerable level.

With respect to the power and redundancy of power supply needed, this reactor system can also be combined to a multiple unit arrangement such as a twin station for example.
V.2 Status of the component development

By comparison of the steam cycle plant-design and the NPH-concepts it can be seen that the main new components for the NPH-applications to be developed are:

- the intermediate helium/helium-heat exchanger (IHX)
- the steam reformer bundle
- hot gas ducts and
- hot gas valves for the secondary cycle.

Two large-scale test facilities are available for testing the components to be discussed here:

The methane reforming plant EVA II in the Nuclear Research Center Juelich (see V.3.1) and the Component Testing Facility (KVK) at INTERATOM in Bergisch Gladbach.

The KVK is currently operated as a single-loop facility. The necessary heat is introduced into the loop via a natural-gas-fired and an electric helium heater. It is discharged via a steam generator. A part of the steam is used to preheat the helium. This regenerative circuit results in a considerable reduction of the energy consumption.

In the initial test phase, which is underway at present, it is planned to concentrate mainly on tests for the hot gas ducts, the hot gas valves and on the component test of the "hot header" of the He/He heat exchangers.

In order to test the two types of 10 MW heat exchangers, the KVK will be converted to a double-loop facility to simulate the primary and secondary system of a NPH-reactor.

The fabrication of both variants of IHX began in 1982 and has been performed without any problems up to now. They will be delivered in 1985 and 1986 respectively for test purposes.

With reference to the heat exchangers, the "hot header" is the most critical component with regard to its loads. In contrast to the two complete heat exchangers, which, as modular systems, need only to be subjected to a functional test, it appears necessary to subject the hot header on an original scale to a simulated life test including extreme loads due to accidents. On the basis of the available experience with large scale plants, it was concluded that an operating time of 3000 hours is adequate for a heat exchanger test. The test object for the "hot header" was delivered in November 1983 (Fig. 24) and has been installed in the KVK. The tests in the KVK were started in February 1984 and are still underway.

Production of the 5 MW test component of the steam reformer is on schedule, delivery is planned for the end of 1984. The planning of the instrumentation for the test component was coordinated with the Nuclear District Heating (NFE) project. Work on the major problem, the spacers for the reforming tubes, has led to a new solution: each reforming tube will be surrounded by a cladding tube as flow guidance and mechanical holding device.

For the primary hot gas ducts there are two variants under discussion:

- a graphite-or carbon fibre reinforced carbon (CFC) -
  tube with fibre insulation and
- a coverplate concept also using CFC as basis material together with fibre insulation.

A graphite tube as a liner has been tested in the high pressure channel of KFA Juelich to determine the effective coefficients of thermal conductivity and was subsequently installed in the ADT-loop of HRB for long-term tests at operating temperature. These tests started recently.
Two constructions with metallic liner, both having the same basic concept, are available for the secondary loop. The tests on the behaviour of the hot gas duct in a horizontal position were completed after 2900 operating hours in KVK. The results for the temperature distribution and heat losses are very positive. The test in a vertical position commenced in September 1983. At present work is centred on the constructive and analytical examination of the design of further subcomponents of the primary and secondary hot gas ducts, such as bends, compensators and T-pieces.

Initially two variants were investigated for the hot gas valve, a ball valve and an axial valve. But, since the end of 1981, work has only been concentrated on the latter. As far as the constructive design is concerned, the test object has been prepared for the component test. The complete manufacturing documents have been submitted and the preliminary examination has been completed. Manufacture of the sub-component has almost been finished. It will be delivered and installed in May of this year.

A scaled-down version of the axial valve, which has been adapted to suit the KVK, is part of the operating equipment of the KVK. Testing of this valve has already supplied valuable results.

The above mentioned component development is accompanied by a comprehensive material program for the development and qualification of metallic and ceramic high temperature materials. An additional program is underway to establish HTR-specific design codes as licensing base for future HTR-plants.
Nuclear long distance energy (Kernforschungsanlage Juelich/Rheinische Braunkohlenwerke)

The R&D-work on the special aspects of the EVA/ADAM system is summarized in the project "NUKLEARE FERNENERGIE" (NFE), where the most important stages of the two energy conversion processes are theoretically analyzed and experimentally investigated.

A decisive experimental step before building a first nuclear plant is the 10 MW test and demonstration facility on a technical scale called EVA II/ADAM II pilot plant.

EVA II is the designation for a bundle of up to 30 reforming tubes heated by helium of 40 bars and 950 °C. The power is transferred to the helium by an electrical heater. In this pilot plant, the behaviour of a complete reforming tube bundle and some components in connection with it are being tested in a closed helium circuit under various conditions.

In the methanator ADAM II the synthesis gas produced in the steam reformer is converted to methane again and the heat transported by the thermochemical cycle is released at 650°C.

The complete facility EVA II/ADAM II has been constructed by Lurgi Kohle and Mineralöltechnik GmbH, Frankfurt.

In the following the facility EVA II is described. The helium circuit represents a complete primary loop of an HTR for process heat application. The core is simulated by an electrical heater with a maximum power input of 10 MW.

The helium is heated in the electrical heater up to 950°C. After passing the hot gas duct the heat is transferred successively primarily to the steam reformer tubes where the helium is cooled down to 650°C and then to the steam generator for process steam production where the helium is cooled down to 350°C. An integrated circulator transports the helium back to the electrical heater by following a coaxial flow principle.

As process gas a mixture of methane and steam enters the steam reformer. By means of the endothermic conversion to synthesis gas it absorbs the heat transferred from the helium.

The steam reformer bundle, tested in EVA II consists of 30 tubes. The tube dimensions correspond to those in the conventional technique with a length of 11 m and an internal diameter of 100 mm.

Further characteristics of the steam reformer bundle tested up to now are:
- test of 4 different alloys for the reformer tubes
- Raschig-ring catalyst as conventionally applied
- reformer tubes with internal return pipes
- baffles (disc-and-doughnuts) to intensify the heat transfer.

The steam generator is an induced-single-circulation boiler. Electrical heater, steam reformer and steam generator are linked together by coaxial hot gas ducts. The thermal insulation separating the hot and cold helium flows is made of carbon bricks. On both sides the insulation is covered by a metallic liner. Just the colder outside liner is gastight.
The tasks for the plant EVA II/ADAM II are as follows:

- tests of the heat transport by a thermo-chemical cycle
- tests of a complete helium loop equivalent to that of a nuclear process heat plant
- tests of steam reformer bundles in a representative size and design
- functional tests of a steam generator under HTR conditions
- investigation of operational behaviour on normal operation, partial load and break-down conditions
- description of the operating characteristics by mathematical models.

The main tasks of the investigations in EVA II do not aim primarily at the life time of the components but at operation characteristics and behaviour in a wide range of different process parameters. The main parameters have been changed between the following limits:

- electrical power input: \(3.8 - 11\) MW
- helium mass flow rate: \(1 - 4\) kg/s
- helium pressure: \(15 - 40\) bar
- helium temperature: \(800 - 950\) °C
- methane mass flow rate: \(0.18 - 0.66\) kg/s.

All test runs of the facility EVA II in connexion with ADAM II could be performed successfully. In the course of the test program the following tasks have been worked out additionally:

- change of catalyst by vacuum extraction
- replacement of a single reforming tube without removal of the whole bundle
- disassembling and reinstallation of the steam reforming bundle.

In total the helium system has been operated for 7,000 hours. The complete EVA/ADAM heat transport cycle reached 5,660 hours. During this time \(6.8 \times 10^4\) GJ heat have been released from ADAM II at 650°C. This heat was used for the most part for district heating purposes.

V.3.2 HKV

Hydrogasification of lignite
(Rheinische Braunkohlenwerke)

In the Federal Republic of Germany approx. 11 billion tonnes of coal equivalent (tce) brown coal can be mined economically. These minable brown coal reserves, comparable with Iran's oil reserves, form a mainstay of Germany's present and future energy supply. Yielding an annual output of approx. 120 million tonnes this coal is mined by the Rheinische Braunkohlenwerke AG (Rheinbraun) in opencast mine operations. 85% of this output is used for power generation in base-load power plants meeting about 25% of West Germany's power requirements. Since it can be expected that today's major fossil energy sources, viz. oil and natural gas, will run short and become even more expensive in the long run, Rheinbraun is developing processes with the aim of opening up markets to brown coal in addition to power generation where presently oil and natural gas are the main suppliers.

In terms of long-term market potentials the production of SNG promises particular advantages. Therefore, Rheinbraun is developing coal hydrogasification using a pressurized fluidized-bed process. This development is done within the framework of the project referred to briefly as "Prototyp-anlage Nuklear Prozeßwärme" (PNP) and receives substantial
financial support from the Federal Ministry of Research and Technology and the State of North-Rhine-Westfalia.

The process is based on the conversion of coal with hydrogen into raw gas which has a high content of methane, i.e. the main component of natural gas. Since this reaction is an exothermal one it proves to be most favourable to use a fluidized-bed gasifier ensuring a reliable temperature control. The hydrogen required for this process can be provided in two ways:

One way is the conversion of residual char obtained during coal hydrogasification with oxygen and steam, e.g. using the High-Temperature Winkler process (HTW) which the Rheinische Braunkohlenwerke has further developed on the basis of the pressureless Winkler process.

Another way is the partial conversion of methane produced during hydrogasification with steam in a steam reformer using a nickel-containing catalyst. The heat required for this endothermal reaction is provided by a high temperature nuclear reactor (HTR). The residual char from hydrogasification occurs as a by-product which is particularly suitable as a low-sulphur fuel for use in power plants and in the industrial heat market.

From 1976 to September 1982 a semi-technical test plant for hydrogasification of coal was operated for approx. 27,000 hours with more than 12,000 hours under gasification conditions. During that time approx. 1,800 tonnes of dry brown coal were processed. The longest continuous operating period under gasification conditions was 780 hours during which approx. 85 tonnes of dry brown coal were gasified. Carbon conversion rates of up to 82 % and methane contents of up to 48 % vol in the dry raw gas (free of \(\text{N}_2\)) were obtained.

In the course of the scale-up to commercial coal gasification plants a pilot plant was erected as the last intermediate step prior to industrial scale-up of the process. Compared with the semi-technical test plant this plant has a 25 fold scale-up gasification unit and comprises additional plant components, e.g. the total raw gas treatment required in the process for the production of SNG meeting the market requirements.

The test data obtained from the semi-technical plant made it possible to begin the basic engineering and licensing

FIG. 25
procedures for the pilot plant on the 01.07.1977. Detail engineering was started on the 01.08.1978.

The first contracts for the pilot plant were awarded in mid-1979. Construction began in September 1979 on the site of the Union Rheinische Braunkohlen Kraftstoff AG in Wesseling. The construction phase was completed in 1982.

During the second half of 1982, performance tests were carried out on completed parts of the plant, parallel to the construction work; the complete plant was commissioned in May 1983.

Fig. 25 shows a view and Fig. 26 a simplified flow scheme of the HKV pilot plant, which at present consists of the following units: coal preparation, gasification, amisol scrubbing and cryogenic gas separation. A test plant for waste disposal is under construction and will be tested as the 5th unit in the autumn of 1984.

The pilot plant is designed for a maximum operating pressure of 120 bar; at this pressure, the maximum coal feed is approx. 10 tonnes per hour of dry brown coal with a residual moisture content of 2%; at this rate of coal feed, a maximum methane production of approx. 8,000 m$^3$ per hour is expected.

The test program includes 18 operating phases with a total of 52 test settings. An average test period of 3 days was taken for each setting. During this period, stationary operating conditions were to be set to produce useful test results.

The effect of the following test parameters is under investigation in this program:

- coal throughput from 4 to 10 t/h (wt)
- gasification pressure from 65 to 120 bar
- gasification temperature from 850 to 950°C
- fluidized bed height from 400 to 600 cm
- ash content of the feed coal from approx. 4 to 15% by wt.

The tests are intended to determine the effects of the considerable increase in the scale-up of the gasifier compared to the semi-technical test plant. A further step is to optimise the process, including use of the
greater flexibility of the gasifier, with regard to the fluidized bed height, coal throughput and gasifier pressure. In addition, test of auxiliary components, in particular raw gas treatment as well as gas separation and its connection with the gasifier, is an important development target for the entire coal hydrogasification process.

From May 1983 up to March 1984, 8 test runs were carried out. During gasification with brown coal, the gas treatment units were always in operation.

The most successful continuous test-run to date was performed between the end of January and the end of February 1984. In 756 hours, at a gasifier pressure of 80 and 95 bar, a throughput of almost 4,000 tonnes of dry brown coal was achieved and approx. 1.8 million m$^3$ (STP) of CH$_4$ produced. Problems with the coal feeding systems were solved by specific measures.

Despite a scaling-up factor of 25 compared with the semi-technical plant, the pilot plant achieved a very high level of availability during the first year of operation.

A first comparison of the pilot plant results with corresponding data of the semi-technical plant was quite satisfactory.

V.3.3 WKV

Steam gasification of hard coal
(Bergbau-Forschung)

Within the cooperation of Bergbau-Forschung GmbH (BF), Rheinische Braunkohlenwerke AG, and German companies of HTR research and industry, BF develops the process for steam gasification of coal where nuclear heat from the HTR is to be used directly as process heat. In view of the almost double specific gas yield compared with conventional gasification this approach promises three major advantages:

- saving of available coal reserves,
- reduction of coal specific emissions,
- lower gas production costs compared with those of conventional gasification.

The flowsheet on Fig. 27 shows the HTR connected to the gasification plant. Helium is used as heat carrier in the primary HTR circuit and heated up to 950°C. The heat is transferred via a heat exchanger to a secondary circuit which also uses helium as operating medium. The heat from the secondary circuit helium will then be coupled into the gas generator as well as into steam generation and superheating. The gasifier was designed as a horizontal, elongated fluidized bed reactor into which heat is transferred via metallic heat exchanger tubes immersed into the bed. The pressure in the helium loops and in the gasifier atmosphere is the same, around 40 bars. The coal to be gasified is fed in at one end, moves through the vessel as gasification with steam takes place, and at the other end the residue with its high ash content is drawn off.
The key problems to be solved for the new gas generator type include materials and design questions, and working out of relevant operating data by laboratory and pilot plant tests.

1. HTR
2. Heat exchanger
3. Steam generator
4. Gas generator
5. Gas cleaning

Technical scale experiments started with the semi-technical test plant which was commissioned in 1976 (Fig. 28). The gas generator is a cut-out version of the full-scale generator. The fluidized bed itself has a cross-section of close to 1 m² and can be operated up to height of 4 m. The helium in this plant is heated electrically. The plant has run about 24,500 hours in hot operation including 12,600 hours of gasification with daily reaction rates of up to 6.4 t/day of carbon.

Due to the fact in the long term in Germany only baking types of coal are available for gasification, a technical solution had to be found to the problem of feeding such coal. Therefore the method of jet-feeding was advanced to the state where agglomeration of the feed coal can be prevented. Finely ground caking coal is injected pneumatically into the bottom of the fluidized bed. Here, it is quickly mixed with the partly gasified char particles whereby agglomeration is suppressed. The liberated volatiles - especially the tar - are completely transformed on the way upward through the fluidized bed so that no tar has been found in the raw gas.

On the basis of laboratory tests a number of runs has been performed in the semi-technical plant with potash and other catalytic substances to accelerate the rate of reaction. For this the catylist has been fed along with the coal in a dry mixture into the gasifier by means of the injection feeder. A doping with 2 or 3 % K₂CO₃ to coal has a relatively modest effect. However, if 4 % potash is added a steep increase of reaction rate by a factor of about 20 can be achieved.

The technical feasibility of the the nuclear gasification concept is dependent on the availability of metallic material for the heat exchanger tubes in the gasifier.
VI. Fuel cycle activities

Mixed thorium/uranium oxide of 93 % uranium enrichment had been the reference for the pebble-bed type High Temperature Gas-Cooled Reactors (HTR) in the Federal Republic of Germany until 1979 and the 15 MW(e) experimental power plant AVR uses, and the 300 MW(e) prototype HTR plant THTR, will use this type of fuel accordingly.

In 1979, however, non-proliferation aspects, difficulties envisaged with the longterm supply of high enriched uranium plus the aspects of utilizing the existing PUREX technology for the recovery of uranium and plutonium from spent HTR fuel led to a change to Low Enriched Uranium (LEU) fuel. Follow-on reactors will be fueled with uranium oxide with initial enrichment of around 10 %.

The activities concentrate on the development of processes and equipment for LEU fuel manufacture and on the qualification of UO₂ fuel. Back-End fuel cycle activities comprise work on interim storage, reprocessing and terminal storage of discharged HTR fuel. The limited amount of spent fuel from the THTR, plus that from the AVR plant (about 3.5 million pebbles by 2005), will be put to terminal storage.

For LEU fuel of follow-on plants both options are kept open, i.e. reprocessing or terminal storage. R&D work follows this strategy with programs on both lines in parallel.

R&D work on fuel fabrication and performance

Fuel fabrication

The LEU reference particle consists of a 500 μm diameter UO₂ kernel with TRISO coating containing a 35 μm thick silicon carbide interlayer. The key problem in the fuel development is optimum retention of gaseous and solid fission products. This requires low defect rates during the manufacture of particles and spherical fuel elements. Two important defect mechanisms have been identified:

- particles having extremely odd shapes cannot withstand the pressure applied during moulding even with optimum overcoating
- adjacent particles experience high local stresses during moulding when insufficient overcoating is applied.

To achieve the target defect fraction several classification steps for kernels, coated particles and overcoated particles are used. With these techniques the fabrication particle defect fraction has been reduced below the design value of 60 x 10⁻⁶.

Progress has also been made in fuel process development; new components were designed for continuous fabrication, and interfaces between different process steps were improved. Further work concentrates on the final specification of all process steps, increase of coating batch sizes, recycling of reagents, and waste treatment.

Fuel irradiation testing

The first set of irradiation tests with low-enriched UO₂ TRISO particles has been completed. The release of short-lived fission gases remained at low enough levels to demonstrate that no particle had failed in all tests comprising more than 10⁵ particles.

The goal of the development is to show that irradiation induced particle failure fraction remains below 2 x 10⁻⁴ during normal operation and that the coatings remain retentive of metallic fission products.

The evaluation of all irradiation and postirradiation data from the previous set of experiments with high-enriched (Th,U)O₂ fuels has been completed.
Here, the necessary data have been established for prediction of particle failure and fission product release.

To supply fuel specific data for safety studies, new furnaces have been built which simulate core heat-up events. In one furnace irradiated fuel is heated with the temperature increasing up to 2500°C. The on-line measurement of Kr 85 indicates the extent of coating failure, which for loose TRISO particles increases from 1% at 2000°C to 80% at 2500°C. In another type of furnace, the fuel is annealed at constant temperature, e.g. at 1600°C and 1800°C, for prolonged periods.

R&D work on spent fuel treatment

During 15 years of operation with the Experimental Power Plant (AVR) much experience has been accumulated on spent fuel handling. More than 125,000 fuel elements, mostly of the HEU-mixed oxide and mixed carbide fuel type, in a wide range of burn-up states, have been discharged from the reactor into small stainless steel canisters with a capacity of 50 fuel elements each. PIE of fuel elements from random & specific samples together with their operational history have yielded an invaluable data base on spent fuel characteristics to which the results from fuel irradiation tests may be added.

Interim storage of spent fuel elements

The small canisters with fuel elements discharged from the AVR have been received and stacked in a pool facility at KFA since 1973. Canister seals are replaced at 4 year intervals. The maximum capacity of this storage facility is 65,000 fuel elements, but was increased by adding space for 28,500 fuel elements in another pool facility at KFA. Since then all past or present planning for interim storage of spent HTR fuel elements concentrates on dry facilities with air cooling systems to take into account the relatively small decay heat production per volume unit.

Operation of the AVR required an increase in storage capacity for spent fuel. Therefore at KFA, a first dry storage facility for 72,000 fuel elements of 2 years minimum cooling time was built and has been operating since the end of 1981. Utilizing an existing concrete shielded cell it houses the fuel in stainless steel canisters of 1000 element maximum capacity in a close-packed array by means of a metal rack comprising vertical tubes with a surrounding frame. A monitoring program shows that H3 and Kr 85 release from this facility which is now loaded up to 80% is still negligible (H3 ≤ 1 mCi/d; Kr 85 ≤ 4 mCi/d, the effective detection limits).

Planning for further capacity increase, as required from 1986 onwards, has started and is based on the transport/storage cask concept. An appropriate 2 yr demonstration and measuring program on two prototype casks loaded with 1950 spent AVR elements each contained in two canisters started in 1982. Results have verified the design with regard to radiation, activity containment and heat dissipation.

For the interim storage of spent THTR fuel the transport/storage cask technique was adopted, too, with one cask housing one of the existing and licensed THTR spent fuel canisters.

Reprocessing

The reprocessing option for spent LEU fuel includes preparation of heavy metal ash by burning off the graphite in oxygen, dissolution and solvent extraction according to
the PUREX-processing scheme followed by the standard LWR refabrication process. Because of know-how available from LWR for the latter steps the effort has concentrated on the head-end consisting of fuel element crushing followed by single step fluidized-bed burning.

Cold pilot scale tests with both (Th,U)O₂ and UO₂ fuel have been performed at KFA and GA Technologies in the modified head-end of the JUPITER facility and the GA pilot plant. They confirmed that the fluidized-bed burning process can treat various HTR fuel types and exhibits a large flexibility.

The JUPITER tests with (Th,U)O₂ BISO and TRISO-coated fuel have confirmed the trouble free performance of the upgraded facility for this fuel. In a 110 hours demonstration run with 3000 (Th,U)O₂-BISO-fuel spheres a burning rate up to 6.2 kg graphite (30 fuel elements per hour) has been reached, thus exceeding the original design value by a factor of 2. The product contained less than 0.1 % carbon and no fines accumulation has been observed.

A demonstration run with (Th,U)O₂-TRISO-fuel spheres has shown equally good facility performance. 2600 fuel elements were processed in 60 hours with a burn rate exceeding 8 kg carbon per hour. The product contained less than 0.1 % C and less than 0.15 % SiC.

Also UO₂ TRISO fuel processing in demonstration runs with a total of 376 kg carbon burned (i.e. nearly 2000 fuel elements equivalent) brought good results. The burn-rate has been further increased to 9.4 kg carbon per hour. The product discharged during operation contained less than 1 % carbon.

The experiment in the GA pilot plant on the 20 cm burner comprising about 500 kg of FRG UO₂ TRISO fuel has simultaneously confirmed the suitability of the one-step burning process. The results show that at burn rates up to 7 kg carbon per hour steady state operation is possible in both carbon lean and carbon rich mode. The product contained practically no carbon.

Supporting data for "hot" tests on a pilot scale are going to be provided by hot laboratory tests with about 250 irradiated spheres of the AVR-6 reload (BISO coated LEU fuel) in KFA laboratory equipment. Laboratory reprocessing of irradiated Ft. St. Vrain block type fuel will follow about a year later. Prior to this exercise some data will come from burn-back experiments with DRAGON fuel at UKAEA Winfrith.

Final Storage

The reference concept for spent fuel treatment for AVR and THTR is final storage in the national repository. The present reference concept for THTR fuel is based on final storage containers developed for LWR fuel but, for economic reasons, other techniques are being analyzed and will be further developed. One of the more promising concepts is based on the present reference design for heat generating waste from reprocessing: fuel element hulls and slurry from the feed solution step. This technique comprises canister storage in sealed boreholes of 300 m depth. With components for this concept being designed and tested a field demonstration test is under preparation and this will involve a limited number of spent AVR fuel discharged in 1981. The test will be carried out in the ASSE salt-mine and last about 5 years from the beginning of 1987.
The necessary database for licensing any of the final storage concepts will involve specific information of long-term fuel characteristics and behaviour. Results from experiments on corrosion effects and fission product and ω-emitter release from the fuel under normal storage conditions and in presence of brine solutions are required. Orientation tests in a relevant q-solution started in 1981/82 on a total of 38 high burnup AVR fuel elements, 24 of which were deliberately crushed. Cesium and strontium (BISO carbide fuel) represented the main activity in the brine. Fractions of up to $3 \times 10^{-5}$ of the total Cs inventory were found in q-solution after 2 years at 40°C or 90°C on intact, $4 \times 10^{-4}$ on crushed fuel elements.

An expanded second program phase has started on fuel elements more fully characterized for fission product distribution and is rendering more systematic results on a variety of fuels including LEU elements with TRISO coated fuel. This program also looks for corrosion effects and the release of ω-emitters into the brine. In the tests finished so far no corrosion effects could be detected.

Fuel cycle strategy

For the spent HEU-thorium fuel from AVR and THTR final disposal is definitely planned.

Planning for the future (Fig. 29) involves: THTR follow-on plant(s) will be designed for LEU fuel. The spent fuel will be put to interim storage with the decision pending to either

- allocate the fuel elements for final disposal
- or
- keep the elements for reprocessing. Reprocessed uranium and plutonium will be fed into LWR or FBR.

Later HTR plants will be designed for uranium fuel but would also allow uranium/thorium fuel to be used. Fuel cycle planning will be the same as for the THTR follow-on plants.
STATUS OF RESEARCH AND DEVELOPMENT ON
VERY HIGH TEMPERATURE GAS-COOLED REACTOR IN
JAPAN

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Japan Atomic Energy Commission (JAEC) submitted its revised
"Long-term plan for the development and utilization of nuclear energy"
in June, 1982. The new plan specifies that an experimental Very High
Temperature Gas-cooled Reactor (VHTR) will be constructed as the first
step of the VHTR development, aiming at a start of its operation around
1990. The plan also proposes that decreasing the design temperature at
the reactor outlet to around 950°C may be realistic with regard to the
present state of art, such as the status of material technology and
needs for earlier construction of the experimental reactor.

Japan Atomic Energy Research Institute (JAERI) is now carrying out
various kinds of research and development including the design of the
experimental reactor under the design conditions suggested in the
revised long-term plan, with the collaboration of Japanese industries.

A summary of the progress made in research and development is
given.

The detailed design of the experimental reactor consists of three
stages. The first stage of design was finished in March 1981, in which
main efforts were directed to settling the overall system concept,
designing detailed structure of main components in the cooling circuit
and determining the reactor performance for the irradiation use. The
second stage was completed in March 1984, in which the total system
design, structural design of components and safety analysis were
performed. Basic specification of the experimental reactor are shown in
the Table. The specifications were established through an extensive
study to optimize the various design parameters of the reactor.

Specifications of Experimental Reactor

<table>
<thead>
<tr>
<th>Characteristics</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal Outlet (MW)</td>
<td>50</td>
</tr>
<tr>
<td>Outlet Coolant Gas Temperature (°C)</td>
<td>950</td>
</tr>
<tr>
<td>Core Effective Height (m)</td>
<td>5.6</td>
</tr>
<tr>
<td>Core Equivalent Diameter (m)</td>
<td>2.75</td>
</tr>
<tr>
<td>Number of Fuel Columns</td>
<td>54</td>
</tr>
<tr>
<td>Number of Control Rod Columns</td>
<td>7</td>
</tr>
<tr>
<td>Total Number of Columns</td>
<td>61</td>
</tr>
<tr>
<td>Number of Fuel Elements per Column</td>
<td>8</td>
</tr>
<tr>
<td>Total Inventory of Uranium (ton)</td>
<td>2.5</td>
</tr>
<tr>
<td>Inside Diameter of Reactor Vessel (m)</td>
<td>6.2</td>
</tr>
</tbody>
</table>

The main activities of fuel research are in-pile loop and capsule
irradiations, out-of-pile experiments on fission product release and
transport studies related to accidents and development of improved
silicon carbide and zirconium carbide coatings. The in-pile gas loop
UOG-1 is the only facility that can be used for irradiation of full size
fuel rods. One fuel assembly has been irradiated every year.
Irradiation of the seventh fuel assembly containing fuel compacts made
of two different matrix graphites was finished in June 1983.
The experiments on graphite materials properties are continued. The effect of neutron irradiation on the corrosion of IG-11 graphite by water vapor in helium is obtained. The corrosion rate of the graphite decreased slightly by neutron irradiation. In addition to the studies on graphite materials, screening test on carbon materials was carried out to select favorable thermal barrier materials and further studies for improvement of the candidate carbon materials are in progress.

Three different categories of heat resisting alloys for structural applications have been handled. The outline of the status is summarized in Table.

<table>
<thead>
<tr>
<th>Application</th>
<th>Material</th>
<th>Program Status</th>
</tr>
</thead>
<tbody>
<tr>
<td>First Generation</td>
<td>Hastelloy XR (Nuclear Grade)</td>
<td>Tests for Design Data and Q.A. Oriented Studies</td>
</tr>
<tr>
<td>Structural Components</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Second Generation</td>
<td>Ni-Cr-W (Developmental)</td>
<td>Alloy Optimization Design and Screening Tests</td>
</tr>
<tr>
<td>Structural Components</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Neutron Absorber</td>
<td>Incoloy 800 (Developmental)</td>
<td>Screening Tests and P.I. Creep Behaviors</td>
</tr>
<tr>
<td>Sheathing</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Long-time performance tests of the first generation alloy (Hastelloy XR) have accumulated the experimental endurance data base of over $1 \times 10^4$ hours in corrosion, carburization, creep, stress-rupture and thermal aging characteristics. The results of the standard strain-controlled fatigue tests in simulated reactor environment have indicated that the performance at 900°C under strain rate of 0.1%/s is reasonably high relative to the literature data on several other common heat resisting alloys including commercial Hastelloy X. Among the minor elements existing either as impurities or additives in Hastelloy XR, boron (B) was found to be most significant in influencing the creep performance of Hastelloy XR. For the second generation superalloy development, five experimental Ni-Cr-W alloys with different Cr/W ratio were examined for exploring the optimum ratio for balanced creep strength and corrosion resistance. Post-irradiation tensile and creep tests of alloys for neutron absorber sheathing have confirmed the better performance of Fe-base relative to Ni rich alloys.

Experiments of pressure retaining low alloy steel have been conducted in order to acquire the material properties data required for the evaluation of the integrity of the pressure boundary components of the experimental reactor. These studies include evaluation of embrittlements due to neutron irradiation, thermal and stress agings, which are mainly assessed by the Charpy impact and fracture toughness test. Low cycle fatigue test with a certain hold time and cyclic crack growth test are also carried out.

Reconstruction program of SHE, Semi-Homogeneous Experimental Critical Facility, was approved by Japanese government early in 1981. The object of the program is to obtain experimental verification for the design accuracy of the experimental reactor fueled with low enriched uranium. The detailed design and its construction method were licensed by the government in May 1983. The facility is now under construction and will be made critical in April 1985.
In the field of the nuclear instrumentation for the reactor control, high-temperature fission counter-chambers have been developed. These chambers withstood accelerated irradiation tests at 600°C, a long term in-reactor operating test at 600°C for 1000 days, and over-heat tests at 800°C for about 500 hours in a simulated accident condition. Also, the gamma-compensated ionization chambers, CICs, have been tested at the temperature between 400°C and 500°C for more than two years in the research reactor JRR-4. As for the in-core temperature monitoring system, platinum-molybdenum thermocouples have been fabricated in trial for use in the temperature range between 1000°C and 1350°C.

The dynamics simulator of the VHTR with associated control system has been developed by using a hybrid computer on real-time or faster than real-time basis for dynamics analysis and control system synthesis. A heat transfer experiment on a simulated fuel rod has been continued using helium gas as coolant. A study related to the safety of VHTR was also carried out. Coolant flow reversal due to loss of forced circulation was studied experimentally and analytically. A correlation for the reversed flow rate was obtained and was found to agree well with the experiment.

Coolant flow tests have been continued to determine the flow characteristics in the experimental reactor. Crossflow tests have been performed with two-block model and with one-column model. And also with a 1/2.75-scale bottom-core model, the leak flow rate into the hot plenum through the gaps between plenum blocks and/or fixed reflector blocks, has been measured.

Helium Engineering Demonstration Loop (HENDEL) is designed as a large scale model testing facility for the high temperature components of the experimental reactor. The loop consists of Mother, Adapter and Test sections. Mother section circulates purified helium gas of the specified flow rate (0.4 and 4.0 kg/s) and pressure (4.0 MPa). Besides the helium gas purification, helium storage and cooling water system, it has helium circulators, heaters, coolers, mixing tank and gas filter. Adapter section heats or cools helium gas up to 1000 or 400°C. It consists of high temperature heaters and coolers. The Mother and Adapter are designed to supply helium gas to the test sections under the simulated VHTR operation conditions of pressure, temperature, flow rate, etc. The fabrication of the fuel stack test section was completed in March 1983 and its test operation has been continued. The in-core structure test section is now under construction.

Thermochemical processes have been studied to produce hydrogen from water by using the heat to be supplied from a VHTR. A process involving nickel, iodine and sulfur, as well as another process involving methanol, iodine and sulfur has been investigated. The latter, the methanol process, is thermally efficient and expressed by reactions below.

\[
\begin{align*}
\text{SO}_2 + x\text{I}_2 + 2\text{H}_2\text{O} & = 2\text{HI} + \text{H}_2\text{SO}_4 \\
\text{CH}_3\text{OH} + \text{HI} & = \text{CH}_4 + \text{H}_2\text{O} + 0.5(x-1)\text{I}_2 \\
\text{CH}_3\text{I} + \text{HI} & = \text{CH}_4 + 0.5(x+1)\text{I}_2 \\
\text{CH}_4 + \text{H}_2\text{O} & = \text{CO} + 3\text{H}_2 \\
\text{CO} + 2\text{H}_2 & = \text{CH}_3\text{OH} \\
\text{H}_2\text{SO}_4 & = \text{H}_2\text{O} + \text{SO}_2 + 0.5\text{O}_2 \\
\text{H}_2\text{O} & = \text{H}_2 + \frac{1}{2}\text{O}_2
\end{align*}
\]

65
According to the long-term plan, JAERI projected the schedule for VHTR development as seen in the figure. Before the construction of the experimental VHTR, the design modification for cost reduction and the licensing procedure are planned. In connection with those activities, the related R & D, the HENDEL program and the nuclear process heat application studies will be continued.

### Schedule of Exp. VHTR Development

<table>
<thead>
<tr>
<th>Item</th>
<th>Year</th>
<th>81</th>
<th>82</th>
<th>83</th>
<th>84</th>
<th>85</th>
<th>86</th>
</tr>
</thead>
<tbody>
<tr>
<td>Experimental VHTR (50MWt)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Operation</td>
<td>Design &amp; Construction</td>
<td>Test</td>
<td>Licensing</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Related R&amp;D</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>OGL-1 (In-Pile Gas Loop)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>SHE (Critical Assembly)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>HENDEL (Test Loop)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Nuclear Process Heat Application System</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Design &amp; Construction of Experimental Plant</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

**Note:**
- V: End of Construction Work
- T2: In-Core Structure Test Section
- M+A: Mother & Adapter Loop Section
- T3: Core Flow Test Section
- Ti: Fuel Stack Test Section
- T4: Heat Removal Test Section

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**THE SWEDISH SITUATION WITH VIEWS ON THE 5th IWGGCR**

Studsvik Energiteknik AB,
Nykoeping, Sweden

Due to the outcome of a referendum on nuclear power Sweden participates with only a manyear/a effort on gas-cooled reactors. This follow-up activity was not considered significant enough to justify a separate progress report, but the following material is submitted on the request of the meeting in order to have the Swedish presentation added to the minutes:

The referendum in 1980 is still being interpreted so that Sweden should not have any other reactors beyond the 10 now in operation and the two ready for fueling.

This has reduced our efforts on GCRs to one manyear as part of a general follow-up of the global nuclear power development. (The fusion program is somehow exempted from this and consequently there are 15 Swedes at JET and a $3M/a fusion budget in Sweden).

As for available facilities we still have the in-pile gas-loop in R2 and the facilities for PIE, but the PCRV experiments have now been "knocked" down.

(I am however expecting a growing nuclear energy debate as many argue that a referendum is not binding as politicians must be able to change their minds as new facts surface and the public can always react to these changes in free elections every three years (the next time in Sep of 1985). The conservatives and the liberals are going in this direction and the Center Party does not like coal, nor new hydro.)

Table 1 shows the energy production prospects for Sweden in a global perspective.

---

*Presented by R. Ekholm.*
As Sweden is consuming more than its share of the finite fossil fuel resources and since the hydro is practically already tapped and the new non-nuclear sources not developed or too expensive one may conclude that the nuclear power sector has still to be expanded several times in capacity even in Sweden. Strategic reasons support this as does a synfuels production capability and capacity. It is therefore in my personal view alarming that when the uranium mining company Ranstads Skiffer AB ceased recently to exist and the land harbouring 0.3 Tg of U (25 000 EJ worth of or the equivalent to the global energy production up to around 2020) is for sale, this hardly made the headlines in the Swedish press. The Ranstad shale contains also Kerogene and an operation of 6 Tgore/a (6 Million tonnes of shale/yr) could have produced 10% of the nations need for gasoline for the automobiles.

The major part of the mentioned many years/a has comprised my survey study of the prospects for using nuclear energy for energy carrier production (elsewhere and in Sweden). I am aiming for getting the report distributed to you in Oct. (To host a specialist mtg in Sweden would have to be done on that same budget.)

There are some results/conclusions in that report that I think are useful to bear in mind when considering GCRs. Table 2.2.1 in it shows that the very modest global development to 1000EJ/a in 2020 would lead to the exhaustion not only of the fossil but also of the assured uranium fuel resources unless significantly improved nuclear reactors with regards to their fuel utilization are quickly introduced on a large scale.

While all conceivable or available reactor production capacity needs to be employed, it is unfortunate that the thermal CO-
cooled reactors like the AGRs are not helpful in assuring a full utilization of the U. The He-cooled are however and they have clear advantages also over the HTRs. Their dev. status, versatility of applications (incl plant sizes) give confidence in the feasibility of a rapid and significant market penetration provided that they are backed by a positive political will.

The table shows also that in the long run not even the what I have called "admirable production targets" would suffice to deal with the global demand for energy. Obviously one needs to save and extend the fossil fuel resources and here again the HTRs could be a major instrument.

Table 1

<table>
<thead>
<tr>
<th>SOURCE</th>
<th>GLOBALLY</th>
<th>SWEDEN</th>
<th>REMARKS</th>
<th>SWEDEN 1983</th>
</tr>
</thead>
<tbody>
<tr>
<td>GAS</td>
<td>120</td>
<td>0.24</td>
<td>Not synfuels. Sweden's share assumed as 10M Swedes/5G World population = 0.002</td>
<td>0.1</td>
</tr>
<tr>
<td>OIL</td>
<td>270</td>
<td>0.54</td>
<td></td>
<td>1.0</td>
</tr>
<tr>
<td>COAL</td>
<td>90</td>
<td>0.18</td>
<td></td>
<td>0.1</td>
</tr>
<tr>
<td>SUM</td>
<td>480</td>
<td>1.0</td>
<td>Finite resources</td>
<td>21.1</td>
</tr>
<tr>
<td>HYDRO</td>
<td>100</td>
<td>0.3</td>
<td></td>
<td>0.2</td>
</tr>
<tr>
<td>SOLAR COLLECTORS</td>
<td>70</td>
<td>0.1</td>
<td></td>
<td>0.0</td>
</tr>
<tr>
<td>WIND</td>
<td>100</td>
<td>0.2</td>
<td></td>
<td>0.0</td>
</tr>
<tr>
<td>BIOLOGAS</td>
<td>120</td>
<td>0.5</td>
<td></td>
<td>0.2</td>
</tr>
<tr>
<td>GEOTHERMAL</td>
<td>160</td>
<td>0.3</td>
<td></td>
<td>0.0</td>
</tr>
<tr>
<td>FEAT</td>
<td>0.2</td>
<td></td>
<td>If 10% of Swedish resources are consumed over 70 years</td>
<td>0.0</td>
</tr>
<tr>
<td>SUM</td>
<td>450</td>
<td>1.3</td>
<td></td>
<td>0.2</td>
</tr>
<tr>
<td>NUCLEAR</td>
<td>360</td>
<td></td>
<td>(Cheap &amp; abundant energy)</td>
<td>0.2</td>
</tr>
<tr>
<td></td>
<td>&gt; 1000</td>
<td>1.7</td>
<td>Actual energy production 1983</td>
<td>&gt; 3</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Reasonable production in 2020 (personal view)</td>
<td></td>
</tr>
</tbody>
</table>
I am thus arguing in Table 3 for the importance of quickly testing out these opportunities as they seem the most promising of the major available nuclear options, (economic demonstration).

What is important to note however is that even though it has become increasingly clear even for the most enthusiastic proponents of the LMFR that a global impact of that system is hardly to be expected before 30 years or more, if ever, it is difficult to convince people that the pressures for introducing more HWRs, advanced LWRs and HTRs has therefore become the more acute. The fact that nearly $2G/a is still being spent on the LMFR development and that there soon will be close to 3GW of such operating plants weighs more than the order of magnitude lower figures for the HTRs regardless of the technical and other arguments for the latter. The fact that the UK is considering to replace the AGR dev with PWRs is also interpreted as proof of the inferiority of GCRs.

Table 3  THE MAJOR NUCLEAR ENERGY ALTERNATIVES IN THE GLOBAL PERSPECTIVE

<table>
<thead>
<tr>
<th>R &amp; D CO.-</th>
<th>HE-OOLED.</th>
<th>SPELLATION</th>
<th>HE-OOLED.</th>
<th>FUSION</th>
</tr>
</thead>
<tbody>
<tr>
<td>CAS</td>
<td>CO.-</td>
<td>AGR</td>
<td>CO.-</td>
<td>ARM</td>
</tr>
<tr>
<td>---------------------------------</td>
<td>-----------</td>
<td>------------</td>
<td>-----------</td>
<td>-------</td>
</tr>
<tr>
<td>BAD FOR DEMONSTR. G$</td>
<td>--</td>
<td>LESS THAN 5 FOR EACH OF THE CASES</td>
<td>--</td>
<td>30</td>
</tr>
<tr>
<td>ECON. POT.</td>
<td>GOOD</td>
<td>GOOD SYMBIOSIS</td>
<td>RISKY</td>
<td>GOOD</td>
</tr>
<tr>
<td>SAFETY</td>
<td>SUFFICIENT</td>
<td>BEST SUP.</td>
<td>BEST B</td>
<td>SUP.</td>
</tr>
<tr>
<td>ENVIRONMENTAL RANKING</td>
<td>5 (SUPP.)</td>
<td>2</td>
<td>3</td>
<td>4</td>
</tr>
<tr>
<td>TIMING</td>
<td>WE'RE GLAD</td>
<td>BEST</td>
<td>PROMISING</td>
<td>GOOD</td>
</tr>
<tr>
<td>RAD 1984 by taxing G$ a</td>
<td>1</td>
<td>0.2</td>
<td>0.0</td>
<td>0.1</td>
</tr>
<tr>
<td>OBJECTIVE</td>
<td>NUCL CAP</td>
<td>CITY, HEAT &amp; PROPULSION</td>
<td>MAIN LINE</td>
<td>PAST NUCL</td>
</tr>
<tr>
<td>PRIORITY</td>
<td>3</td>
<td>1</td>
<td>2</td>
<td>4</td>
</tr>
<tr>
<td>BAD G$</td>
<td>7</td>
<td>25</td>
<td>8</td>
<td>7</td>
</tr>
<tr>
<td>G$ a</td>
<td>0.7</td>
<td>2</td>
<td>1</td>
<td>0.4</td>
</tr>
<tr>
<td>*</td>
<td>TOTAL 5G$ a</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
I suppose we are here in the first place because the Agency can, with its authority of international consensus on technical matters help the member states to refrain from resorting to extreme and costly measures such as transporting hot water hundred or more kilometers when one can build perfectly safe reactors in urban areas, or bury unprocessed valuable waste in expensive copper containers 500 m unretrievably in rocks when recycling is a better way to fully utilize our resources and minimize the environmental impacts and so on.

But to do that there must first exist internationally approved and demonstrated better options to offer. This applies to the GCR development as well.

When one looks at the actual status of things in this area one finds that the British are building CO\(_2\)-cooled reactors which perform well, the Germans have developed and demonstrated the wonderful spherical fuel concept that permits a minimum of excess reactivity in the reactor, on-load fueling, high burnups and high coolant temperatures etc, the Americans on their part have developed the prismatic HTR fuel that permits fabrication of larger fuel blocks, allows better natural up-flow core cooling, i.e. larger reactor sizes that can be inherently safe against fuel melt-downs and/or which permit higher core power densities for compact plants and so on.

This is a typical situation in the development of reactors with the same type of coolant and there are always strong forces trying to make everybody just concentrating on how to utilize this organisation to push their own marginal interests. At best then the different members learn something from their foreign colleagues, but that does not solve eg our basic problem of bringing out the GCRs as the number one global priority in the reactor development aiming at solving the energy crisis by abundant nuclear power along the arguments that I have presented under pt 5 on the agenda, Table 3. That is, I think the objective of this WR if you are like myself convinced that a full out effort would not only bring all members here benefits but the whole world.

With the global energy systems becoming more and more refined in the future one will be able to buy more custom made optimized reactors (still mass produced) for special applications. I could therefore visualize (as pointed out in my Status Report on page 10/20 of NR-82/185) several first plants fitting a coordinated international program.

In my personal opinion the items of size discussions and optimizations considering also market aspects, easy introduction (launching of commercial capacity all over the world) and short delivery times (through standardization and factory production) and design comparisons for this could become (if they are not already) bottle necks/obstacles to the development unless broad international consensus can be reached about these key issues. They seem thus most suitable topics for a CRP and should be addressed as quickly and openly as possibly with a maximum of international exchange of views.

To illustrate what I mean, I think that we should firstly agree on a desirable menu of GCR applications like:

- Electricity producing plants
- Cogeneration plants
- High temp process heat plants
- Direct cycle plants
- Urban located plants/remote plants
- Fast breeders

At the same time one should select the fuel (spherical or block), the type of pressure vessel, IHX or not, plant size (modular or large/sizes) and so on from the matrix of most interesting options.

Secondly one could map out who wants to concentrate ones efforts on what of the needed options and organize them in a technically and economically practical time coordinate for a basis of cooperation discussions.

In doing this one would thirdly need to agree on criteria for:
- When to use IHX
- What pressure vessel is optimal for the different cases
- What fuel is
- and so on (the SG is e.g. a key component for any HTR).

After that one can fourthly adjust ongoing R&D efforts in these new directions/objectives and rationalize the efforts for the agreed goals by means of specialist mtgs and CRPs.

Actually I think that we in fact are proceeding like this, but the whole picture has not been painted and I realize that it is difficult for members heavily committed to specific lines to express these thoughts. These are matters that one is sensitive to, but since we in Sweden do not have a program and I am assessing nuclear power quite unbiassed I guess it is easiest for me to speak this frankly.

There has been many cost comparisons in the past regarding the capital costs for large HTRs and GCFRs showing some percent (less than 15%) higher costs than for LWRs and even lower costs in the case of the direct (gas turbine) cycle plants (especially when cooling towers are needed).
The main unknown seems today to be the fuel cycle costs, including the unknown times when what fuel cycle services will be available and the relations with respect to burnups and conversion ratios. As long as these questions are difficult to answer it is hard to make the case for the HTRs in Sweden.

These questions should undoubtedly be part of a CRP and/or the IAEA CARES study and a specialist meeting on a subject like "FUEL CYCLES & COSTS CONSIDERING MAIN PARAMETERS" would be very useful.

At the very useful components specialist mtg in Düsseldorf one did not try to address the problems of the rational of using the IHXs. Do we need it for safety against high temp graphite/fuel corrosion by coolant moisture, to prevent process gas contamination etc or just for various technical advantages or what? I think we would be served by a more systematic study of this issue. That is necessary for understanding the rational of various plant design proposals. Obviously component development and safety studies must be equally integrated as technical and cost arguments. Should one have a specialist meeting on "PLANT DESIGN CONSOLIDATION WITH REGARDS TO SAFETY AND COMPONENTS DEVELOPMENTS"?

With this perspective on the GCR option I think that this UGR should work out and make a clear statement for the execution of a list of recommended priority tasks that really address the key issues. From that list one could then choose the CRPs and the objectives of pressing specialist meetings.

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**SWISS ACTIVITIES IN THE FIELD OF GAS-COOLED REACTORS IN 1983**

Schweizerische Interessengemeinschaft zur Wahrnehmung Gemeinsamer Interessen an der Entwicklung Nuklearer Technologien (IGNT), Würrenlingen, Switzerland

Swiss industrial companies and the Swiss Federal Institute for Reactor Research (EIR) have been involved in the development of Gas Cooled Reactors since 1968. A significant contribution has been made to the HHT-Project (High Temperature Reactor with a Helium Turbine) in close cooperation with German partners between 1973 and 1982. Both the Swiss and The German partners intend to continue the collaboration within the framework of the German project HTR-500. A four year-working program for the Swiss partners was established in the beginning of 1983. However, since the request for financial support addressed to the Swiss Government has not yet been granted, work has been performed only to a limited extent so far. The activities of the Swiss partners in 1983 are summarized below.

**EIR**

The effort made at the EIR in the HTR-field is devoted mainly to generic type of work and HTR-base programs. The annual budget amounts to approx. 3 Mio Sfr.

Plant layout calculations and sensitivity analyses have been performed to study the influence and the interdependence of main parameters on efficiency, component size and plant cost.

Theoretical and experimental work on heat exchangers and cooling towers has been continued. Studies and tests have been and are continuing to be carried out on the improvement of heat transfer in dry-cooling-tower-elements by wetting the exchanger surface or moistening the incoming air.

* Presented by G. Sarlos.
A special effort is being made to investigate the various effects of down-flow evaporation at low mass flow rates e.g., start-up and shut-down conditions in helix-type steam generators.

Materials research continues to be carried out in the fields of gas/metal reaction kinetics, LCF, HCF and fracture mechanics studies with special emphasis on the effects of crack initiation and propagation.

Bonnard & Cordel

Tests on creep and shrinkage of various concrete-types at temperatures up to 120°C have been continued. A test for the determination of the characteristics of the 'liner-anchorage by struts, especially in singular positions, has been prepared.

Elektrowatt

The activities of Elektrowatt comprise investigations on the dynamic behavior of the core structures under seismic excitation. Work in 1983 was concentrated on the development of a code for the description of the static and dynamic behavior of an agglomeration of pebbles based on the laws characterizing the contact between discrete pebbles (static and dynamic stiffness, friction, elastic and plastic behavior etc.). Tests for the determination of these laws have been defined.

A specification of tests to be performed in the vibration test rig of HRB at Jülich has been worked out. Theses tests will make it possible to verify the calculation procedure for the dynamic behavior of the pebble bed.

Motor Columbus

Motor Columbus was involved in theoretical investigations of the behavior of the botten structures of pebble bed reactors. Questions which should be answered are the distribution and seize of crevices between the graphite blocks under external forces and cyclic thermal load. The calculation procedure will be verified by tests performed by HRB.

Sulzer Brothers

Mounting of the steam generators for the THTR-300 was completed. Installation of connecting pipework and pipe supports was continued. Feedwater throttling valves were installed and adjusted. Main steam valves and by-pass valves were supplied and installed. Structural analysis of steam generators and penetration closures was completed. A steam generator commissioning manual was issued. Analysis and experiments were performed for special transients involving superheater flooding.

For the HTR-500 project, a manufacturing study for the expansion zone of the live steam lead-out tubing was performed. The design of a gas-guiding device and main shroud was optimized. Cost of steam generator and core auxiliary heat exchanger were analyzed. An experimental screening program for INCOLOY 800 welds was executed.

In the framework of the Process Heat Program, design and analysis work for the helium to helium intermediate heat exchanger (IHX) was continued. Main components for a 10 MW helical tube IHX test unit were manufactured and delivered. Tube protection coatings for operation at 950°C were experimentally investigated under thermal cycling and wear conditions.
The major share of the capacity for electrical generation from nuclear power will continue to be provided by gas-cooled reactors with the Magnox reactor providing 3.8 GW and the AGR capacity increasing to 8.8 GW. The major investment of £7,000M in AGRs will provide almost all the nuclear power at the turn of the century.

The 18 commercial Magnox reactors, on 9 sites, comprise a mature system, having been commissioned between 1962 and 1971. All these reactors reach their 20 year accounting lives, extensions to their licenses are being sought. The original Magnox reactors at Calder Hall and Chapelcross are still operating successfully some 24-28 years after being commissioned.

There are now 14 Advanced Gas-Cooled Reactors in operation or under construction at six sites. The two stations which are fully operational at Hinkley Point and Hunterston have increased their averaged load factor to 68% during the last year, compared with 58% the previous year. At each of the stations, Dungeness B, Hartlepool and Heysham I, one reactor is being brought up to full power while the second is being commissioned. The 4 AGRs being built at Heysham II and Torness are over half-way through their construction phase and are on schedule, for both time and cost.

In the fiscal year 1982/83 the total electricity output from the CEGB's nuclear stations in England and Wales increased by 25% on the previous year. These improvements stem from the return to service of 6 Magnox reactors which had been shutdown for inspection and from the improved load factors of the AGR stations, all six reactors (two at each station) were shut down in 1980-81. Full inspections were carried out and work put in hand to establish that it was safe to return the reactors to operation. By means of detailed metallographic examination it was shown that the defects were originally present when the stations were built and had not grown in the subsequent 15-20 years. By the use of an assessment procedure developed by the Board's laboratories to examine the structural integrity of plant containing defects, it was shown that all the bellows units to be returned to service had adequate strength for the remaining lives of the stations.

The strength of the bellows units containing defects was confirmed by pressure tests on units removed from the gas ducts. Some units were pressure cycled to simulate fatigue in future operation and then pneumatically pressure tested to failure. These tests confirmed that the bellows units had sufficient strength to withstand pressures several times greater than they will experience in operation. All six reactors shut down for inspection have been returned to service, with the approval of the Nuclear Installations Inspectorate.

Following tests on coolant-circuit gas relief valves of the type used at Berkeley power station, it has been possible to specify modifications which will improve their performance and allow the reactors to operate at increased gas pressure. This will increase station output by about 10 MW, worth £800,000 a year.

During the overhaul of the turbine units at Wylfa power station, modifications were made to the reheat system which improved the efficiency of the heat cycle and restored some 14 MW of previously lost output capability. A record annual output from this station was achieved in 1983. Reactor 2 at Trawsfynydd was taken off load on 3 September 1982 for routine overhaul after 640 days of continuous operation during which it produced 3.0 TWh. This lengthy period of continuous production fell just short of a world record.

In 1983 the stations at Berkeley and Bradwell celebrated 21 years of generating electricity. Hunterston 'A' station, operated by the South of Scotland Generating Board, maintained a lifetime load factor based upon design output of 81% which is one of the highest for commercial nuclear power stations in the world.

The assumed life of the Magnox stations has been increased to 30 years, as the result of research and development work.

Advanced Gas-cooled Reactors

In order to realise the greatest benefit from the £7,000M invested in the AGRs, research and development effort is directed to obtaining maximum output and economic life from each station without detriment to safety, to achieving on-load refuelling at the highest power justifiable, to develop the fuel design to attain increased burn-up, and to bring the stations being built or commissioned to timely and satisfactory operation. Some 650 qualified scientists and engineers are concerned in this work at an annual cost of about £50M. The benefit of achieving all the objectives is about £2,000M discounted at 5% over the life of the stations.

Recent technical achievements include:

(a) increased confidence in the attainment of a graphite life of 22 full power years, by the successful operation of one of the reactors at Hinkley B for a year in a coolant which inhibits graphite corrosion without producing any carbon deposition on the fuel pins;
(b) The target irradiations of 18 GWd/t have been achieved on both initial charges and on replacement fuel. All the fuel now being loaded is enriched to achieve 21 GWd/t and from 1985 will be enriched to achieve 24 GWd/t. Further improvements to fuel design to allow higher ratings are in progress;

(c) On-load refuelling at 30% power level has become routine, with consequent improvement in reactor load factors; while a stronger fuel element has been developed which is able to withstand the buffeting it receives during refuelling at higher power levels;

(d) An economic assessment showing that the attainable benefits from the items above and small attainable increases in power level, will reduce the cost of power from AGRs by 14%;

(e) Injections of particles and methyl iodide in an operating reactor have each demonstrated half lives of less than a minute, which is of benefit in safety assessments.

Future major objectives include:

(a) The development of an improved fuel design to enable higher ratings and burn-up to be achieved;

(b) The study of the release of fission products from failed pins in a loop in PLUTO, and the development of our understanding of the transport behaviour of fission products in fuel pins, and in the coolant circuit;

(c) The study of the ability of the fuel to withstand very large transients leading to pellet-clad interaction;

(d) Graphite irradiations to extend the range of information on graphite weight loss, neutron damage and the interaction between these processes. The modelling of graphite as a structure will be complemented by the construction of scale models of a section of the graphite core, and by the monitoring of the reactor cores;

(e) Fuel management and performance studies to secure higher output, complemented by the use of instrumented fuel elements in the operating reactors;

(f) Engineering studies on the integrity of components and structures, on the corrosion of structural steels, on boiler and turbine performance and on the inspection of vessel internals with the aim of extending station lives;

(g) Studies to enable the reprocessing or dry-storage of AGR fuel.

HIGH TEMPERATURE GAS-COOLED REACTOR (HTGR) PROGRAM IN THE UNITED STATES OF AMERICA*

US Department of Energy,
Washington D.C.,
United States of America

PROGRAM STATUS

The current focus of the HTGR program in the United States is the development and evaluation of several HTGR designs and size options to meet the power generation requirements and economic limitation of U.S. utilities in the 1990 timeframe.

Utility/User interest and support for the HTGR has continued to grow through Gas-Cooled Reactor Associates (GCRA) which, in addition to its project development and evaluation activities, provides program coordination services to Department of Energy (DOE). Program planning, coordination and control has continued to advance and stabilize through the implementation of program level procedures and the computer-based Summary Level Program Plan (SLPP) from which formal work statements are extracted for all major program contracts and subcontracts. In addition, the SLPP is the initiating basis for more detailed Program Control Documents (PCDs) that identify the design and contractor interfaces.

The Conceptual Baseline Documentation for the 2240 MWt HTGR-Steam Cycle/Cogeneration (SC/C) integrated plant is nearing completion. Major program and GCRA resources have been applied in the development of the Overall Plant Design Specification, the Conceptual Design Report, the Long-Term Project Summary Plan, the Licensing Plan and the Lead Project Strategy Plan. United Engineers & Constructors has provided the major balance-of-plant design resources for the respective system and subsystem design descriptions plus a major resource for the overall plant, cost, and schedule estimates. Detailed economic studies that include baseload generation, several cogeneration, and site-specific application studies have been conducted to evaluate the costs, risks and benefits associated with deploying the 2240 MWt plant.

During the past year, there have been several introspective studies and surveys, including a broad utility survey by GCRA, on the future of the Nation’s nuclear power industry. A recurrent message from these efforts is the need for smaller, simpler nuclear powerplants that will ease the current regulatory and financing difficulties of nuclear plants in the United States. Within the HTGR program, smaller alternatives to the 2240 MWt plant have been identified that offer added dimensions of

* Presented by I. Helms
simplicity that may translate to overall performance and economic advantages. In one approach, a 1170 MWT/428 MWe design has been configured that was originally conceived as a 2-loop version of the reference 4-loop plant. However, with an emerging market interest in plants of this size range, an optimized, integrated-type HTGR in the 400-500 MWe range is being developed. Some of the key areas that will be addressed are:

1. A simpler approach to decay heat removal through the use of passive systems.

2. The number and size of primary system coolant loops, the configuration of the Pre stressed Concrete Reactor Vessel (PCRV) and the consideration of the use of a confinement rather than a containment type secondary enclosure.

3. The optimum approach to multiple unit plants, and the sharing of common systems and facilities, plus greater use of modularity and conventional plant construction standards.

Stone and Webster has been selected to provide a major balance-of-plant design resource for the mid-size integrated HTGR.

In another approach to a smaller, simpler HTGR system, a modular HTGR plant has been conceived as independent reactor units of approximately 250 MWT/100 MWe that can be constructed as a cluster of modules or sequentially constructed to more closely match load growth requirements and reduce the utility’s financial exposure. An intent of the modular HTGR concept is to achieve greater inherent safety by removing the decay heat by passive mechanisms such as radiation, conduction, and natural convection to minimize reliance on active engineered safety systems. The concept is based on limiting the core size and power density such that the fuel particle coatings retain the fission products to an acceptable degree under all circumstances, including the loss of all forced circulation or the failure of the pressure vessel. The economic challenge of the modular HTGR concept is for the design simplicity afforded by the passive safety attributes to offset the economy of scale advantages of larger, integrated nuclear plants. To do so, it is intended that the nuclear steam supply system will be shop fabricated to nuclear standards and the balance-of-plant field-erected to predominantly conventional fossil plant standards. This approach is expected to improve the low productivities, reduce the schedules and reduce the field-erection costs encountered on most of the recent nuclear projects in the United States.

The modular HTGR concept is in the early stage of evaluation in the United States. Some of the key issues that are being addressed are:

1. Sufficient design activity must be conducted to support economic analyses that will examine sensitivities in the costs and schedules for the various modular HTGR configurations versus the integrated HTGRs and the other relevant alternatives.

2. The safety goals must be specifically defined and translated to licensing requirements and engineering terms. Key issues are the development and acceptance of design basis events, source terms, and allowable fission product releases.

3. The design and development programs required to demonstrate the unique, inherent safety features must be defined. Central to the modular HTGR concept is the use of a fuel for which high integrity and low fission product release can be assured under all postulated reactor conditions. These design requirements place provisions on fuel manufacturing and performance that require extensive verification.

Bechtel provides the major balance-of-plant design resource for the modular HTGR concept.

The primary emphasis of the HTGR program during the balance of 1984 and 1985 is to establish which of the HTGR concepts (2240 MWT/855 MWe integrated plant, mid-size integrated plant or modular) is optimally suited for initial deployment. In concert with the design and technology development activities, key institutional issues of licensing approach and criteria development, vendor/industrial team development, international cooperation, and utility/user project development activities are underway.

A keystone to the development and deployment of HTGR is the successful operation of the Fort St. Vrain (FSV) HTGR Generating Station. Since the last meeting of the International Atomic Energy Agency (IAEA) Working Group on Gas-Cooled Reactor, FSV achieved a good record of availability and capacity factors during the second half of 1983. FSV is to resume operation after refueling and a major maintenance outage this spring.

The work in the principal U.S. industries and laboratories involved in the program is described in the following sections:

HTGR PROGRAM AT GA TECHNOLOGIES, INC. (GA)

RECENT DEVELOPMENTS:

HTGR Applications Program

The principal activity at GA this past year has been design of the 2240 MWT HTGR-SC/C Lead Plant. The conceptual design has been completed. A major achievement in the design and documentation development has been the use of a functional analysis based on DOE's "Integrated Approach" which insures that top level goals for plant performance, availability, economics, safety, and investment risk are manifest at the system and component design levels.

Based on direction from DOE and information and encouragement from many U.S. utilities, an increasing effort is being applied to the study of
smaller, passively safe HTGR's. Thus, design activities are being initiated on an integrated HTGR with electric output up to 400 MWe. The design goal at this system is to blend the best features of modular HTGR's with those of the 2240 MWe HTGR. There is also an effort on a 250 MWe modular HTGR for use in multiples, with interest primarily on a 4-module (400 MWe) version. This effort is aimed at the selection of design variants, including prismatic block, pebble-bed reactor cores with steel, and prestressed concrete reactor vessels, as well as type and location of steam generators, circulators, control rods, and fuel handling systems. An intriguing aspect of the modular HTGR is the potential for developing a passively safe design. The 250 MWe core at a power density of 4.2 w/cc is small enough to enable passive decay heat removal via conduction, radiation, and natural convection while maintaining fuel temperatures below levels that would cause significant fission product release. This feature represents a major advance in the reduction of risk to the health and safety of the public over the already significant advantage of the large HTGR's.

**HTGR Technology Program**

Successful design and deployment of HTGR's in the 1990's is seen as being critically dependent on a vigorous program for new technology development to reduce technical, licensing, and economic risks. Major technology programs currently underway include:

- Manufacturing test programs to improve fuel quality,
- In-pile tests of fuel particle fission product retention during normal operation and loss-of-flow accident conditions,
- Flow distribution testing of redesigned fuel elements,
- Code development for modelling plant seismic response, plant transient response, acoustic vibrations, moisture ingress incident and quantification of safety and investment risk,
- Tests of the reactor vessel thermal barrier response to acoustic and flow vibrations,
- Flow distribution testing of the core exit plenum and cross ducts,
- Dynamic testing of the main circulator bearing and seal assembly to verify that design changes will prevent water in-leakage,
- Seismic testing of the helically coiled steam generator.

Work is proceeding on FSV incorporation of Fort St. Vrain construction and operation experience into the data base, as well as assimilation of research results from Electric Power Research Institute programs. In the area of international cooperation, important ongoing programs are in place for HTGR fuels, fission products, graphite development, and on spent fuel treatment and a new program on metallic materials has been initiated with the Federal Republic of Germany (FRG).

**HTGR Fuel Recycle Development Program**

During the past year, a GA cold pilot plant fluidized bed combustor was successfully used to burn simulated U.S. low enriched fuel particles and 100 percent broken low enriched fuel particles produced in the FRG. The Joint U.S./FRG Fuel Processing Demonstration (JFFD) tests confirmed the superiority of an in-bed-fines recycle system in preventing fines buildup. Other work performed during the period included the completion of the testing of individual components of the GA burner off-gas treatment system, an evaluation of the feasibility of separating HTGR fuel rods from the graphite block to produce discrete high level and low level waste streams, and preliminary laboratory preparations of volume reduced HTGR fuel cement stabilized waste forms.

The cold pilot plant work at GA is complemented by hot laboratory work being conducted by the FRG at KfK Forschungsanlange (KFA). In 1983, GA began the shipment of cold and irradiated HTGR fuel rods and fuel element segments to KFA for crushing and burning in the facility in FY 1984. Preliminary static burning tests have already been performed.

**PLANNED EFFORTS FOR THE REMAINDER OF THE YEAR:**

The balance of efforts in 1984 is aimed at bringing the work on the larger 2240 MWe HTGR design to a logical conclusion in order to focus more effort on the smaller 400 MWe HTGR design to a logical conclusion in order to focus more effort on the smaller 400 MWe HTGR and the 100 MWe modular HTGR and concepts. As these concepts evolve, they will be evaluated using the DOE Integrated Approach. Top level requirements will be specified and trade studies performed to determine base design features. At the end of U.S. Fiscal Year 1984, a reference integrated plant concept and a reference modular plant concept will be selected for future study and evaluation in Fiscal Year 1985.

**FORT ST. VRAIN**

In October 1982, the U.S. Nuclear Regulatory Commission (NRC) gave the plant a full release to operate at 100 percent power. However, the Public Service of Colorado (PSC) elected to not exceed 70-75 percent power until late in 1983. After a good run during the latter half of 1982, the plant was shut down in January 1984, for its third scheduled refueling and major maintenance outage. It is noteworthy that fuel handling only required 27 days in a scheduled 90-day outage. Major maintenance was performed on the main turbine generator set, 3 boiler feedpumps and 2 feedwater heaters while major changes were made to the plant auxiliary electrical system and a helium circulator was replaced with a spare.

Very low personnel radiological exposure at FSV continued in 1983 with a total collective dose less than 1 mrem-rem, about two orders of magnitude lower than the average of the U.S. power reactor industry. Additionally, for the first time since FSV started nuclear operations in 1974, low level solid waste was shipped from the station. Approximately 850 cubic feet of irradiated graphite reflector blocks and reserve shutdown material containing only 18 curies of activity was removed for disposal.
HTGR PROGRAM AT OAK RIDGE NATIONAL LABORATORY

The HTGR program at the Oak Ridge National Laboratory (ORNL) involves a broad range of technology development areas in support of design. Most of the efforts are concentrated on the steam cycle lead plant concept, but important activities are also directed toward advanced, very high temperature systems. In addition, studies are conducted relative to economics and evaluations. The areas of activity at ORNL are summarized below.

Shielding Studies. The current major focus of this work is to experimentally determine the extent of neutron streaming through the coolant holes in the bottom reflector and core support block and to compare these results with design predictions. A streaming experiment, conducted layer by layer through a mockup of the reflector/support structure, is about 75 percent complete; completion of the experiment and the majority of the associated analysis are expected by years end.

Core Support Performance Test. A series of engineering tests have been designed to evaluate the performance of the graphite core support posts and seats under flow, temperature, and environmental conditions appropriate to the HTGR. TEST ZERO, just completed, qualified both the impurity measurement and control systems, and demonstrated that measurable and predictable corrosion rates of large graphite specimens can be achieved in the engineering-scale test. Preparations for TEST ONE, with nuclear grade graphite post/seat combinations under load, will continue through the remainder of the year. These tests are performed in the Component Flow Test Loop (CFTL) facility at ORNL.

Physics. These studies are directed at detailed analysis and evaluation of reactor designs, development, and application of analytical capabilities including 3-D neutronics and thermal-hydraulics. Recent activities have involved the programming of depletion/perturbation theory into user codes and review and assessment of nodal diffusion theory methods. During the remainder of the year, the core physics effort will involve analysis of annular pebble bed cores.

Concrete Development. Development and testing of high-strength concrete mixes to minimize PCRV size and cost are in progress. The goal is an 8000 psi ASME approved mix. 9200 psi strength has been achieved with the first of the four area-representative aggregates. All mix development and associated testing will be completed this year.

Structural Ceramics Testing. Compression testing of full-sized ceramic core support pads is being conducted to qualify these components for service. All room temperature tests on two grades of alumina have been completed and work with fused silica will be finished within a few months. Development of strain measuring instrumentation for elevated temperature tests will be the major area of attention during the remainder of the year.

Fuel Materials Qualification. This activity supports the technology base for manufacture of safe, reliable, and economic coated particle fuel. Major ongoing tasks are the design, construction, and operation of irradiation experiments and post irradiation examination (PIE) of the irradiated fuels. Preparation of fuel hydrolysis experiments HRB-17 and HRB-18 is almost complete and irradiation will begin this year. All PIE of UCO fuels irradiated in HRB-16 will also be completed this year.

Fission Product and Actinide Studies. These studies are currently devoted to the transport of plutonium in and from fuel block graphite to coolant gas, and to the chemical form of Cesium and Iodine in HTGR systems. Experimental measurements in these areas will be nearly completed this year.

Structural Metals Studies. This work is aimed at development of materials data and technology needed for design of HTGR components. Focus is on high temperature mechanical properties, thermal stability, corrosion, and joining technology. A corrosion study of 2-1/4 Cr-1 Mo steel to provide predictive capability for long-term decarburization was recently completed. Heding parameters for the steam generator superheater tube-to-tubesheet joint will be finalized this year.

Graphite Qualification. Graphite grades are being developed and/or studied for use in fuel blocks and reflector properties before and after irradiation and oxidation behavior. Determination of the effects of irradiation on fracture toughness is a current major effort continuing through the year. Another fracture mechanics irradiation capsule, HFP-4, will be constructed by years end.

Advanced System Alloys. This task is aimed at development and qualification of alloys for use in HTGR systems with core outlet temperatures in excess of 850°C. Current efforts involve creep-rupture testing of advanced nickel-base alloys and evaluation of the applicability of nickel aluminide ordered materials in an HTGR. The latter activity will reach a "go-no go" decision point at the end of the year.

High Temperature Design Criteria. This year's goal is to complete an assessment of the applicability of ASME Code Case N-47 to the design of very high temperature HTGR components. Shortcomings of the Code will be identified and development needs described.

HTGR Plant Economics. Cost data have been collected to permit comparison of busbar generation costs for large HTGR's, modular HTGRs, LWRs, and coal-fired plants. Preliminary results have been obtained and analysis will continue throughout the year.

Advanced Reactor Systems Engineering. Analysis of annular core designs for 1000 to 1200 MWe HTGR's is in progress. A 2-D depressurized core heatup case is set up to determine the limit on annulus thickness imposed by restrictions on peak fuel temperature. Cooperative work with FRG on the response of the AVR to a depressurized core heatup accident is in progress.
Summary and Conclusions. The ORNL Gas-Cooled Reactor program has made significant progress in:

1. Providing generic technical data needed for High Temperature Reactor (HTR) commercialization, covering areas such as fuels, metals, ceramics, graphite, concrete, and fission products, and actinides;
2. Performing important component testing of the core support structures involving graphites and ceramic structures;
3. Experimentally determining that either the present shielding design of the 2240 MWt HTGR needs to be revised, or materials have to be qualified for higher thermal fluences, or both;
4. Developing improved reactor physics design tools;
5. Performing technical and cost evaluation studies covering HTGRs with improved passive safety features; and
6. Carrying out cooperative studies with FRG in fuels, graphite, metals, and reactor-analysis areas under the U.S./FRG Gas-Cooled Reactor Technology Umbrella Agreement.

HTGR PROGRAM AT GENERAL ELECTRIC (GE)

For over 10 years, GE has been involved in technical activities concerning the HTGR. GE's major HTGR efforts are in the areas of:

- Design, technical, and economic evaluation of advanced HTGR's (e.g., the HTGR-Reformer plant, and the modular reactor);
- Testing of advanced HTGR alloys which are principally planned for use in the HTGR-Reformer plant;
- Development and assessment of applications for advanced HTGR's; principally for the HTGR-Reformer plant;
- Development, analytical methods, and design criteria for analysis of advanced HTGR systems and components; and
- Development of computer programs for simulation and analysis of plant transient behavior, and control system characteristics.

In all of these activities, GE's work is closely coordinated with related work being performed by other U.S. HTGR program participants. These efforts are mutually supporting and provide the basis for an overall evaluation of the various applications and reactor plant concepts for the national U.S. program.

In the time period beyond 2005, the HTGR-Reformer plant could be available industrially for the supply of heat and industrial gases (hydrogen and carbon monoxide) to such processes as:

- Distributed process steam use (the thermochemical pipeline concept)
- Oil shale processing
- Upgrading (hydrogenation) of oil and syncrudes, and
- Chemical synthesis of methanol/ammonia

In the area of these advanced plant applications, GE's studies of these processes (with the support of other U.S. participants), indicate that the HTGR-Reformer plant would be generally competitive with fossil fuels beyond the year 2005. In addition, use of the HTGR for production of industrial heat and gases results in reduction of fossil fuel usage and pollution emissions.

By the year 2010, we believe that the new and replacement applications would provide a 400-500 GWe potential market in the United States, assuming a 1.5 percent per year growth rate in the use of industrial heat and gases.

The modular HTGR (200 to 300 MWt reactor modules) is visualized by GE as a plant concept which would meet non-electric industrial requirements for:

- Power availability,
- Economics,
- Flexibility to meet varying lead growth and "mix" of energy products (e.g., steam and industrial gases), and
- Safety and ease of operation. Study under the U.S. HTGR program has indicated that the multi-module HTGR-Reformer complex may be economically superior to the monolithic HTGR-Reformer (single large reactor) concept.

More recent U.S. program efforts have focused on the modular HTGR for nearer-term steam and electricity production for comparison with the monolithic HTGR plant. For the remainder of this year, GE's efforts on the modular reactor will focus on plant dynamic simulation, operation, and maintenance assistance in the establishment of overall plant requirements and work on alternative, advanced modular reactor concepts.

In the area of reactor design, GE has performed nuclear/thermal/hydraulic analyses of the modular pebble bed fueled reactor. Early conclusions (subsequently reinforced by other U.S. program results) have indicated that the pebble bed is well-suited to the modular reactor concept.

In addition to analysis of process heat applications, and technical integration of advanced HTGR studies, GE has devoted detailed technical efforts to the design and evaluation of the reformer component. These activities include:

- Design and structural analysis of the reformer;
- Development of structural analysis methods and criteria for very high temperature (800°C - 950°C) component design, beyond the ASME code case N-47;
- Development of an analytical (computer program) means to predict the hydrocarbon impurity content (H₂, CH₄, H₂O, CO, CO₂) in the primary helium circuit and reactions with the heat exchanger alloys;
- Performing tests to determine the mechanical properties of alloys for use in high temperature components; and
- Development of component design and catalyst concepts for the steam-methane reforming and methanation processes.
In performing this work, GE has maintained close liaison with similar programs in the FRG and in Japan.

While these efforts are far from being completed, the initial indications are that Inconel 617 and Incoloy 800H are viable reference materials for reformer design, with the expectation of a 5-10 year component life, and possibly even longer, with proper control of the impurities in the primary helium circuit. Efforts in these areas are planned to continue through the remainder of this year.

Advanced Materials Program

In previous GE screening programs sponsored by DOE, some 30 commercially available alloys were evaluated for use in advanced HTGR systems. The alloys were evaluated on the basis of environmental compatibility, thermal stability, high temperature mechanical properties and the effect of simulated HTGR helium on mechanical properties. Alloy 800H and Inconel 617 were chosen as the best candidate alloys for high temperature heat exchanger (e.g., steam/methane reformer) components for systems.

For these two alloys, several questions must be addressed. Environmental compatibility, particularly with respect to increases in carbon levels (carburization) due to gas/metal interactions, remains of concern. Because carburization can seriously degrade mechanical properties, the kinetics of carburization, as a function of temperature and gas composition, needs to be determined. Also, the effect of various degrees of carburization on important mechanical properties needs to be quantitatively defined, to enable limits on carburization to be set. Work on both these areas is underway in the GE program.

The higher strength of Inconel 617 is needed for steam reformer components that must operate at temperatures approaching 950°C. This alloy, however, has poor thermal stability which results in poor toughness and ductility at temperatures below 600°C after long time exposures at higher temperatures. Studies are underway to determine the causes of the thermal instability. Efforts to increase the alloy's thermal stability by controlling grain size, by lower temperature heat treatment after the final solution annealing operation, and by alloy chemistry control (being done by the alloy producer) also are in progress. The effects of these treatments on the thermal stability and on elevated temperature creep strength are being evaluated.

Work also is underway to define the design tensile stress allowables for Inconel 617 to 1900°F (1038°C) and for Alloy 800H from 1400°F (760°C) to 1900°F. This work involves collecting and evaluating all the available elevated temperature creep (and tensile) data for these alloys, analyzing the data by computerized methods, and developing creep equations, based on creep models, where necessary. Work to determine the limitations of fatigue and creep fatigue also is underway. Long time, high temperature creep tests, low cycle fatigue tests and creep fatigue tests, to supply the data needed (in addition to available test data) are being carried out.

To provide a backup position, alternate alloys are being evaluated in search of an alloy with better elevated temperature strength than Alloy 800H and better thermal stability than Inconel 617. The alternate alloys include new, commercially available alloys as well as several developmental alloys. An alloy development program also is underway to develop an alloy for direct cycle steam reformer applications with good elevated temperature stability, adequate thermal stability and environmental compatibility, and the necessary fabricability, and weldability.

Considerable work is underway to upgrade our simulated HTGR helium testing facilities to minimize the amount of controlled impurity (e.g., H2O, CO, CH2) gas depletion during flow through test retorts, by decreasing the amount of hot metallic surfaces in the test retorts, and by increasing the test gas flow rates. For large retorts, where depletion cannot be held to acceptable limits, gas sampling probes to determine gas compositions inside the retorts, are being installed. Over the past year, considerable success with these efforts has been achieved.

Over the past few years, considerable progress has been made in defining the important parameters in gas/metal interactions for reactions of structural alloys in simulated HTGR helium and in determining several important problems potentially affecting the service lives of these alloys. Considerable work remains to be done to completely bound the gas/metal interaction problems, to determine required limits on system primary coolant compositions, necessary limitations on carburization and thermal instability of structural alloys, the long time/elevated temperature properties of the structural alloys, and the design stress allowables for these alloys.

STATUS OF GAS-COOLED REACTORS DEVELOPMENT IN THE UNION OF SOVIET SOCIALIST REPUBLICS

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Abstracts

The report offers the review of the state of works on the high temperature thermal gas-cooled reactors (HTGR) and gas-cooled fast reactors (GCFR) over a period of 1983-1984.

INTRODUCTION

One of present problems together with a speed-up progress in the nuclear electricity production is an expansion of the scope of atomic energy for a domestic and industrial heat supply and of power production for industrial processes.

The fraction of fuel-power resources consumed in our country for production of all kinds of heat energy (low- and high-potential heat, steam, hot water) amounts to about 50% of the total volume of organic fuel consumption, and its considerable portion is the most deficit kinds of fuel: oil and natural gas.

Use of nuclear energy sources to produce low-potential heat in the USSR is realization on a basis of light water type reactors by creating nuclear power plants for heat supply (NPPES) and nuclear central heating and power plants (NCHPP).

At the same time, large-scale consumers of oil and natural gas spent to produce high-potential heat energy (500-1000°C) are various power-consuming branches of industry (chemical, petrochemical, metallurgical et al.).

Among possible nuclear power sources with such a temperature potential the most promising are high-temperature reactors with a helium coolant (HTGR), making it possible to obtain temperature at the reactor output up to 1000°C.

As is clear from investigations carried out and experience of operation of such reactors, in a number of world countries they feature good neutron-physical performance, the effective use of nuclear fuel, high nuclear and radiation safety.

This is provided due to the helium coolant's inertia and phase stability, the reactor core's high heat capacity, the use of fuel as particles with a multi-layer protective coating, as well as the layout of the main primary circuit equipment in the prestressed concrete reactor vessel.

Owing to these and a number of other factors (lesser heat losses, the possibility of dry cooling towers employment, etc.) the HTGR has a lesser environment impact and can consider as a promising power source for densely populated and industrial regions.

Calculated experimental investigations

In 1983-1984 the calculated and experimental investigations were in progress in the USSR to prove designs of pilot and pilot industrial plants with gas-cooled reactors (VGR-50, VG-400, OR-300), as well as investigations on energy-technological application of HTGR.

The study of HTGR physical characteristics is carried out at I.V. Kurchatov Institute with critical benches "GROG" and "ASTRA". A complex of investigations was performed on the bench "GROG" on testing a graphite stacking homogeneity and measuring its diffusion characteristics [1]; the effect of technological deviations of the bench elements on assembly neutron-physical parameters [2] was determined as well.
On the critical bench "ASTRA" the physical start-up of a methodic assembly was accomplished. There was run a set of experiments on the influence of various materials on critical parameters, in the investigation of energy release and space-energy distribution of neutrons in the methodic critical assembly. The calculation results are in agreement with experimental data. Reactor tests of fuel elements and a coated particles in ampoule channels and helium loop RS-100 continued on the experimental reactor HR.

While testing spherical fuel elements in the channel "Kashtan-3" there were attained burnups exceeding substantially design values. In this case, the release of fission products is within permissible limits.

The channel designed for thermocycling spherical fuel elements in order to imitate transient conditions of a reactor operating in the multiple fuel circulation mode was prepared and installed into the VVR-C reactor. The thermocycling of fuel elements is carried out by shifting a fuel element assembly over a core height of the experimental reactor operating at a constant power.

The second resource experiment on testing the spherical fuel elements and HTGR materials in the KVG-2 channel with the helium loop PG-100 was completed in 1983. At the helium temperature up to 800°C, fuel temperature up to 1200°C and other test parameters representative for VGR-50 and VG-400, the burnup of 17% fima was reached, which is two times exceeds the burnup under operating conditions.

In 1983 the channel KVG-1 of the loop PG-100 was prepared, in which 16 spherical fuel elements fabricated by various technologies were irradiated. The following maximum parameter values were attained; fuel temperature - 900°C; burnup - 13% fima; neutron fluence - 2.3.10^21 n/cm^2 (E > 0.18 MeV). All the spherical fuel elements have kept their integrity. Reduction in spherical fuel elements volume due to the radiation shrinkage and corrosion is within the limits of 0.2-5%, corresponding to the decrease in diameter by 0.08-1 mm, which is less than permissible.

Besides, the fuel element lot irradiated previously in the HR reactor ampoule channel was investigated under the following conditions: fuel temperature - 1000-1300°C, burnup - 10-14% fima. All the fuel elements kept their integrity. The graphite cladding strength was practically invariable. Under the conditions mentioned the maximum speed of a core shifting ("amaeba" effect) in some coated particles amounted to 7.3 mkm/year. In the majority of its no core shifting was made.

The complex of investigations carried out by the present time made it possible to elaborate substantiated requirements for coated particles and fuel elements for their industrial production.

In 1983 calculated investigations were performed on the problems of radiation safety of gas-cooled reactors. Fig.1 presents data characterizing service conditions of main units of the HTGR primary circuit from which it follows that when requirements on the fuel element tightness are observed (P < 10^-5), the equipment can be served without use of special protective means.

In terms of the analysis of experimental data on the GFP leakage dependence of the fraction of damaged coated particles, at the fixed contamination of various FE construction elements with fuel (covers of coated particles, graphite matrix, fuel element claddings without uranium, etc.), obtained with the use of the "weak" irradiation method, the requirements for VGR-50 and VG-400 fuel elements were substantiated; providing their needed tightness (Fig.2).

With a view to develop an experimental basis for mastering the problems of HTGR technological utilization, the chemothermal bench (CTB) is creating at the institute, designed to study a joint operation of the helium and chemical circuits. The bench (Fig.3,4) has...
a helium circulation circuit with a flow rate of 10-15 g/s and a pressure up to 10 atm. The bench chemical circuit provides heating up a water and methane and their supply to the conversion apparatus model, as well as cooling, collection and release of reaction products. The thermoconversion pipe was made according to the PPM scheme with location of a catalyst in a circular gap between the main and return tubes. The pipe working part length is 3 m, the diameter - 116 mm. The bench is equipped with the system for check and analysis of conversion gas and helium coolant content.

The investigations of properties of construction materials and nickel - and niobium-based heat-resistant alloys satisfying the HTGR operating conditions, the investigation of radiation damage and corrosion resistance of a graphite used for reflectors, the investigations of helium technology (corrosion of metals, mass-transfer), bench testing on heat-exchange, spherical dynamics and other works continued over the past period of time to substantiate HTGR designs under development.

The pilot energy-technological installation with VGR-50 reactor [6]

The installation is designed for mastering production processes of high-potential heat, steam and electric energy, as well as for utilization of fission product gamma-radiation energy. The installation's main characteristics are given in Table 1.

The research and experimental design work were in progress in 1982-1984 to substantiate a design of this installation.

The complex of calculated research was carried out to specify heat-physical parameters of a reactor taking into account additional experimental data and investigations on neutron-physical characteristics of the in-reactor control system (IRCS). The space-energy distribution of neutron fluxes in a graphite stacking, a pressure vessel and other frames of a reactor was made more precise. A maximum neutron fluence with energy E > 0.5 MeV on the reactor vessel does not exceed maximum quantity and amounts to ~ 6,5 \times 10^{12} n/cm² for the 30 - year operation.

The significant stage of work was performed to prove the graphite stacking workability. The calculated research of strength characteristics carried out under conditions of real temperatures and neutron fluences have demonstrated that the graphite used in RBMK reactors can be employed to create a stacking (as a whole). Exceptions are pylons of the reactor core central part and a number of lower reflector units where more strong sorts of graphite should be employed. These investigations will be carried on.

In order to master the process of utilization of the VGR-50 reactor high potential heat energy the modification of high-temperature intermediate heat-exchanger built in the primary circuit was elaborated (Fig.5), from which heat is to be delivered to a thermoconversion apparatus. This device will allow for the steam catalytic conversion of methane to be checked out and master with the use of the nuclear reactor thermal energy.

For the period of time elapsed investigations were continued on controls, safety, fuel elements as well as on substantiation of workability of the primary circuit main equipment on helium benches.

The VG-400 pilot-commercial energy-technological installation.

The installation VG-400 is designed for a complex production of high-potential heat (950°C) and electric energy, making it possible to use it in various energy-consuming branches of industry (Fig.6). The installation main characteristics are given in Table 2.

The research and experimental designing works were in progress in 1983-84, to prove engineering decision of this installation design, based on the VG-400 high-temperature helium reactor with the output of 1060 MW(t)[7].
Corrosion resistance of fuel elements was investigated on the high-temperature helium bench CBH. A high degree of ensuring the reactor installation safety in emergency situations was confirmed by calculated investigations and experimentally.

Experimental investigations on radiation transfer were completed on a mock-up of the VC-400 installation upper end reflector, making it possible to appraise basic calculated procedures on protection, to give recommendations on neutron fluence values and gamma-quantum values on construction elements.

The starting load and the installation operation in transient conditions were substantiated, drawing optimal planning methods; temperature fields in the regulating system rods were determined. Estimated research on emergency seal failure of the CRS channel and at emergency ejection of the central dipping shim rod were carried out. Fulfilment of standardized requirements on ensuring the safety in these operating conditions was shown.

Works are being carried out to substantiate the design of VG-400 prestressed reinforced concrete high-pressure vessel, as well as thermal insulation of intra-vessel elements.

The research on thermal physics, a helium coolant technology, protection, radiation safety and other works were continued to substantiate the installation design. The creation of VG-400 installation will make it possible to accomplish a representative checking and mastering of the main reactor and energy technological equipment which will feature a needed scale primary standardization for energy technological plants of commercial scale.

The BGR-300 pilot-commercial installation with fast neutron reactor

During the past period the experimental designing and research works were continued on the helium breeder which was considered as an alternative to the breeder with sodium coolant. Efforts are concentrated on the development of experimental designing HPP project with an electric capacity of 300 MW with the fast neutron breeder-reactor BGR-300 (Fig.7). The use of rod type fuel elements with a cladding artificial roughness is provided for the reactor core design. In the design a maximum degree use is made of technical decisions based on an experience of elaborating the breeders with sodium coolant in a part of reactor core (fuel, jackets) and the HPGR in a part of the main equipment (steam generators, reinforced concrete vessels, gas blowers, etc.), making it possible, while substantiated experimentally, to focus main attention on mastering the specific questions associated with workability of design elements and materials under conditions of increased temperatures and coolant pressure taking into account actions of fast neutron high fluences.

To ensure safety conditions of the installation's operation, the loop reserving is provided for an auxiliary cooling circuit as well as injecting facilities and feasibility to operate at a natural circulation under emergency situations.

The development of the main equipment and reactor's construction elements are being continued and experimental investigations of single units and elements of a steam generator, PCRV, etc. are being carried out as well.

In 1983-84 investigations were in progress aimed to develop nuclear power plants for distant heat supply (HPP DHS) on the basis of promising HPGR with the output of 2500 MWe (W). Based on the estimated-optimised complex elaborated at I.V.Kurchatov Institute an optimization of schemes, technological and heat energy parameters of such plants was performed, optimal technical-economic characteristics of basic composite elements of the distant heat
supply complexes were determined on the basis of NPPDHS. The analysis of optimal characteristics dependence on NPPDHS's location conditions (a thermal load density, closeness of heat-energy users, an attached electric load, substituted energy source expenditures, etc.) was run.

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7. Koskin Yu.I., Komarov E.V., Kiryushin A.I. et al. The atomic power-technological plant VG-400, the schehe and features, ibid, p.115-118.


Table I

Main characteristics of the installation with reactor VGR-50

<table>
<thead>
<tr>
<th>Description</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Power</td>
<td>50 kW (e.)</td>
</tr>
<tr>
<td>Helium temperature</td>
<td></td>
</tr>
<tr>
<td>- at reactor output</td>
<td>810°C</td>
</tr>
<tr>
<td>- input</td>
<td>280°C</td>
</tr>
<tr>
<td>Helium pressure</td>
<td>4 MPa</td>
</tr>
<tr>
<td>Reactor core size</td>
<td>D/H = 2.8/40 m</td>
</tr>
<tr>
<td>Fuel element</td>
<td>sphere 60 mm</td>
</tr>
<tr>
<td>Fuel elements number</td>
<td>in reactor 125000 pc</td>
</tr>
<tr>
<td></td>
<td>in installation 300000 pc</td>
</tr>
<tr>
<td>Fuel enrichment</td>
<td>21%</td>
</tr>
<tr>
<td>Burnup</td>
<td>100000 MW day/t</td>
</tr>
<tr>
<td>Energy cooling loops number</td>
<td>4</td>
</tr>
<tr>
<td>Radiation circuit gamma-power</td>
<td>300-400 kW</td>
</tr>
<tr>
<td>Steam parameters</td>
<td></td>
</tr>
<tr>
<td>- pressure</td>
<td>9 MPa</td>
</tr>
<tr>
<td>- temperature</td>
<td>539°C</td>
</tr>
<tr>
<td>Reactor pressure vessel</td>
<td>steel</td>
</tr>
<tr>
<td>Table 2. Main characteristics of the Installation VG-400</td>
<td></td>
</tr>
<tr>
<td>----------------------------------------------------------</td>
<td></td>
</tr>
<tr>
<td>Thermal power - 1060 MW</td>
<td></td>
</tr>
<tr>
<td>Helium temperature:</td>
<td></td>
</tr>
<tr>
<td>- at reactor core output - 950°C</td>
<td></td>
</tr>
<tr>
<td>- &quot;&quot; - input - 350°C</td>
<td></td>
</tr>
<tr>
<td>Helium pressure - 5 MPa</td>
<td></td>
</tr>
<tr>
<td>Reactor core volume - 6.4/4.8 m</td>
<td></td>
</tr>
<tr>
<td>Fuel element - sphere 60 mm</td>
<td></td>
</tr>
<tr>
<td>Number of fuel elements in reactor core - 8.10^5 pc</td>
<td></td>
</tr>
<tr>
<td>Initial fuel enrichment - 6.5%</td>
<td></td>
</tr>
<tr>
<td>Burnup - 70,000 MW.day/t</td>
<td></td>
</tr>
<tr>
<td>Fuel lifetime - 320 eff.days</td>
<td></td>
</tr>
<tr>
<td>Cooling loops number - 4</td>
<td></td>
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<tr>
<td>Steam parameters:</td>
<td></td>
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<tr>
<td>- pressure - 17.5 MPa</td>
<td></td>
</tr>
<tr>
<td>- temperature - 535°C</td>
<td></td>
</tr>
<tr>
<td>Reactor vessel - prestressed reinforced concrete</td>
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</table>

<table>
<thead>
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<th>Table 3. Main characteristics of reactor BGR-300</th>
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<tbody>
<tr>
<td>Variant I Variant II</td>
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<tr>
<td>Reactor thermal power, MW</td>
</tr>
<tr>
<td>- helium 810</td>
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<tr>
<td>- helium 810</td>
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<tr>
<td>Coolant pressure, MPa:</td>
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<tr>
<td>- helium 16</td>
</tr>
<tr>
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</tr>
<tr>
<td>Coolant temperature:</td>
</tr>
<tr>
<td>- at reactor input, °C</td>
</tr>
<tr>
<td>- output, °C</td>
</tr>
<tr>
<td>- at reactor input, °C</td>
</tr>
<tr>
<td>- output, °C</td>
</tr>
<tr>
<td>Reactor core volume, m^3</td>
</tr>
<tr>
<td>- 2.31</td>
</tr>
<tr>
<td>- 1.44</td>
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<tr>
<td>Fissile isotopes load, kg</td>
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<tr>
<td>- 948</td>
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<td>- 753</td>
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<tr>
<td>Fuel element type</td>
</tr>
<tr>
<td>- rod</td>
</tr>
<tr>
<td>- rod</td>
</tr>
<tr>
<td>Fuel element diameter, mm</td>
</tr>
<tr>
<td>- 6.9</td>
</tr>
<tr>
<td>- 6.9</td>
</tr>
<tr>
<td>Fuel element cladding maximum temperature, °C</td>
</tr>
<tr>
<td>- 800</td>
</tr>
<tr>
<td>- 800</td>
</tr>
<tr>
<td>Fuel lifetime, eff.days</td>
</tr>
<tr>
<td>- 750</td>
</tr>
<tr>
<td>- 615</td>
</tr>
<tr>
<td>Burnup, %</td>
</tr>
<tr>
<td>- 10</td>
</tr>
<tr>
<td>- 10</td>
</tr>
<tr>
<td>Reproduction coefficient</td>
</tr>
<tr>
<td>- 1.5</td>
</tr>
<tr>
<td>- 1.61</td>
</tr>
<tr>
<td>Doubling time, years</td>
</tr>
<tr>
<td>- 10</td>
</tr>
<tr>
<td>- 7</td>
</tr>
</tbody>
</table>
Fig. 1. The dependence of a dose power of the primary circuit equipment on the f.e. tightness degree.
1. Steam generators boxes
2. Rooms for mechanism of distribution, reflector loading, rejection and injection.
3. Reflector canyon
4. Mechanism to control of spherical element
   - 7-day exposure
   - 30-day exposure
   - 7-day exposure after steam generator decontamination

Fig. 2. The dependence of Xe-133 leakage on the fuel fraction in damaged micro fuel elements
1 - Xe-133 relative leakage
2 - fuel fraction in damaged micro fuel elements

Fig. 3. The CTB schematic:
1 - working section; 2 - composite tube; 3 - heater;
4 - purification system; 5 - gas blower; 6 - cooler;
7 - test pipe; 8 - regenerator; 9 - SGM heater; 10 - cooler;
11 - separator; 12 - mixer; 13 - gas heater; 14 - compressor;
15 - filter; 16 - analyzer; 17 - steam generator; 18 - pump;
19 - accumulator; 20 - distillator.
a - conversion gas;
b - natural gas;
c - water

Fig. 4. CTB general view

Fig. 5. High-temperature helium heat-exchanger

Fig. 6. Reactor VG-400:
1 - reactor core; 2 - discharge channel; 3 - RCV;
4 - main gas blower
5 - steam generator; 6 - working ionization chamber;
7 - CRS side actuator; 8 - CRS dipped actuator;
9 - starting up ionization chamber; 10 - loading tubes;
11 - high-temperature intermediate heat-exchanger;
12 - prestressing system

Fig. 7. Reactor installation BGR-300
1 - main gas blower; 2 - steam generator;
3 - reactor vessel; 4 - reactor cover; 5 - locking unit;
6 - heat exchanger; 7 - reactor core; 8 - auxiliary gas blower; 9 - CRS drive.
Рис. 1. Зависимость мощности дозн на оборудовании I контура от степени герметичности телов:
1. Боксы III
2. Насосные механизмы различных, нагревающие охладители, отбирающие в нагнетатели
3. Нагреваемые баки
4. Механизм контроля НЗП

- выдержка 7 суток
- выдержка 30 суток
- выдержка 7 суток после демонтажа III

Рис. 2. Зависимость утечки Xe-133 от доли топлива в поврежденных микротелах

Рис. 3. Принципиальная схема завершенного стенда XСУ:
1 - рабочий участок; 2 - кол-труба; 3 - нагреватель; 4 - система очистки; 5 - газодувка;
6 - коллектор; 7 - турник тел; 8 - регенератор; 9 - высоконапорный шнур; 10 - конденсатор;
11 - спектрометр; 12 - смеситель; 13 - подогреватель газа; 14 - компрессор; 15 - фильтр;
16 - анализатор; 17 - парогенератор; 18 - насос; 19 - насос; 20 - дестиллятор

Доля топлива в поврежденных микротелах, %
Рис 4. Общий вид стенда Н33

Рис 5.
1. Активная зона
2. Канал разгрузочный
3. Корпус железобетонный
4. Главная газодувка
5. Парогенератор
6. Позиционная камера рабочая
7. Исполнительный механизм СУЗ боковой
8. Исполнительный механизм СУЗ изгружной
9. Позиционная камера пусковая
10. Труба загрузочная
11. Высокотемпературный промежуточный теплообменник
12. Система циркуляционного охлаждения
Рис. 7. Реакторная установка БГР-300:
1 - газодувка основная; 2 - парогенератор;
3 - корпус реактора; 4 - крышка реактора;
5 - запорное устройство; 6 - теплообменник;
7 - активная зона; 8 - газодувка вспомогательная;
9 - привод СУЗ.

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