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Gas Cooled Fast Reactor Background, Facilities, Industries, and Programmes

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Abstract

This report was prepared at the request of the OECD-NEA Coordinating Group on Gas Cooled Fast Reactor Development and it represents a contribution (Vol.II) to the jointly sponsored Vol.I (GCFR Status Report). After a chapter on background with a brief description of the early studies and the activities in the various countries involved in the collaborative programme (Austria, Belgium, France, Germany, Japan, Sweden, Switzerland, United Kingdom and United States), the report describes the facilities available in those countries and at the Gas Breeder Reactor Association and the industrial capabilities relevant to the GCFR. Finally the programmes are described briefly with programme charts, conclusions and recommendations are given.

Schneller Gasgekühlter Reaktor: Entwicklungsgeschichte, Anlagen, Industrie und Programme

Kurzfassung

Dieser Bericht wurde auf Anforderung der OECD-NEA Coordinating Group on Gas Cooled Fast Reactor Development erstellt und ist ein Beitrag (Band II) des gemeinsam getragenen Bandes I (Schneller Gasgekühlter Reaktor - Statusbericht). Einem Kapitel über die Geschichte mit Kurzbeschreibung der frühen Untersuchungen und der Tätigkeit in den verschiedenen, am Kooperationsprogramm beteiligten Ländern (Belgien, Deutschland, Frankreich, Großbritannien, Japan, Österreich, Schweden, Schweiz, Vereinigte Staaten von Amerika) folgt eine Beschreibung der in diesen Ländern sowie bei der Gas Breeder Reactor Association verfügbaren Anlagen und der für schnelle gasgekühlte Reaktoren in Frage kommenden Industriekapazitäten. Schließlich werden die Programme anhand von Tabellen kurz beschrieben, Schlußfolgerungen gezogen und Empfehlungen ausgesprochen.

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I. BACKGROUND

1. Introduction

Fermi and Zinn started already in 1944 to consider the possibility of using fast breeder reactors, capable to increase the uranium energy reserve of the world by a factor 50 in comparison with the case of using thermal converter reactors only. The beginning of construction of the first fast reactor, Clementine, in Los Alamos, was 1946. The reactor was cooled by mercury. The second fast reactor, EBR1 in Idaho, was started in 1949. The coolant was NaK. These reactors and the next following had metallic fuel and the core power was relatively small. This lead to very high power densities and small coolant passages in the core. In these conditions, and considering the technological development of that time, there was no other possibility as to choose a liquid metal as coolant. Water was excluded for neutron thermalisation reasons and a gas looked as too a poor heat transfer medium to be able to cool a very small core with tremendous power densities.

At the beginning of the sixties however it was found that oxide fuel was better than metal, due to the experience gained with Light Water Reactors, which showed that with oxide fuel it was possible to reach higher burn-ups, the fuel could withstand higher temperatures, and it was more compatible with the coolant and the cladding. First BR5 in Russia and Rapsodie in France used UO_2-PuO_2 as fuel. The thermal conductivity of the oxide fuel is much less of that of the fuel metal alloys, thus the linear power rating possible with oxide fuel is also considerably smaller than with metal fuel. The use of worse coolants than liquid metals is then possible.

The power and the size of the cores of the modern reactors has increased steadily. It is now a generally recognized fact that a reactor power plant can be economical only if it is of very large size, at least greater than 500 MWe and may be even as high as 1000-2000 MWe. Greater core sizes allow more space for coolant passages, and this fact also tends to favour worse coolants. Indeed the pumping power required to cool the core is inversely proportional to the fifth power of the size of the coolant channels.

At the same time the development of gas cooled reactors - Magnox and AGR's in England, Magnox reactors in France, High Temperature Reactors in U.S.A., Germany and England - showed that high pressure gas can be used as coolant of the core of thermal reactors. An increase in pressure is very effective in improving the heat transfer properties of a coolant gas. Indeed the thermal performance of a coolant, defined as the ratio of the extracted heat to the required pumping power, is proportional to the square of the gas pressure.

Especially important for the increase of the gas pressure was the development of the prestressed concrete pressure vessels. These have been initially developed in France mainly due to difficulties and extra costs of welding, and afterwards annealing, very thick steel vessels on the site. Since then, both the size and the working pressure of the concrete vessels have increased steadily in France and in England, mainly for CO_2 cooled thermal reactors of the Magnox or AGR type. In Germany the THTR (Thorium-Hochtemperaturreaktor) is presently under construction. The primary helium coolant circuit of this prototype reactor of 300 MWe power is contained in a concrete pressure vessel of 16 meters inner diameter. The helium working pressure is 40 Atms. In the United States the construction of the 330 MWe High Temperature Reactor prototype of Fort St. Vrain is completed. The helium working pressure is 50 Atms. A 1:3 scale model of a concrete pressure vessel with 2.5 meters inner diameter has been built in Germany /1/ and tested successfully at full pressure (100 Atms) and temperature (300°C). In Sweden a 1:3.5 scale model of a concrete pressure vessel with 2 meters inner diameter and for a maximum working pressure of 85 Atms /2/ has been subjected to tests at temperatures of up to 300°C, as well as to cold pressure tests up to 215 Atms without any damage /3/. This model has as an interesting feature, a large concrete removable lid, which is being proposed both for the General Atomic, the Gas Breeder Reactor Association and the KWU designs of a Gas Cooled Fast Reactor.

The concrete pressure vessels for big dimensions and high pressures can be made considerably safer than steel vessels. Their enormous mass makes a sudden catastrophic failure extremely unlikely. The steel tendons, which take up all the tension stresses, are made highly redundant, can be checked, tested and, if required, replaced during or after reactor operation. The failure characteristics of a concrete vessel are such that the decrease of pressure due to leakage through craks in the wall is very slow. Indeed, once the pressure in the vessel has decreased, the tendons subjected to less stress close up the bigger crackings in the concrete. The tendons are designed to withstand an accidental condition with fully pressurized concrete cracks and, in recent designs for very high pressures, a venting system in the concrete wall to detect and reduce this accidental condition has been proposed. Finally, the inner steel liner, which makes the concrete vessel leak tight, is kept under compression only and at low temperatures and low thermal gradients by means of a thermal insulation and a water cooling system.

Another technological improvement coming from the development of the Advanced Gas Reactor is the application of artificial roughening to the surface of the fuel element pin, in particualar the use of partially roughened pins, originally suggested by Fortescue for the Gas Breeder /4/ and adopted for the AGR type power station of Hinkley Point B in England /5/. Partial roughening allows a considerable increase in power density in the core and/or a reduction in the required pumping power. Rough surfaces are only present in a relatively short axial portion of the fuel pins where wall temperatures are the highest (about 3/4 of the core length, which means about 35% of the total pin length), thus avoiding supplementary pressure drops where they are not required.

While the early attempts started towards the end of the second world war in the United States dictated the choice of the coolant of a Fast Breeder Reactor, the only practical possibility being at that time a liquid metal, subsequent technological improvements have made the use of a gas as a fast reactor a much more real possibility. These improvements originated by the development of light water reactors (oxide fuel), of sodium cooled fast reactors (development of thin fuel pins and of subassemblies) and of gas cooled reactors (prestressed concrete pressure vessel, artificial roughening of fuel pin surfaces). But that early choice influenced the research and development programmes of all technologically advanced nations, which are now based mainly on the Sodium Cooled Fast Breeder Reactor.

2. Early Studies

One of the first studies performed in the frame of the Karlsruhe Fast Breeder Project was concerned with helium-cooled fast breeders /6,7/. One of the main results, which were reported briefly at the 1963 Argonne Conference /8/, was that indeed high ratings of the order of 0.5 to 1 MWth/kg fissile material needed for fast breeders could be attained.

At the 1964 Geneva Conference Fortescue and coworkers from GGA published the results of their studies of a GCFR of 450 MWe. The reactor was helium-cooled at 68 Atms and the oxide fuel was contained in stainless steel clad pins with artificial roughness on the surface to improve the heat transfer between pin and helium coolant /4/.

In October 1965 Dalle Donne published a comparison between helium, CO_2 and superheated steam as coolants of a large fast reactor /9/. The main conclusions of this study were that, although steam is a better heat transfer medium, helium- and CO_2-cooled reactors were better breeders and, with sufficiently high gas pressures ($\geq$ 70 Atms), reasonable performances could be obtained. Furthermore, while the coolant void coefficients of He and CO_2 were positive but always below one dollar for pressures below 100 Atms, the void coefficients with steam cooling were positive and considerably larger (between 5 and 9 dollars).

In 1967 Wirtz presented the conclusions of some preliminary studies on gas cooled fast reactors at the 3rd FORATOM Congress in London /10/. After twelve years the main conclusions of the paper remain still valid, namely: "The idea of extrapolating a high temperature helium cooled thermal reactor to a high temperature helium cooled fast reactor seems appealing. Many reactor components are practically unchanged, the core of course is different, and the helium pressure is considerably higher, with all the problems that go with it". "If one assumes, that, starting from a certain date, the majority of reactors built will be fast, there is no reason to think that only one type of fast reactor will be constructed, like there is not only one type of thermal reactor being made now". "A gas breeder seems to have a lot of potential and seems to be the best reactor in the long run........ Fuel costs appear to be comparable to those of sodium breeders, and capital costs even lower than those of a steam breeder". After this paper at the Foratom Congress the interest in gas cooled fast reactors was raised again in Karlsruhe. New technical improvement were considered, such as the feasibility of large prestressed concrete pressure vessels for high pressures (100 Atms), the use of partially roughened fuel element surfaces, the development of new vanadium alloys with good creep properties under fast flux irradiation at high temperatures and the possibility of using gas turbine cycles /11/. Furthermore in 1967 various studies were performed on gas breeders in Europe. The Belgian firm

Belgonucléaire performed a study on a CO_2-cooled fast reactor with CO_2 gas turbine /12,13/. A study was performed in Sweden as well, with helium as coolant and steam turbines /14/. In the meantime, the Gulf group had continued its studies on the gas breeder /15-26,28/. Some of these were performed in collaboration with the Swiss Federal Institute of Reactor Research /27/.

In 1967 the USAEC asked the Oak Ridge National Laboratory with the assistance of the Argonne, Los Alamos, and Pacific Northwest Laboratories and of the American firms Babcock + Wilcox, General Electric, Gulf General Atomic, and Westinghouse to perform a study on the alternate (to sodium) coolants for fast breeder reactors. The main results of these studies have been published in 1968 and 1969 /29, 30/. The main point of the conclusion was: "On the basis of the design evaluated and the combined criteria of low power consts and good breeding capability, GCFR's have the highest potential of the concepts considered. Steam-cooled reactors, on the other hand, suffer either from higher power costs (85 and 180 Atms SCBR's) or low breeding ratio (250 Atms SCBR)".

In 1968 two specialist teams were set up by the European Nuclear Energy Agency to evaluate the merits of steam and gas as alternative coolants to sodium for a fast breeder reactor. The results of these studies have been published /31,32/. The ENEA Specialist Group, which met in Winfrith to assess gas cooling, examined the proposals of GCFR's, mainly those of the GGA, Sweden, Karlsruhe, and Belgonucleaire groups, which have been mentioned above, and in addition a gas-cooled fast reactor with coated particle fuel proposed by the UKAEA, which had not yet been reported in the literature up to that time and which was described in two papers later in 1968 /33,34/. It was not possible to reach an agreement in the conclusions of the Working Team, which had to evaluate the two studies on gas and steam in comparison with sodium as coolant of large fast power reactors. One body of opinion held that the development of an alternative coolant was admissible only as a back-up solution in the event of difficulties with the large-scale application of sodium technology. An equally strong body of opinion held that gas cooling had ample scope for sharing the future fast reactor market with sodium and that there was merit in maintaining the principle of choice, which has evolved in the present-day thermal reactor market. This latter conclusion was confirmed by a subsequent Swedish study /45/. Nor further interest on steam cooling was shown at that time by any country participating at that study.

In April 1969, the Steering Committee for Nuclear Energy of the Organisation for Economic Co-operation and Development set up a Working Group on Gas-Cooled Fast Reactors with the objective of exchanging information on current activities in the field of GCFRs. The OECD European Nuclear Energy Agency Working Group, whose membership was open to all countries interested or potentially interested, met four times (May 1969, December 1969, November 1970), and September 1971) with the participation of the following countries and organisations: Austria, Belgium, France, the Federal Republic of Germany, Italy,Japan, the Netherlands, Portugal, Spain, Sweden, Switzerland, the United Kingdom, the United States of America, the Commission of the European Communities, Foratom, and (after December 1969) the Gas Breeder Reactor Association.

Seven of the above-mentioned countries (Austria, Belgium, the Federal Republic of Germany, the Netherlands, Sweden, Switzerland, and the United Kingdom), wishing to have more detailed exchanges of information between countries actively engaged in GCFR development, decided in July 1969 to set up a more restricted group for such exchanges, outside the framework of ENEA. This group, known as the "Zurich Club", composed of national nuclear research organisations, sponsored specialist meetings on fuel, heat transfer, physics, design, and safety.

The Winfrith study of the "Zurich-Club" meeting stimulated the interest and the work in Europe on the GCFR, as it is shown by the many publications from Germany /35,38,39,52,58,59,60,61,62,64,68,69,70,74,75,76,80/, Great Britain /33,34/, Switzerland /36,41,42,44,48,51,67/, Sweden /45/, and Belgium /47/. The work in Germany was centered on the evaluation of various fuels of GCFR's, on safety /35/ and on improvement of the neutron physics calculations with the objective to obtain more information on reactivity coefficients (void, steam inleakage, etc.) /39,70,75/. Originally the reference design was based on fuel pins clad in an especially developed vanadium alloy (V,3Ti,1Si), which allowed a maximum clad temperature of 850^{o}C and a helium temperature of 700^{o}C. The helium was flowing directly to gas turbines /52/. Design studies on the gas turbine circuit connected with a GCFR showed that this concept is feasible and the dimensions of the components reasonable (1000 MWe turbine: length 25 m, maximum outer diameter: 5.5 m, recuperative heat exchanger: 6 units, length: 18 m outer diameter: 4.4 m) /68,74/. Lately, however, experimental investigations have shown that the oxide fuel would, at high temperatures and in presence of temperature gradients in the fuel, oxidate the vanadium cladding unduly /69,80/. Vanadium based cladding would therefore be compatible with oxide fuel only in presence of a suitable oxygen getter in the fuel or, perhaps with carbide fuel.

The work in Great Britain was based on a GCFR with ceramic coated particles /33,34/. These coated particles have been originally developed for High Temperature (thermal) Gas-cooled Reactors. For fast reactors the pyrolitic graphite cannot be used as fuel cladding material due to lack of dimensional stabiliby in presence of large fast fluences and high temperatures. Silicon carbide was proposed in its place. Coated particles with pyrolitic SiC outer coating for GCFR application were developed and tested. The problems (pressure distribution in the fuel element, mechanical stresses, central ceramic porous tube) connected with the fuel element itself, were recognized, but not fully tackled.

The Swiss Federal Institute for Reactor Research since 1968 was mainly involved in the study of GCFR's with direct cycle helium turbine at relatively moderate gas temperatures (600^{o}C), obtainable possibly with steel clad pins /36,41,44,51/. In Sweden a rather detailed comparison study between helium, steam, and sodium as coolants of a Fast Reactor was performed /45/, while in Belgium the accent was on a GCFR with CO_2 cooling and direct cycle gas turbine /47/.

3. The German Gas Breeder Memorandum

In August 1969 the German Federal Ministry for Education and Science requested the two nuclear centers at Karlsruhe and Jülich to prepare a study on the feasibility and the economics of a GCFR. This study (the so-called "Gas Breeder Memorandum") was carried out by the two centers with the collaboration of the German nuclear industry, which included the following companies: AEG, BBC, BBK, GHH, Krupp and Siemens. The Gas Breeder Memorandum has been published /58/. Summaries of it were presented at the Bonn Reaktortagung of 1971 /59,60,61/. The study was performed by five working groups (fuel elements, physical criteria, components, safety, economics). Three concepts were chosen as representative of the main possible options:

a) GCFR with steam turbine, oxide fuel in steel clad pins ("vented fuel"), primary system integrated in prestressed concrete pressure vessel (this concept is based on the GGA concept /18,26,46/)

b) GCFR with gas turbine, oxide fuel in vanadium pins ("strong clad") (this concept is based on the Karlsruhe concept /52,68/).

c) GCFR with steam turbine, oxide fuel in coated particle form (this concept is based on the UKAEA concept /33,34/).

These alternatives were calculated again in the context of the study based on consistent assumptions and methods. The heat transfer correlations used were the same, and so was the method to calculate the hot spots in the core. In all the cases the fuel density was assumed to be 84% of theoretical and the main discharge burn-up 75000 MWD/t. The nuclear calculations were performed with the then available cross section set of Karlsruhe, the so-called MOXTOT set. The main results of these calculations are listed in Table I together with the data of an advanced sodium breeder and a steam-cooled fast reactor, which had been calculated with similar assumptions.

The study came to the conclusion that the GCFR with steel clad vented fuel pins was the type with the minimum amount of required further development work, especially because the fuel element could be based on the current work for the sodium breeder and the reactor components on the development of the High Temperature Thermal Reactor. On the other hand, the reactor offered a performance comparable to that of a sodium-cooled reactor with probably smaller electricity generating costs. The calculated electricity generating costs of steam were also favourable, but the plutonium doubling time appeared to be too high.

The conclusions of the German study were endorsed by the ENEA Working Group on Gas-Cooled Fast Reactors at its November 1970 meeting.

4. The Gas Breeder Reactor Association

In December 1969, the European Association for the Gas-cooled Breeder Reactor [in short Gas Breeder Reactor Association : GBRA] was established by a group of European industrial companies.
The Association set up an engineering working team which was located at its headquarters in Brussels.

During the first two years, alternative coolants [He and CO_2] and fuel element designs [pins and coated particles] were examined on the basis of three different 1000 Mwe designs with steam turbine cycle and primary circuit integrated in prestressed concrete pressure vessel :

- GBR 1 : based on steel-clad mixed oxide vented fuel pins, cooled by helium [81]
- GBR 2 : based on silicon-carbide coated fuel particles, cooled by helium [82]
- GBR 3 : based on silicon-carbide coated fuel particles, cooled by CO_2 [82]

The three designs are summarized in [83] and their main characteristics are given in Table II.

GBRA also performed comparative cost calculations [84], [85], [86], [87], [88], [89]. The main conclusions of these were that a GCFR with fuel pins would have the same capital costs as that of a Thermal High Temperature Reactor, while the helium and CO_2 reactors with coated particles would have 7 % and 9 % lower costs respectively. However, the fuel-cycle costs with coated particles would be higher than with fuel pins due to the lower doubling time and this would compensate almost completely the gains in capital cost.

Moreover, it later appeared that a coated particle and the corresponding fuel assembly, designed for a high fast neutron fluence were very difficult to develop and should be considered as long term proposals.

Since 1972 therefore, the GBRA effort has been primarily devoted to the study of a vented pin, helium cooled design : GBR 4 : a 1200 Mwe commercial reference design aimed at assessing all questions related with design, performance, safety, economics, demonstration plant and R & D programme definition.

In 1972, GBRA was invited by the NEA participating members to represent the European industrial design activity and agreed to integrate its programme with that of NEA.

Since that time, GBRA has performed studies and generated reference documents in the following fields :

- design [90]
- performance [91], [92]
- safety [92],[94], [95]
- economics[96], [97]
- R & D [98],[99]
- case for the GBR [100]

The main characteristics of GBR 4 are given in Tabelle III. This reactor, moreover, can be adapted to various types of fuel strategies and its rating can be improved if required as shown on Table IV. Fig.1 and 2 show a vertical section of the GBR 4 Nuclear Stream Supply System and the GBR 4 Emergency Cooling System respectively.

GBRA also performed a comparison of GBR 4 with a LMFBR calculated with consistent assumptions. The main results are that the LMFBR has a Pu doubling time and a breeding gain of 18 years and 0.2 respectively, against 12 years and 0.4 for GBR 4 [101].

The safety analyses performed by the GBRA were submitted to an ad-hoc group of experts gathered by the Commission of the European Communities in order to obtain on a Community level a first assessment of the GBR safety. The group of safety experts did not identify any fundamental reason which would prevent a Gas-cooled Breeder Reactor like GBR 4 achieving a satisfactory safety status.

Since 1972, the GBRA signed with the Joint Research Centre of Euratom at Ispra three contracts on key aspects of the GBR development :

- Ability of a PCV to contain a nuclear excursion
- Reliability of a PCV
- Wear and friction between pins and support grids in a GBR assembly.

There have also been agreements for information exchange between GBRA and General Atomic Company, San Diego, USA.

Reports on the Association's activities have regularly been presented to the ENEA Worling Group on Gas-Cooled Fast Reactors and Co-ordinating Group on Gas-Cooled Fast Reactor Development.

5. OECD/NEA Co-ordinating Group on Gas-Cooled Fast Reactor Development

(ENEA was transformed into NEA in April 1972)

At its September 1971 meeting, the ENEA Working Group on Gas-Cooled Fast Reactors recommended to the OECD Steering Committee for Nuclear Energy to set up a Co-ordinating Group on Gas-Cooled Fast Reactor Development. Such a group was created in October 1971, with initial membership of the governments of Austria, Belgium, the Federal Republic of Germany, the Netherlands, Sweden, Switzerland and the United Kingdom. The Co-ordinating Group has been joined at later dates by France, Japan, the United States of America, and the Commission of the European Communities. The Gas Breeder Reactor Association was invited to participate in the work of the Co-ordinating Group.

The role of the Group is to facilitate co-ordinating of the work of the Member countries and organisations in a collaborative programme, and to exchange information relevant to gas-cooled fast reactors. In order to achieve the first objective, a Co-ordinator is appointed by the Group to advise on distribution of work amongst participants so as to secure the most effective use of available resources, while information exchange takes place mainly through specialist meetings. This coherent programme covers the majority of the work required for clearing up the questions related to feasibility of the fuel elements, plant safety and design, and component development.

6. Activities in Austria

Austria, represented by Österreichische Studiengesellschaft für Atomenergie, joined the OECD-NEA GCFR development program in the field of particle fuel element technology and component development work, in particular the development of a PCPV.

6.1 Particle Fuel Element Technology

Fabrication studies of GCFR particles with alternative outer coating instead of SiC.

Out-of-pile study of the effect of an oxidizing coolant (CO_2) on cracked GCFR particles.

Out-of-pile studies on the compatibility of broken GCFR particles at high burn-up on adjacent SiC coated particles.

Experimental and theoretical study on the pressure build-up in GCFR particles.

Measurement of thermal expansion coefficient of SiC.

This work was discontinued by end of 1974.

6.2 Prestressed Concrete Pressure Vessel /102 - 108/

One of the main components of the GCFR is the prestressed concrete pressure vessel. To increase the operational safety and economy of PCPVs the development of a PCPV with hot liner and ajustable wall temperature was made to the central point of the joint R & D-Project of the Austrian Industry and the Österreichischen Studiengesellschaft für Atomenergie - "Prestressed Concrete Pressure Vessel - High Temperature Helium Test Rig". The development is based on extensive analysis of possible failures and accidents of PCPVs with cold liner and shall offer solution for following problems and requirements:

- inspection of the liner
- repair of the liner
- location and limitation of leaks
- sufficiently high number of allowable operating cycles during life time according to valid regulations.

This can be achieved by using a hot liner without inner insulation, and by limitation of the stress in the liner during operation to elastic compression, by adapting the adjustable wall temperature to the operating conditions of the liner, by the selection of a suitable liner material and the development of corresponding concretes. Leak detection and limitation can be achieved by a venting system just behind the liner and a steel leak barrier between insulating and structural concrete.

According to this concept a large scale prototype vessel was built at the Research Center Seibersdorf.

The main values are:

overall diameter	3.6 m
internal diameter	1.5 m
overall height	12 m
pressure	100 bar
temperature	300° C

The vessel has an upper steel lid which is constructed in the way of a removable prestressed concrete cover and several axial and radial penetrations. The testing of the vesssl has already been started.

To demonstrate the application of this concept to a PCPV of a GCFR a study is carried out by Reaktorbau Forschungs- und Baugesellschaft and VÖRST-ALPINE in collaboration with KWU.

7. Activities in Belgium

The irradiation of the German 12-rod vented pin bundle is performed in the Belgian reactor BR2 in Mol with the collaboration of the CEN at Mol. The main responsibility of CEN were the nuclear calculations relative to the irradiation experiment. See also 9.1.

8. Activities in France

Coated particles for GCFR application have been manufactured and tested in Rapsodie. Models of coated particle fuel assemblies have been constructed. Feasibility studies of single vented pin tests in Rapsodie and of full scale fuel elements in the helium loop Carmen II have been performed.

9. Activities in Germany

In 1971 the two German nuclear centers at Karlsruhe and Jülich agreed on a joint research and development program based on the conclusions of the Gas Breeder Memorandum. The limited funds availably are concentrated on the reference design concept with helium cooling, steel clad vented pins, oxide fuel and steam turbine cycle. The main activities within this program are:

- A joint irradiation test of the Jülich Nuclear Center and the German firm KWU, with the collaboration of the Karlsruhe Nuclear Center and of the Belgian Nuclear Center at Mol, of a 12 vented pin bundle in the Belgian reactor BR2.

- A joint study of the Karlsruhe Nuclear Center and KWU on the design and safety asepcts of a 1000 MWe GCFR it steam turbine cycle, integrated primary helium circuit and vented steel clad fuel pins.

Another major item is the heat transfer work in Karlsruhe. The Heat Transfer Laboratory of the Institute of Neutron Physics and Reactor Engineering of the Karlsruhe Nuclear Center is performing since 1963 research covering many aspects of the heat transfer with gas cooling, especially heat transfer with pins with artificially rough surfaces (see for istance references /109-118/). As mentioned earlier in the paper, heat transfer is much more important for a GCFR than for a LMFBR, because typically the temperature difference between fuel pins surface and coolant is of the order of $10^{o}C$ for a LMFBR and it can be up to 20 times as much as for a GCFR. This has as a consequence that for a GCFR it is necessary to know the heat transfer coefficient with considerable more precision, if one wants to avoid large uncertainties in the fuel pin clad temperature prediction. Furthermore the thermal performance of the presently developed "two dimensional" roughness ribs is only one fourth of the maximum theoretically obtainable, which shows that the research work on rough surfaces can lead to further great improvements. This could be achieved for istance by the use of "three dimensional" roughness ribs, for two types of which very promising results have been obtained /119/.

KWU has an information exchange agreement in the field of GCFR's with the U.S. firm General Atomic Company. Similar tripartite agreements have been recently signed between Karlsruhe, KWU and GA in the field of safety and between Jülich, KWU and GA for the BR2 irradiation experiment.

9.1 The BR2 Irradiation

The objective of this irradiation experiment in the BR2-reactor in Mol, Belgium, is to provide information on in-pile behaviour of a fuel element pin cluster, especially as far as two major points are concerned, which are not investigated within the LMFBR program. Namely

a) the in-pile behaviour of the pin venting system

b) the in-pile behaviour of the rough and smooth pin surfaces and of the spacer grids in a relatively dry helium atmosphere and in presence of temperature and power variations.

Fig.3 shows the test fuel element and illustrates the functioning principle of the venting system to a separate helium circuit and a fission gas plant.

Table V shows the main data of the test fuel element and of the helium loop. More detailed information is given in reference /120/.
Experiments on an electrically heated mockup fuel element have been successfully performed in the High Pressure Helium Loop of the Karlsruhe Nuclear Center /115/. These experiments have allowed together with the computer code SAGAPØ /115/ the correct prediction of the bundle dimensions and of temperature and pressure distributions in the in-pile experiment.
First irradiations in the BR2 reactor with a dummy fuel element (HELM1: γ-heating only) and with enriched uranium as fuel (HELM2) have been sucessfully performed. The HELM3 irradiation experiment (U-enrichment 75 % and 93%, Pu/U = 15%) was started in April 1978. Because of the beryllium-matrix replacement in the BR2 reactor the HELM3 experiment has been interrupted at the end of 1978. A burn-up of 28000 MKd/t has been so far achieved. It is hoped to continue this experiment in 1980.

9.2 GfK-KWU Design and Safety Studies

These studies have been reported more extensively in references /121/ to /123/. Here only the main results will be reported.

Fig. 4 shows a vertical section through the Nuclear Steam Supply System of the 1000 MWe GCFR reference design (GSB-1). The main data of this design are shown in Table VI, those of the NSSS in Table VII. In Table VIII are listed the safety related nuclear characteristics.

The transients experienced during depressurization accidents for various depressurization time constants and containment back pressures are depicted in Fig. 5 . For these studies it was assumed that the circulation speed of the blowers had remained unchanged, that the scram occurs simultaneously with the initiating event and that all loops are available for decay heat removal.

The shortest depressurization time of 100 sec. of Fig.5 corresponds to the breach of the largest penetration of pressure vessel, i.e. the failure of the seal of the steam generator plug, and it is considered as the Design Basis Accident.

Since the fuel pins are pressure-equalized, it is assumed that a maximum clad temperature of about 1200°C can be tolerated before limiting conditions would occur. Associated with a hot spot temperature of 1200°C in the core is a mixed mean reactor outlet temperature of about 1000°C which is tentatively assumed as an acceptable upper once-in-a-lifetime-limit for the boiler structure. Further calculations have shown that during the DBA depressurization accident up to four of the eight main loops can be lost without reaching these limit temperatures.

A detailed reliability analysis for the DBA /124/ lead to the conclusion that the probability that the decay heat cooling system formed by the 8 main coolant loops and the 4 auxiliary loops would not be capable of maintaining the fuel can maximum temperature below 1200°C, is less than 10^{-4} per demand. If we assume that the chance of breaking the seal of a steam generator plug of the GCFR is as small as that of a double ended rupture of a coolant pipe in the PWR system, that is 10^{-4} a^{-1} or less, then we obtain a chance of not meeting the emergency cooling requirements of less than 10^{-8} a^{-1}, which is equivalent to that at present estimated for the PWR in Germany.

In the frame of the GfK-KWU collaborative work the computer code PHAETON2 has been developed in Karlsruhe, which allows to calculate the transients in a GCFR during normal or accidental operations, such as loss of pressure and/or flow taking into account of natural convection in helium. Flow reversals in the core in a originally down-flow core have been calculated with PHAETON2 /125, 126/.

Although up to present time no realistic chain of events has been detected that would lead to accidents beyond the DBA, some work is being carried out in the FRG in the fiel of hypothetical accidents. This was mainly done because similar analyses have been performed for the German LMFBR SNR 300. To have an idea of the reactor response to large hypothetical reactivity insertions, a calculation was performed of the energy release due to a reactivity ramp of 60 $/sec. The energy release calculated with the Karlsruhe disassembly code KADIS was 22 000 MW sec /127/. The program ARES of Interatom allowed the calculation of the stresses caused by this energy release on the concrete pressure vessel. The maximum calculated strain on the prestressing tendons and on the liner was 0.3%, showing that the concrete pressure vessel could withstand this release energy very well. Subsequent calculations, where due account was taken of the effect of helium inside and outside the fuel pins, lead to an energy release value of about 9000 MW sec for this highly improbable accident.

In the frame of the hypothetical accident studies problems associated with handling of gross core melting have been investigated. The analysis of the temperature distribution in a slab of molten GCFR core and blanket material shows that a relatively small fraction of the total decay heat generated can be removed across the lower surface of the melt. The remainder of the heat is radiated off its upper surface. As a result of this analysis it is concluded that it would be very desiderable to protect the internals in the reactor cavity. A mean to do that effectively is the use of an internal core-catcher in the reactor cavity of the concrete pressure vessel, just below the reactor core /128/. Recently a design proposal of a core-catcher based on Borax, which appears to be feasible has been put forward /129/.

Recently KWU, under the sponsorship of GfK Karlsruhe and KFA Jülich, has embarked upon defining an alternate design concept for a commercial-size GCFR that tries to eliminate the concerns with the previous design: The new design concept (Fig. 6 and 7) emphasizes complete access to all reactor internals even if this were to be associated with a cost penalty. The fully access feature is considered almost mandatory if one wishes to minimize the component development program and if one wishes to proceed as rapidly as possible from a demonstration plant to one of commercial size. Key features selected for the current studies are:

- Upflow core in satellite PCRV with bare liner
- Straight line refuelling through a rotating plug
- Complete access to the reactor cavity after unloading of the core and underwater-removal of all internals
- Elevated steam generators to permit effective natural circulation
- Electric blower drive
- Independent heat sink incorporated into main loops
- High moisture content in primary coolant tolerable.

10. Activities in Japan

The GCFR works have been continued at Japan Atomic Energy Research Institute since 1973 when Japan joined the OECD-NEA GCFR development programme.

(1) Materials

The primary emphasis of the activity has been placed on the evaluation of performance reliability of structural metals, Ni-based alloy, exposed to reactor service conditions for prolonged time. The studies include:
- Compatibility of materials with helium environment,
- Creep, fatigue and their interactions in reactor enviroment,
- Study of post-irradiation mechanical properties under the influences of neutron, heat and stress.

(2) Coated particle fuel

In order to find an appropriate fabrication process of ZrC coated fuel particles, the chloride process, the iodide process and the bromide process have been investigated by using alumina microspheres, and the bromide process was chosen as the standard fabrication process.

(3) Thermohydraulics

The prevention techniques of He-laminarisation and transition at low Reynolds number have been studied by means of thermal augmentation techniques. Enhanced heat transfer by roughened cladding surface at low Reynolds number has been demonstrated by parallel channel with wire promoter. A flow visualisation technique using streak line method has been developed to analyse the flow pattern around promotors at low Reynolds number.

(4) Core performance and fuel cycle

Applicability of thorium-cycle to GCFR has been investigated, comparing with uranium-cycle. The study indicated that use of U-233 in the core was not preferable, because of very low breeding gain and very large positive steam entry reactivity effect. A computer code has been developed to calculate composition, radioactivity, decay heat and γ-ray spectrum of a large number of nuclides in fast reactor fuel cycle. Neutron streaming effect on GCFR core performances has been investigated using the experimental results on Na-voided cores of LMFBR installed in FCA at JAERI.

11. Activities in Sweden

11.1 Background

The national research center, Studsvik Energiteknik AB and ASEA-ATOM represent Swedish organisations that have been active in the development of the GCFR. This interest dates back to 1964-69 when alternative fast breeders were assessed by Studsvik in collaboration with the industry, mainly ASEA, see reference /130/. Following the termination of the R&D on the steam-cooled fast breeders based on the domestic water reactor technology, somewhat more emphasis was devoted to the potentially promising GCFR through

- the participation of Studsvik in the activities that resulted in the establishment of the OECD-NEA-GCFR Development Coordination in which Sweden has continued to take part;

- the participation of ASEA-ATOM since 1969 in design studies of the European Gas Breeder Reactor Association (GBRA).

The rather modest GCFR effort which initially varied between 5 and 10 manyears per year has since 1975 been reduced to one manyear/year due to lack of funding.

11.2 Activities

The following activities related to GCFR have been carried out:
- Pin development: creep, swelling and ductility studies of irradiated stainless steel and creep studies of OU_2. 3-pin NaK-capsule irradiation in a thermal reactor.
- Prestressed concrete reactor pressure vessel development: a large model of PCRV for a design pressure of 85 bar has been constructed and failure tests of lids and of bottom slabs as well as liner venting system tests have been performed.

A list of Swedish reports used as contributions to the NEA-GCFR program is given in the References /130 - 171/.

11.3 Programme

Currently one of the tasks of Studsvik Energiteknik AB within the government financed Swedish Energy R & D program is to follow the international development of advanced reactor systems in order to provide the necessary background for national energy policy decisions. In addition a minor R & D effort on FBR safety is jointly financed by the government, utilities and ASEA-ATOM.

Should a viable GCFR-programme be established based on the NEA-GCFR coordination initiative, then most certainly ASEA-ATOM and other Swedish companies with activities in the nuclear field would be interested in participation.

12. Activities in Switzerland

The Swiss GCFR activities are concentrated at the Swiss Federal Institute for Reactor Research at Würenlingen. The main items of this programme are listed below /172/.

12.1 GCFR Core Thermal-Hydraulics

These studies include:

- Rough surface thermal-hydraulics: these refer to measurements in single rough pins contained in a smooth annulus and to the development of transformation methods for the application of the obtained experimental results to bundle geometries.
- Investigation of spacer influence: on rod surface temperature and on pressure drop
- Computer code development: the CLUHET and SCRIMP codes for the heat transfer analysis of rod bundles are under development. The code SCRIMP is available /173/.
- AGATHE-HEX code verification experiment: an electrically heated bundle of 37 rough rods has been tested in high pressure CO_2.

12.2 Experimental Reactor Physics Studies

The zero energy reactor PROTEUS was used during the period April 1972 to April 1979 for a wide range of studies on the neutron physics of plutonium-fuelled GCFR lattices. This work involved the measurement of integral neutron reaction rates, differential neutron spectra and the reactivity worths of a variety of lattice components. Its aim was to check the performance of nuclear data sets and calculation method used in the design of fast breeder reactors.

The experimental programme included:

1) Measurements in a typical GCFR benchmark lattice.

2) Investigation of the effects of specific power reactor features, e.g. measurement of reaction rate distributions in the vicinity of a B_4C control rod.

3) Investigation of the reactivity changes produced by the accidental entry of steam into a GCFR lattice.

4) Construction of a series of lattices with unit K-infinity to check the capture cross-sections of reactor structural materials by means of null reactivity measurements.

5) Measurement of reaction rates and reactivity worths in an axial UO_2 blanket of the fast reactor lattice.

6) Measurement of neutron spectrum and threshold reaction rates at various depths in iron and steel shields placed adjacent to the fast lattice American ENDF/B-4 data set or the British FGL5 set.

7) Investigations concerned with proliferation-resistant nuclear fuel cycles. These involved measurements made in a series of thorium-bearing cores which included uniform lattices and also configurations with central zones on axial blankets of thorium oxide or thorium metal. The results of these various measurements were generally used to check the validity of calculations based on the American ENDF/B-4 data set and the British FGL5 set.

12.3 Nuclear Performance and Safety Studies

Theoretical studies in the core physics area at EIR concern
- the steam entry reactivity effect,
- the flooding of the core for emergency cooling and fuel changing,
- the recycling of actinides, particularly Np-237,
- the performance of alternate fuel cycles.

A thorough analysis of the steam entry reactivity effect of the General Atomic 300 MWe prototype showed that the effect is very sensitive not only to the basic nuclear data (i.e. to data uncertainties) but also to the geometry of the core, the fissile enrichment of the zones, the control absorbers and the burnup. For steam densities in the coolant channels up to 0.03 g/cm^3 the overall effect was calculated to be negative. A regional break-down indicates that core zones without control absorbers can give positive contributions. A considerable fraction of the total effect can be attributed to the negative influence of the blankets.

The flooding of the core for emergency cooling and fuel changing has been studied for a simplified model of a GCFR. To compensate the excess reactivity relatively high absorber concentrations are necessary. The required amount of poisoning is such that the neutron balance in the dry lattice (and thereby the breeding ratio) would be affected noticeably. It is therefore preferable to poison the H_2O coolant rather than the fuel. Considering the performance, the availability and the price, samarium was found to be the most favourable absorber to use. More detailed calculations are planned.

Various studies are concerned with the aspects of recycling Np-237 in fast reactors. A concept resulting from this work includes a GCFR which operates as a "Np-237 burner" and a "Pu-238 breeder". Pu-238 (produced by Np-237 capture) has a possible application in Pu-238 "spiked" fuel elements, which are thought to be more proliferation resistant than ordinary plutonium fuel elements. The study showed that the Np-237 burner Pu-238 breeder has favourable steam entry and burnup characteristics. Work on this modified fuel cycle in the GCFR is being continued.

Further work on special aspects of alternate fuel cycles in the GCFR is in progress. Continuous efforts are being made to validate cross section data and reactor codes with the help of benchmark calculations and comparisons with experiments.

12.4 The Development of Mixed Carbide Fuel

Although not specific to the GCFR the Swiss Federal Institute for Reactor Research, since 1967, has carried out a vigorous programme of development on a mixed carbide fuel which can provide information of relevance to GCFR fuel development. The fuel is produced by a wet-chemical (gelation) technique developed partly at EIR and vibrofilled into fuel pins to give a smeared fuel density of approaching 80 % theoretical. Aspects of the work common to all fuel studies and of use to the GCFR are: the fabrication and handling of Pu containing fuels, development of advanced methods of fuel fabrication using particle concepts, access to and use of irradiation facilities and examination of irradiated fuels and fuel pins, and the development of a sphere-pac fuel performance code. Some preliminary studies have also been made of the use of carbides in the GCFR when it was shown that the full potential of this fuel would be realized only with an advanced high temperature clad material. With such cladding significant improvements of breeding performance are possible.

Production of fuel which is based on an oxide process with the addition of carbon is still on a laboratory scale but studies have commenced on the conceptual design study of a pilot fabrication plant. Irradiation tests have been successfully carried out up to 950 w/cm and 650^{o} clad with stoichiometric fuel and burn-ups of $\sim$7% fima reached with no sign of failure. It is hoped in the future to extend these tests to bundle experiments in realistic fast reactor conditions as well as continuing detailed parameter studies.

13. Activities in the United Kingdom

In the years immediately following the NEA Study of 1968, the UK interest in GCFRs was in a system which could profit from the developing HTR technology on fuel as well as engineering components. Although a steam-raising reactor was taken to be the first objective, there clearly was interest also in the possibilities of higher temperature operation. It was further recognized that CO_2 was a possible alternative coolant, and this was included in the work.

Clearly a major part of the work had to be concerned with fuel development, and specimen particles were produced which were irradiated. The first series of tests was carried out in the R2 reactor at Studsvik with the cooperation of AB Atomenergi and tested the burnup capabilities of the particles. The last series was carried out in RAPSODIE by collaboration with CEA, thus aiming to give some information on damage flux effects, but was limited in scope due to particle failures and also to the reducing interest in the particle version of the GCFR. Neutron damage effects in a GCFR rule out outer graphite coatings and require reliance to be placed on the silicon carbide particle coating retaining its properties. Tests on silicon carbide shells in DFR had given encouraging results, but this line of attack was not followed beyond the initial programme.

In parallel with this development work on particles, compatibility studies were undertaken examining interaction effects between coolant and particle coatings, and between coatings and structural materials, using some cases the technique developed at Harwell of simulating neutron demage by use of fission fragments to enhance the demage rate, as well as tests being made in a VEC.

The consideration of incorporation of particles into fuel assemblies led to study of heat transfer in particle beds and stability of flow, particularly at low flow conditions, to engineering design and fabrication tests of a feasibility nature in collaboration with CEA, and also to a range of safety and circuit activity investigations. It was in the course of this work that the problems of developing a satisfactory fuel assembly arrangement emerged more clearly, and at the end of 1975, when it was also becoming apparent that work in the HTR field was slowing down, the decision was taken to discontinue the examination of the particle version.

Since that time UK studies have looked at the pin-fuelled GCFR. During the whole of the joint programme, of course, there have been interchanges on heat transfer and fluid flow aspects of pin bundles, the UK contribution developing from the programme of work on AGRs. This has been a very active R&D area which, it is believed, has been of considerable mutual benefit. In the materials field, general work on coolant compatibility with both cladding and circuit materials has continued which has helped to define the desirable levels of H_2/H_2O combinations, taking account also of the need to ensure no serious rapid coolant/fuel reactions in the event of a leaky fuel pin developing. More recently results have become available from a series of compatibility and tribological tests carried out under contract in industry covering a range of materials at various temperature and pressure conditions.

A significant part of the work has been on engineering design studies of the pin concept during the last four years. An examination was made of the behaviour during transients of the pin pressure-balancing system to see if there were adverse conditions which could develop causing releases into the main coolant circuit. In addition, the effects of local core blockages were studied, from which it appeared that within a fuel cluster, if a local blockage could form, it could lead to excessively high temperatures. However, it did seem possible that suitable designs of fuel assembly "wrapper" might be evolved which would prevent inter-assembly propagation. As an alternative to the vented-pin system, the capabilities of sealed pins were examined, as it was thought that pins of this type might not only be of interest in themselves, but might be needed to form the "driver" fuel section of a first experimental reactor. This investigation, though showing that sealed pins may be feasible, brought out their performance penalties and other limitations.

Core catchers have been the subject of another investigation. In the initial stages a study was made of alternative principles with the objective of defining the performance requirements. Subsequent work has shown that it appears feasible to meet the technical specification which emerged from this initial study with certain types of core catcher design.

The results of the UK programme in these various fields have all been reported to the appropriate Specialists' Meetings.

14. Activities in the United States

14.1 The General Atomic Company Design of a 300 MWe GCFR Prototype

The American firm General Atomic Company has performed further studies on the GCFR, based on the original suggestion of Fortescue and coworkers. The work was centered on the detail design of two GCFR prototypes of 300 and 750 MWe respectively /174,175/. Subsequently it was decided to choose the 300 MWe design as a reference design (see Fig.8). This reference design is characterized by:

- a primary helium circuit completely contained inside a concrete pressure vessel
- a hanging core with flow downward through the core
- main heat exchangers and blowers, and auxiliary systems contained in pods in the concrete separated from the main reactor cavity
- steam drive circulators
- vented fuel pins.

Extensive safety investigations and discussion with regulatory authorities in the U.S. have been performed on a slightly different previous design of a 300 MWe prototype /176/. These discussions have indicated that there are no principal difficulties regarding the safety of the GCFR prototype and also the areas where further investigations are required. In January 1976 an accident probability analysis and design evaluation of the GCFR 300 MWe prototype was performed at the Massachussetts Institute of Technology, which came to same conclusions /177/. General Atomic has recently decided after exaustive discussions with KWU to change their design. The new design is characterized by upflow through the core, electrically driven main circulators and control rod penetrations in a concrete plug above the core (see Fig.9).

GCFR UTILITY PROGRAM MEMBERS - 1978

INVESTOR-OWNED UTILITIES
- Arizona Public Service
- Baltimore Gas & Electric
- Central Illinois Light
- Cincinnati Gas & Electric
- Green Mountain Power
- Gulf States Utilities
- Illinois Power
- Northeast Utilities
  - The Connecticut Light & Power Co.
  - The Hartford Electric Light Co.
  - Western Massachusetts Electric Co.
  - Holyoke Water Power Co.
- Public Service of Colorado
- Public Service of Oklahoma
- Puget Sound Power & Light
- San Diego Gas & Electric
- Sierra Pacific Power
- Southwestern Public Service
- Union Electric
- Utah Power & Light
- Washington Water Power
- Philadelphia Electric
- Empire State Electric Energy Research (7)
  - Central Hudson Gas & Electric Corp.
  - Consolidated Edison Co. of N.Y., Inc.
  - Long Island Lighting Company
  - New York State Electric & Gas Corp.
  - Niagara Mohawk Power Corp.
  - Orange and Rockland Utilities, Inc.
  - Rochester Gas and Electric Corp.
- East Central Nuclear Group (ECNG) (14)
  - American Electric Power
    - Appalachian Power Company
    - Indiana and Michigan Electric Co.
    - Ohio Power Company
  - Allegheny Power System
    - Monogahela Power Company
    - Potomac Edison Company
    - West Penn Power Company
  - Ohio Edison Company
    - Pennsylvania Power Company
  - Columbus & Southern Ohio Electric Co.
  - Southern Indiana Gas & Electric Co.
  - Cleveland Electric Illuminating Co.
  - Louisville Gas & Electric Company

OVERSEAS UTILITIES
- Denmark: Elsam
- Finland: Imatran Voima
- Sweden: AB Kaernkraft, AKK
  - South Swedish Power
  - Skandinaviska Elverk Voxnan Power
  - Krangede Power
  - Stora Kopparberg Bergvik-Ala
  - Stockholm Energy "Svarthalsforsen" Lanforsen
  - Balforsen Power
  - Gullspang Power
  - Uddeholm
  - Bergslagen United Utilities

MUNICIPAL SYSTEMS
- Tacoma Public Utilities
- Seattle Lighting

FEDERAL, STATE & DISTRICT SYSTEMS
- Guadalupe-Blanco River Authority
- PUD No. 1 of Clark County
- PUD No. 1 of Franklin County

RURAL ELECTRIC COOPERATIVES
- Bailey County
- Bandera
- Cherokee County
- Lamar County
- Lighthouse
- Lynthegar
- Medina
- Surprise Valley
- Tri-County Electric
- Umatilla
- South Texas Electric
  - Jackson
  - Karnes
  - Nueces
  - San Patricio
  - Victoria County
  - Wharton County

14.2 The Helium Breeder Associates

In June 1976, the GCFR Utility Group, which represented about 35% of the US electrical generating capacity, proposed to ERDA (now DOE) a Program Definition and Licensing Phase (PDLP) program for support. The scope of work to be performed under the PDLP is given in Table IX. In November 1976, the Utility Group organized Helium Breeder Associates (HBA) to manage the PDLP activities (see attached list of HBA membership). In October 1977, under ERDA Contract, HBA produced a Gas-Cooled Fast Breeder Reactor Commercialization Study /178/ whose main conclusions are reported down below .

"The following conclusions and recommendations are made by HBA based on the results of the GCFR commercialization study:

1. Based upon current nuclear plant capacity projections and cost information, LWRs will dominate the nuclear plant additions for the remainder of this century, and the GCFR could capture the breeder market during the first two decades of the twenty-first century.

2. External conditions necessary for successful commercialization of the GCFR include (a) a clearly stated national breeder policy which recognizes the role of the breeder in ensuring a viable and long-term nuclear power option; (b) an expeditious licensing process; (c) a commitment to provide the required of an industrial infrastructure capable of supporting commercial deployment.

3. Participants in the GCFR management organization should include U.S. and foreign electric utility companies as well as other organizations in the nuclear industry. National laboratories, vendors, and engineering firms would provide their services as subcontractors to the management organization. The role of the government would be to fund the majority

of the initial phase of the program by multiyear contracts between ERDA and the management organization of end-users and to participate in program definition and monitoring.

4. Three commercialization strategies have been developed to provide a commercial GCFR breeder option. The sequence of plant construction for each strategy is as follows:

   Strategy I: Demonstration → Prototype → Commercial
   Strategy II: Experimental → Prototype → Commercial
   Strategy III: Prototype → Commercial

   Helium Breeder Associates currently favors Strategy I.

   If the GCFR is not successfully commercialized, the U.S. utility industry will not have the option of purchasing a technologically different breeder. <u>The GCFR option increases the likelihood that commercial breeders will be available in the U.S.</u>

5. The GCFR provides a viable breeder option which is technically different from the LMFBR. Because breeders are essential for ensuring long-term nuclear power, the U.S. must develop more than one breeder concept and can afford reasonable programs to do so. It is therefore recommended that development work on the GCFR be concentrated on efforts which would lead to commercial plants.

6. In general, proposed component development programs in support of the 300 MWe(e) demonstration plant are adequate. However, it is recommended that a greater effort be expended in the areas of planning, scheduling, and systems integration for the demonstration plant.

7. A major technical impact on the GCFR program is not expected if the Clinch River breeder reactor (CRBR) program is cancelled, provided the fast flux test facility (FFTF) continues and the reference fuel cycle does not change. The need for a helium loop in the FFTF for possible additional testing of GCFR fuel should be reviewed.

8. A successful GCFR program must clearly reflect the needs of the end-user and have private sector leadership. Long-range working relationships and well defined objectives, program, and priorities must be clearly understood and accepted by all participants. It is recommended

that an organization representing the interests of the end-user manage the entire GCFR program.

9. To ensure program control and flexibility, the program should be funded in multiyear phases. Each phase should have well defined objectives and milestones and be structured so that the participants are committed for the duration of each phase. Thus, the program could be stopped or redirected at the end of each phase without incurring outstanding commitments or risks.

10. To the greatest extent possible, well defined and binding contracts should be drawn up between all participants, including the user management organization, ERDA, the national laboratories, the equipment, and the foreign participants.

11. Operation of the first commercial plant could occur as early as 2010 if there are two preceding plants (Strategies I or II in item 4) or as early as 2000 if there is one preceding plant (Strategy III). As part of Phase I, it is recommended that the strategy selection be reevaluated based on data generated during this phase and the then current status of external factors such as the national energy policy.

12. The national nonproliferation policy requires clarification before its total effect on GCFR commercialization can be assessed.

It is concluded by HBA that development of a GCFR option is mandatory for a viable breeder program in the U.S. and that this can be accomplished by an end-user management organization using a phased program."

14.3 The Department of Energy (DOE).

The Department of Energy of the U.S. Government (formerly AEC and ERDA) has increased in the last few years the yearly budget for the GCFR. One of the main reasons for this increase in interest in the GCFR is the Gas Cooled Reactor Assessment prepared for ERDA by Athur D. Little Jnc, with the Assistance of United Engineers and Constructors Inc., and S.M. Stoller Cooporation/179/. In the conclusions of this study, which are reported below, on the various different gas-cooled reactors, the most favored line was the GCFR.

"GCFR

The Gas-Cooled Fast Reactor is a breeder reactor currently under development at General Atomic. The present development approach is to marry the helium gas cooling technology developed over the last twenty years as presently exemplified in the Fort St. Vrain reactor and the fast reactor fuel technology from the liquid metal cooled Clinch River Reactor. In addition to ERDA support, this program is presently enjoying the support of a large number of utility companies with the Southwest Public Service Company looking towards operation in 1988 of a demonstration plant on their system in West Texas.

The changes that result from substituting gas-cooling for liquid metal cooling promise to improve neutron economy to the point where the technical performance, as reflected in fuel cycle, operating and maintenance costs, and doubling time (a critical consideration) are all reduced. Simultaneously, the removal of the requirement for an intermediate heat exchanger which also results from a substitution of gas-cooling for liquid metal cooling provides potential for capital cost reduction.

Uncertainties in the development program for the reactor include questions relating to requirement for a special fuel test facility and the reliability of the shut-down cooling system. Both of these will have to be addressed in detail and resolved in the development program.

It is anticipated that the LMFBR program will fulfill our requirement for an effective breeder reactor. However, since the satisfaction of this requirement is critical to future national energy supplies, a strong case can be made for measures to insure timely success of the national breeder reactor program. For this reason the gas-cooled fast reactor program should be an integral part of the overall national breeder program as a backup to the LMFBR. GCFR technology should be developed to provide timely access to an effective breeder if the LMFBR program should falter.

Technical assessment indicates that this reactor could not be available for commercialization until the 1990's with a lead plant in place not much before the end of the century. Economic assessment indicates that it would produce power least expensive of all the alternatives considered, although this assessment involves the uncertainities inevitable to the very early development status of the GCFR.

Findings and Recommendations

As a result of the technical and economic assessment of the four gas-cooled reactors carried out under this study, we present the following findings:

1. Gas-cooled reactor technology provides the potential to realize economic, conservation, safety and environmental benefits relative to alternative nuclear and coal fueled electric power plants by about 1987 and in the more distant future. Therefore, it is important that the research, development and demonstration of these concepts be pursued. Those concepts which demonstrate high economic and technical promise will then be in a position to be commercialized if the necessary qualified industrial base for commercialization exists at that time.

2. If gas-cooled reactor development is to continue in the United States, ERDA must formulate a national gas-cooled reactor program and take leadership in funding the program and directing its execution by industry and government. The leadership role has until now been divided among private industry, the federal government, and electric utilities, with the private sector assuming relatively more responsibility and cost than was true for comparable stages of development of light water technology. We see no mechanism by which it will be feasible for the private sector to continue to play this role; the principal problem is that it is difficult, if not impossible, to channel a sufficient stream of the expected benefits of a successful program to compensate private investors on a timely basis for the large outlays and considerable risks which the program entails over a long time horizon.

The U. S. program should be planned and coordinated with those of other nations, such as West Germany, France, Japan and the United Kingdom to bring about information exchange, avoid unnecessary duplication and minimize the total cost of gas-cooled reactor development and commercialization. Close cooperation with West Germany would be particularly beneficial since its gas-cooled reactor program has closely paralleled the U. S. program.

Such a program would have to be flexible in that its magnitude and scope would be importantly influenced by the following:

a. The size and scope of foreign gas-cooled reactor development programs.

b. The degree of investment by U.S. industry.

c. The price and availability of uranium and their effect on resource conservation policies.

d. The effectiveness of the LMFBR as a breeder.

3. From the viewpoint of national requirements, the highest priority in the gas-cooled reactor program should be assigned to the GCFR.

The single greatest promise of the fission process in terms of satisfying our long-term needs is to make large quantities of energy available from normally nonfissionable U-238 and thorium through the use of breeder reactors. The importance of the development and successful commercialization of a technically and economically effective breeder reactor is an overriding national consideration. The GCFR, presently in an early state of development, is perceived as having potential for both superior technical and economic performance to the LMFBR which is in a more advanced state of development. It is therefore concluded that GCFR development should be pursued initially as a backup to the LMFBR program with the possibility of its becoming our primary approach to an effective breeder if the LMFBR program falters."

Following this study the U.S. Department of Energy has then adopted the Project Definition and Licensing Phase proposed by General Atomic with a year delay and is considering a large research and development program which follows the suggestion of the Helium Breeder Associates to start with a demonstration plant of 300 MWe and then go to a prototype of about 1000 MWe and later to the commercialisation plant. The main items of the DOE program are illustrated in Tables IX to XIX.

14.4 Activities at the U.S. National Centers

In 1978 the US Department of Energy accepted a stretched out version of the PDLP as the basis for the FY 1979 and FY 1980 budgets. DOE has contracted with HBA for technical management of the GCFR industrial contractors and coordination of the National Laboratories GCFR activities and for integration of these efforts with the private sector funded work into one program. The main items of the DOE program are illustrated in Tables X through XIX.

The following activities for the GCFR have been started at the U.S. National Centers:

Argonne National Laboratory (ANL):

- fuel and materials development /180/
- safety analysis especially of core disruptive accidents /181/
- Zero Power Reactor critical experiments and their evaluation /182/-
- Post-Accident Fuel Containment studies
- Direct Electrical Heating (DEH) safety tests ranging from single pin to few-rod clusters, from ambient pressure to GCFR design pressures /183/.

Oak Ridge National Laboratory:

- concrete pressure vessel and closure investigations, inclusive of experimental model tests /184/
- irradiation of capsules containing fuel pins /185/
- fabrication of the Core Flow Test Facility (CFTL): this high pressure helium loop will allow the testing of large bundles (up to 91 rods) of electrically heated rods, both in steady state and during fast transients /184/
- shielding studies especially for the core supporting grid and for the liner of the concrete pressure vessel /184/.

Los Alamos National Laboratory:

- Direct Electrical Heating (DEH) safety tests ranging from few-rod clusters to the simulation of interactions among a few adjacent full-size fuel elements, to investigate the behaviour of melted clad and grid material between fuel elements and the time and mode of fuel element dropout from the the core region following melt-through of the fuel element duct /183/.

Idaho National Laboratory:

- Gas Reactor In-Pile Safety Test (GRIST) loop: GRIST-2 is a transient overpower test loop intended to determine fuel behaviour under high power transient conditions. It is being designed to test fuel assembliesä up to 37-rods starting at design power conditions. The TREAT reactor was selected as the driver core. At present, the reactor is being modified (TREAT Upgrade) to provide greater transient power capability for both the LMFBR and GRIST programs .

Conceptual design of the circulator test facility for testing the main and auxiliary helium circulators under full power operating conditions.

Pacific Northwest Laboratory
- Creep rupture testing of GCFR cladding in a flowing helium loop containing controlled amounts of impurities

Hanford Engineering Development Laboratory
- Mechanical properties of irradiated GCFR structural and shielding materials.

II. FACILITIES, INDUSTRIES

1. Austria

1.1 Facilities

The main facility which could be useful for the development of the GCFR is a model of a PCPV with a hot liner. This is being constructed jointly by the Österreichische Studiengesellschaft für Atomenergie (contact Dipl.-Ing. Walter Binner) and the Reaktorbau Forschungs- und Baugesellschaft (contact Dipl.-Ing. J. Német).

The main values of the PCPV are:

pressure	100 bar
temperature of the liner	300°C
overall diameter	3.6 m
internal diameter	1.5 m
overall height	12 m
internal height	10 m

Subsequent to the main vessel tests, a High Temperature Helium Test Rig will be erected inside the vessel for testing of materials, material combinations and structural assemblies under conditions up to 100 bar and 1000°C.

1.2 Industries

By the Reaktorbau Forschungs- und Baugesellschaft experience is available for the design of concrete pressure vessels for high pressure and with a hot liner.

In addition considerable experience has been accumulated by the Austrian Industries as supplier of Nuclear Power Plant Components.

Maschinenfabrik Andritz:

Pumps, mechanical components

Simmering-Graz-Pauker AG:

Heat exchangers, steam generators, structural steel

Vereinigte Edelstahlwerke:

Heat exchangers, stainless steel parts, purification plants, tanks

VÖEST-ALPINE:

Heat exchangers, pressure vessels, structural steel work, mechanical components, prestressing steel

Waagner-Biro AG:

Heat exchangers, structural steel work, mechanical components, steam generators

Felten & Guilleaume:

prestressing steel

2. Belgium

2.1 Facilities

The Belgian National Nuclear Research Center (C.E.N./S.C.K. - B 2400 - MOL - BELGIUM; contact J. PLANQUART) presents the following possibilities:

- Research and development departments such as the Reactor Physics Studies, Metallurgy, Chemistry, Safety.
- High flux material testing reactor BR2 and its facilities such as hot cells and the zero power facility.
- Technology and Energy department with a specialised group for conception, manufacturing and exploitation of large loops and instrumented capsules and rigs.

The main experimental facilities in operation at the CEN/SCK are:

- The in-pile helium loop "GSB" for irradiation of a 12 rod vented fuel element (see also German programme):

fuel element power : 285 kW
fuel element gas inlet temperature : 250 °C
fuel element gas outlet temperature : 500 °C
maximal rod surface temperature : 680 °C
main gas flow rate : 0,225 $kg \cdot s^{-1}$
loop pressure : 60 bar
average linear heating rate in the maximum of the neutron flux: 450 $W \cdot cm^{-1}$

- The out-of-pile helium loop "He1" for material and component testing in controlled environmental conditions of temperature, pressure and impurity levels:

  Main helium flow: up to 2,7 $g \cdot s^{-1}$

  Pressure:

  - compressor outlet and test section: up to 68 bar
  - compressor inlet and purification system: up to 18 bar.

  Maximum obtainable test section temperature: 1100 °C
  Test section maximum diameter: 200 mm
  Maximum purification flow: 0,65 $g \cdot s^{-1}$
  Detectable impurities: O_2, N_2, CO, CO_2, H_2, CH_4, Ne, Kr, Xe, H_2O
  Controllable impurity levels: from 1 to 1000 Vpm of O_2, H_2, H_2O, CO, CO_2
  Minimum detection threshold in either impurity: 0,1 Vpm.

  The main features of the oven are:

  - useful diameter of the test cavity: 200 mm
  - heated length of the test cavity: 2000 mm

2.2 Industries

a) DESIGN AND STUDY OFFICES

- BELGONUCLEAIRE - BRUSSELS
  (Contact J. CHERMANNE)

Experience in the field of fast and thermal reactors, and in the field of fuel cycle, especially plutonium fuel.
Different codes are available which can be used for calculations relative to the development of the GCFR.

COMETHE: prediction of fuel element lifetime performance

RUST and TRUMOC: statistical and probabilistic hot spot analysis

CRASH: clad stress and distorsion analysis

BEAM: pin bowing calculations

SMAC: probabilistic assessment of fuel pin reliability

NADIA, FIESTA: steady state thermohydraulic calculations

DIFLAC: overall core hydraulic balance

STRAW: structural analysis for wrapper

SWAMB: dynamic performance code giving thermomechanical equilibrium of a bundle with wire-wrap spacer.

- GBRA - BRUSSELS
  (Contact J. CHERMANNE)
  See corresponding chapter.

- BELGATOME - BRUSSELS
  (Contact GAUBE)
  Study office from the Utilities and the Industry.

b) FUEL INDUSTRY

BELGONUCLEAIRE - DESSEL
(Contact P. VAN DEN BEMDEN)
Mainly for plutonium fuel.

c) INDUSTRY FOR LARGE COMPONENT

A.C.E.C. - CHARLEROI
(Contact P. KEES - Nuclear Division)

COCKERILL - SERAING
(Contact F. BRAIBANT)

d) OTHER INDUSTRIES working in the field of nuclear energy

E.N.I. - AARTSELAAR
Contact J.P. RO?BAUX).

3. France

3.1 Background facilities

The "*Département des Etudes Mécaniques et Thermiques*" (D.E.M.T.) of "*Commissariat à l'Energie Atomique*" (C.E.A.) has in SACLAY a large variety of test facilities for studies in the reactor field. Those which are more used for the Gas Cooled Reactors (C.G.R.) Programs are shortly descripted below :

1/ CARMEN COMPLEX

In the CARMEN complex facilities, it is possible to test in actual size different reactor components at the same pressure and temperature conditions as in the reactor.

CARMEN 1 circuit is most devoted to channel testing and CARMEN 2 circuit, more powerful than CARMEN 1, is connected with different peripheral test rigs which are :

- Multichannels vessel,
- Steam generator test bench,
- Hot duct bench.

Both these circuits can work with carbon dioxide or with hélium.

The main values are :

CARMEN 1 : power : 250 kW
gas flow : 0,650 m^3/s
discharge head : 1 200 m
pressure : 50 bars
temperature : 500°C

CARMEN 2 : power : 3 000 kW
gas flow : 4 m^3/s
pressure : 100 bars
temperature : 450°C

Multichannel vessel :

inside diameter : 1,250 m

heigh : 15 m

pressure : 47,5 bars

temperature : 450°C

Steam generator test bench :

water flow : 1,4 kg/s ; max. : 3 kg/s

steam pressure : 185 bars ; max. : 220 bars

steam temperature : 510°C ; max. : 550°C

2/ TEST RIG FOR FIBROUS INSULATION

A criterion of performance of the thermal barrier is that blanket material under compression shall not relax, in order to prevent helium channelling through and behind the blanket.

The spring value must keep a sufficient value. A test rig was built to test full scale panels at a high temperature simulating accident conditions and for long term test in normal conditions.

These tests are performed in helium atmosphere of reactor purity but the pressure conditions are slightly in excess of one atmosphere.

The thermal performances and resilience or relaxation characteristics can be measured during and of course at the end of the test.

The dimensions of the three boxes of this test rig are :

2,6 x 2,6 x 0,5 m

heating power is 60 kW

and maximum temperature : 1 260°C.

3/ AIR CIRCUITS AT ATMOSPHERIQUE PRESSURE (MISTRAL)

USES :

- The assembly of large scale, easily accessible models for detailed investigation of aerodynamic and thermal phenomena.
- The models are easily and rapidly produced, temperatures and pressures being moderate.

Number of units	Max. temp. °C	Max. flow-rate kg/s	Compression ratio	Blower power kW	Exchanger power kW
1	120	8	1.20	260	500
1	120	8	1.35	370	500
2	50	1,2	1,20	33	80

(Flow rates are held absolutely constant by sonic venturis)

4/ FRICTION TESTING MACHINES "HETRIX 1 AND 2"

These tests facilities are devoted for friction testing of materials in helium atmosphere with race

Characteristics	HETRIX 1	HETRIX 2
Gas	hélium	hélium
Temperature	500 °C	1100°C
Pressure	1,3 bars	1,5 bars
Stroke lenght (half cycle)	5 to 20 mm	5 to 20 mm
Speed	0,04 to 60 mm/min	0,04 to 240 mm/min
Load	10 to 330 N	150 to 2500 N
Average specific pressure	70 bars	520 bars

5/ VESUVE AND TOURNESOL SHAKE-TABLES

The dynamic testing facilities are used for studing of structures or components behaviour when subjected to vibratory excitation (seismic shock, for example) and for qualification of electrical equipment in accordance with the recent standards.

	VESUVE	TOURNESOL
a/ Table		
- Dimensions	3,1 x 3,1 m	2 x 2 m
- Weight	4,2 t	1,2 t
- First resonant frequency	> 200 Hz	> 200 Hz
b/ Reaction mass		
- Weight	> 500 t	> 180 t
c/ Hydraulic jacks		
- 1 horizontal jack		
. Force	350 KN	100 KN
. Stroke	± 100 mm	± 125 mm
. Speed max.	1 m/s	1 m/s
- 2 vertical jacks		
. Force per jack	/	100 KN
. Stroke	/	± 100 mm
. Speed max.	/	1,2 m/s
d/ Sample		
- Weight	20 t	10 t
- Height	12 m	12 m
- Center of gravity height	< 3 m	< 1,5 m

3.2 Background industries

DESIGN AND CONSTRUCTION OF ADVANCED REACTORS

CEA (Commissariat à l'Energie Atomique)

NOVATOME

SYFRA (System Society)

DESIGN AND CONSTRUCTION OF NUCLEAR COMPONENTS

TECHNICATOME

ACB (Ateliers et Chantiers de Bretagne)

CREUSOT-LOIRE

PRE-STRESSED CONCRETE REACTOR VESSELS

SPIE - BATIGNOLLES

Société des Grands Travaux de Marseille

CITRA

CAMPENON BERNARD

BOUYGUES

STEAM GENERATORS

CCM (Compagnie de Construction Mécanique, procédés SULZER)

CREUSOT-LOIRE

STEIN Industries

CIRCULATORS

CCM

RATEAU

HISPANO-SUIZA (Division de la SNECMA)

CEM (Compagnie Electro-Mecanique)

FUEL ELEMENTS

CEA

COGEMA

4. Germany

4.1 Rigs and Loops

The following rigs and loops are available at KfK Karlsruhe (contact: Dalle Donne)

- Air rig for heat transfer experiments in annuli with rough rods:

max heating power:	80 KW
maximum air flow:	0.5 kg/sec
maximum wall temperature:	1000°C
maximum air outlet temperature:	700°C
range of air pressure:	1-5 bar

- Air rig for measurements of rough rod temperatures in correspondence of spacer grids:

maximum air flow:	0.3 kg/sec
maximum air pressure:	4 bar
maximum air temperature:	300°C

- Air rectangular wind channel for measurements of drag and velocity distribution at rough wall:

maximum air flow:	5 kg/sec
maximum available pressure drop:	0.1 bar
ambient temperature	

- Air rectangular wind channel for measurements of velocity and turbulence distribution at rough and smooth walls in rod clusters:

maximum air flow:	5 kg/sec
max. available pressure drop:	0.1 bar
ambient temperature	

- Water loop for measurements of velocity distribution and pressure drop in rectangular channels with smooth and rough walls:

max. water flow:	6000 ℓ/min
max. pressure drop at test section:	15 bar
ambient temperature	

- Helium loop for heat transfer and pressure drop measurements in rod clusters:

max. heating power:	600 KW
max. helium flow:	1.2 kg/sec
max. blower power:	140 KW
max. helium temperature:	520°C
max. helium temperature at blower:	250°C
max. helium pressure:	50 bar

High frequency induction heater for core-catcher tests (for istance borax).

At KFA Jülich (contact: Krug) a large loop for isothermal high temperature, high pressure endurance tests is available. Also measurements of pressure drop are possible.

At KWU-Erlangen (contact: Peehs) a rig is available for long duration relative movement tests between rods and spacer grids especially for rough rods.

4.2 Reactors

At KfK Karlsruhe the zero-power fast reactor SNEAK (contact: Helm) is available, which allows measurements with various fuel element configurations of reactivities, of control rod worths, void reactivity effects and steam entry effects.

4.3 Codes

The following codes relavant for GCFR calculations are available at KfK:

SATURN: lifetime performance of fuel element pins
SHOSPA: statistical and probabilistic hot spot analysis
SAGAPO: especially developed for GCFR fuel elements: calculation of temperature and pressure drop in clusters of rough rods
THESIS, THEKA, THEDRA: steady state thermohydraulic codes for fuel elements
PHAETON: transient thermohydraulic code (inclusive of neutronic point-kinetics representation of core) for calculation of accidental conditions in core, primary (helium) and secondary (water, steam) circuit.
KADIS: code for core-disruptive accident calculations
KfK-INR: code for neutronic calculations in fast reactor cores.

4.4 Industries

At present the German firm Kraftwerk Union in Erlangen (contact: C.A. Goetzmann) is working with a small group for the GCFR. KWU has of course experience in design and construction of large components and complete designs of water reactors. Other German firms who have experience in the reactor field are:

- Interatom, Bensberg: sodium cooled fast reactors,
- GHT, Bensberg: gas cooled thermal reactors,
- HRB, Mannheim: gas cooled thermal reactors,
- BBC, Mannheim: gas cooled thermal reactors,
- Krupp, Essen: concrete pressure vessels.

5. Japan

At the Japan Atomic Energy Research Institute in Tokai-mura (contact: Hirata) the following facilities are available, which are or can be used for the development of GCFR :

High Temperature Helium Gas Loop (HTGL) and Secondary Hydrogen Gas Loop

These loops are used for resource testing on heat transfer and hydrodynamic characteristics of fuel element and for verification of the possibility in reducing hydrogen permeation through the tubes of a He/H_2 heat exchanger. The main parameters of these loops are as follows:

	HTGL	Hydrogen Gas Loop
Maximum operating pressure	42 kg/cm^2G	42 kg/cm^2G
Maximum operating temperature	1000 °C	900 °C
Maximum flow rate	100 g/sec	30 g/sec
Diameter of main piping	7.5 - 15.2 cm	2.5 - 10.2 cm
Heater input	270 kW	50 kW

Fig. 10 shows the flow diagram of these loops.

High Temperature In-Pile Gas Loop (OGL-1)

The OGL-1, installed in the material testing thermal reactor (JMTR) is used for irradiation testing of fuel elements and structural materials and for the study of fission gas behaviour. The main parameters are as follows :

Outlet helium gas temperature	1000 °C
Maximum gas pressure	35 kgf/cm^2
Maximum gas flow rate	100 g/sec
Gas circulater head	4 kgf/cm^2
Heater input	150 kW

Irradiation sample size	80 mm dia, 750 mm length
Maximum heat generation of irradiated sample	400 W/cm
Available irradiation neutron flux	6 10^{13} n/cm^2 sec (thermal)
	9 10^{12} n/cm^2 sec (fast)

Fig.11 shows the flow diadram of OGL-1 loop.

Helium Engineering Demonstration Loop (HENDEL)

The HENDEL loop is being designed as a large-scale model testing facility for the demonstrative opration of high temperature components, such as the heat exchanger, piping, valves, core support structure which are operated under the severe conditions. Its operation will be started in 1981. Fig. 12 shows the flow diagram of HENDEL loop. The test conditions of the loop considered are as follows :

Test items / Test conditions	Fuel stack test section	Reactor structures test section	Large flow rate test section	High temperature components test section
Helium gas temperature (°C)	1000 (max.1200)	1000 (max. 1200)	~400	1000
Helium gas Flow rate (kg/sec)	0.4	4.0	4.0	2.6
Helium gas pressure (kg/cm^2G)	40	40	40	40
Tested components and test objects	Fuel stack, Control rod	Core support structure	In-vessel flow distribution, stop valve, Core lateral restraint structure	Intermediate heat exchanger, High temperature piping, Steam generator, Emergency isolation valve

Fast Critical Assembly (FCA)

The FCA is used for reactor physics measurements of reactivities, control rod worths, void reactivity effects and steam entry effects. Fig. 13 shows the views of FCA and its material drawer.

Computation code

The following computer codes relevant for GCFR calculations are available at JAERI.

RELAM : Heat transfer coefficient of turbulent gas heated by a high heat flux

TRAN : Transient hydrodynamics and heat transfer of turbulent flow

THYDES : Steady state thermohydraulic code for fuel element

GAKIT : Transient thermo-hydraulic code for calculation of accidential conditions

PIGEON-CITATION : Neutronic calculation codes for fast reactor

APOLLO : Fuel cycle analysis code based on two-dimentional diffusion approximation.

Industry

Kawasaki Heavy Industry (contact: R. Tanaka) is working with a small group for assessment of a symbiosis between VHTR and GCFR, in collaboration with JAERI's GCFR group.

6. Sweden

The following is a list of organizations with useful expertese, experience and facilities in the mentioned areas of services:

Studsvik Energiteknik AB, S-611 82 Nyköping

TESTS IN IN-PILE LOOP 4 (HTR) OF R2 (a 50MW ORNL-type reactor) with 6 test positions, He/Ne-cooled at 1.5-3x10^{14} n/cm^2s thermal fluence and 2-4.5x10^{14} fast fluence (5-13W/g gamma heating in steel).
Contact: Mr K Saltvedt.

MATERIALS AND WELDING DEV. AND TESTING (incl irradiations).
Contact: Mr K Pettersson.

CORROSION AND DEPOSITION EXPERIMENTS IN WATER, STEAM AND GASES
Contact: Mr W Hübner.

PRESSURE DROP MEASUREMENTS, SPACER TESTS, VIBRATION STUDIES, VISUAL FLOW STUDIES IN LARGE MODEL TANKS, HEAT TRANSFER ETC EXPERIMENTS.
Contact: Mr B McHugh.

ANALYSIS OF PROCESS AND CONTROL SYSTEMS, NUCLEAR INSTRUMENTATION, SIMULATORS.
Contact: Mr P Blomberg.

COMPONENTS DEVELOPMENT AND TESTING (INCL PCRVs).
Contact: Mr S Menon.

FUEL CYCEL STUDIES.
Contact: Dr E Hellstrand.

AB ASEA-ATOM, Box 53 S-721 04 Västerås 1

MECHANISMS (e.g. Control Rods)

Sandvikens Jernverks AB, S-811 01 Sandviken 1

STAINLESS STEEL DETAILS

Skånska Cementgjuteriet AB, Fack S-103 40 Stockholm

CIVIL ENGINEERING CONTRACTOR

Armerad Betong AB, Fack S-171 04 Solna

CIVIL ENGINEERING CONTRACTOR

Spännarmering AB, Internordisk, Box 106 S-161 26 Bromma

PRESTRESSING SUPPLIER

Strängbetong AB, Box 9205 S-102 73 Stockholm

PRESTRESSING SUPPLIER

Uddcomb Sweden AB, Fack S-371 01 Karlskrona

LINER, PENETRATIONS AND OTHER STEEL DETAILS

7. Switzerland

7.1 Facilities

At the Eidg. Institut für Reaktorforschung in Würenlingen (contact: Markòczy) the following facilities have been used or can be used for the GCFR development:

Rohan test rig:

test section geometry:	annulus
coolant:	air
pressure:	1.2-2 bar
coolant temperature :	in 20°C
	out 90°C
max. air velocity:	73 m/sec
heating power:	1.3 KW
max. wall temperature:	190°C

- Prospect experiment: allows determination of spacer pressure drop and velocity distribution in a bundle
- Megaere experiment: allows to study mixing and cross flow effects between subchannels in air flow
- Agathe loop: for heat transfer experiments with rough clusters:
  - coolant: CO_2
  - coolant pressure: 1-60 bar
  - coolant temperature: 30-500°C
  - Maximum coolant flow: 4.5 kg/sec
  - Heating power: 0 to 1000 KW
- Zero energy reactor Proteus:

This assembly is a coupled fast-thermal system in which thermal driver zones surround a fast zone which is large enough to produce a central neutron spectrum closely approximating that in a GCFR. The 500 mm diameter central zone contains about 2000 fuel pins on a 10 mm pitch hexagonal lattice. Available for filling into the pins are sealed capsules of mixed PuO_2/UO_2 fuel pellets, or of depleted UO_2 blanket material, or of sintered ThO_2 particles. The driver zones contain 5% enriched UO_2 moderated partly by D_2O and partly by graphite. At the maximum power level of 1 kW the thermal and fast neutron fluxes assembly are each approximately $5\times10^9 n\cdot cm^{-2} sec^{-1}$.

The computer codes CLUHET and SCRIMP for calculations of temperatures and pressure drop in clusters of rough and smooth rods are in operation.

7.2 Background Industry

Sulzer Brothers Ltd. Winterthur. GCFR Steam Generator Research, Development and Design.

8. United Kingdom

Various heat transfer rigs with air cooling for tests on single rough pins and CO_2 loops for tests on rod clusters are available in Windscale (contact: Wilkie). Rigs for compatibility studies among various materials are in operation in Harwell (contact: Bennet). Two helium loops, one at 41 bar and one at 0.4 bar with controlled amounts of impurities in the helium atmosphere are available in the Nuclear Power Company at Whetstone (contact: Knowles).

In the following papers general information is given on 8 different helium loops which could be used for compatibility, tribology, wear and fretting, corrosion, vibration and pressure drop tests. Information is also given of three CO_2 loops which could be used for pressure drops vibration and thermal insulation studies.

Reliability and transient codes, originally developed for the LMFBR are being modified for GCFR application, as well as methods to investigate the effects of local core blockages. Heat transfer codes to calculate temperatures and pressure drop in rod clusters have been developed at the UKAEA Establishment of Windscale (HOTSPOT) and at the CEGB center in Berkeley (SCANDAL), mainly for the AGR type of fuel element, but can be easily modified for GCFR application.

NPC HIGH PRESSURE LOOP

LOCATION: R&D LABORATORIES, NUCLEAR POWER COMPANY, WHETSTONE, LEICESTER, ENGLAND

STATUS: Not in use

PRINCIPAL USE: The loop has been used to investigate the compatability of materials as part of a Gas Cooled Fast Breeder Reactor feasibility study.

FACILITY DESCRIPTION: Samples of materials are loaded into autoclaves and exposed to a fast flow of high pressure helium containing controlled amounts of impurities. The autoclaves and the loop are made of stainless steel and the helium circulators are totally enclosed. Impurities in the helium are monitored by a Helium Ionisation Chromatograph and a High Pressure Electrolytic Hygrometer. The concentrations of impurities (including most permanent gases) are controlled by removing them in a by-pass purification circuit or adding them manually.

The main parameters of the loop are in Table 1.

TABLE 1

Operating gas	helium
Operating pressure	4.1 MN/m^2
Total flow	3.2 g/s
Maximum Temperature	850°C
Minimum impurity level	less than 5 ppb total

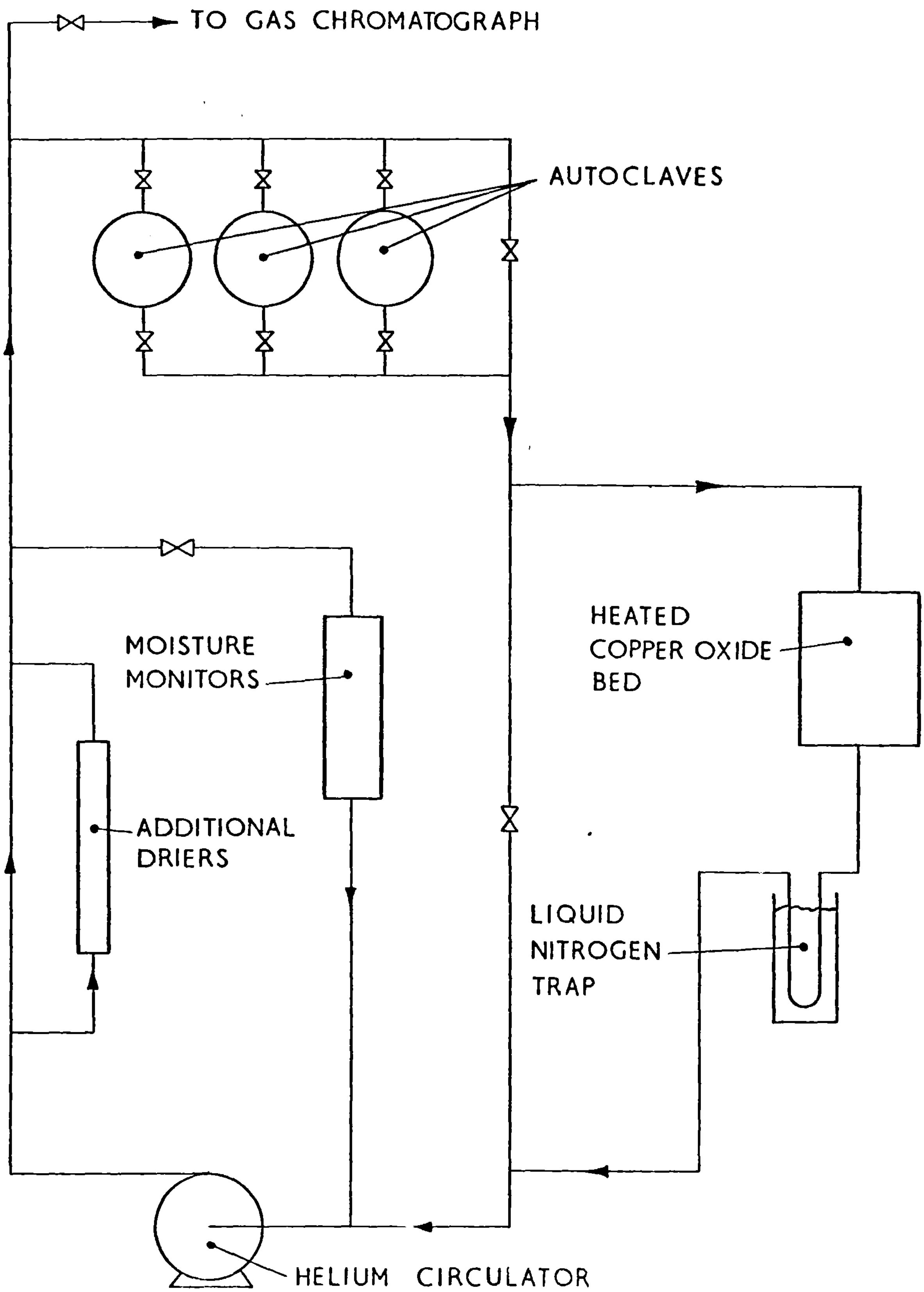


DIAGRAM OF NPC HIGH PRESSURE LOOP

NPC LOW PRESSURE LOOPS

LOCATION: R&D LABORATORIES, NUCLEAR POWER COMPANY, WHETSTONE, LEICESTER, ENGLAND.

STATUS: Not is use

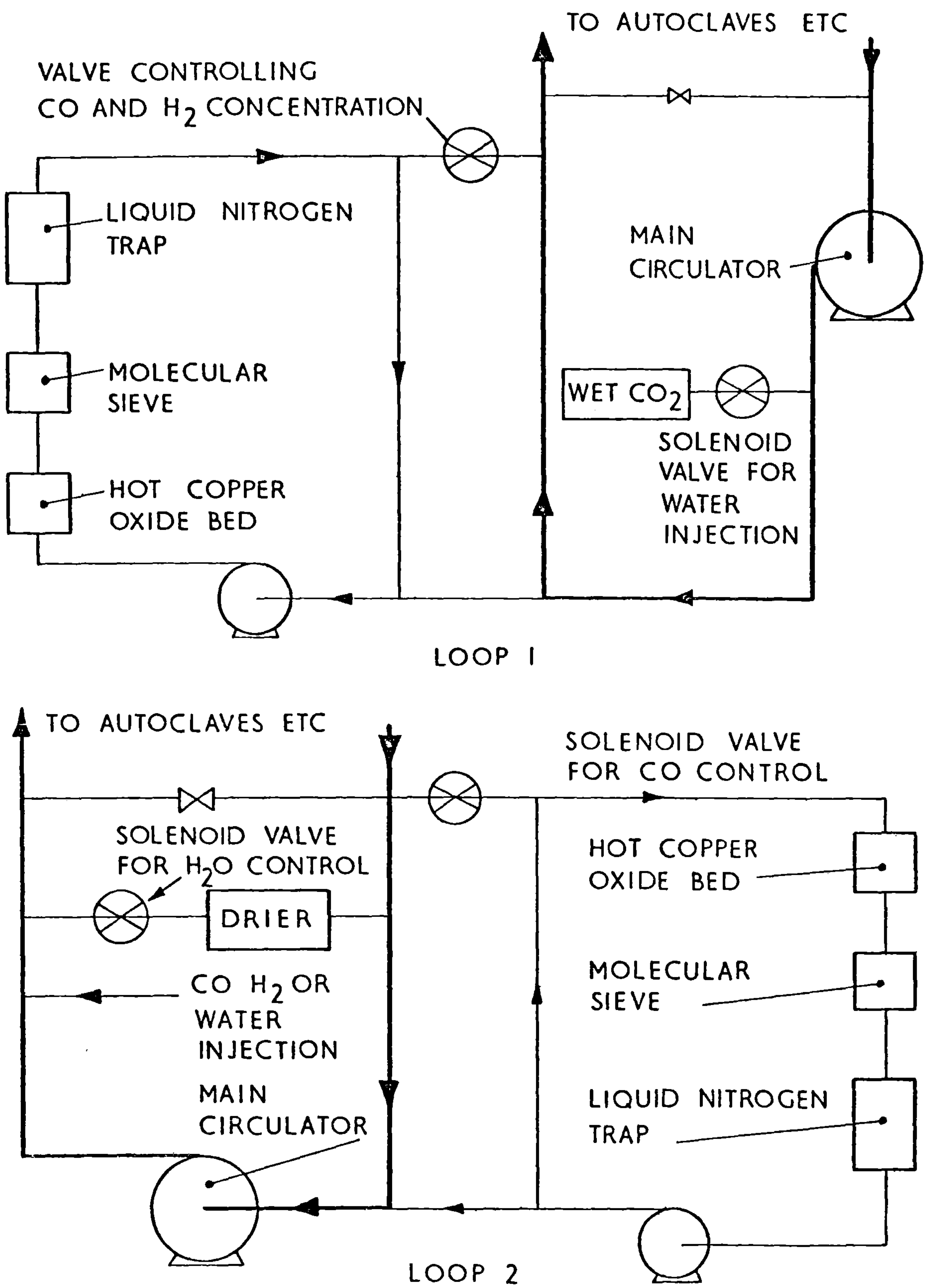
PRINCIPAL USE: The loops have been used on compatibility and tribology programmes as part of a Gas Cooled Fast Breeder Reactor feasibility study.

FACILITY DESCRIPTION: There are two low pressure loops very similar in function but one is made in stainless steel and the other has copper pipework. Each loop has a circulator which pumps helium at near atmospheric pressure through an interchangeable range of autoclaves or tribology test rigs. A subsidiary circuit on each loop contains a second circulator and a purification plant. Impurity levels are measured by a Helium Ionisation Chromatograph and levels are controlled automatically by connecting and disconnecting the purification circuit.

The main loop parameters are in Table 1.

TABLE 1

Number of loops	two
Operating gas	helium
Operating pressure	about 170 kN/m^2
Pressure rise across main circulator	about 200 kN/m^2
Maximum flow (loop 1 - copper)	0.8 g/s
(loop 2 - stainless steel)	2.1 g/s
Maximum autoclave temperature	850°C



FLOW DIAGRAMS OF NPC LOW PRESSURE LOOPS

NPC HELIUM TEST LOOP

LOCATION: R&D LABORATORIES, NUCLEAR POWER COMPANY, RISLEY, WARRINGTON, LANCS

STATUS: The facility has been put into reserve in a state of near completion.

PRINCIPAL USE: The rig was designed for supplying and recirculating pure dry helium to purpose-built environmental component test rigs such as materials exposure rigs, wear and fretting rigs insulation rigs and mechanism testing rigs.

FACILITY DESCRIPTION: The loop consists of two parallel recirculating streams each containing a copper oxide converter bed, circulator, molecular sieve and cryogenic trays. A valve network allows beds to be interchanged on line to permit continuous operation during bed regeneration.

Rigs are supplied from a terminal manifold which has at present four tapping points. Each outlet has a by-pass which can be set to match the test rig circuit resistance and a service manifold is provided for vacuum and initial pressurising purposes.

Gas purity is monitored on a sequential system of solenoid operated selection valves which route gas samples to two sampling trains. Controls include an early alarm of purifier bed saturation or excessive loading. The rig environment is protected by control systems which diverts the flow and ultimately shut the rig down.

The main rig parameters are shown in Table 1.

TABLE 1

Operating pressure	1 to 5 MN/m^2
Maximum flow	80 g/s per circuit
Maximum temperature at purifier inlet	80°C
Gas purity	total impurities less than 0.5 ppm by volume

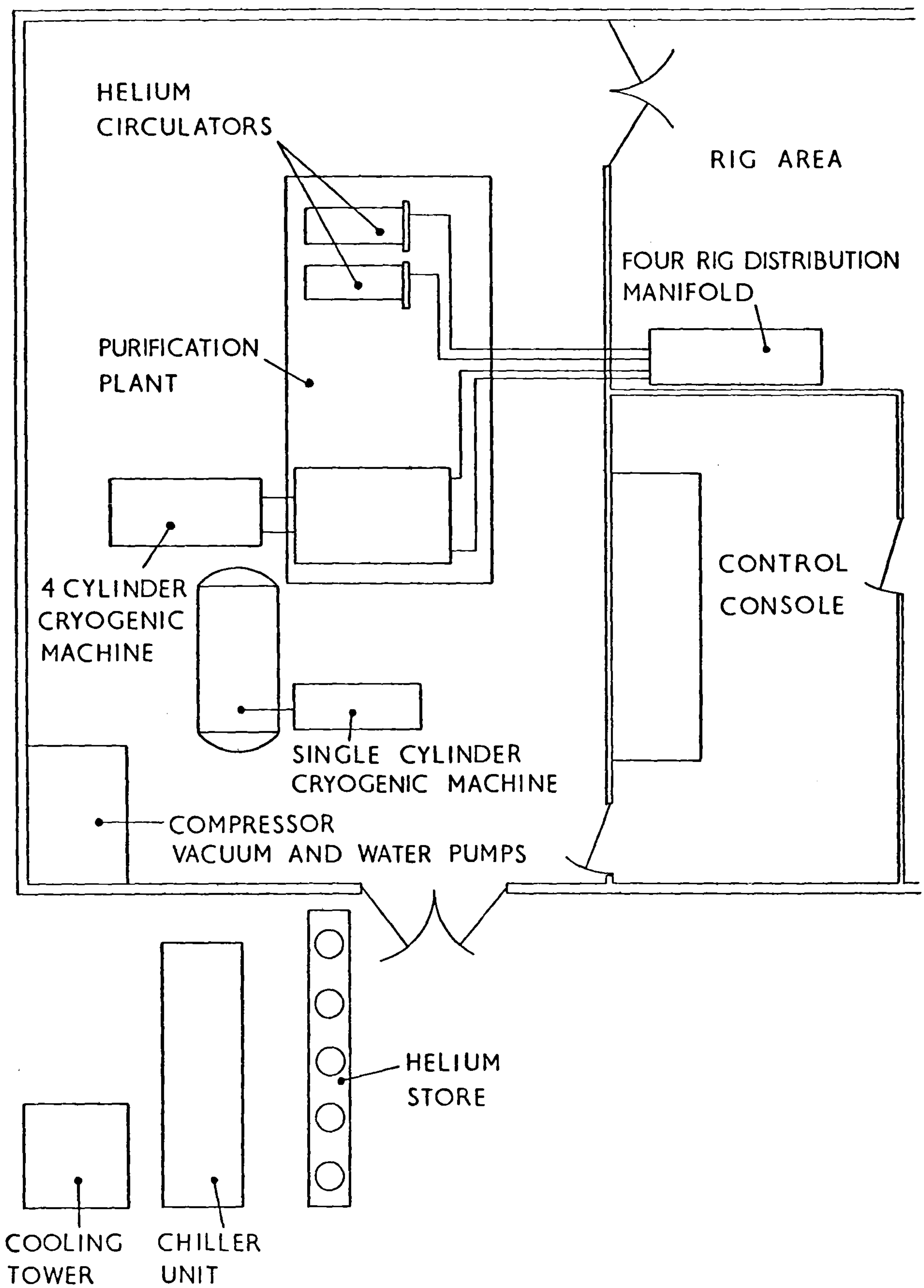


DIAGRAM OF NPC HELIUM TEST LOOP PLANT ROOM

HELIUM LOOP FACILITY

LOCATION: N E I CLARKE CHAPMAN POWER ENGINEERING LTD, GATESHEAD TYNE AND WEAR, ENGLAND

STATUS: Available for use

PRINCIPAL USES: The facility is used to test materials and components in helium containing controlled amounts of impurities.

FACILITY DESCRIPTION: The main circuit consists of a reciprocating diaphragm compressor which circulates helium through a network of 25 mm bore pipes to three test sections. The main test section is mounted inside a furnace and is 2.5 m long and 0.6 m internal diameter. During fatique and fretting experiments specimens inside the test section are vibrated through a penetration in the side of the vessel. There are provisions for larger test sections to be connected to the circuit.

An important feature of this facility is the ability to control the concentration of impurities in the helium. The control is automatic and operates on a continuous bleed purification principle so that unwanted impurities do not accumulate. A proportion of the gas is continuously by-passed though a purification circuit which removes all impurities, then the desired composition is restored by adding carbon dioxide, carbon monoxide, methane, water and hydrogen. The rate of addition is controlled by an electronic injection unit operating on signals from an analysis circuit containing hygrometers, infra red gas analysers, a flame ionisation meter and a gas chromatograph.

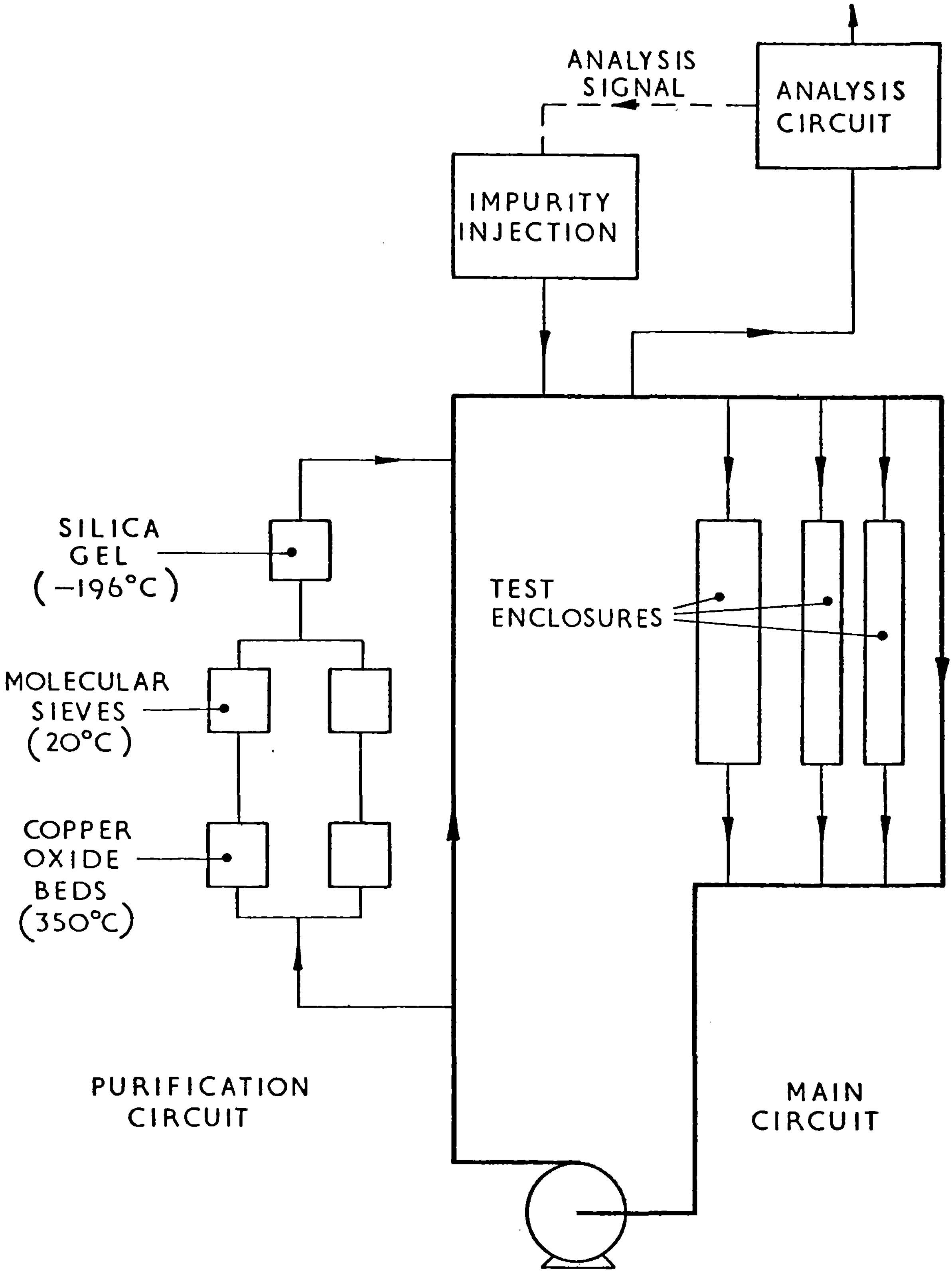
The main rig parameters are shown in Table 1 and the impurity control range in Table 2.

TABLE 1

Material of construction	316 Stainless Steel
Operating gas	Helium
Operating pressure	up to 1.1 MN/m^2
Maximum flow rate	0.016 kg/s
Maximum test section temperature	$800^{\circ}C$

TABLE 2

IMPURITY	CONTROL ACCURACY %	CONTROL SPAN ppm (vol)
Carbon dioxide	± 2	10 to 50,000
Carbon monoxide	± 3	10 to 4,000
Methane	± 10	1 to 10,000
Water	± 10	2 to 400
Hydrogen	± 10	10 to 1,000



SIMPLIFIED FLOW DIAGRAM OF THE
HELIUM LOOP FACILITY

THERMOBALANCE FACILITY

LOCATION: SPRINGFIELDS NUCLEAR POWER DEVELOPMENT LABORATORIES
UKAEA, SALWICK, PRESTON, ENGLAND.

STATUS: Operational

PRINCIPAL USE: This facility is used in the assessment of corrosion kinetics of reactor materials. It enables the weight change of specimens in a hot helium environment to be monitored continuously.

FACILITY DESCRIPTION: A thermobalance consists of a beam which supports a single specimen inside a helium filled furnace. If the weight of the specimen changes the beam becomes out of balance but this is automatically corrected by an electromagnetic restoring force. The current necessary to maintain balance is a measure of the weight change so the weight can be recorded continuously.

This facility has two thermobalances. One with a silica furnace tube is suitable for pressures up to atmospheric and temperatures up to 1000°C. The other with an Inconel tube can be used at up to 4 MPa and 800°C. A flow of helium is maintained through the furnace tubes either from gas cylinders or from the adjacent Helium Corrosion Test Facility which provides helium containing controlled amounts of impurities.

The main facility parameters are shown in Table 1

TABLE 1

Balance Working Load	10 g
Sensitivity Ranges	1 μg 5 μg 10 μg 50 μg 100 μg
Corresponding Spans	20 mg 100 mg 200 mg 1 g 2g
Gas Pressures	Sub atmospheric to 4 MPa
Maximum Temperatures	800 and 1000°C
Maximum Gas Flow	150 cm^3/min
Typical Gas Flow	20 to 50 cm^3/min

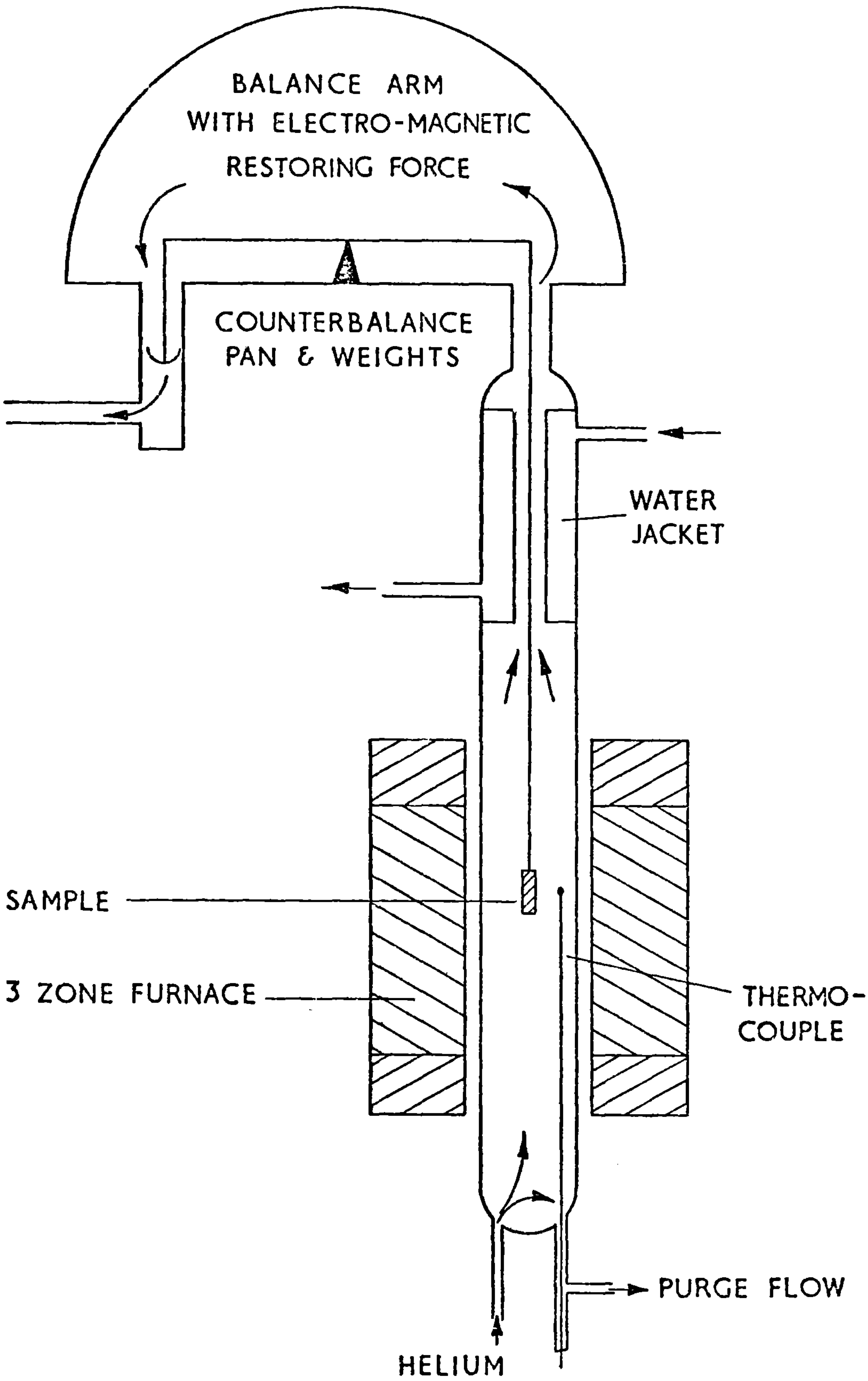


DIAGRAM OF THERMOBALANCE

HELIUM CORROSION TEST FACILITY

LOCATION: SPRINGFIELDS NUCLEAR POWER DEVELOPMENT LABORATORIES, UKAEA, SALWICK, PRESTON, ENGLAND.

STATUS: Operational

PRINCIPAL USE: The facility provides a controlled helium based environment for specimens of reactor materials so that the effect of corrosion on their chemical physical and mechanical properties can be investigated.

FACILITY DESCRIPTION: The facility consists of two basically similar but independently controlled and monitored closed loops around which helium is circulated. Impurities such as water carbon monoxide and hydrogen can be added to the helium and their levels controlled independently between 50 and 5000 ppm by volume. The loops operate over a range of pressures and flow rates. Specimens in sealable silica tubes are housed in reaction vessels surrounded by furnaces which maintain the temperature uniform to within 5°C.

There are a total of five test sections (2 on loop 2, 3 on loop 3) and any one can be isolated for specimen change or inspection without disturbing the others.

The main loop parameters are shown in Table 1.

TABLE 1

Working Volume of Furnaces	2 off 50 mm diameter 254 mm long 3 off 70 mm diameter 254 mm long
Maximum Furnace Power	9.5 kW
Maximum Furnace Temperature	1000°C
Main Loop Temperature	Ambient
Working Pressure	Sub atmospheric to 0.18 MPa
Total Loop Flow	0.2 g/s (loop 2) 0.65 g/s (loop 3)

AUXILIARY EQUIPMENT: Gas composition is continuously monitored externally by programmed gas chromatography and either rig can be connected to a thermobalance which is described separately.

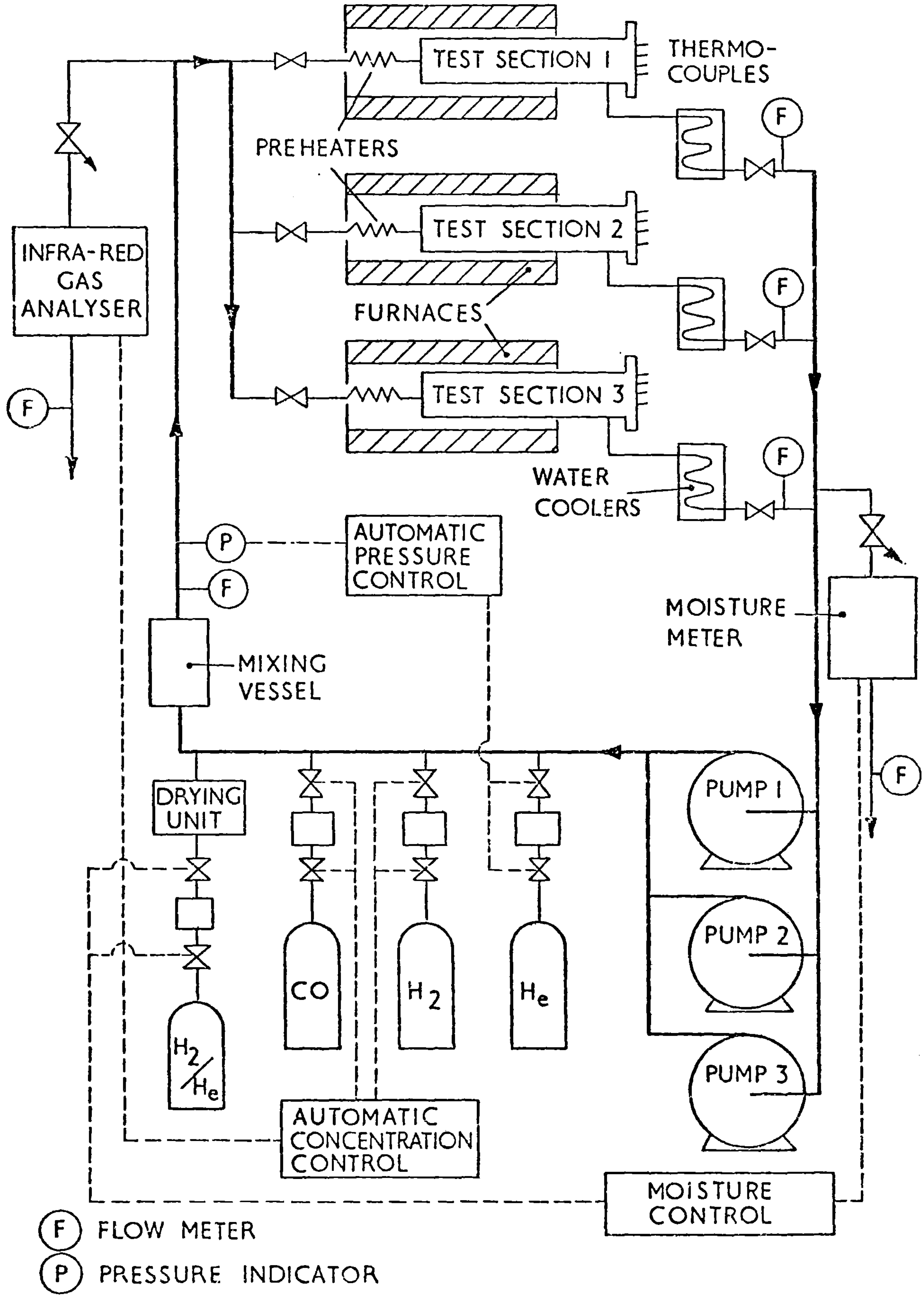


DIAGRAM OF HELIUM CORROSION TEST FACILITY (LOOP 3)

HELIUM TRIBOLOGY RIGS

LOCATION: RISLEY NUCLEAR POWER DEVELOPMENT LABORATORY, UKAEA, RISLEY, WARRINGTON, ENGLAND

STATUS: Operational

PRINCIPAL USES: The tribology rigs are used to study the friction, wear and fretting behaviour of materials in a high temperature helium environment.

FACILITY DESCRIPTION: There are four rigs each operating independently but all supplied with helium from a central manifold which contains a clean up system to keep impurities other than moisture below 2 ppm by volume. All the rigs operate at atmospheric pressure.

The main parameters of the test sections are listed in Table 1. Some test sections are interchangeable so that up to four can be operated at any time.

TABLE 1

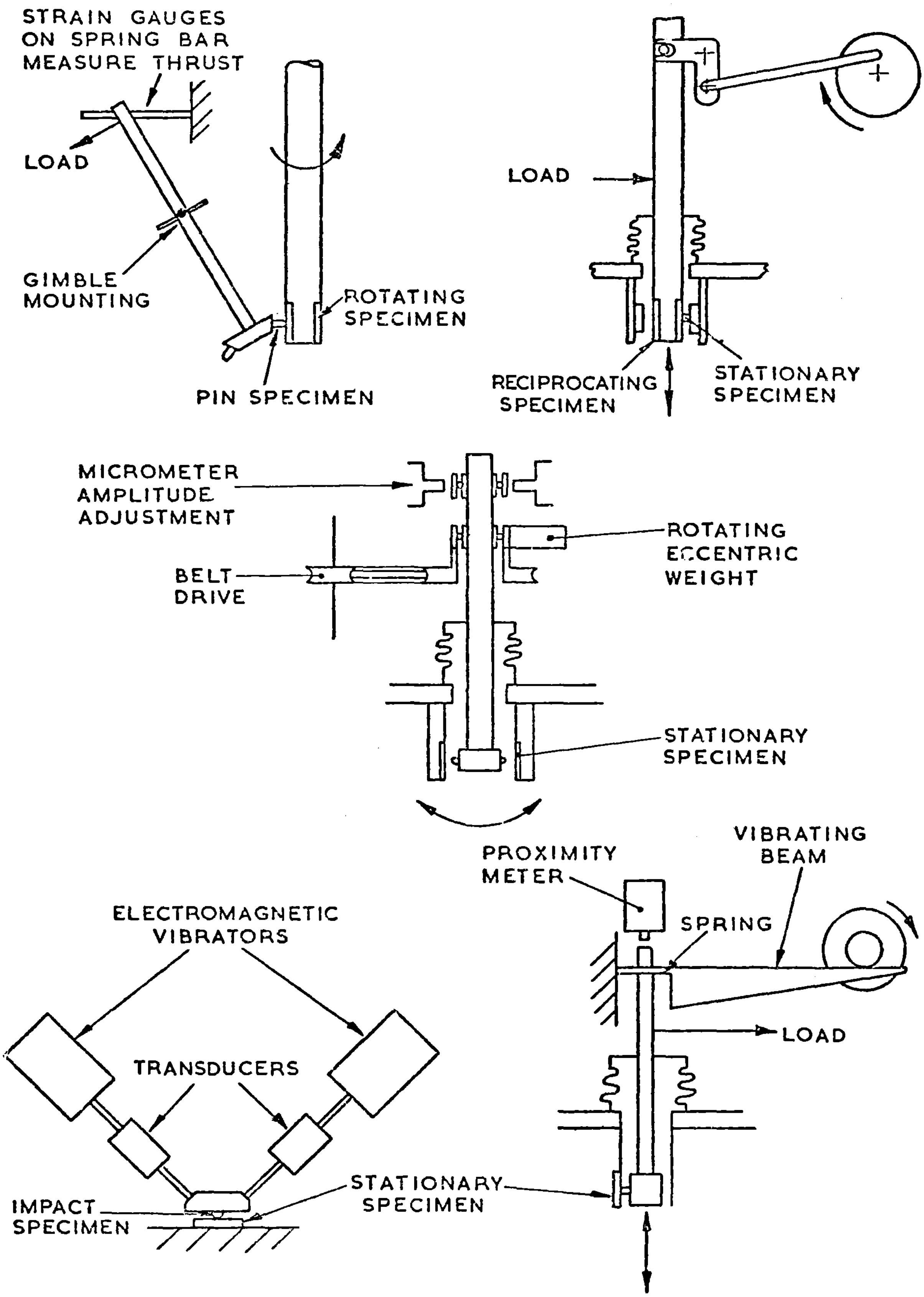
Rubbing Pairs Rig	
Operating temperature	up to 800°C
Motion	Rotational unidirectional
Specimen shape	Cylindrical
Operation	Specimen rubs against any one of eight rounded or flat ended pins
Purpose	Friction and Wear studies
High Temperature Rig	
Operating temperature	up to 1000°C
Motion	Reciprocating
Specimen shape	Flat on flat or cross cylinder
Operation	Up to four pairs of material can be accommodated
Purpose	Friction and Wear studies
Slide Impact Mechanism	
Typical Operating temperature	650°C
Motion	Oscillatory
Operation	A rotating eccentric weight causes a specimen to hammer a stationary specimen at right angles to its surface with a controlled amplitude
Purpose	Impact fretting studies

Impact Slide Mechanism

Typical Operating temperature	650°C
Motion	Reciprocating unidirectional either pure sliding or impact sliding with a pulsed, sinusoidal or random frequency
Operation	The mechanism employs twin electromagnetic vibrators operating at right angles which are controlled by power amplifiers providing motion along and vertical to the specimen surface
Purpose	Rubbing or impact fretting studies

Rubbing Fretting Mechanism

Typical operating temperature	650°C
Motion	Reciprocating
Operation	The mechanism allows specimens to be rubbed together under load at small amplitude in the absence of any superimposed impact
Purpose	Fretting studies



PRINCIPLE OF OPERATION OF HELIUM TRIBOLOGY RIGS

SCOT LOOP

LOCATION: SPRINGFIELDS NUCLEAR POWER DEVELOPMENT LABORATORIES
UKAEA, SALWICK, PRESTON, ENGLAND.

STATUS: Loop No 1 Operational   Loop No 2 Under Construction

PRINCIPLE USE: The SCOT loop is used for the flow and acoustic vibrational testing of gas-cooled reactor fuel stringers at simulated reactor conditions.

FACILITY DESCRIPTION: The loop consists of two test sections (Scot loop 1 and Scot loop 2) which share a set of circulators, heaters and regenerative heat exchangers. One test section is used whilst the other is loaded. The operating gas is carbon dioxide.

The facility is housed in a 30 m high building and fuel stringers up to 22 m long can be accommodated if they can be split for loading into the test sections.

The main parameters of the facility are shown in Table 1.

TABLE 1

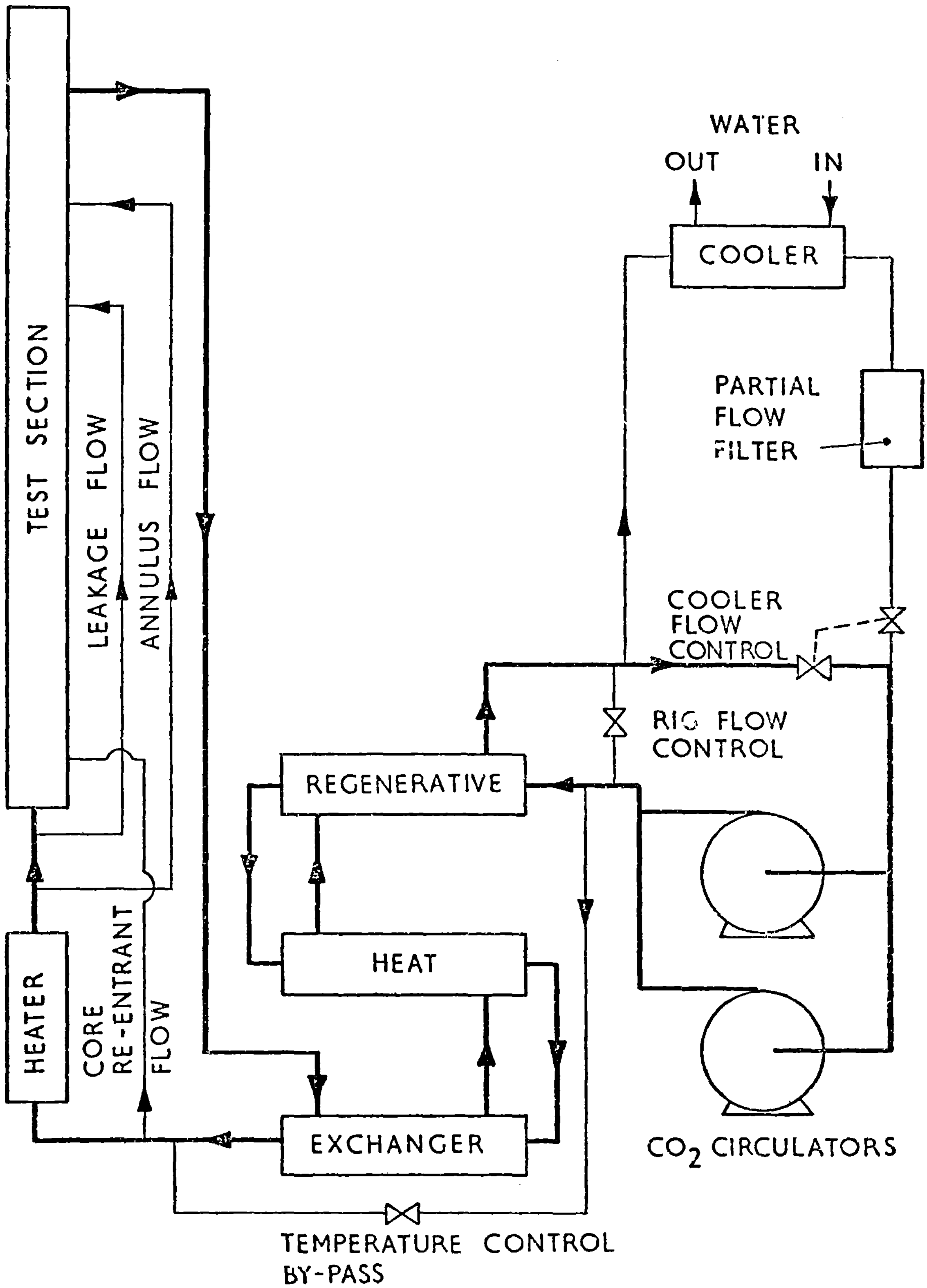
Operating gas	Carbon dioxide
Gas flow	15 kg/s
Total circulator power	800 kW
Heater power	800 kW
Test section diameter	0.38 m at smallest section
Test section height	22 m of which 15 m is exposed to gas flow
Maximum operating conditions	Scot loop 1 200°C at 2 MPa Scot loop 2 325°C at 4 MPa or 425°C at 2.6MPa

AUXILIARY EQUIPMENT: The facility has a small by-pass circuit for moisture removal.

Equipment is available to record and analyse signals from vibration transducers.

REFERENCE

[1] ETHERINGTON, C., JONES, C H., "Scot - development of a rig for the out-of-pile testing of reactor fuel elements and components", Journal of British Nuclear Energy Society, 11 3 (1972) 291.



FLOW DIAGRAM OF SCOT LOOP
(SHOWING ONLY ONE TEST SECTION)

CAGR RIG

LOCATION: WINDSCALE NUCLEAR POWER DEVELOPMENT LABORATORIES
UKAEA, SEASCALE, ENGLAND

STATUS: Operational

PRINCIPAL USE: The rig was built to study the behaviour of fuel stringers during on load refuelling. It has also been used for the assessment of thermal insulation and to study vibrations of reactor components under plant conditions.

FACILITY DESCRIPTION: The rig consists of a pressure vessel 24 m long and 1.4 m diameter through which carbon dioxide is circulated by three variable speed blowers. A stand pipe 2.5 m long and 0.5 m diameter can be mounted on top of the vessel to simulate a charge shoot. The rig is heated by the heat of compression in the blowers and the temperature is controlled by three large coolers.

Test sections are hung from a support ring in the main vessel which has penetrations along its length providing access for experimental equipment, instrument connections or viewing with a television camera.

The main rig parameters are listed in Table 1.

TABLE 1

Operating Gas	Normally carbon dioxide but nitrogen or other gases could possibly be used.
Maximum test section size	23 m long 1.24 m diameter
Maximum flow rate (CO_2)	180 kg/s
Gas pressure	0.34 MPa (min) 4.5 MPa (max)
Gas temperature	100°C (min) 350°C (max)
Circulator power	No 1 1.3 MW No 2 1.3 MW No 3 1.5 MW

AUXILIARY EQUIPMENT: Local to the rig are assembly bays served by 25 ton cranes in which test sections can be prepared in a vertical or horizontal position.

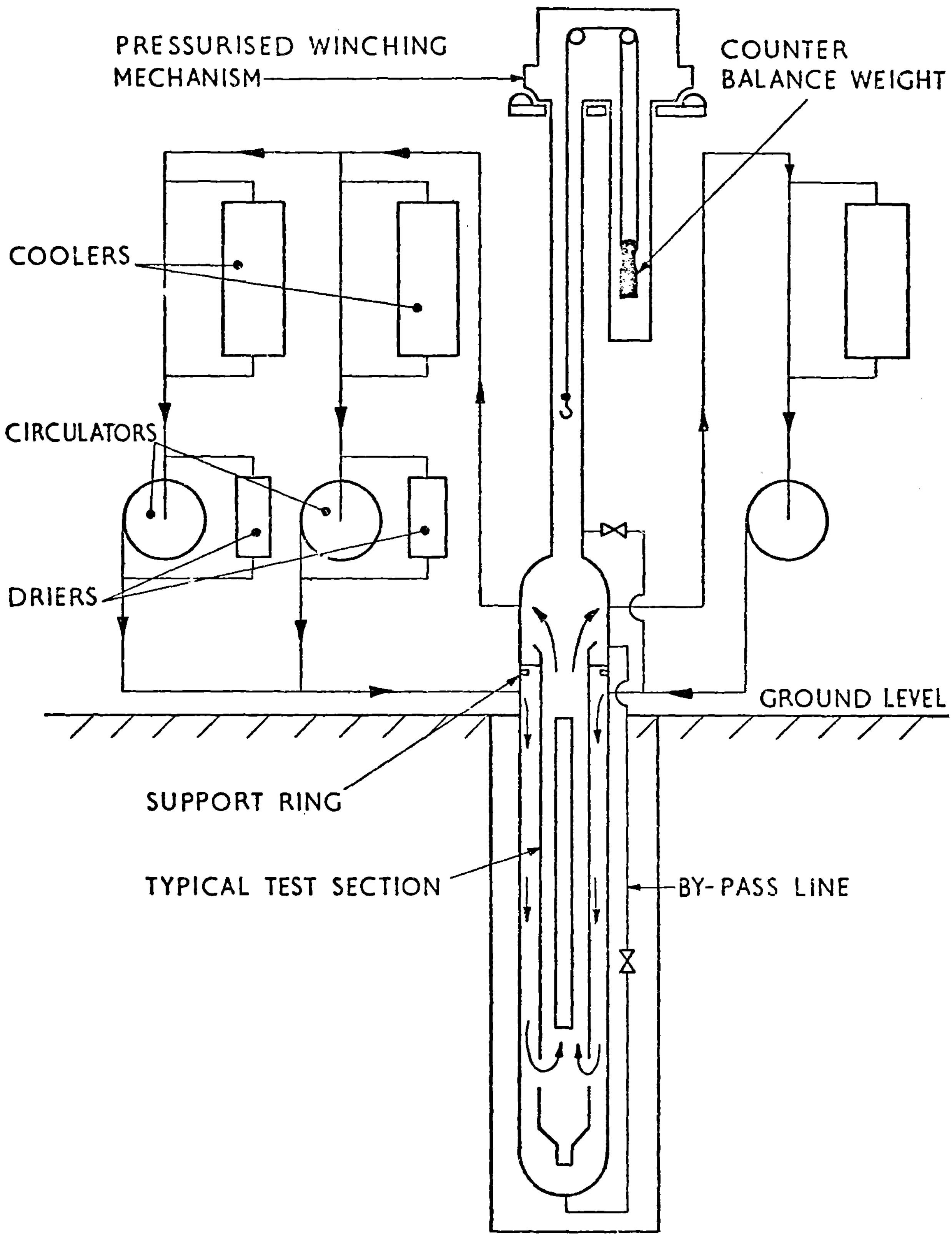


DIAGRAM OF THE CAGR RIG

HIGH PRESSURE CO_2 LOOP

LOCATION: RISLEY NUCLEAR POWER DEVELOPMENT LABORATORY, UKAEA, RISLEY, WARRINGTON, ENGLAND

STATUS: Operational

PRINCIPAL USES: (a) Formerly used for vibration and rattling experiments in gas-cooled reactor fuel stringers.

(b) Now used for general pressure loss tests on fuel sub-assemblies for sodium cooled fast reactors.

(c) Pressure loss checks over a wide range of Reynolds numbers on fuel element flow restrictors.

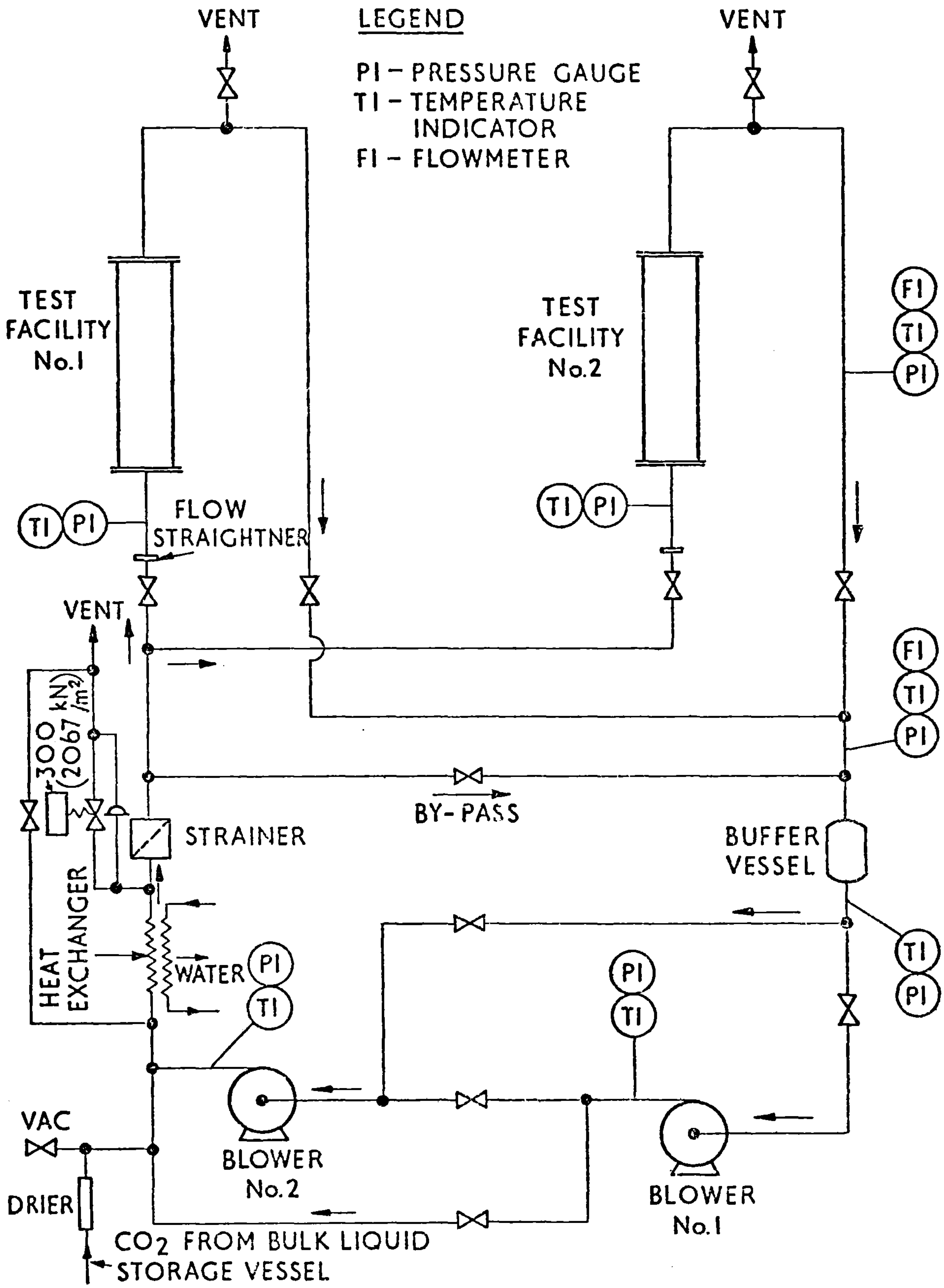
DESCRIPTION: The test loop is laid out to form four vertical legs of 15 cm and 25 cm pipework; both upward and downward flow can be obtained. The vertical lengths can extend to 18.5 m if required.

CO_2 at various pressures from atmospheric to 1730 kN/m^2 can be circulated at up to 0.24 m^3/s. Gas temperatures can be held steady by the use of the built-in gas-to-water shell-and-tube heat exchanger.

Flow measurement is by an eddy shedding device giving an accuracy of $\pm$0.2%. Pressure loss measurements can be made to within 0.1% accuracy.

TEST CAPABILITY: The two gas circulators can each provide a flow of 0.12 m^3/s at 1730 kN/m^2 pressure with 104 kN/m^2 rise across the machine, and they can be operated individually or in series or parallel. The upper operating gas temperature is limited to around 60°C.

AUXILIARY EQUIPMENT: Working platform at various levels. Peripheral read-out and recording equipment, and the necessary engineering back-up services.



HIGH PRESSURE CO_2 LOOP

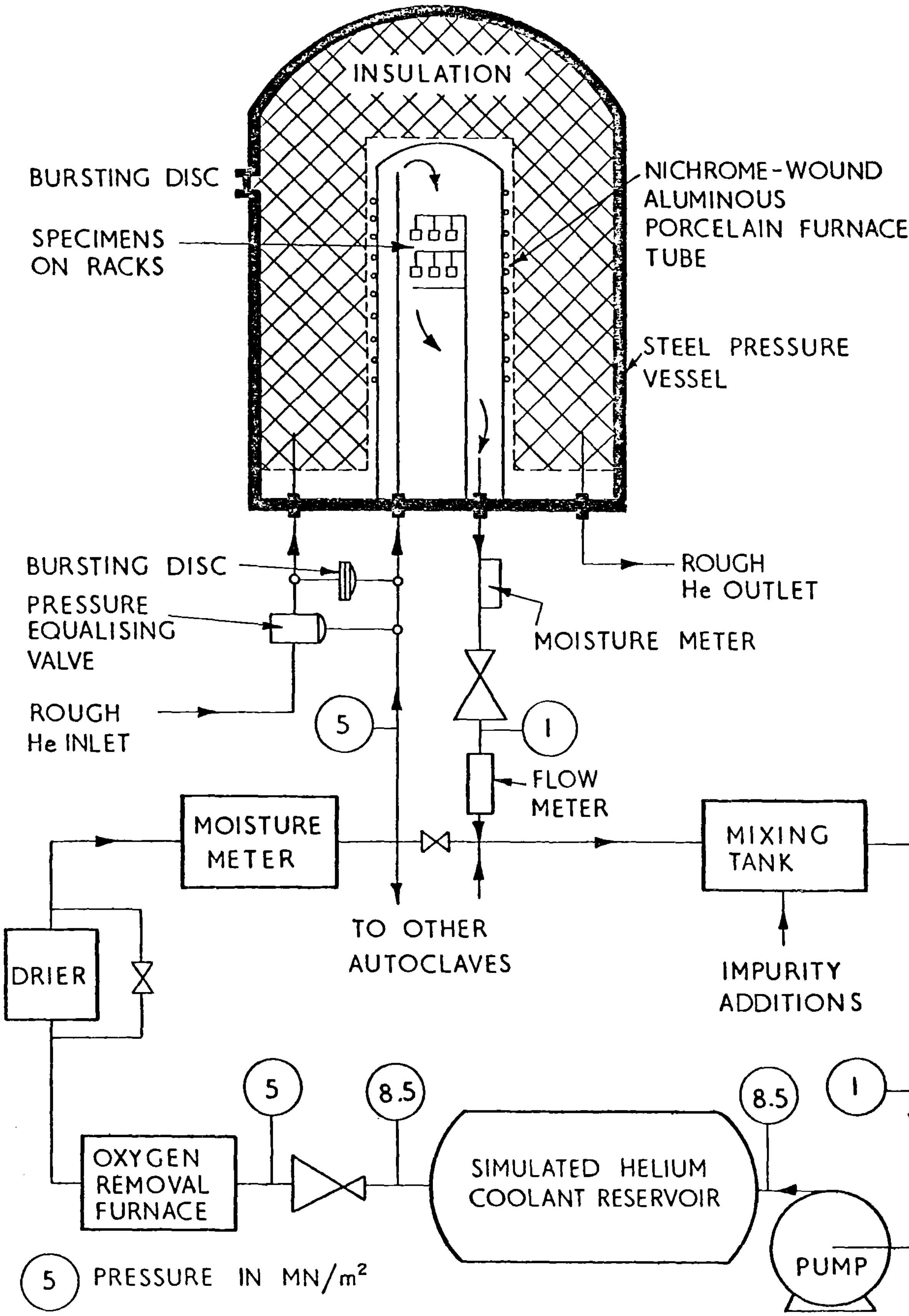
CERL HIGH PRESSURE HELIUM LOOP

LOCATION: CENTRAL ELECTRICITY RESEARCH LABORATORIES, CEGB, LEATHERHEAD, SURREY.

STATUS: Dismantled but could be re-assembled.

PRINCIPAL USES: The rig was used for compatibility testing of materials and small assemblies for the primary circuit of an HTR. Controlled impurity helium gas was also supplied to creep test machines for studies of mechanical properties in helium.

DESCRIPTION: The rig consists of a high pressure (5 MN/m^2) helium recirculating loop which feeds six autoclaves with helium via a manifold. The maximum flow rate is 0.1 g/s and the autoclaves operate at up to 850°C with hot zones of approximate dimensions 200 mm x 60 mm diameter. Impurities are monitored by gas chromatography and moisture meters and are maintained at the chosen levels by either direct injection to increase the level or passage through a by-pass loop containing molecular sieve in liquid nitrogen to decrease the level.



CERL HIGH PRESSURE HELIUM LOOP

9. United States

The following codes are available which could be used for GCFR work:

LIFE: lifetime performance evaluation of fuel element pins

FRAP-T, DEFORM2, MARC: for transient performance evaluation of fuel element pins.

TEPC, ANSYS: analysis of clad stress and distorsion (inclusive of clad interaction with fuel and spacers)

PECT, PEFT (General Electric): probabilistic assessment of fuel pin reliability

COBRA, CINDA: steady state thermohydraulic codes for fuel assembly calculations

FLOMAX, COBRA3C, HEATING2: for transient thermohydraulic calculations

SAS-GAS: code for accident calculations in the core (inclusive of clad and fuel movement)

VENUS: code for core-disruptive accident calculations

The following facilities have been used in the frame of GCFR-US programme:

ORR reactor for irradiations of pin capsules in a thermal flux

EBR2 reactor for irradiations of pin capsules in a thermal fast flux

ZPR IX (ANL): zero power fast reactor used for the determination of water reactivity in a GCFR core

A large Core FLow Test Facility (CFTL), which will allow tests of large bundles of rods (up to 91 rods) in steady and transient conditions is in construction at the Oak Ridge National Laboratory. Figure 14 is an illustration of the CFTL layout for an upflow core configuration.
Steel melting and relocation test (SMART) facility - Los Alamos Scientific Laboratory (LASL) - Purpose of this test facility is to demonstrate the out-of-pile behavior of a GCFR core assembly in the event of a loss-of-core coolant flow or pressure and subsequent shutdown of reactor power to the level resulting from decay heat alone.

PNL Helium Loop: high temperature circulating helium loop for testing GCFR cladding and structural materials in a controlled impurity atmosphere.

Direct Electrical Heating Loop: ANL apparatus to investigate the behavior of fresh and irradiated fuel subject to transient heating similar to GCFR hypothetical accident conditions.

At the Idaho National Laboratory the Gas Reactor In-Pile Safety Test loop (GRIST2), a transient overpower test loop to study the fuel behaviour under high power transient conditions, has been planned.

10. Gas Breeder Reactor Association

Full members of GBRA are the following organisations:

AB ASEA-ATOM, Västeras, Sweden

Belgonucleaire S.A., Brussels, Belgium

Brown Boveri-Sulzer Turbomaschinen A.G., Zürich, Switzerland

Centre d'Etude de l'Energie Nucléaire,
Studiecentrum voor Kernenergie Brussels, Belgium

Hochtemperatur Reaktorbau GmbH, Köln, Germany

B.V. Neratoom, Den Haag, Netherlands

Nucleare Italiana Reattori Avanzati, Genova, Italy

Technicatome, Paris, France

The Nuclear Power Group Limited, United Kingdom

and associate members the following:

Atomkraftkonsortiet Krangede AB & Co, Sweden

Central Electricity Generating Board, United Kingdom

South of Scotland Electricity Board, United Kingdom

Statens Vattenfallsverk, Sweden

Vereinigte Elektrizitätswerke Westfalen A.G., Germany

III. PROGRAMMES

The GCFR research and development programmes of the countries participating the NEA-GCFR Collaborative Programme are illustrated in the attached **Table A to GBRA.**

IV. CONCLUSIONS AND RECOMMENDATIONS

During the past years various designs have been proposed for large GCFR. Two have been studied in considerable detail. The main characteristics of these designs are given in the Table below:

Design	GA (Fig.9)	GBRA (Fig.1)
Coolant	helium	helium
Coolant pressure (bar)	88	90
Fuel pin venting	yes	yes
Pin diameter (cm)	0.72	0.7
Surface roughening	yes	yes
Coolant flow direction in core	upward	upward
Fuel handling	from below with fuel manipulator	from above with fuel manipulator
Blower drive	electric motor	electric motor
PCRV type	pod boiler	pod boiler
PCRV liner type	cold	cold
Core-catcher	Borax:outside reactor cavern	not yet decided, probably inside reactor cavern

One can see from the Table that, with the exception of the fuel handling system and of core-catcher design, the two designs are very similar.

The editor of this report feels that there is time for further design improvements and simplifications in the design of the GCFR. These improvements seem to be more appropriate with modern trends and requirements. One could be the use of a single cavity PCRV (see recent AGR's ordered in the UK). Another, the use of bigger and longer pins. These bigger pins (about 1 cm in diameter; similar in size to the PWR fuel pins) would cause an increase of plutonium inventory, but the fuel cycle costs would decrease considerably. Furthermore the use of large pin lattices (p/d=1.5, again similar to those of PWR cores) would still allow to reach the high breeding gains typical of the GCFR, but at the same time it would permit a dramatic decrease of the hot spot problems given by geometrical tolerances, bowing and differential swelling.

A further simplification could be, if at all possible, the elimination the venting system.
The editor of this report is also of the opinion that presently the most important R. and D. items still to be investigated for the GCFR are the following:

a) Out of pile tests of electrically heated bundles with a large number of pins, subjected to bowing, to simulated swelling and pin displacements. These tests should be performed in steady state and in transient conditions.

b) Corrosion tests of cladding tubes with selected roughnesses in presence of helium with known amounts of H_2O/H_2 impurities.

c) Irradiation experiments of roughened pins in a fast flux to investigate stress concentration and ductility problems.

d) Investigation of alternative (round) fuel elements, if one sticks to rough fuel pins of small diameters (The round fuel elements would make the hot spot problems at the subassembly walls easier in case of swelling and shroud deformation).

e) Investigation of the possibility of using a hot linear in the PCRV.

f) Core-catcher investigations.

g) Development and test of large helium blowers.

Table I: Main Parameters of Helium-cooled Breeder Reactors of 1000 MWe Compared to Advanced Sodium- and Steam-cooled Types

Concept No.	1	2	3	Advanced Na-Breeder	Steam Breeder
Cycle	Steam turbine	Gas turbine	Steam turbine	Steam turbine	Steam turbine
Fuel	Oxide	Oxide	Oxide	Oxide	Oxide
Fuel element	Fuel pin (vented)	Fuel pin (sealed can)	Coat. particle	Fuel pin (sealed can)	Fuel pin (vented)
Max. lin. power rating in pin W/cm	430	440	---	530	420
Mean discharge burn up MWd/t	75 000				
Inlet coolant pressure kg/cm^2	70	100	70	10	150
Mixed mean coolant temp. at reactor outlet °C	600	706	675	580	500
Max. hot spot temp. at clad midwall °C	755	850	950	700	720
Core fissile inventory kg Pu^{239} . Pu^{241}	3140	2770	1800	1630	2860
Breeding ratio	1.44	1.32	1.19	1.29	1.15
System lin. doubling time yrs^{+}	13.2	17.8	31.8	14.5	32.3
Specific investment $/kWe	162	145	162	170-240	$152^{*)}$
Fuel cycle cost mills/kWh	1.3	1.5	1.5	0.875	$1.4^{*)}$
Electricity cost $mills/kWh^{+}$	5.2	5.05	5.4	5.0-6.5	$5.2^{*)}$

$^{+}$ Load factor 0.7

All costs are for the spring 1970; $^{*)}$ estimated costs.

Table II

Comparative characteristics of the three first GBRA designs

	Unit	GBR 1	GBR 2	GBR 3
Electrical output	Mwe	1,000	1,000	1,000
Primary gas pressure	bar	120	120	60
Total gas pumping power	Mwe	107	78	88
Inlet coolant temperature	°C	260	260	260
Mixed mean outlet temperature	°C	587	700	650
Steam pressure/temperature	bar/°C	115/540	115/540	115/540
Number of loops	-	8	6	8
Fissile inventory (system)	kg	4,310	2,800	3,070
Breeding gain	-	0,43	0,36	0,42
Doubling time	y	13	16	16

- T A B L E III -

PARAMETERS FOR 1200 MWe G B R 4

Temperature (core outlet)	°C	560
Temperature (core inlet)	°C	260
Coolant pressure (core inlet)	bar	90
Core pressure drop	bar	2.4
Pumping power	MWe	124
Net efficiency	%	35
Peak linear rating	W/cm	400
Mid-cycle fissile enrichment	%	13.2
Peak burn-up	MWd/kg	100
Peak fluence (E > 0.1 MeV)	$10^{-23}cm^{-2}$	2.5
Refuelling interval (0.75 LF)	y	1
Core fuel in-pile time (0.75 LF)	y	3
Burn-up reactivity	%	0.6
Start-up fissile core inventory	kg/MWe	3.92
Breeding ratio	-	1.40
System doubling time	y	11.8
Net fissile Pu production	kg/MWe.y	.287

- TABLE IV -

MAIN PERFORMANCE DATA FOR ALTERNATIVE

1200 MWe GBR DESIGNS WITH He-COOLED MIXED OXIDE PIN FUEL

PARAMETER	DESIGN	GBR -4 Reference	GBR -4 High rated	ADVANCED GBR TARGET
Fuel element technology :				
- random peak clad hot-spot temperature	°C	720		785
- peak mixed oxide fuel burn-up	MWd/kg	100		100
Thermal plant data :				
- peak coolant (He) pressure	bar	90		120
- reactor pressure drop	bar	2.4		4.4
- reactor coolant exit temperature	°C	560		615
- total He-circulator power	MWe	126		126
- steam cycle		non-reheat		reheat
- thermal plant net efficiency		0.35		0.38
Core data :				
- core fuel in-pile time at LF = 0.75	years	3	2	1.5
- refuelling interval	years	1	1	0.5
- inner fuel can diameter	cm	0.70	0.53	0.53
- peak linear fuel pin rating	W/cm	400	350	450
- initial fissile core inventory	kg/MWe	4.1	3.2	2.3
- total initial fissile system inventory at out-of-pile time 0.75 years	kg/MWe	5.1	4.4	3.5
- net fissile plutonium production at LF = 0.75 and 2 % losses	kg/MWe year	0.29	0.28	0.27
- breeder system doubling time at LF = 0.75 and 0.75 years out-of-pile time	year	12.2	10.9	9.0

Table V: Main Data of the BR2 Irradiation Experiment

Test Fuel Element Bundle Data	
Number of pins	12
Pin outer diameter	8 mm
Pin pitch	11.1 mm
Fuel	$(U/Pu)O_2$
Cladding material	stainless steel 1.4981
Pin surface	artificially roughened
Max. linear pin rating	450 W/cm
Max. clad surface temp. (hot spot)	680°C
Burn-up objective	60000 (100000) MWd/t
Loop Data	
Cooling gas	helium
Operation pressure	60 bar
Mass flow	0.25 Kg/sec
He inlet temperature	255°C
He outlet temperature	510°C

Table VI: Main Data of 1000 MWe Reference Design (GSB-1)

Coolant pressure	120 bar
Coolant inlet temperature	273°C
Coolant outlet temperature,	555°C
Core height	148 cm
Core H/D	0.5
Pin diameter	8.2 mm
Pin pitch	11 mm
Hot spot temp., mid clad	700°C
Max. linear rating	492 W/cm
Fissile rating core	0.78 MWth/Kg
Breeding ratio	1.40
Core plutonium fissile inventory	3230 Kg
Linear doubling time	11.8 yrs
Plant net efficiency (wet cooling tower)	37 %

Table VII: Main Data of 1ooo MWe Nuclear Steam Supply System (GSB-1)

Vessel	PCRV, Pod Boiler
Core and Blanket	
type of support	top clamped, in individual standpipes
flow direction	downward
refueling access	from beneath
No. of Main Loops	8
diameter of boiler cavity, m	3.5
closure design	doubly retained concrete plug with flow limiter
Coolant Circulation	single stage axial blowers, series-steam driven
blower power, MW	8 x 16.5
No. of Auxiliary Loops	4
coolant circulation	elect. driven radial blowers, single stage
blower power, MW (depressurization cond.)	4 x 1.4
secondary containment pressure, bar (depressurization cond.)	3

Table VIII: Safety Related Nuclear Characteristics. 1ooo MWe GCFBR (GSB-1)

Av. enrichment Pu_{fiss}.	12.7 %
Core conversion ratio	o.87
Reactivity loss per cycle	1.6 $
Doppler effect, Tdk/dT	.oo61
β_{eff}	o.324 x $1o^{-2}$
Helium void reactivity	o.88 $
Cladding expansion reactivity coefficient	-o.227 x $1o^{-5}$
Fuel expansion reactivity coefficient	-o.126 x $1o^{-5}$
Power coefficient (prompt)	-1.5 x $1o^{-6}$ MW^{-1}
Total control requirements	9.o $
Number of control rods	12
Worth of 1 rod	o.83 $
Number of shut down rods	2 x 3
Worth of 1 rod	3.3 $

Table IX:

PROGRAM DEFINITION AND LICENSING PHASE
SCOPE OF WORK

- PERFORM PRELIMINARY SITE EVALUATION STUDIES
- PREPARE PRELIMINARY SAFETY ANALYSIS REPORT
- PREPARE ENVIRONMENTAL IMPACT STATEMENT
- COMPLETE APPROXIMATELY 65% OF THE OVERALL PLANT ENGINEERING DESIGN WORK
- DETERMINE PLANT SIZE, R&D PROGRAM, COST AND SCHEDULE
- OBTAIN SITE CONSTRUCTION PERMIT
- ESTABLISH INSTITUTIONAL RELATIONS AMONG PARTICIPANTS, INCLUDING COST SHARING AND RISK ROLES
- PERFORM R&D AND SAFETY RESEARCH REQUIRED TO ACCOMPLISH ABOVE

Table X:

	FY-79	FY-80	FY-81	FY-82	FY-83	FY-84	FY-85
DESIGN: NSSS		Complete Design Definition			Complete Preliminary Design		Complete 70% Engr'g Criti-cal Systems
CORE		Complete Design Definition					Complete Preliminary Design
BOP		Select A/E			Complete Preliminary Design		Complete 70% Engr'g Criti-cal Systems
LICENSING:			Start PSAR & ER	File ER	File PSAR		
DEVELOPMENT CORE						Complete CFTL Loop	Complete Prel. CFTL Tests
PHYSICS & SHIELDING	Complete Grid Plate Shielding Test						Complete Core Shielding Analysis
SAFETY				Complete Smart Test			Install GRIST II Test Train
COMPONENT					Complete PCRV Clos. Tests Circulator Test Facility Schedule 44 Approval	Complete PCRV Thermal Barrier Tests	

Table XI:

US GCFR FUNDING - FOUR YEAR FORECAST
(DOLLARS IN MILLIONS)

FISCAL YEAR	PDLP	SAFETY	TOTAL PROGRAM
1977	13.5	3.3	16.8
1978	14.4	3.6	18.0
1979	21.1	4.6	25.7
1980	22.4	6.2	28.6
TOTALS	71.4	17.7	89.1

Table XII:

CORE DEVELOPMENT PROGRAM

KEY ACTIVITIES

- o FUEL AND MATERIALS DEVELOPMENT - ANL, GA (FRG)
- o THERMAL-HYDRAULIC - ORNL, GA (SWISS, FRG)
- o PRESSURE EQUALIZATION SYSTEM - GA (FRG)
- o FABRICATION TECHNIQUES - GA (FRG)
- o IN-PILE AND OUT-OF-PILE TESTING - ORNL, ANL, GA (FRG, SWISS, BELGIUM)
- o DESIGN AND ANALYSIS - ANL, GA, ORNL (FRG, SWISS)

FUNDING LEVELS (IN THOUSANDS)

CONTRACTOR	1977	1978	1979	1980
GA	1910	1815	2940	4100
ANL	950	900	1100	1250
ORNL	1865	2250	4000	4500

MAJOR ACTIVITIES (1977-1979)

- o COMPLETE DESIGN AND START FABRICATION OF CFTL
- o INITIATE BR-2 (HELM) EXPERIMENT
- o FABRICATE AND IRRADIATE F-5 FUEL ASSEMBLY
- o DETERMINE PROPERTIES OF IRRADIATED RIBBED CLADDING
- o EVALUATE U-Th FUEL CONCEPTS
- o COMPLETE POST IRRADIATION EXAMINATION OF F-1, F-10, AND GB-10

Table XIII:

PHYSICS AND SHIELDING PROGRAM

KEY ACTIVITIES

- o TOWER SHIELDING FACILITY EXPERIMENTS - ORNL, GA (FRG)
- o ZPR CRITICALS - ANL, GA (FRG, SWISS)
- o METHODS DEVELOPMENT, DESIGN AND ANALYSIS - ORNL, GA, ANL (FRG, SWISS)

FUNDING LEVELS (IN THOUSANDS)

CONTRACTOR	1977	1978	1979	1980
ORNL	150	600	875	750
ANL	150	100	100	100
GA	465	450	750	570

MAJOR ACTIVITIES (1977-1979)

- o PERFORM GRID PLATE SHIELDING EXPERIMENTS
- o COMPLETE ANALYSIS OF ZPR CRITICAL EXPERIMENTS
- o PLAN ENGINEERING MOCKUP CRITICALS
- o ALTERNATE CORE DESIGN STUDIES (PROLIFERATION RESISTANT CORES)
- o PERFORM RADIAL SHIELDING EXPERIMENTS

Table XIV:

COMPONENTS AND SYSTEMS PROGRAM

KEY ACTIVITIES

- o NSSS COMPONENT DEVELOPMENT - GA (FRG, AUSTRIA, SWEDEN, BELGIUM)
- o NUCLEAR ISLAND DESIGN - A-E (FRG)
- o REACTOR SYSTEMS ENGINEERING - GA (FRG)

FUNDING LEVELS (IN THOUSANDS)

CONTRACTOR	1977	1978	1979	1980
GA	2575	2650	5750	8000
A-E	0	100	400	500
ORNL	150	150	160	210

MAJOR ACTIVITIES (1977-1979)

- o EVALUATE ALTERNATE DESIGNS AND SELECT REFERENCE US 300 MW(E) PLANT DESIGN
- o PERFORM CONCEPTUAL DESIGN STUDIES OF NSSS COMPONENTS
- o INITIATE NUCLEAR ISLAND DESIGN
- o CONSOLIDATE US-FRG NSSS DESIGNS INTO ONE COMMON REFERENCE DESIGN
- o PERFORM SCALE MODEL PCRV CLOSURE TESTS
- o INITIATE CIRCULATOR TEST FACILITY CONCEPTUAL DESIGN
- o COMPLETE CONCEPTUAL DESIGN OF REFERENCE PLANT

Table XV:

GENERAL ATOMIC COMPANY SAFETY PROGRAM

PROGRAM: GCFR SAFETY TEST PROGRAM

FUNDING LEVELS: (IN THOUSANDS)

1976	1977	1978	1979	1980
$ 100	$ 125	$ 100	$ 200	$ 250

MAJOR ACTIVITIES (1977-1979):

1977
- o COORDINATE THE GRIST-2 TEST AMONG GA, ANL, AND EG&G
- o DEFINE THE DMFT PROGRAM AT LASL

1978
- o PROVIDE PRELIMINARY GRIST-2 TEST ASSEMBLY DESIGN
- o PLAN THE DMFT AND DAC TESTS AT LASL

1979
- o DEVELOP DETAILED GRIST-2 TEST PROGRAM
- o ANALYZE AND INTERPRET DMFT AND DAC TEST RESULTS

1980
- o DEFINE STEEL MELTING AND RELOCATION TEST PROGRAM AT LASL
- o CONTINUE GRIST-2 SUPPORT

Table XVI:

GENERAL ATOMIC COMPANY SAFETY PROGRAM

PROGRAM: GCFR REACTOR SAFETY, ENVIRONMENTAL AND RISK ANALYSIS

FUNDING LEVELS: (IN THOUSANDS)

1976	1977	1978	1979	1980
$ 205	$ 660	$ 580	$ 945	$1500

MAJOR ACTIVITIES (1977-1979):

1977
- o OVERALL GCFR SAFETY PROGRAM PLAN
- o RELIABILITY ANALYSIS OF THE DECAY HEAT REMOVAL SYSTEM
- o ASSESSMENT OF POST-ACCIDENT FUEL CONTAINMENT (PAFC) WITHIN THE PCRV

1978
- o PRELIMINARY ANALYSIS OF THE LOSS OF DECAY HEAT REMOVAL ACCIDENT
- o EVALUATION OF CORE CATCHER CONCEPTS
- o ASSESSMENT OF PAFC EXTERNAL TO THE PCRV

1979
- o IDENTIFY POTENTIAL DESIGN IMPROVEMENTS BASED ON RISK ANALYSIS RESULTS
- o IDENTIFY R&D PROGRAMS TO REDUCE RISK UNCERTAINTIES
- o DEFINE DESIGN REQUIREMENTS FOR PAFC

1980
- o PREPARE REVISED SAFETY PROGRAM PLAN
- o SUBMIT LICENSING AMENDMENTS TO NRC FOR REVIEW
- o ANALYZE NATURAL CIRCULATION CAPABILITY OF UPFLOW CORE

Table XVII:

ANL SAFETY PROGRAM

PROGRAM: GCFR SAFETY ASPECTS OF FUEL AND CORE

FUNDING LEVELS: (IN THOUSANDS)

1976	1977	1978	1979	1980
$ 660	$ 700	$ 610	$ 825	$1000

MAJOR ACTIVITIES (1977-1979):

1977
- o ANALYZE THE EFFECTS OF HIGH BURNUP AND ABSORBED HELIUM ON ACCIDENTS
- o ANALYZE THE POST-ACCIDENT CORE DEBRIS BEHAVIOR (FUEL-GRAPHITE AND FUEL-CONCRETE INTERACTIONS)
- o COMPLETE HIGH PRESSURE, FLOWING HELIUM, DEH-TEST CHAMBER

1978
- o DEMONSTRATE EFFECTS OF FUEL SWEEPOUT DURING HIGH-RATE TOP ACCIDENTS
- o CDA ANALYSIS FOR A GCFR DEMO PLANT DESIGN

1979
- o INTEGRATION OF EARLY LASL AND ANL TEST RESULTS INTO ACCIDENT ANALYSES
- o PRELIMINARY DESIGNS AND TEST SPECIFICATIONS FOR THE GRIST-2 TEST TRAINS

1980
- o MODIFY HOT CELL TO ACCOMMODATE TESTING OF IRRADIATED FUEL

Table XVIII:

LASL SAFETY PROGRAMS

PROGRAM AND FUNDING (IN THOUSANDS)	FY 1977	FY 1978	FY 1979	FY 1980
DUCT MELT AND FALL AWAY TESTS (DMFT)	$ 270	$ 600	$ 675	$ 510
DEPRESSURIZATION ACCIDENT (DAC)		40	100	200

MAJOR ACTIVITIES (1977-1979)

1977
- o DESIGN TEST FIXTURES
- o MATERIAL CHARACTERIZATION UNDER TEST CONDITIONS

1978
- o EFFECT OF He PRESSURE AND NATURAL CONVECTION ON FUEL MELT-DOWN
- o CLADDING AND DUCT WALL MELT-DOWN BEHAVIOR

1979
- o RUN FULL LENGTH SUBASSEMBLY DMFT TESTS
- o MODIFY FACILITY FOR GUARDED CORE MODULE TESTS
- o REVISE TEST PROGRAM TO ACCOMMODATE NEW REFERENCE UPFLOW CORE DESIGN

1980
- o COMPLETE FULL LENGTH SUBASSEMBLY TESTS
- o COMPLETE CONSTRUCTION OF TEST FACILITY AND TEST LOOP
- o RUN GUARDED CORE MODULE TEST

Table XIX:

SPS SITE SELECTION PROGRAM

PROGRAM: PRELIMINARY SITE EVALUATION AND ENVIRONMENTAL IMPACT STUDY

FUNDING LEVELS: (IN THOUSANDS)

1977	1978	1979
$ 400	$ 280	$ 0

MAJOR ACTIVITIES (1977-1979):

1977
- o SELECT RECOMMENDED SITE AND TWO OR MORE ALTERNATIVE SITES
- o DETERMINE PRELIMINARY SEISMIC ACCELERATION AND FOUNDATION GEOLOGY
- o ORDER AND ERECT METEOROLOGICAL TOWER; INSTALL AND CALIBRATE INSTRUMENTS

1978
- o COLLECT ENVIRONMENTAL AND SITE-SPECIFIC DATA FOR PSAR CHAPTER 2 AND ER CHAPTER 2

1979
- o SITE EVALUATION WORK TERMINATED SOUTHWEST PUBLIC SERVICES WITHDREW SITE OFFER

Table A CONTRIBUTIONS OF AUSTRIA TO THE NEA-GCFR R&D PROGRAM

Coordinator : W BINNER

ACTIVITY	RESPONSIBLE	OBJECTIVES	COMMENTS
1. Prestressed concrete vessel-helium station	WITT RS	To gain experience in the construction and performance of PCPV suited for gas cooled reactors with:	The vessel of 1.5 m i.d. and 8 m i.h. has a steel lid and several o.3-o.7m i.d. penetrations. Steady state and cycling pressure and heating tests are performed with water, air and helium.
1.1 Construction and testing of a large prestressed concrete vessel (PCPV) model with hot liner.	NEMET RFB	- a "hot liner" in direct contact with the coolant at max.3oo-35o°C; - a coolant design pressure of 1oo bar	After the construction of the PCPV, preliminary tests on concrete had to be performed. Experiments at 1oo bar and with hot gas (3oo°C, 1oo bar) are carried out.
1.2 Construction and testing of components in a He-Loop.	WITT RS	In the PCPV a He-Loop with process heat components will be tested.	With an electrical heater of 1 MW the He is heated up to a temperature of 1ooo°C at 1oo bar. At the outlet of the steam generator the gas is cooled down to 3oo°C.
2. Coated particle development			
2.1 Fabrication studies of GCFR particles with alternative outer coatings instead of SiC.	KOSS ME	Development of a back-up coating using metals or alternative carbides.	Including heat treatment, out-of-pile compatibility tests and examination of the coating characteristics.
2.2 Out-of-pile study of the effect of an oxidising coolant (CO_2) on cracked GCFR particles.	PROKSCH CH	Study of the possible gross failure of cracked particles by kernel swelling due to oxidation at different burnup levels.	work discontinued
2.3 Out-of-pile studies on the compatibility of broken GCFR particles at high burnup on adjacent SiC-coated particles.	PROKSCH CH	Investigation of the effect of fission products and Pu on the protective SiO_2 layer on SiC coatings at temperatures 8oo-11oo°C.	work discontinued
2.4 Experimental and theoretical study on the pressure build up in GCFR particles.	PROKSCH CH	To establish design criteria for GCFR particles by investigation and assessment of gas pressure, gas content and free volume in particles at different burnups and irradiation conditions.	-"-
2.5 Measurements of thermal expansion coefficient of SiC.	KOSS ME	Investigation of the influence of high fast dose in SiC properties.	-"-

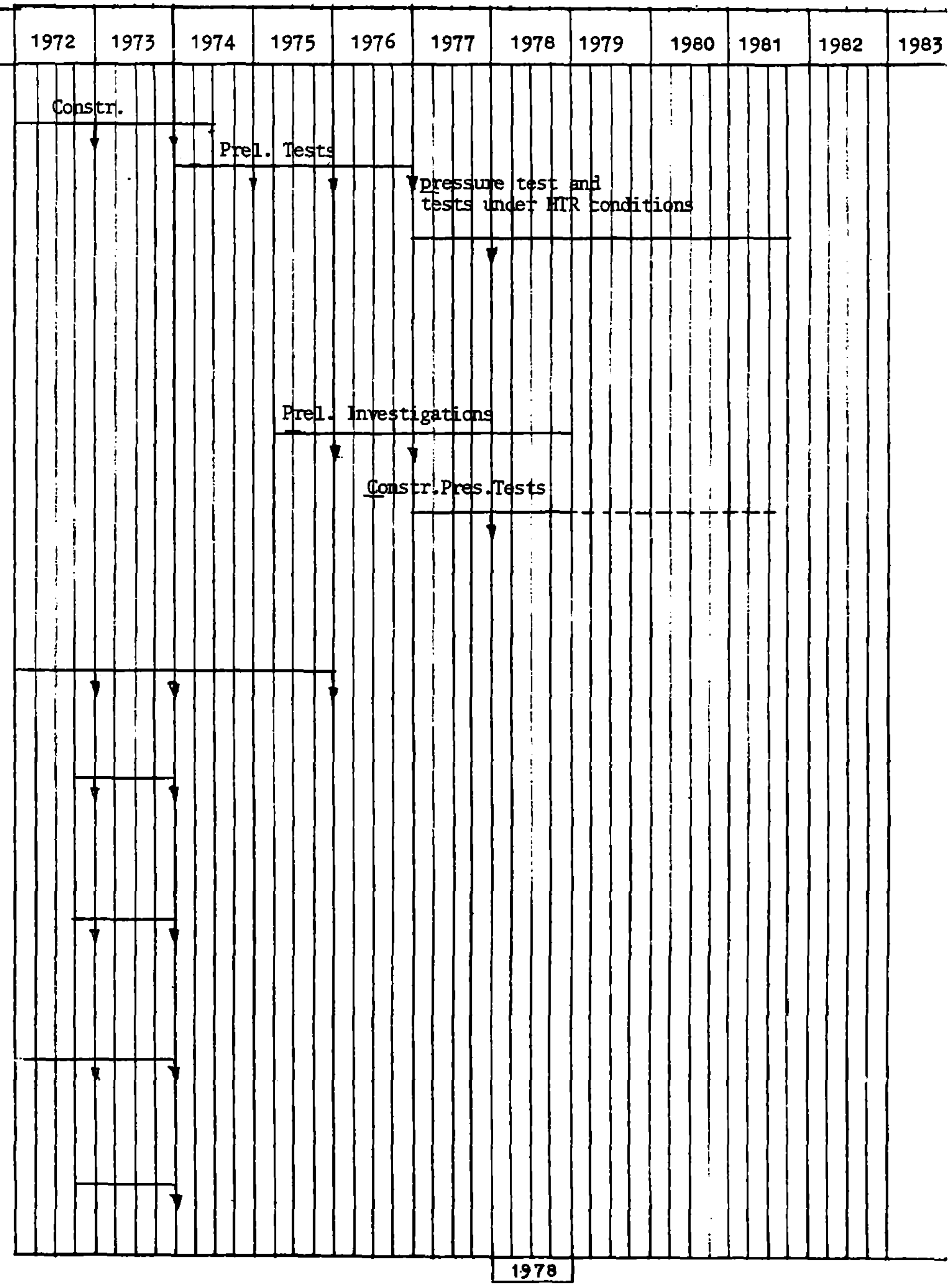


Table B CONTRIBUTIONS OF BELGIUM TO THE NEA-GCFR R&D PROGRAM

Coordinator : J. PLANQUART C.E.N/S.C.K. - MOL

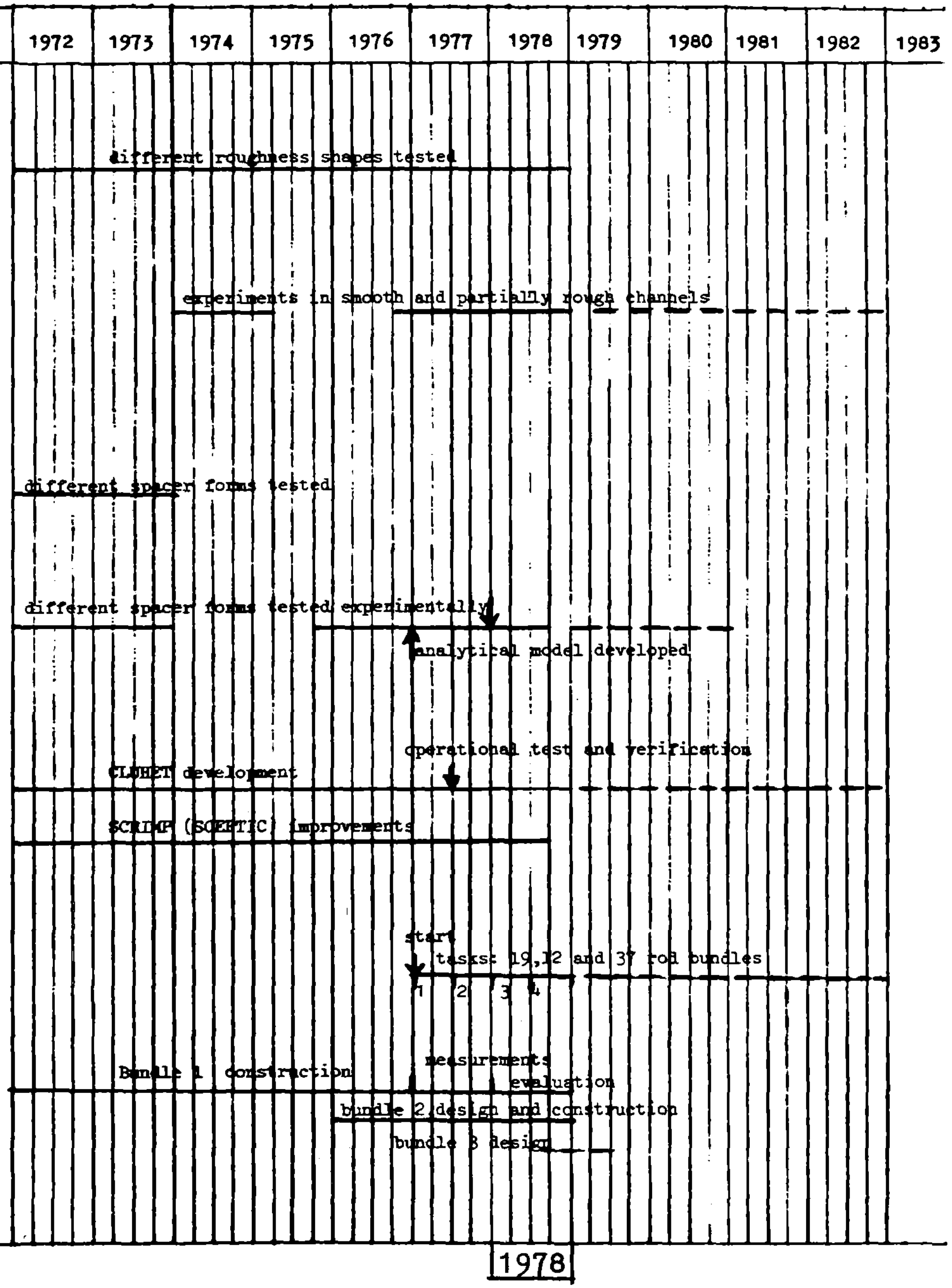
ACTIVITY	RESPONSIBLE	OBJECTIVES	COMMENTS	1972 – 1983
1. Ferritic stainless steels	J.J.HUET	To develop a canning and structural material for fast reactor conditions	To have material with lower swelling rates and better ductility than for austenitic stainless steels.	
1.1. Thin walled tube development		Optimization of powder preparation and fabrication parameters.		
1.2. Irradiation tests	P. VAN ASBROECK			
1.3. Irradiation of fuel pins in BR2.			700°C inside peak temp.: (U,Pu) oxide	
2. Helium Technology	A.FALLA			
2.1. Helium loop "HE 1" (Out of Pile)		-Control and measurement of impurities in helium -High temperature technology -Friction and fretting tests.	Up to 1100°C	Constr. / Operation / Constr. / Tests
3. Fuel assemblies	G.VANMASSENHOVE			Study Design / Construction / HELM 1 / HELM 2 / HELM 3 / HELM 3
3.1. Irradiation of 12 vented fuel pins in BR2	P.VANDERSTRAETEN	-Demonstration of the feasibility of a vented pin assembly and the corresponding fission product trapping system.	In collaboration with FRG (see point 1.3. of table 4).	
3.2. Irradiation of vented fuel rod in BR2	F.MOONS			Design Study
4. Carbide Fuel	A. DELBRASSINE			
4.1. Fabrication development.		Fabrication processes	-Adapt (U,Pu)C fabrication line; -Increase production capacity.	
4.2. Irradiation tests				
a) POM-Type capsules (pellets)(in BR2)		Comparison of various fuel types	Target burn-up > 100,000 MWd/t .	
b) CIRCE type experiments (in BR2)		Capsule behaviour Compatibility swelling	Up to 100,000 MWd/t 1000 W.cm^{-1}.	
c) MFBS irradiation (in BR2)		Fuel pin testing in flowing Na	6 twin pins including 2 Vipac pins from EIR.	
d) DFR irradiation		Fast flux experiment with 3 (U,Pu)C pins	700 W.cm^{-1} .	
5. Other programs (p.m.) - Development of coated particles - Hot-spot effects on particle bed.				

1978

Table CH CONTRIBUTIONS OF SWITZERLAND TO THE NEA-GCFR R&D PROGRAM

Coordinator : Dr. G. Markoczy
Swiss Federal Institute of Reactor Research

ACTIVITY	RESPONSIBLE	OBJECTIVES	COMMENTS
1. CORE HEAT TRANSFER AND FLUID FLOW			
1.1 Surface roughness performance investigations	Hudina	GCFR core heat transfer and fluid flow studies in order to develop analytical models for the prediction of the temperature and pressure distribution in the GCFR fuel elements.	Experiments in air and CO_2 with single rod (annular geometry).
1.2 Analysis of subchannel flow structure and coolant "mixing"	Hudina		Measurements of velocity, pressure drop, shear stress, turbulence intensity and mixing in simple channel geometries.
1.3 Investigation of grid spacer effect on the local heat transfer	Hudina		Experiment with air in annular geometry measuring disturbed and undisturbed heat transfer coefficients.
1.4 Grid spacer pressure drop investigations	Barroyer		Bundle experiments with air under low pressure conditions. Analytical model development.
1.5 Computer codes development CLUHET SCRIMP	Hudina Barroyer Huggenberger		Synthesis of the fundamental phenomeny in the comprehensive thermal-hydraulic design codes.
1.6 Benchmark calculations	Hudina		Calculations with different computer codes and comparison of results with measured information
1.7 Code verification tests	Hudina	Cont.	Experimental verification of the computer codes be dectrially heated instrumented multiple rod bundle tests. Different bundle geometries with smooth and rough surfaces of the rods under investigation.



Cont of Table CH CONTRIBUTIONS OF Switzerland TO THE NEA-GCFR R&D PROGRAM

Coordinator : Dr. Georg Markòczy
Swiss Federal Institute of Reactor Research

ACTIVITY	RESPONSIBLE	OBJECTIVES	COMMENTS	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983
2. NEUTRON PHYSICS MEASUREMENTS IN THE PROTEUS CRITICAL ASSEMBLY.	Richmond	Validation of nuclear data sets and calculation methods through measurements of neutron reaction rates, neutron spectra and reactivity worths.													
2.1. Measurements in typical GCFR lattice	Richmond			—	—										
2.2. Lattice with high steel content.	Richmond				—	—									
2.3. Lattice with "power reactor features" on central axis.	Richmond		A B_4C control rod, a depleted UO_2 column and a sub-assembly wrapper were insented successively and their effects on reaction rate distributions was measured.			—	—								
2.4. Studies of the steam entry accident	Richmond		Steam was simulated by hydrogenous plastic.				—	—							
2.5. Measurement of K-infinity by the null reactivity method.	Richmond		The measurements were used to check the capture cross-sections of structural materials.					—							
2.6. Iron shield benchmark measurements.	Richmond		The shield was placed above the GCFR zone of PROTEUS.						—						
2.7. Measurements in a depleted UO_2 axial blanket.	Richmond								—						
2.8. Measurements in thorium-bearing lattices	Richmond		Both ThO_2 and thorium metal configurations were investigated							—	→				

Cont.

1978

Cont of Table CH CONTRIBUTIONS OF SWITZERLAND TO THE NEA-GCFR R&D PROGRA

Coordinator : Dr. G.Markoczy
Swiss Federal Institute of Reactor Researc

ACTIVITY	RESPONSIBLE	OBJECTIVES	COMMENTS	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983
3. NUCLEAR PERFORMANCE AND SAFETY STUDIES	Wydler														
3.1 Physics code development and validation.	Wydler	To provide access to current basic data libraries. To test and improve data and codes.						3.1							
3.2 Investigation of safety parameters.	Wydler	Determination of steam entry effect, neutron streaming effect, etc.							3.2						
3.3 Fuel cycle studies.	Wydler	Assessment of alternate fuel cycles.								3.3					
3.4 Study of GCFR core flooding for fuel changing and emergency core cooling.	Wydler	Assessment of the effectiveness of various neutron absorbers.								3.4					

Cont.

1978

Cont of **Table** CH CONTRIBUTIONS OF SWITZERLAND TO THE NEA-GCFR R&D PROGRAM

Coordinator : Dr. G. Markoczy
Swiss Federal Institute of Reactor Research

ACTIVITY	RESPONSIBLE	OBJECTIVES	COMMENTS
4. Development of Carbide Fuel	Stratton Bischoff	As previously	
4.1 Fabrication of Carbide fuel particles	Hausmann	Demonstration of fuel quality and raising of throughput. Fabrication of fuel for irrad.tests	
4.2 SAPHIR CAPSULE IRRADIATIONS	Stratton	Detailed parameter tests on (UPu)C sphere pac fuel.	
4.3 DIDO Irradiations	Stratton	Carry out PIE of DIDO-III fuel test	
4.4 Irradiation test in BR-2	Stratton	Irradiation of one or two pins type Mol 11/K5	In collaboration with KfK
4.5 PIE of DFR Irrad.	Smith	Completion of PIE and Report	In collaboration with SCK/CEN Mol
4.6 Examination of (UPu)C recycle	Bischoff	Preliminary studies on waste handling and recycle	
4.7 Preparation for bundle tests	Stratton	Test of (UPu)C under Fast Flux	With foreign partner (restricted)

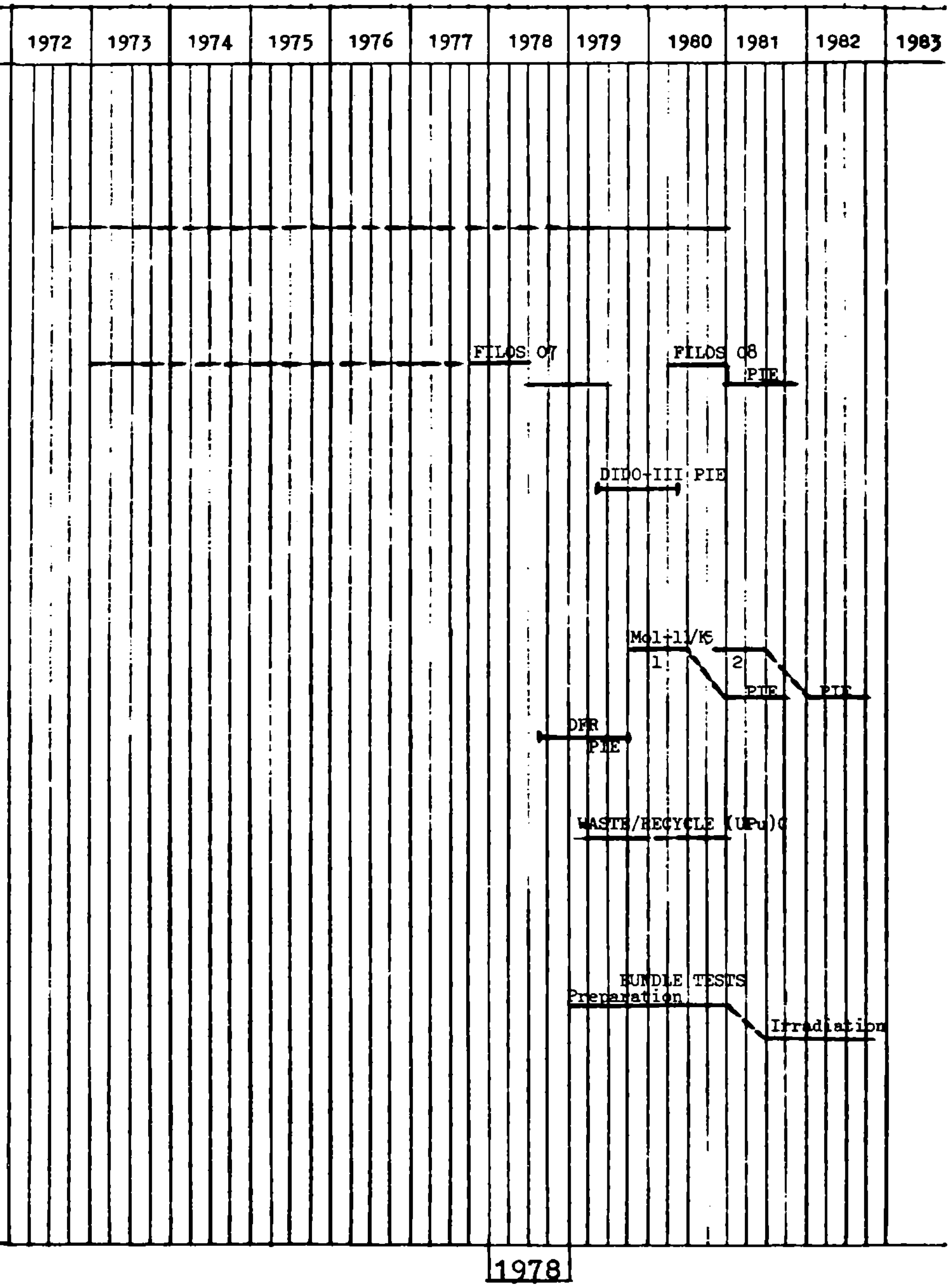


Table **D** CONTRIBUTIONS OF FRG TO THE NEA-GCFR R&D PROGRAM

Coordinators: M. Dalle Donne, KfK, S. Krawczynski, KFA

ACTIVITY	RESPONSIBLE	OBJECTIVES	COMMENTS	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983
1 Fuel Element Development															
1.1 Heat Transfer Experiments		Improved accuracy in the prediction of performance data for different roughening geometries including new rough surface types.													
1.1.1 Heat Transfer Experiments with Rough Rods in Annuli	M. Dalle Donne		Experiments in air rigs with single rough rods in smooth tubes.	Air annulus I operation			Air annulus II Design, construction operation								
1.1.2 Measurements in the wind channel	L. Meyer		Measurements of drag coefficients and velocity profile with single and repeated roughness ribs.			Design and rig construction operation									
1.1.3 Measurements in the water loop	Baumann		Measurements of velocity profiles and pressure drop over rough surfaces.	Design and loop construction operation											
1.1.4 Development of a computer program for rough clusters	Martelli/ Cevolani		Code development for evaluation of tests with clusters and detailed fuel element calculations.	NISA(isothermal) SAGAPO development SAGAPO testing and further development											
1.1.5 Experiments with rod clusters	K. Rehme		Experiments in high pressure helium loop or with models in air rig at atmospheric pressure.	Smooth rod clusters; 19 rough rod clusters; 12 rough rod clusters; turbulence distribution; 3-dim. roughness cluster			(BR2 calibration)								
1.1.6 Effect of spacers on pressure drop and temperature distribution on rough rods	K. Rehme/ A. Hassan		Experiments in water loop (measurements of spacer pressure drop) or in air rig with three rough rods (temp. distribution).	Air rig for temp. distr. on rough rods rig design + constr. operat. at high Re operation at low Re			pressure drop meas. for BR2 grid					pressure drop meas. for 3-dim. rough cluster			
1.2 Material Tests		Basic data for cladding fuel and coolant specifications.													
1.2.1 Fuel-clad compatibility	Schumacher		Compatibility experiments between oxide fuel and various claddings.	Vanadium alloy, steel comp. clad/fuel, fuel/coolant											
1.2.2 Corrosion experiments in helium at a known level of impurities.	Leistikow		Cladding corrosion experiments at relatively low impurity levels.	Test of German steels at 800°C and 27 vpm H_2O + 127 vpm H_2											
1.2.3 Investigations on the effect on fuel rods of higher amounts of steam in the helium coolant	Schumacher		Cladding (smooth and rough) and fuel corrosion tests with helium with relatively high amounts of H_2O and H_2.					Test of 1.4981 (smooth+rough) at 700°C and H_2O 150-500 vpm+H_2/H_2O=0.02÷0.2							

Continues

1978

Cont of Table D CONTRIBUTIONS OF FRG TO THE NEA-GCFR R&D PROGRAM

Coordinators: M. Dalle Donne, KfK, S. Krawczynski, KFA

ACTIVITY	RESPONSIBLE	OBJECTIVES	COMMENTS	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983
1.3 Irradiation experimts															
1.3.1 BR2-Irradiation (HELM)	Krug, Krawczynski (KFA), Stehle (KWU)	Demonstration of the feasibility of a vented pin assembly and of the corresponding fission product trapping system.	The bundle of 12 SS-clad mixed oxide fuel pins with 450 w/cm peak rating is being irradiated at 680°C clad mid wall hot spot temperature up to 60000 MWd/t peak burn up in a helium loop at 6MPa.	Design+constr. of loop and fuel bundle out of pile tests of blowers and dummy fuel element					HELM 1	HELM 2 HELM 3		HELM 3	HELM 4		
1.3.2 GB11-Experiments	Krug, Krawczynski (KFA), Langer et al. (GA)	Irradiation of three single vented fuel-rods with contamined He (H_2O etc.).											GB11		
2. Design and Safety Studies															
2.1 Karlsruhe - Jülich KWU Study	Dalle Donne, Götzmann, Krawczynski	Design of a 1000 MWe GCFR plant under consideration of safety problems in the context of current practice in licensing of LWR, HTR and LMFBR. Design of alternate concepts for a 300 MWe and a 1200 MWe prototyp plant.	The work has been mainlyperformed by KWU under contract from KfK Karlsruhe and subsequently from KFA Jülich.	1000 MWe GCFR Design safety study			300 MWe GCFR prototype design safety and alternate concepts					Alternate concept 1200 Mwe plant			
2.1.1 Development of a computer code to investigate the dynamic behaviour of a GCFR with a steam turbine.	Wilhelm	Digital computer code for accident analysis		Termination of PHAETON I				PHAETON II steam circuit, core parallel channels natural convection							
2.1.2 Cooling of the core melt caused by a core melt-down accident	Dalle Donne, Dorner	Containment of the core+ blankets molten mass within reactor cavity in PCRV.	After studies of various possibilities the investigations have concentraced on the use of borax as diluting material for internal core-catcher.	Temp. distribution in slabs		Molten salts as sacrifical layer			Calc.	Borax small laboratory tests					
2.1.3 Neutron streaming effects and steam entry reactivity calculations	Eisemann Kiefhaber	Determination of neutron streaming effects in reactivity coefficients and of reactivity variations due to steam entry in core region.		Neutron streaming calc.				Steam entry benchmark							
2.1.4 Energy release calculations by hypo-	Jacobs	Evaluation of energy release, fuel temperature and effects on PCRV integrity due to reactor excursion caused by fast ejection of a scram rod.			Kadis calcul. without He+fiss. products			Calc. with He + fiss. products		1978					

Table F CONTRIBUTIONS OF FRANCE TO THE NEA-GCFR R&D PROGRAM

Coordinator : M M ROBIN DEDR/CEA CEN/ SACLAY

ACTIVITY	RESPONSIBLE	OBJECTIVES	COMMENTS
F.1 Coated particle fuel development	C.MOREAU	Optimization of coating	
1.1. Coating design studies			
1.2. Coating fabrication		Optimization of coating parameters ; improvement of SiC layer	
1.3. Coated particle irradiation RAG Party test		Ability to survive full exposure for different coated particle designs	Started on June 1974 ended on March 1975
1.4. RAG Party PIE		- Failure fraction measurements - Mode of failure	2 UKAEA batches survived 3.10^{22} η.cm^{-2}
F.2 Coated particle fuel assembly technology	R.BUJAS J.MALHERBE		
2.1. Design		Improvement of the basket design	
2.2. Feasability		Mock-up machining and assembling	
2.3. Out of pile tests		Experimental measurement of pressure drops	
F.3 Pin fuel assembly			
3.1. Feasability of an instrumented irradiation test in Rapsodie	A.BALLAGNY	- Release of fission products out of vented pin - Trap efficiency studies	Feasability O.K.
3.2. Out-of-pile tests	J.MALHERBE	Preliminary study of a full scale fuel element test in Carmen II circuit	

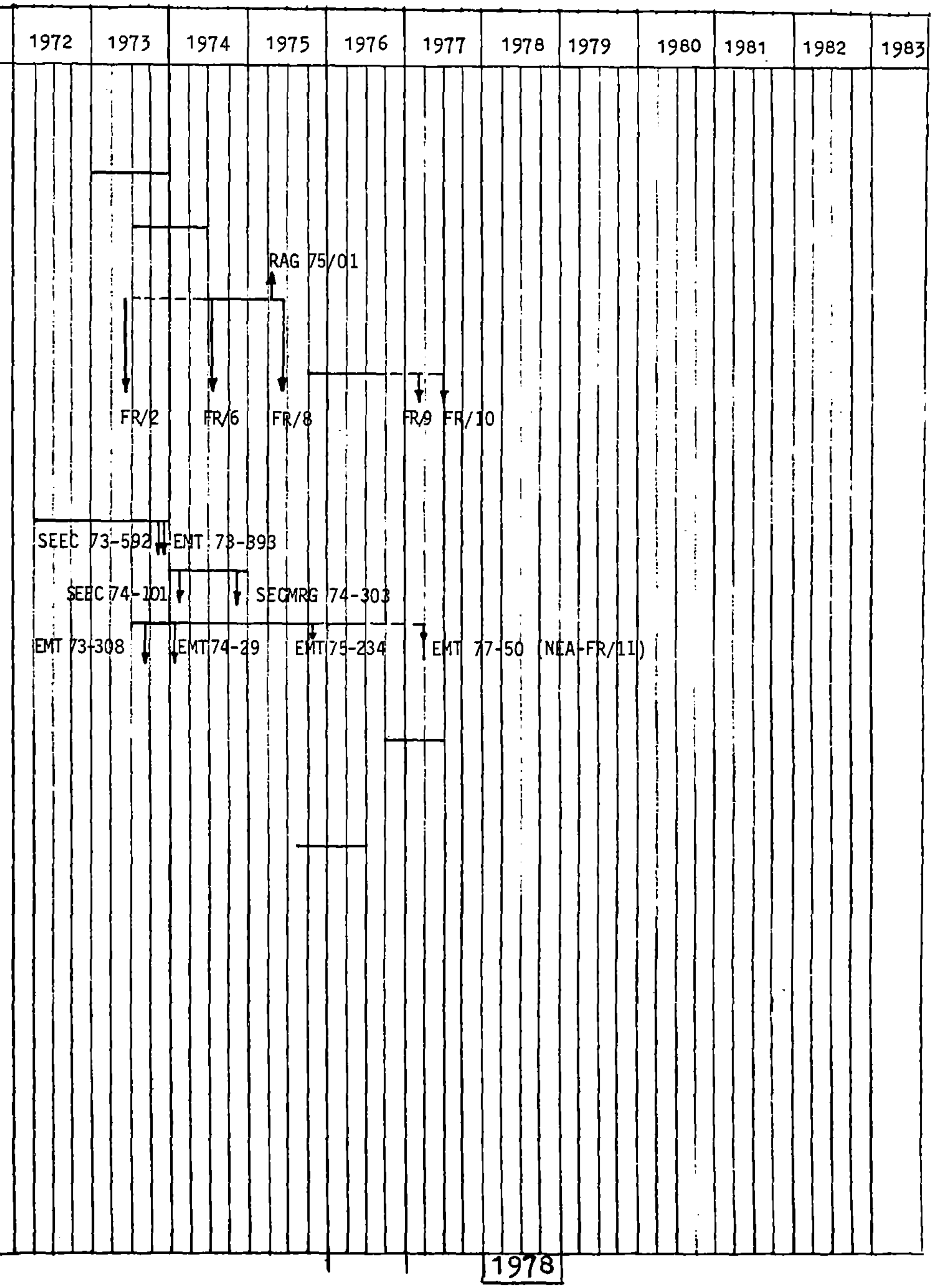


Table J CONTRIBUTIONS OF JAPAN TO THE NEA-GCFR R&D PROGRAM

Coordinator : Mitsuho Hirata, JAERI

ACTIVITY	RESPONSIBLE	OBJECTIVES	COMMENTS	Schedule (1972 · 1973 · 1974 · 1975 · 1976 · 1977 · 1978 · 1979 · 1980 · 1981 · 1982 · 1983)
1 Material development				
1.1 Investigation of corrosion and rubbing effects on roughened pin in moist He	Tatsuo Kondo (MEL)	Basic data of compatibility of cladding materials with moist He and of effects of spacer-pin interactions.	Corrosion behaviour with particular emphasis being placed on morphology of reaction surfaces for coolant corrosion in static and circulating moist He environment.	1.1: JP-03, JP-05, JP-07, JP-15, R
1.2 Property evaluation and modification of potential alloys for higher temperature uses.	Tatsuo Kondo (MEL)	Evaluation of commercial and experimental super alloys for use particularly on mechanical and chemical stabilities.	Corrosion, fatigue, creep and irradiation effects on mechanical properties of super alloys are also investigated.	1.2: JP-03, JP-05, JP-07, JP-15, JP-27 -28, JP-29 +30 +31
2 Coated particle development				
2.1 Study of improved coated particle fuel	Kazumi Iwamoto (FIAL)	Investigation of process of ZrC or alternative coating.	Use of uranium oxide kernal with PyC buffer and ZrC or ZrC-C composition.	2.1: JP-02, JP-03, JP-05, JP-07, JP-15, JP-21 -32, R
2.2 Irradiation of coated particle fuels in thermal reactor.	Kazumi Iwamoto (FIAL)	Investigation of irradiation performance of coated particles under He gas flow.	Irradiation in He in-pile loop, OGL-1, with the material testing thermal reactor, JMTR.	2.2: JP-07, JP-15; OGL-1 contruction, Irrad, JP-32, R
3 Thermohydraulic study				
3.1 Thermohydraulic testing and evaluation of representative roughended pin.	Yoshizo Okamoto (HTL)	Investigation of hydraulics and heat transfer characteristics of roughened pins.	Using turbulence promoter, pressure drop and distribution of local heat transfer coefficients are studies.	3.1: JP-03, JP-05, JP-06, JP-07,12 -13,14, JP-16,17 -18,19, JP-33,34 -35,36 -37
4 Core performance and safety studies				
4.1 Reactor physics experiment and analyses.	Hideo Kuroi (FRPL)	Investigation of special features of GCFR core, to improve accuracy in prediction of core performance and safety-related physics parameters.	Power distribution, streaming effect and steam ingression reactivity worth are measured and analysed.	4.1: JP-22, R
4.2 Core performance and fuel cycle analyses.	Hiroyuki Yoshida (FRDL)	Finding a credible and economical solution to problems of GCFR core.	Assessment of Pu-U and U-Th fuelled GCFRs including their fuel cycle.	4.2: JP-03, JP-05, JP-07 -08, JP-15, JP-23,25, JP-38

Continues

1978

Table J CONTRIBUTIONS OF JAPAN TO THE NEA-GCFR R&D PROGRAM

Coordinator : Mitsuho Hirata, JAERI

ACTIVITY	RESPONSIBLE	OBJECTIVES	COMMENTS	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983
4.3 Safety studies and related R&D works	Hiroyuki Yoshida (FRDL)	Establishment of current status and trend with regard to accepted safety principles for LMFBR and HTGR.	Survey of safety requirement and investigation of engineering safeguards through DBA.	4.3		JP-04	JP-05	JP-07,09-11	JP-15	JP-24	JP-39,40				
5. Coordination of GCFR works	Mitsuho Hirata (OP)	Coordination and annual progress report		5	JP-01	JP-03	JP-05	JP-07	JP-15	JP-20	JP-26				

Notes

<u>Abreviation for laboratories performing GCFR work at JAERI</u>

MEL	:	Materials Engineering Laboratory
FIAL	:	Fuel Irradiation and Analysis Laboratory
HTL	:	Heat Transfer Laboratory
FRPL	:	Fast Reactor Physics Laboratory
FRDL	:	Fast Reactor Design Laboratory
OP	:	Office of Planning, JAERI Headquarters

1978

Table N CONTRIBUTIONS OF NETHERLANDS TO NEA-GCFR R&D PROGRAM

Coordinator : R.A. van der Laken

ACTIVITY	RESPONSIBLE	OBJECTIVES	COMMENTS		1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983
1. Cladding development for pin fuel		To study mechanical properties after irradiation of 0.1-15ppm ^{10}B containing samples	304L and 316L alloys irradiated at 100 and 500°C to $3x10^{20} n/cm^2$ (E 1MeV)		See RCN-NEA/5											
2 Thermohydraulics		Codes development based on experimental and analytical work	Re $1-5x10^5$ Wall temp 750°C		RCN-NEA/4 RCN-NEA/3 RCN-NEA/2 RCN-NEA/1			RCN-NEA/11								
2.1 Rough surfaces																
2.2 Abnormal conditions		Safety analysis (incl. hot spots etc)				RCN-NEA/7		RCN-NEA/10								
3 Miscellaneous					RCN-NEA/5	RCN-NEA/6	RCN-NEA/9									
			(Table 6 (N) was prepared by the Techn Coordinator 1979-02) RE													

Table S CONTRIBUTIONS OF SWEDEN TO THE NEA-GCFR R&D PROGRAM

Coordinator : Gottfried Vieider, Reino Ekholm
Studsvik Energiteknik AB, S-611 82, Nyköping

ACTIVITY	RESPONSIBLE	OBJECTIVES	COMMENTS
1 Pin development			
1.1 Creep, swelling and ductility studies of irradiated ss and creep studies of UO_2.	Tord Jonsson	Investigations of the ss clad performance in FBR to predict failures.	
1.2 3-pin NaK-capsule irradiation in R-2	Ronald Forsyth	Study of compatibility problems due to FP attack on the clad. Inter- and introgranular precipitation and densities are varied.	5.8 mm OD pins of mixed oxide fuel with 316 ss (Sandvik 5R60HV) at 700°C clad inside wall max 50kW/m, thermal irr to about 6.5GJ/g (75 MWd/kg)
1.3 Studies of turbulent flow in rod bundles.	Björn Kjellström	Studies of velocity and temperature distributions in rod bundles.	
1.4 Stabilization of mixed oxide fuel.	Olof Hindbeck	Prevention of Pu migration.	Influence of Pu and Ce valences. Structure and thermal studies. Irrad.
2 Coated particle dev.	Ulf Runfors	Lead exp to the irradiation of GCFR prototype particles in Rapsodie. Investigation of the influence of particle design and production on the performance.	Up to 13GJ/g (150MWd/kg) burnup in continuously swept and fission product monitored capsules.
3 Fuel cycle assessment	Klas Jirlow	Comparison of the nucl. performance characteristics of GCFR and LMFBR cores including blanket optimization.	A 2D fuel cycle code with ENDF/B-III data. Input provided by GBRA.
		Parametric study of fuel cycle costs.	Influence of interest rate and Pu prices.
		Breeder performance and power growth patterns.	Connection btw U-demand and breeder fuel cycle parameters.

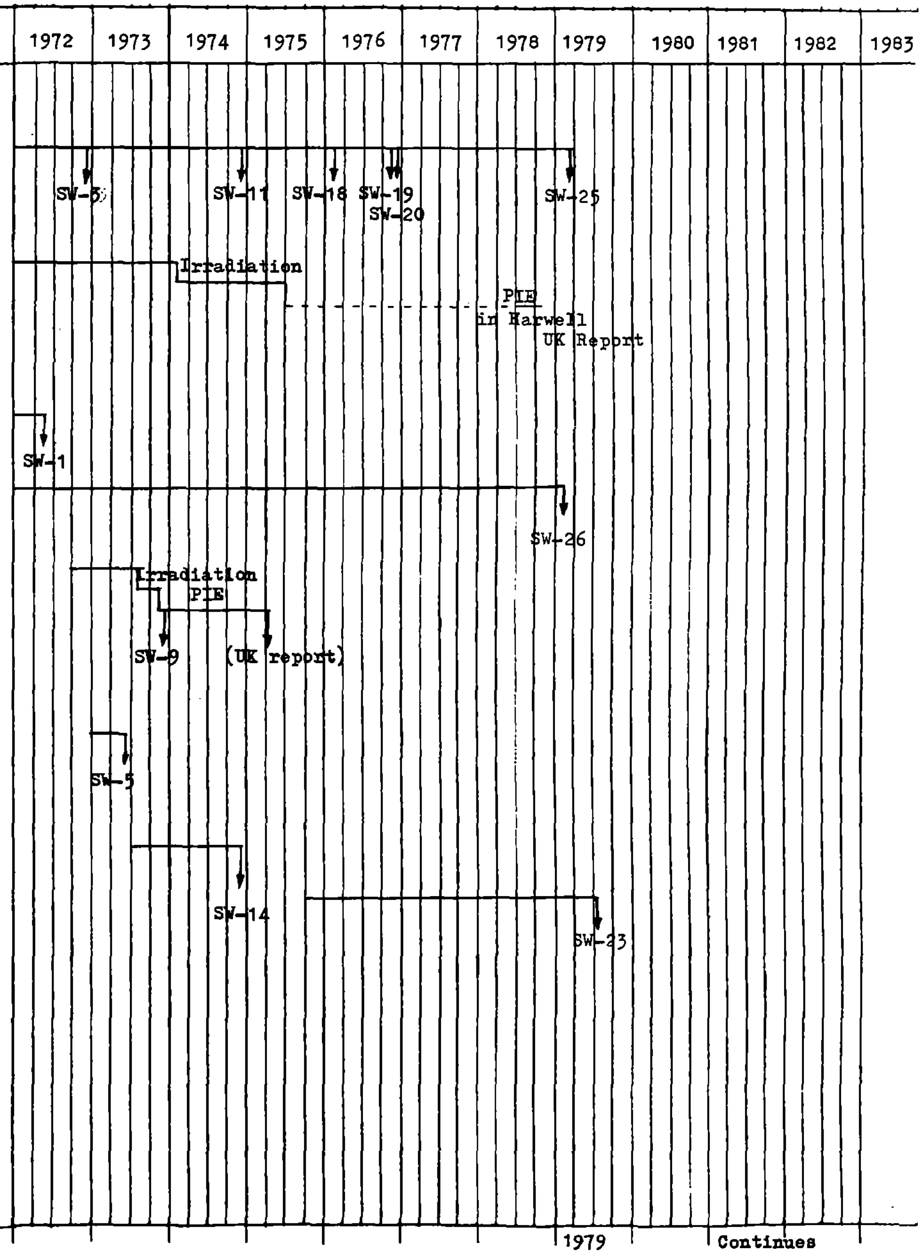


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Table S CONTRIBUTIONS OF SWEDEN TO THE NEA-GCFR R&D PROGRAM

Coordinator : Gottfried Vieider, Reino Ekholm
Studsvik Energiteknik AB, S-611 82, Nyköping

ACTIVITY	RESPONSIBLE	OBJECTIVES	COMMENTS
4 <u>Prestressed concrete reactor pressure vessel (PCRV) development</u>	Shankar Menon	Demonstration of the feasibility of high pressure PCRVs by studies of safety and manufacturing problems on large models at 8.5MPa design pressures.	
4.1 Failure tests of lids and bottom slabs.		Study of the mode of failure and of the safety factors.	Perforated bottom slabs and variations of lid design in 1:10 scale are tested to 40MPa.
4.2 Liner venting system tests.		Performance tests of a system relieving coolant pressures into the concrete to provide a design basis for such venting systems.	Tests of a PCRV model with simulation of liner cracks of different sizes.
4.3 Manufacturing tests in a part of a bottom slab with a large number of penetrations to be positioned at a high accuracy.		Demonstration of the design's feasibility with regard to tolerances on a nearly full scale model.	The GBRA-GCFR vessel has a perforated bottom slab.
4.4 High pressure tests on small models.		Pressurization of special models to failure to determine local safety factors in lid and bottom slab areas.	Tests at Norwegian Technical University of Trondheim.
5 <u>Miscellaneous</u>	Reino Ekholm	Other activities in the support of the GCFR dev.	Coordination work, progress and status reports etc.

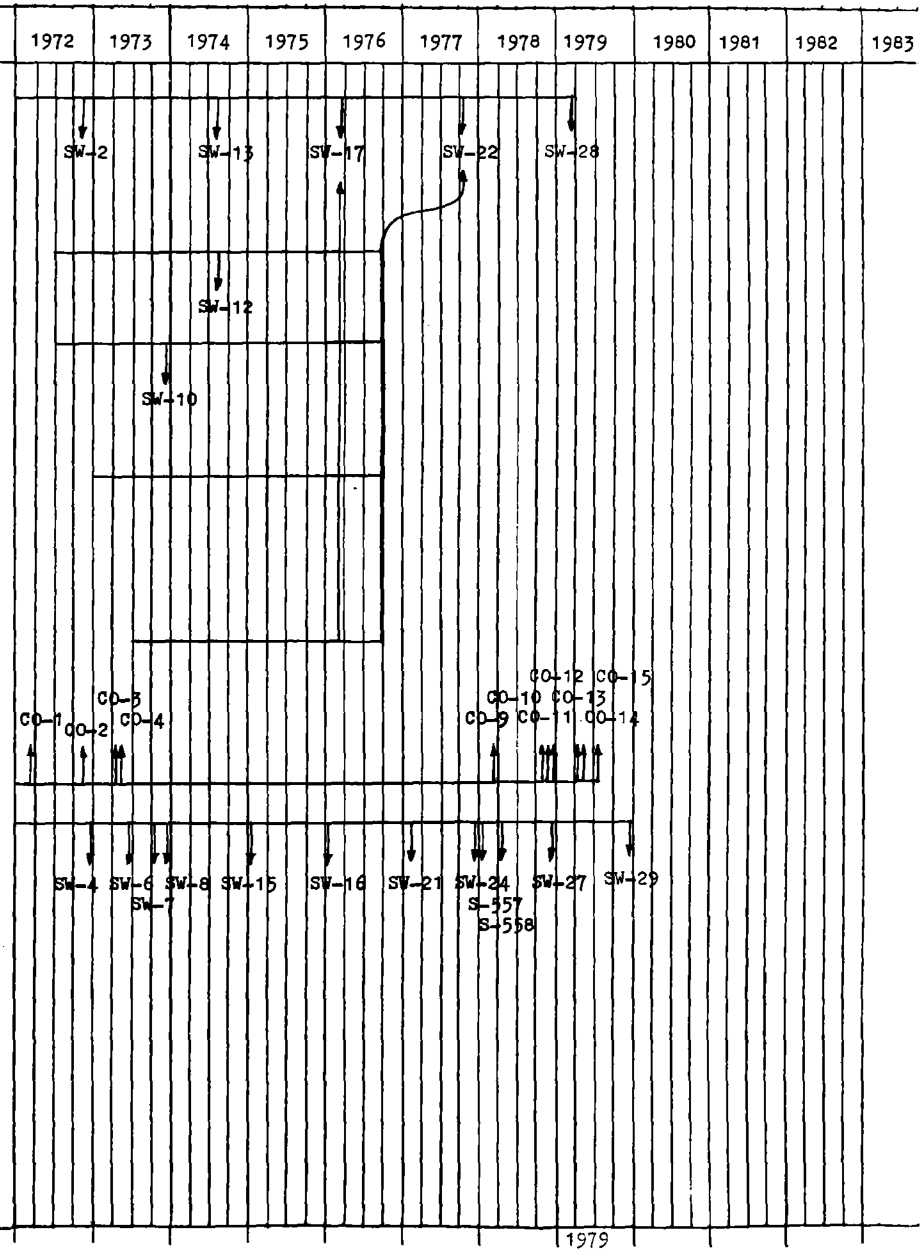

Table UK CONTRIBUTIONS OF UK TO THE NEA-GCFR R&D PROGRAM

Coordinator : J. Smith

FUEL AND MATERIALS INVESTIGATIONS

ACTIVITY	RESPONSIBLE	OBJECTIVES	COMMENTS
1. Fuel development			
(a) Coated particle fuel specification and manufacture	G.W. Horsley	To produce high packing density coated particle fuel consisting of porous gel-route $(U,Pu)O_{2-x}$ fuel kernels enveloped in successive coatings of porous PyC and high density glabrous SiC	~1 mm dia porous spherical $(U,Pu)O_{2-x}$ particles successfully coated with porous PyC and glabrous SiC layers
(b) Irradiation and PIE of coated particle fuel	G.W. Horsley	To demonstrate the ability of the fuel to withstand a burn-up of ~100 GWD/te at temperatures up to 1000°C with an associated fast dose of 6.10^{22} EDN	The burn-up capability has been demonstrated in thermal reactors. Irradiation in fast reactors reached $1.3.10^{22}$ EDN
(c) Assessment of fast dose damage to coating materials	B.E. Sheldon	To examine the influence of fast dose on the physical and mechanical properties of pyrolytic silicon carbide	Pyrolytic SiC expands by ~1.3% in vol. at 5.10^{22} EDN at 500°C with a loss in strength of 14%
2. Materials compatibility			
(a) GCFR materials - coolant compatibility review	J.E. Antill	To evaluate status of existing knowledge and identify potential problem areas in GCFR designs	Design modification accommodated areas of concern where possible. Further experimental programme initiated
(b) Coated particle fuel-coolant interaction	M.J. Bennett	To examine influence of fission recoil damage, simulating fast fluence, upon both the passive oxidation of SiC at 950°C and the transition between active and passive oxidation at 850-950°C	Slight enhancement in passive attack unlikely to affect outer particle layer integrity. Irradiation had no significant influence upon the minimum temperature of onset of active corrosion or the active passive transition pressure over temperature range examined
(c) Containment alloy - CO_2 coolant compatibility	M.J. Bennett	To examine influence of fission recoil damage; simulating fast fluence upon the oxidation of a 20/25/Nb steel in oxygen at 850°C.	Fast fluence would probably have no significant effect on oxidation behaviour in CO_2
			Cont.

1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983

1978

Cont. of Table UK CONTRIBUTIONS OF UK TO THE NEA-GCFR R&D PROGRAM

Coordinator : J. Smith

HEAT TRANSFER AND FLUID FLOW

ACTIVITY	RESPONSIBLE	OBJECTIVES	COMMENTS	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983
1. Stability of flow in particle beds	D Wilkie	Demonstrate the limitations of upward flow designs	Work discontinued as need for design changes became apparent												
2. Heat transfer within particle beds	"	Determination of heat transfer within particle beds due to flow through bed Heat transfer between a particle bed and solid and perforated boundaries	Studies carried out to support tube in shell and particle loaded pin concepts												
3. Heat transfer from roughened surfaces	"	Provide basic data for the selection of optimum heat transfer surfaces. Provide methods for calculation of heat transfer in any type of flow passage	Continue work to improve understanding of pin fuel assemblies												
(a) measurements of friction factor in roughened clusters															
(b) general correlation of flow and heat transfer data											Report				
(c) application of data to practical cases											Report				
4. Heat transfer from rod clusters	"	Provide a method for calculating temperatures throughout cluster fuel elements and provide experimental confirmation	Continue work to improve understanding of pin fuel assemblies												
(a) computer programme development															
(b) 7 pin clusters with smooth, transverse ribbed and helical ribbed surfaces.															
(c) 36 pin clusters with transverse ribbed and helical ribbed surfaces											Report				
5. Effect of bowed pins in clusters	"	Provide a method for calculating the temperature changes due to pins bowing up to touching in groups of 2, 3 or 4	Important further development for pin assemblies												
(a) single pin tests with smooth, transverse and helical ribbed surfaces.															
(b) 7 pin cluster tests with transverse ribbed and helical ribbed surfaces			Cont.								Report				

1978

Coordinator : J. Smith

FUEL AND MATERIALS INVESTIGATIONS (Continued)

ACTIVITY	RESPON-SIBLE	OBJECTIVES	COMMENTS
(d) Containment alloy - wet helium coolant compatibility	M.J. Bennett	To examine the oxidation behaviour of Fecralloy, 20/25/Nb stainless steels, and nimonic alloy PE16, in inert gas containing 7 µatm water vapour and 375 µatm hydrogen, for periods up to 7166 h, at temperatures between 650-1000°C	Behaviour of alloys in general consistent with that observed in fully oxidising environments, such as carbon dioxide
(e) Containment alloy - coated fuel particle compatibility	M.J. Bennett	To examine the solid-solid reactions between SiC and Si_3N_4 and representative alloys, at temperatures up to 1000°C and periods up to 5000 h	SiC was more reactive than Si_3N_4. These solid-solid reactions could impose design limitations on the use of coated particle fuel
(f) Compatibility and tribology loop tests - vented pin systems	A.N. Knowles	Two loops, one at 41 bar, one at 0.4 bar, used to examine cladding wrapper and boiler tube materials. Tests included fretting behaviour	Test atmosphere was helium with 1 vpm H_2O, 10 vpm H_2. Temperature range for cladding 600-800°C; 600°C common temperature for other materials

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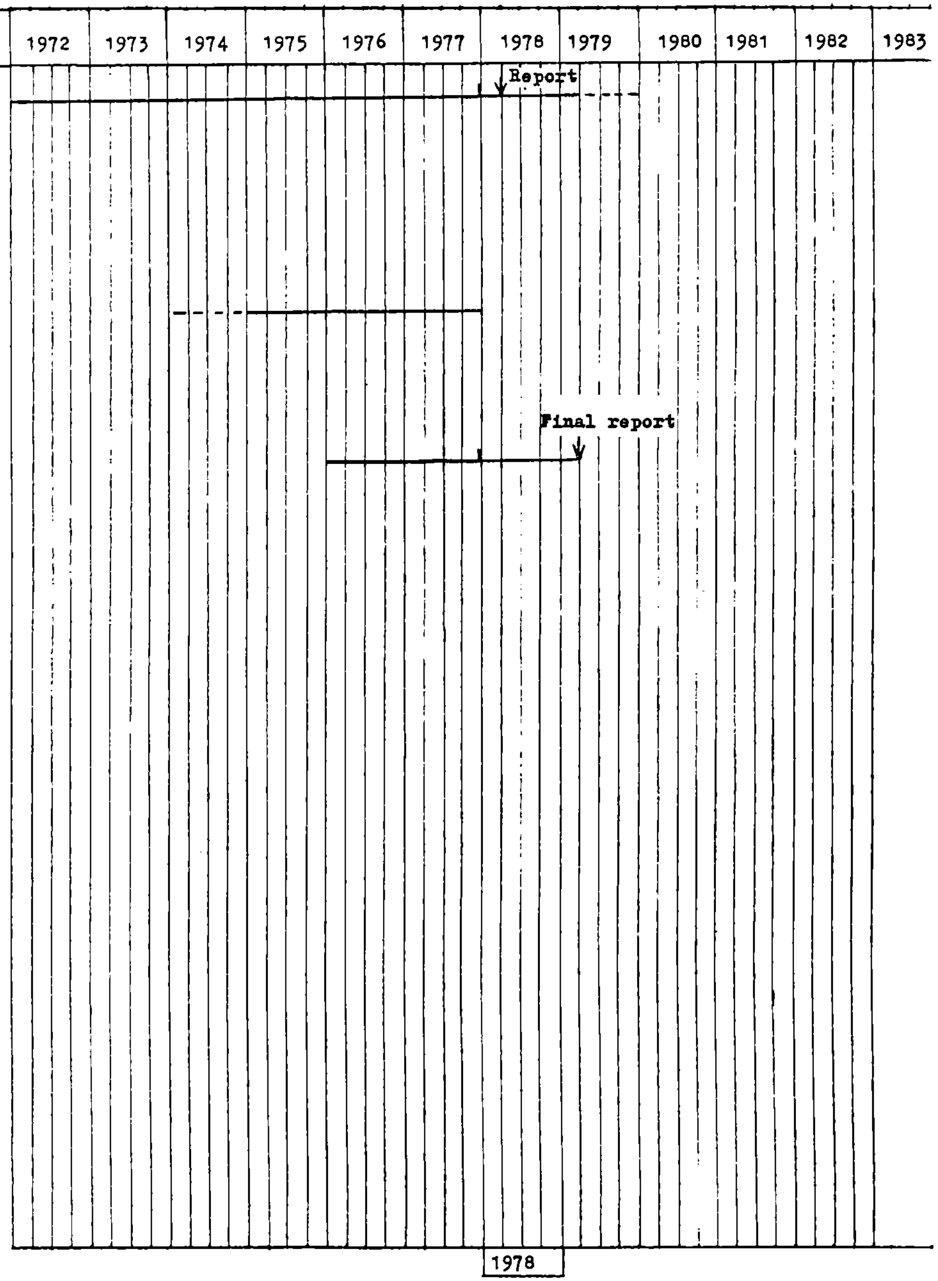

Cont. of Table UK CONTRIBUTIONS OF UK TO THE NEA-GCFR R&D PROGRAM

Coordinator : J. Smith

DESIGN AND SAFETY WORK : GENERAL

ACTIVITY	RESPONSIBLE	OBJECTIVES	COMMENTS
1. Fuel assembly development - coated particle type (a) coated particle type with particle beds (b) coated particles with tubular pins (c) tube in shell coated particle assemblies	C.P. Gratton	Appraisal of different forms of assemblies and selection of a reference solution	Included both analytical and experimental studies Work discontinued as limitations became clear
2. Studies of pin-fuelled cores - sealed pins	A.N. Knowles	Investigation as to whether a suitable sealed pin design could be devised as a starting point for a demonstration or experimental reactor	This work began as a possible backup for a vented pin core in case it was not considered prudent to begin with all-vented assemblies
3. Performance studies of particle-fuelled GCFRs	C.P. Gratton	Reactor physics, control and kinetics investigations	This work terminated at the end of 1974 when particle-fuelled systems studies ceased
4. Safety investigations			
(a) reliability consideration of emergency cooling arrangements for GCFRs	J.Ablitt	Examination of both loss of flow and loss of pressure effects	Joint work with GBRA
(b) transient analyses for vented pin cores	A.N. Knowles	Assessment of peak fuel temperatures and conditions	
(c) theoretical investigation of the effect of local core blockages	A.N. Knowles	To assess subchannel effects, and also inter-assembly effects	This is exploratory using LMFBR methods as a starting point
(d) study of core catchers	A.N. Knowles	Assessment of the general heat removal and performance requirements	Two alternatives considered (GBRA schemes also examined) - no entirely satisfactory solution so far evolved

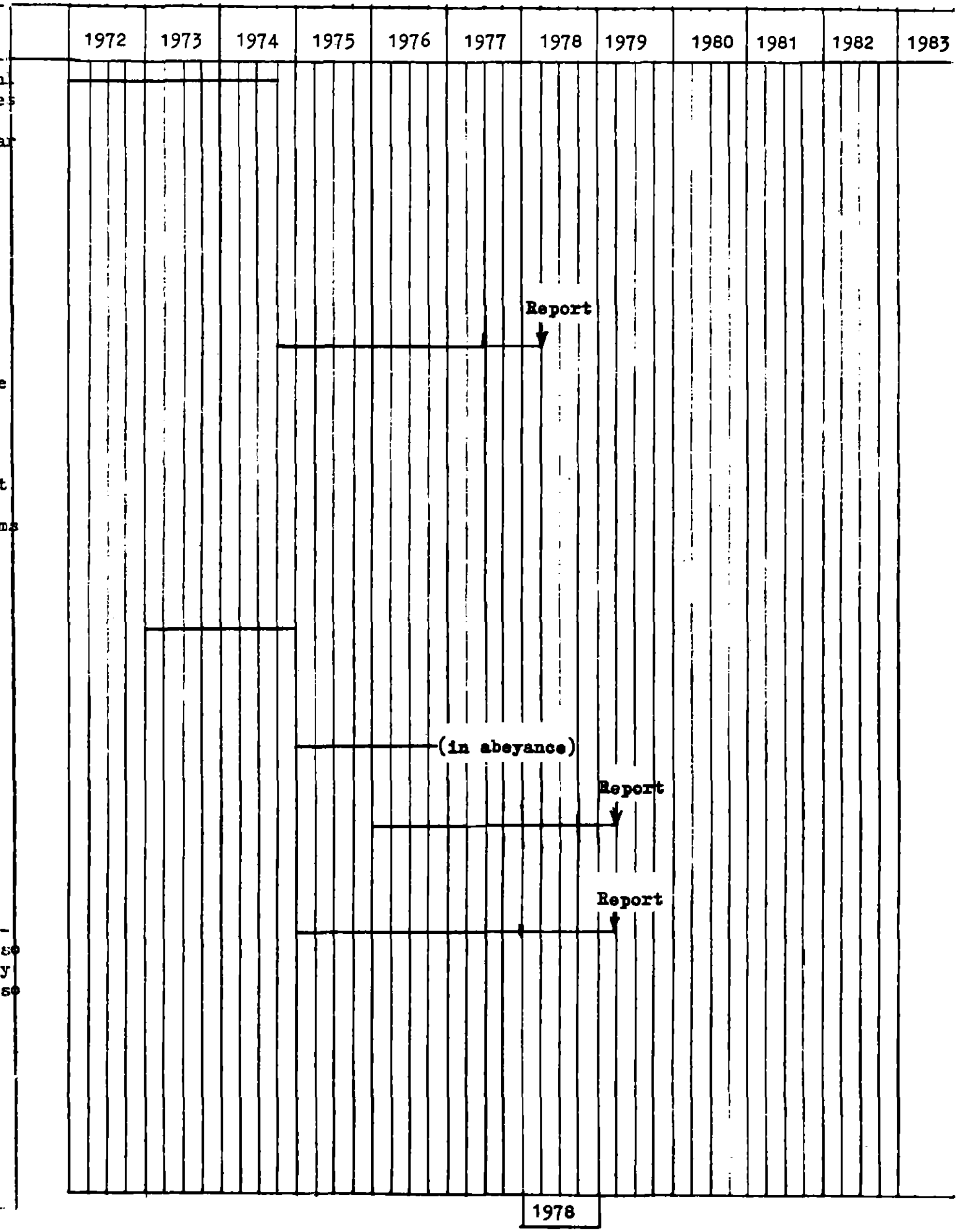

Table **US** CONTRIBUTIONS OF U.S. TO THE NEA-GCFR RAI PROGRAM

Coordinator : Donald E. Erb, DOE

NOTE: It is not practical to make reference to our reports in these charts, instead see U.S. annual reports beginning in 1977.

ACTIVITY	RESPON-SIBLE	OBJECTIVES	COMMENTS	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983
1 Nuclear Steam Supply System Engineering															
1.1 Prestressed Concrete Reactor Vessel		Develop design options and ensure design criteria (including licensing) are met.													
1.1.1 PCRV Configuration	L. Kube		Design efforts at GA.		CONCEPTUAL DESIGNS					ESTABLISH REF. DESIGN				DESIGN 70% COMPLETE	
1.1.2 PCRV Closure Tests	U. Gat/ J. Callahan		Test and analysis program at ORNL.						1/15 SCALE STEAM GEN. CLOSURE			1/4 SCALE CLOSURE TEST			
1.2 Fuel Handling	L. Kube	Develop design options and ensure design criteria (including licensing) are met.			CONCEPTUAL DESIGNS					ESTABLISH REP. DESIGN				DESIGN 70% COMPLETE	
1.3 Reactor Internals	L. Kube	Develop design options and ensure design criteria (including licensing) are met.			CONCEPTUAL DESIGNS										
1.4 Main Helium Circulator	L. Kube	Develop design options and ensure design criteria (including licensing) are met.			CONCEPTUAL DESIGNS										
1.4.1 Circulator Test Facility	L. Kube	Test circulator and drive.	Facility location has not yet been determined.							ESTABLISH REQUIREMENTS			CONSTRUCTION		
1.5 Auxiliary Circulator	L. Kube	Develop design options and ensure design criteria (including licensing) are met.			CONCEPTUAL DESIGNS										
1.6 Steam Generator	L. Kube	Develop design options and ensure design criteria (including licensing) are met.			CONCEPTUAL DESIGNS										
1.7 Systems Engineering	L. Kube	Ensure components are integrated into systems which meet system design criteria (including licensing).							DEVELOP AND USE ANALYTICAL CAPABILITY						
1.8 Plant Dynamics	L. Kube	Determine plant time dependent characteristics for all design conditions for use in ensuring that all design criteria (including licensing) are met.							DEVELOP AND USE ANALYTICAL CAPABILITY						

Cont.

1978

Cont. of Table US CONTRIBUTIONS OF U.S. TO THE [illegible] PROGRAM

Coordinator : Donald E. Erb, DOE

ACTIVITY	RESPONSIBLE	OBJECTIVES	COMMENTS	SCHEDULE 1972 1973 1974 1975 1976 1977 1978 1979 1980 1981 1982 1983
1.9 Control Rod Systems	T. Pitterlie	Utilize a modified CRBR system to meet primary system shutdown design criteria (including licensing).	Lead for this work at W.	MODIF. DRIVE; TEST DRIVE
1.10 Helium Processing Components	L. Kube	Develop design options and ensure design criteria (including licensing) are met.		CONCEPTUAL DESIGNS; ESTABLISH REF. DESIGN; DESIGN 70% COMPLETE
1.11 Control and Electric Components	L. Kube	Develop design options and ensure design criteria (including licensing) are met.		CONCEPTUAL DESIGNS; ESTABLISH REF. DESIGN; DESIGN 70% COMPLETE
2 Fuels and Materials				
2.1 Core Assembly Development	L. Kube	Develop design options and ensure design criteria (including licensing) are met.		CONCEPTUAL DESIGNS; ESTABLISH REF. DESIGN; DESIGN 70% COMPLETE
2.2 Pressure Equalization System for Fuel	L. Kube	Develop design options and ensure design criteria (including licensing) are met.		CONCEPTUAL DESIGNS; ESTABLISH REF. DESIGN; DESIGN 70% COMPLETE
2.3 Core Assembly Design Verification	L. Kube	[illegible]fy that core assemblies [illegible] design criteria [illegible]uding licensing).		CONCEPTUAL DESIGNS; ESTABLISH REF. DESIGN; DESIGN 70% COMPLETE
2.3.1 Core Flow Test Loop	U. Gat/ A. Grindell	Design, build and operate an experimental loop to obtain operational characteristics of core assemblies.	Electrically heated pins used to simulate fuel, blanket, and control rods.	CONCEPTUAL DESIGN; PRELIMINARY DESIGN; CONSTRUCTION; OPERATION
2.4 Fuels and Materials R&D	J. Broido	Provide fuel and material R&D data in support of design, operation and licensing efforts.		GA EXPERIMENTS AT ORNL; F-1 IRRADIATION; F-3 IRRADIATION; F-5 IRRADIATION
2.5 Fuel Rod Engineering	L. Kube	Develop design options and ensure design criteria (including licensing) are met.		CONCEPTUAL DESIGNS; ESTABLISH REF. DESIGN
3 Core Physics and Shielding				

Cont.

1978

Cont. of Table US CONTRIBUTIONS OF U.S. TO THE NEA-GCFR R&D PROGRAM

Coordinator : Donald E. Erb, DOE

ACTIVITY	RESPON-SIBLE	OBJECTIVES	COMMENTS	Schedule (1972 – 1983)
3.1 Nuclear Analysis and Core Physics	J. Broido	Provide validation nuclear information for use in the Nuclear Steam Supply System Engineering efforts		CORE PHYSICS CRITICALS; ENGINEERING MOCKUP
3.2 Shielding R&D	J. Broido/ U. Gat	Provide validated radiation shielding information for use in Nuclear Steam Supply System Engineering efforts.	ORNL Tower Shield Facility is used in this effort.	SHIELDING ANALYSIS; GRID PLATE SHIELD; EFFORTS IN SUPPORT OF ENGINEERING DESIGN
4 Licensing and Safety				
4.1 Risk Analysis	J. Broido	Quantitatively determine risks and identify potential design modifications to reduce risks. Provide methods for integrating reliability considerations into design efforts.		RISK ANALYSIS; RELIABILITY INTEGRATION METHODS; LICENSING SUPPORT
4.2 Core Accident R&D	J. Broido/ R. Sevy	Determine core accident and radioactive material characteristics release for use in design and licensing efforts.		DEVELOP ANALYTICAL CAPABILITY AND APPLY TO DESIGNS; SUPPORT PSAR EFFORTS
4.3 Post Accident R&D	J. Broido/ R. Sevy	Determine the time dependent characteristics of the plant after an accident with particular emphasis on cooling of fuel and radioactive material distribution.		PCRV CONTAINMENT OF MOLTEN FUEL; NRC REPORT ON PAHR
4.4 Safety Test Program				
4.4.1 Out-of-pile Fuel Tests	R. Sevy	Provide design and licensing information based on experiments with fuel which is heated by non-nuclear methods.		MOLTEN POOL EXPERIMENTS; CLAD MELT TESTS
4.4.2 In-pile Tests	R. Sevy	Provide design and licensing information based on experiments with fuel which is heated by nuclear methods.		GRIST EXPERIMENT
4.4.3 Gas Reactor In-pile Safety Test Loop	E. Arbtin	Design and build a test loop for use in TREAT.		DESIGN; CONSTRUCT; INSTALL

Cont.

1978

Cont of Table US CONTRIBUTIONS OF U.S. TO THE NEA-GCFR RAI PROGRAM

Coordinator : Donald E. Erb, DOE

ACTIVITY	RESPONSIBLE	OBJECTIVES	COMMENTS		1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983
4.4.4 Electrical Simulator Tests	D. Hanson	Develop and use electrical simulators for fuel, blanket and control rod safety tests involving elevated clad and duct temperatures including melting.									SMALL CLUSTERS		LARGE CLUSTERS			
4.5 Licensability	J. Broid	Support Helium Breeder Associates in their efforts to determine the licensability of the GCFR.			PSID		DOE SAFETY EVAL	REPORTS				PREPARE PSAR	SUBMIT PSAR			

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Table GBRA CONTRIBUTIONS OF GBRA TO THE NEA-GCFR R&D PROGRAM

Coordinator : J. CHERMANNE

ACTIVITY	RESPONSIBLE	OBJECTIVES	COMMENTS	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983
1. <u>REFERENCE DESIGN</u>	J. Chermanne C. Sacriste J. Yellowlees	Identification of a 1200 MWe commercial reference design used for the assessment of feasibility, performance, safety, economics and R & D questions.	The reference design [GBR4] is a vented pin, 90 bar, He cooled concept with standing core, upward flow and motor driven circulator sets. It is described in "GBR4 DESIGN DESCRIPTION" June 1975												
		Design revision.													
2. <u>SAFETY STUDIES</u>	P. Burgsmuller J.J. Dekais	Safety principles and design solutions acceptable to European safety authorities are defined by : - collection of information from safety authorities and GBRA Member Companies concerning the safety requirements and the related design features for other reactor systems; - design and reliability study of the required engineered safeguards; - detailed analysis of all major physically possible accidents including low probability events such as the core melt down; - preparation of a preliminary safety report; - negociations with European safety authorities concerning the acceptance of GCFR safety principles.	A "GBR4 Safety Working Document" was produced in July 1974. This document was examined by a group of safety experts convened by the CEC. A "Supplement to GBR4 Safety Working Document", answering all the questions of the experts was produced in April 1975. In May 1975, the experts concluded that "no reasons have been identified which would prevent a GBR of the kind proposed by the Association achieving a satisfactory status". The experts indicated various fields of R&D required to support evidences during definite safety qualification : PCV integrity, core catcher, etc.												
		Safety related studies on	These studies were carried out at JRC Ispra.												
	M. Holtbecker	- Ability of a PCV to withstand a nuclear excursion.													
	G. Volta	- Reliability of a PCV liner at 90 bar.													
	J.M. Defalque	- Gross core melt down analysis : a study including the assessment of the mechanistic, nuclear and energetic behaviour of a molten core between start of melting and final containment in the core catcher.	This study has identified very mild reactivity ramps and energy releases (3 $/sec & 8000 MJ) in the worst conditions (complete loss of cooling, no trip, no absorber in the core) This is reported in a												

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1978

Table GBRA CONTRIBUTIONS OF GBRA TO THE NEA-GCFR R&D PROGRAM

Coordinator : J. CHERMANNE

ACTIVITY	RESPONSIBLE	OBJECTIVES	COMMENTS	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983
			"Supplement n° 2 to GBR4 Safety Working Document" February 1978.												
3. PERFORMANCE STUDIES	G. Vieider C. Oppenheim A. Krähe	The performance and economics relative to other reactor systems are evaluated by : - design reoptimization; - detailed physics analysis including fuel management and consistent comparisons to LMFBRs; - final thermohydraulic study including results from the R & D programme; - thermomechanical behaviour of the core throughout life taking into account the variation of swelling, creep, etc, with fast neutron dose; - short and long term economic assessment comparing with the HTR and LMFBR.	Possibilities of using a 1 year refuelling scheme was examined. The influence of uncertainties in the prediction of main fuel technology parameters on the economic prospects of the GCFR investigated. Physics and thermohydraulics performances are assessed in "GBR4 Performance", December 1974. Thermo-mechanics are assessed in "Supplement to GBR4 Performance", June 1976				R	R							
4. ECONOMIC ASSESSMENT	G. Vieider J. Bissell M. Chapelot	- Assessment of the cost of various GBR fuel cycles. - Comparison on an item-to-item basis of GBR4 with two HTR designs	- See "GBR4 Performance" - The conclusion shows that GBR4 and HTR capital costs are the same : "Analysis of relative capital costs between GBR4 and HTR projects", March 1975.				R								
5. R & D PROGRAMMES	J. Chermanne Ph. Van Asbroeck T. Bryant M. Quick	- Assessment of the effort required for the demonstration and later, commercialization of the GBR - Discussion of the "Gas cooled Breeder Reactor Research, Development & Demonstration Planning Guide" with various utilities, manufactures and research organizations in the European Community.	- "Gas cooled Breeder Reactor Research and Development programme" November 1974. - "GBR core material selection and related phenomena", June 1975. - "Gas cooled Breeder Reactor Research, Development & Demonstration Planning Guide", May 1978, a document prepared for the CEC. Cont.			R	R		R						

1978

Table GBRA CONTRIBUTIONS OF GBRA TO THE NEA-GCFR R&D PROGRAM

Coordinator : J. CHERMANNE

ACTIVITY	RESPONSIBLE	OBJECTIVES	COMMENTS		1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983
6. EXPLORATORY DESIGN STUDIES OF THE COATED PARTICLE CONCEPT	J. Yellowlees	Study of core concepts with particle fuel, selection of a reference design with considerable fuel cycle cost saving compared to earlier particle designs. The overall benefits from the development of this design are evaluated.	Work concentrated on : - fuel assemblies using SS-particle container; - special safety aspects of directly cooled particle designs; - thermohydraulics R & D work by the von Karman Institute, Brussels. In 1974, taking into account the results of a first fast flux irradiation of coated particles and the relatively poor nuclear performance reached with the coated particle design, it was decided to concentrate on the vented fuel pin design and consider the coated particle concept as a long term development.													
7. CASE FOR THE GBR	R.D. Vaughan J. Chermanne	Identify the reasons why the GBR should be developed.	"The case for the GBR and the construction of a Demonstration Plant in Europe", January 1974.													

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(1) Reactor core and blanket
(2) Steam generator unit
(3) Circulator unit
(4) Emergency cooling loop
(5) Refueling pantograph
(6) Lower cavity
(7) Helium and fission gas treatment plant
(8) Shielding

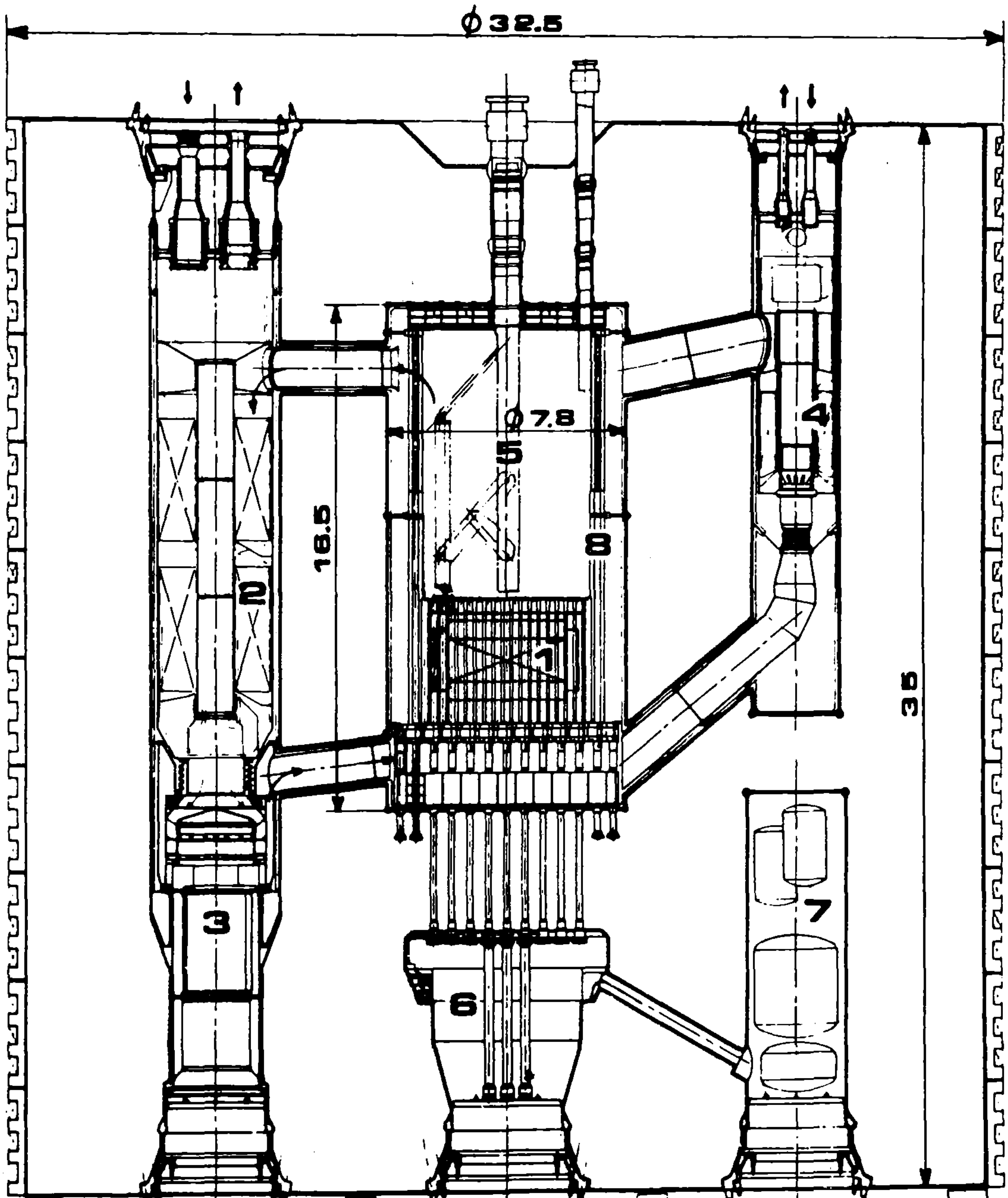


Fig.1: GBR 4 Nuclear Steam Supply System - Vertical section.

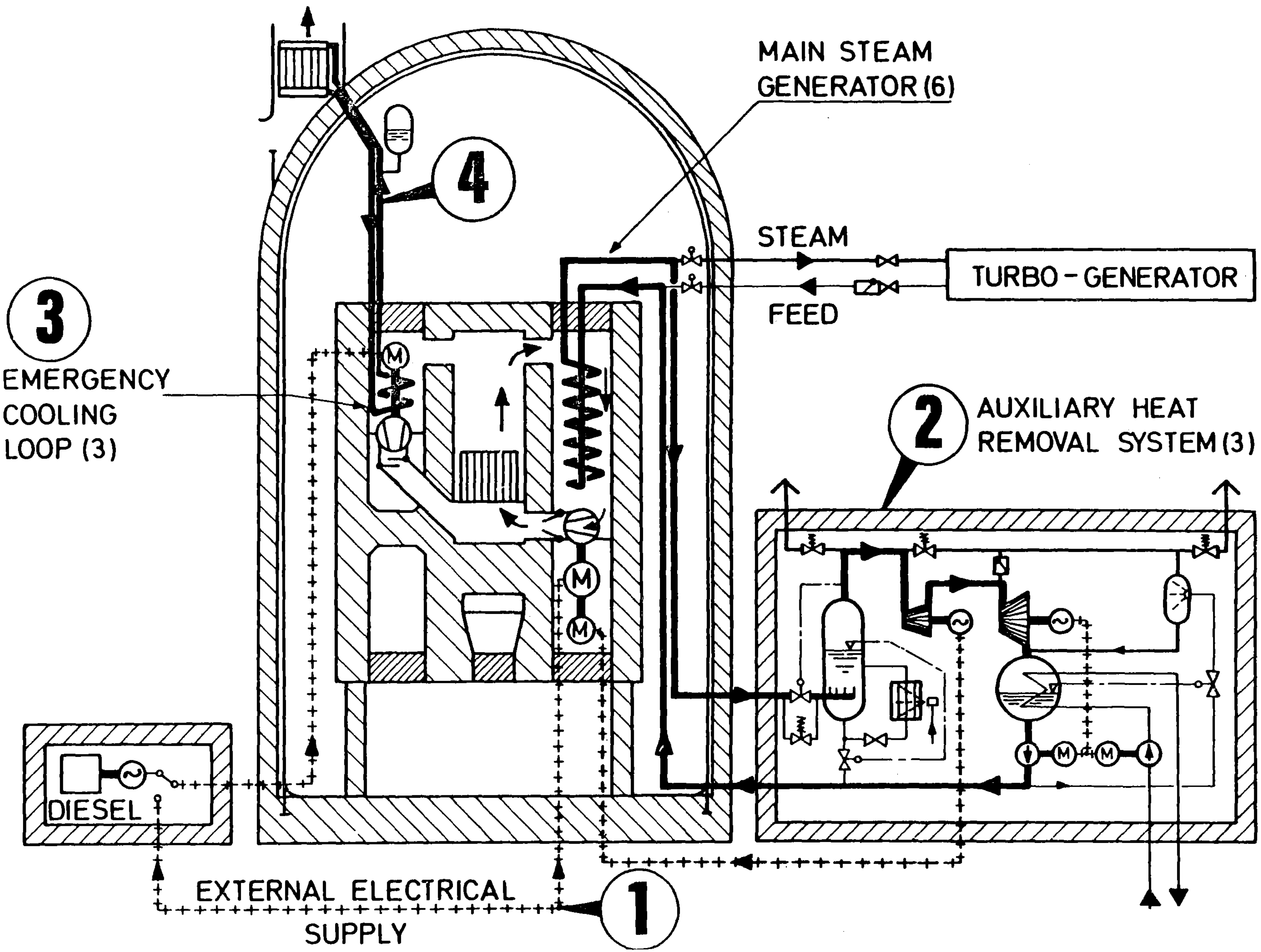


Fig. 2: GBR4 - Emergency Cooling System

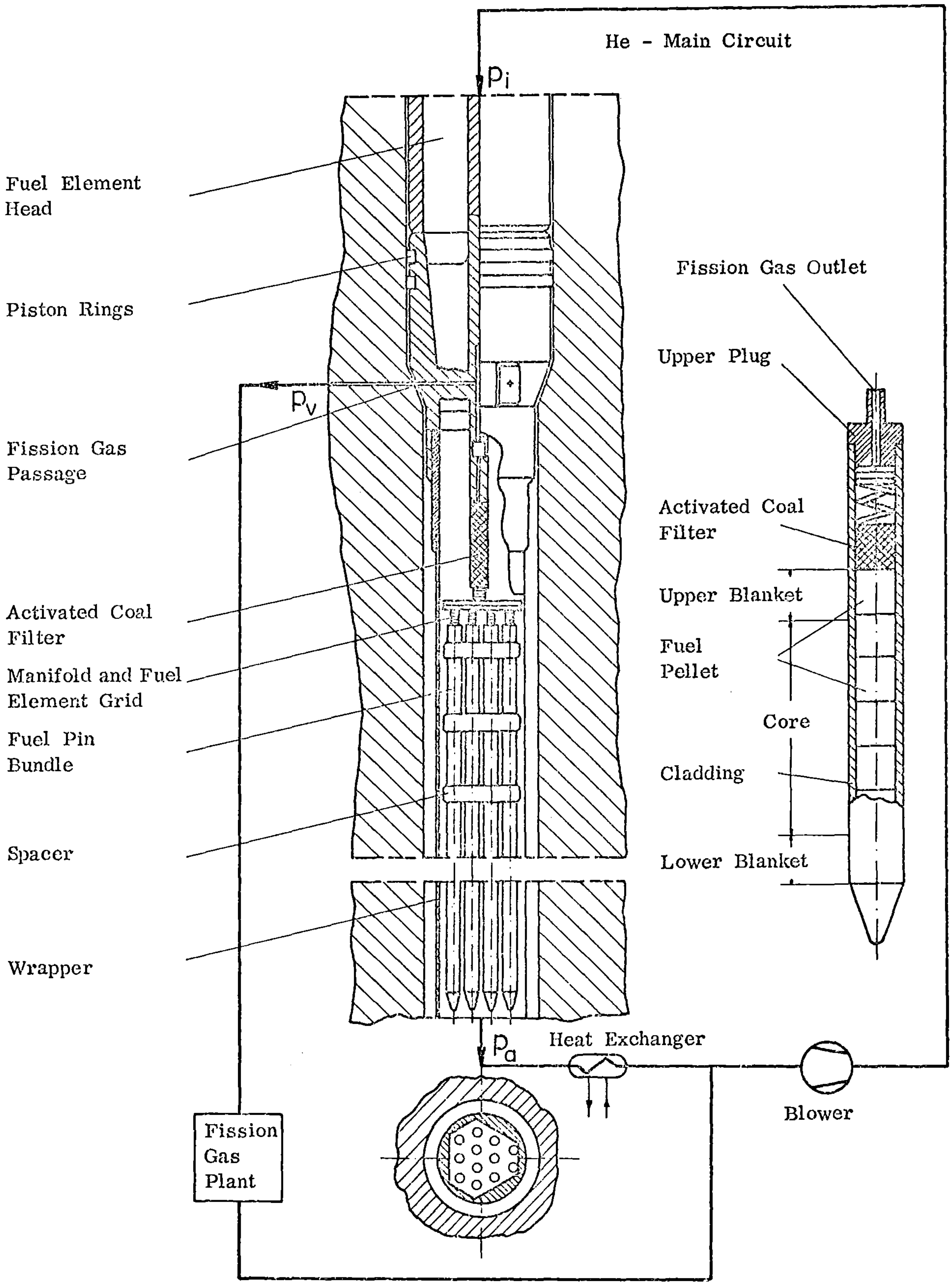


Fig.3: **Test fuel element (schematic) for the BR2 irradiation experiment**

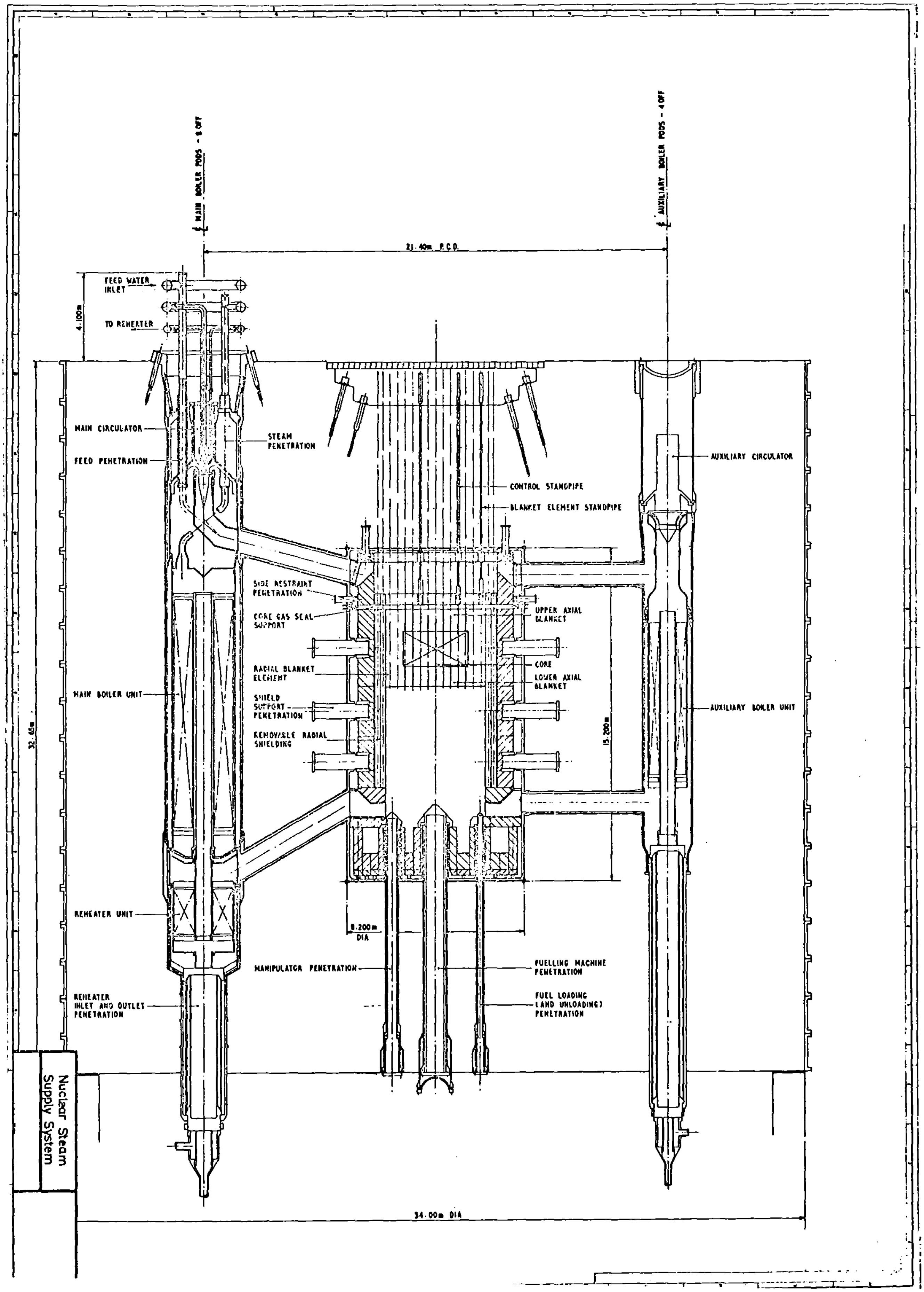


Fig.4: Vertical section through the Nuclear Steam Supply System of the GfK-KWU 1000 MWe GCFR reference design (GSB-1)

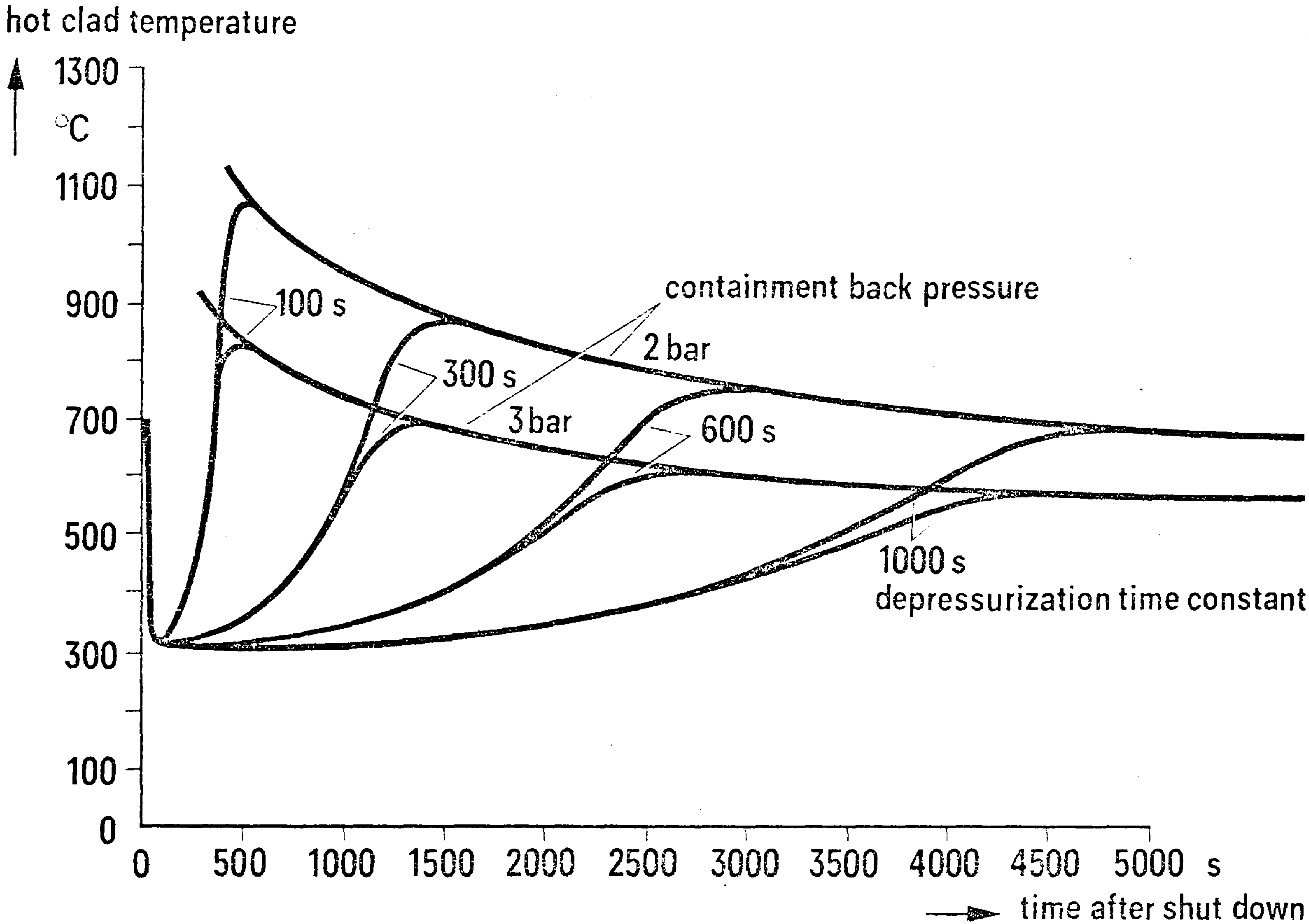


Fig.5: **Peak cladding temperatures after depressurization accident (GSB-1)**

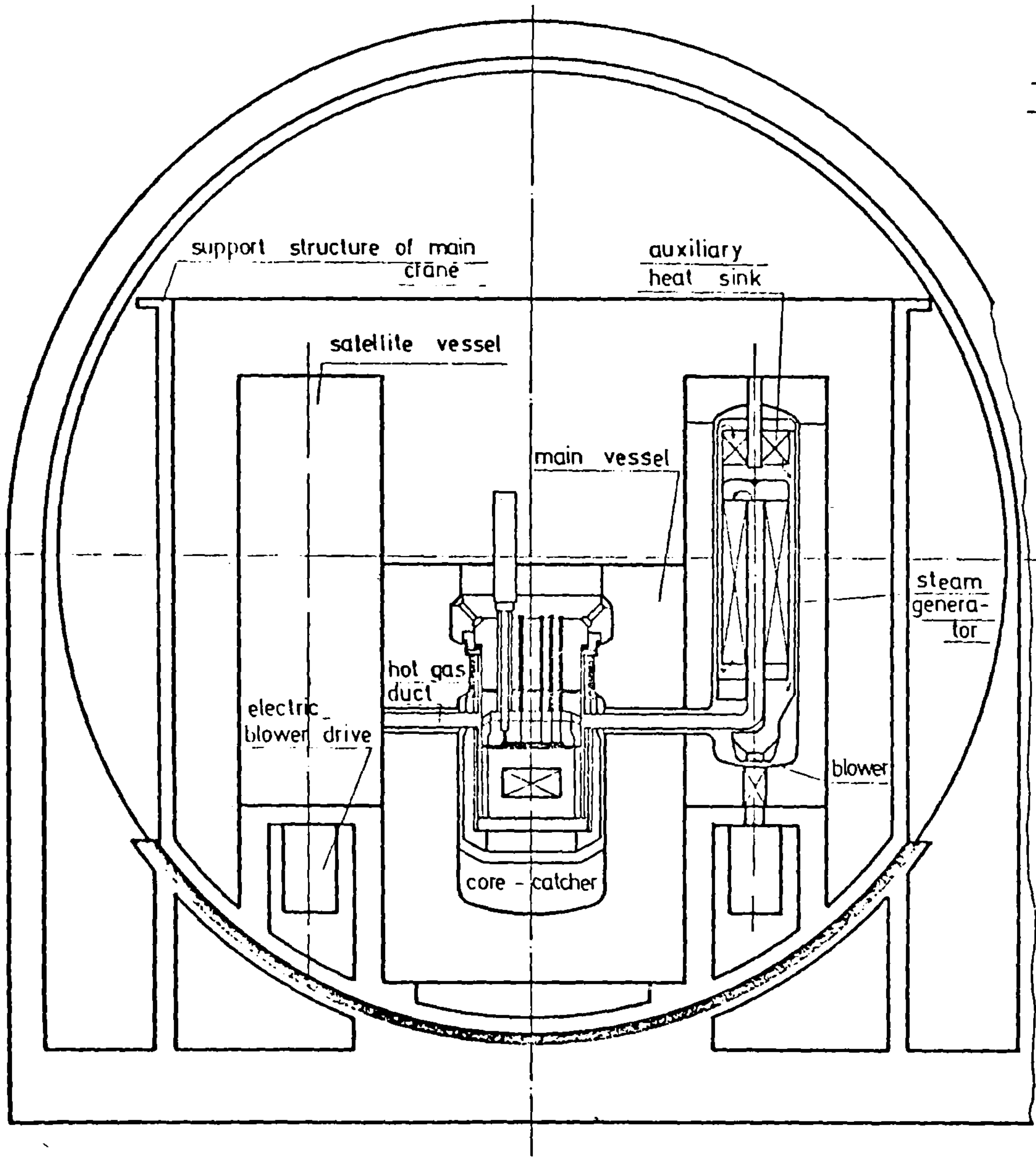


Fig.6: General Arrangement GCFR 1200

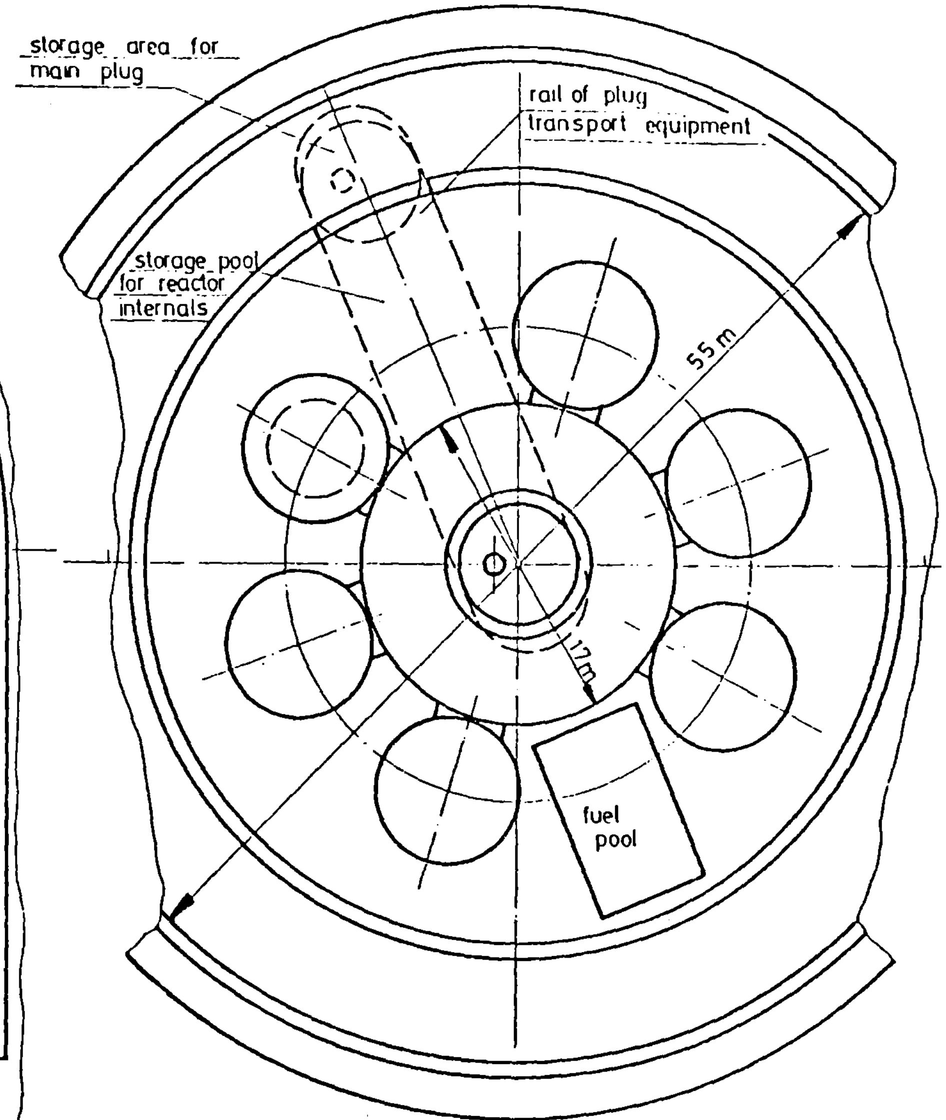


Fig.7: PCRV General Arrangement Plan View

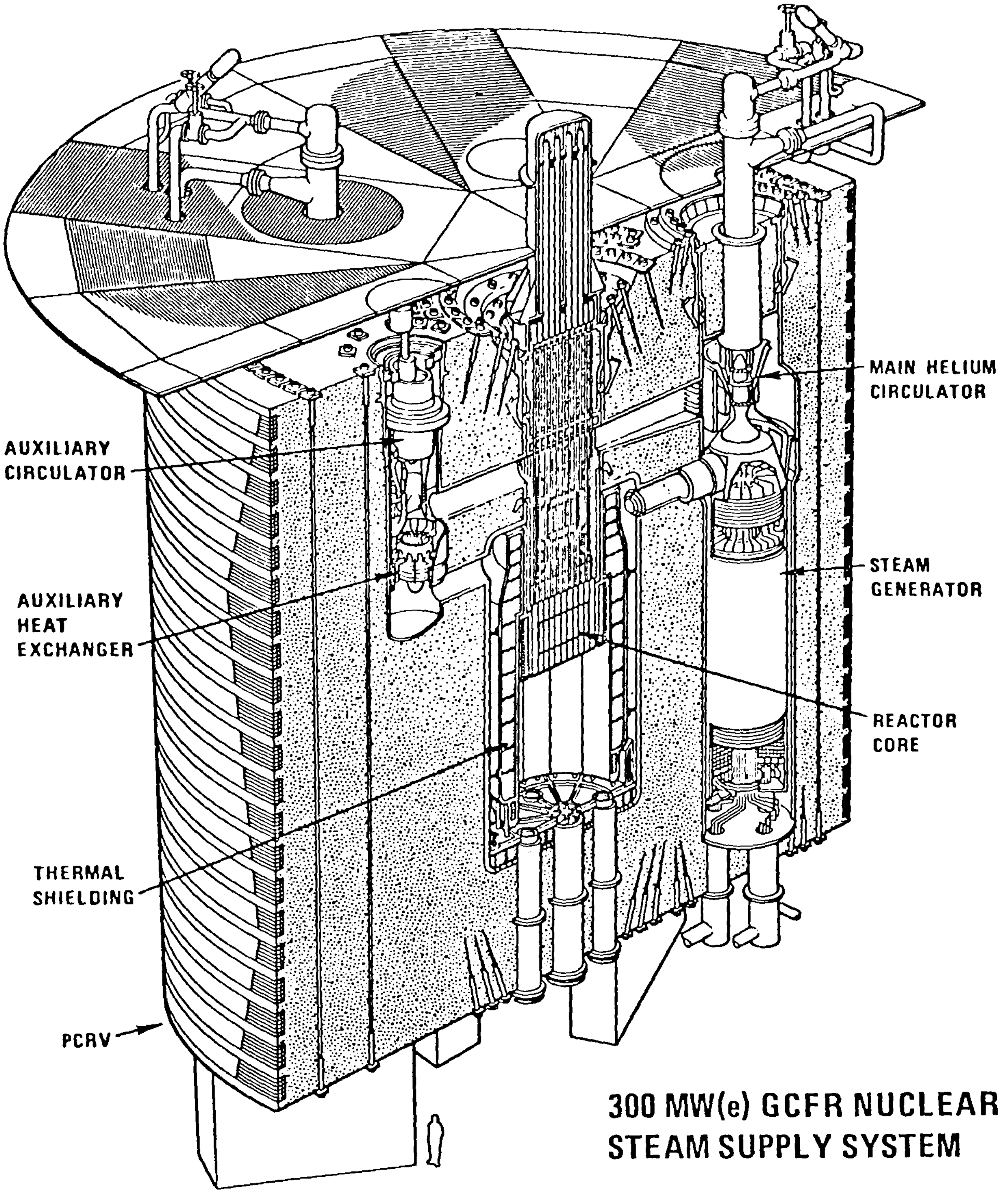


Fig.8: General Atomic GCFR 300 MW(e) prototype: cut through concrete pressure vessel showing the nuclear steam supply system

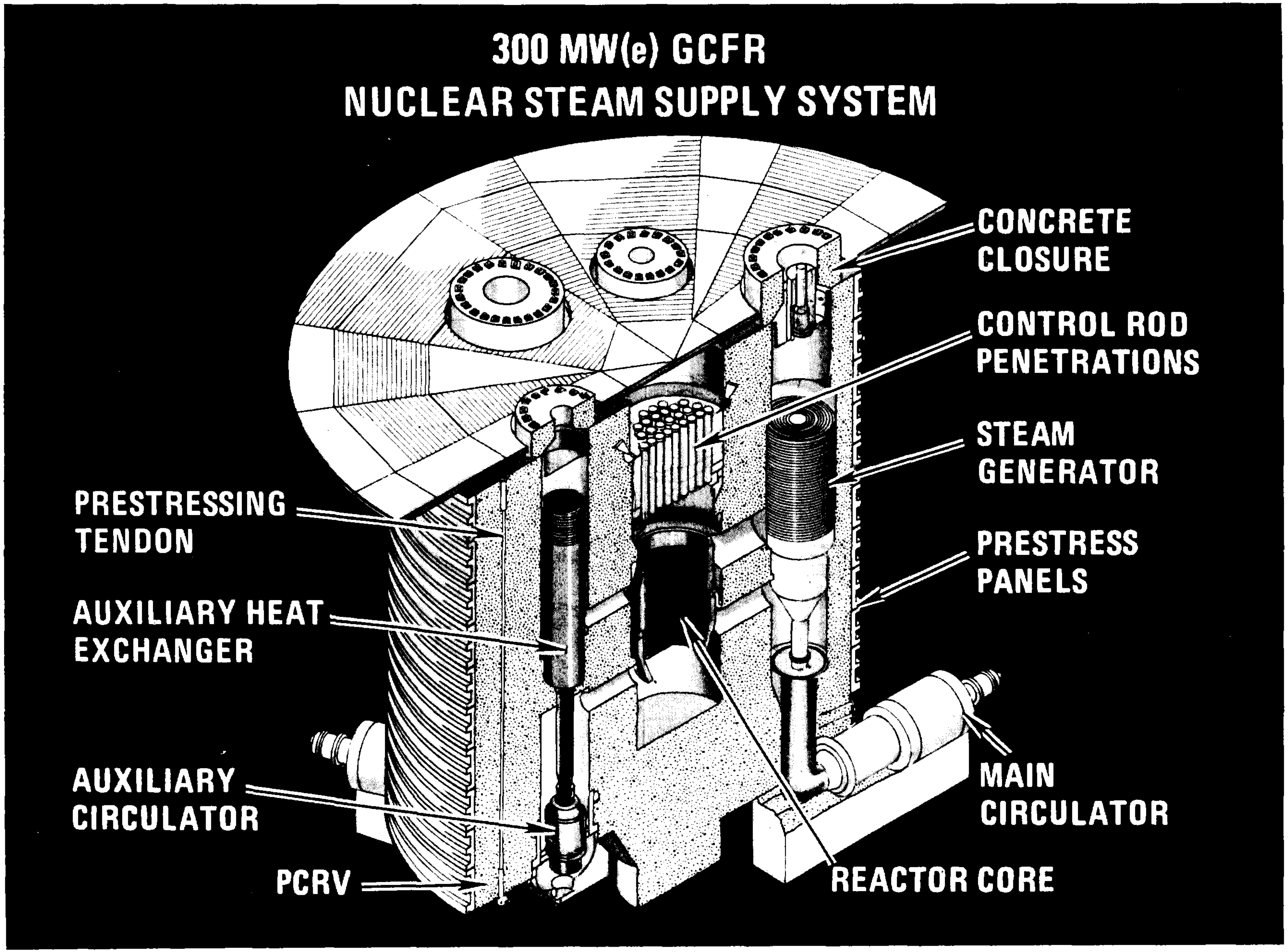
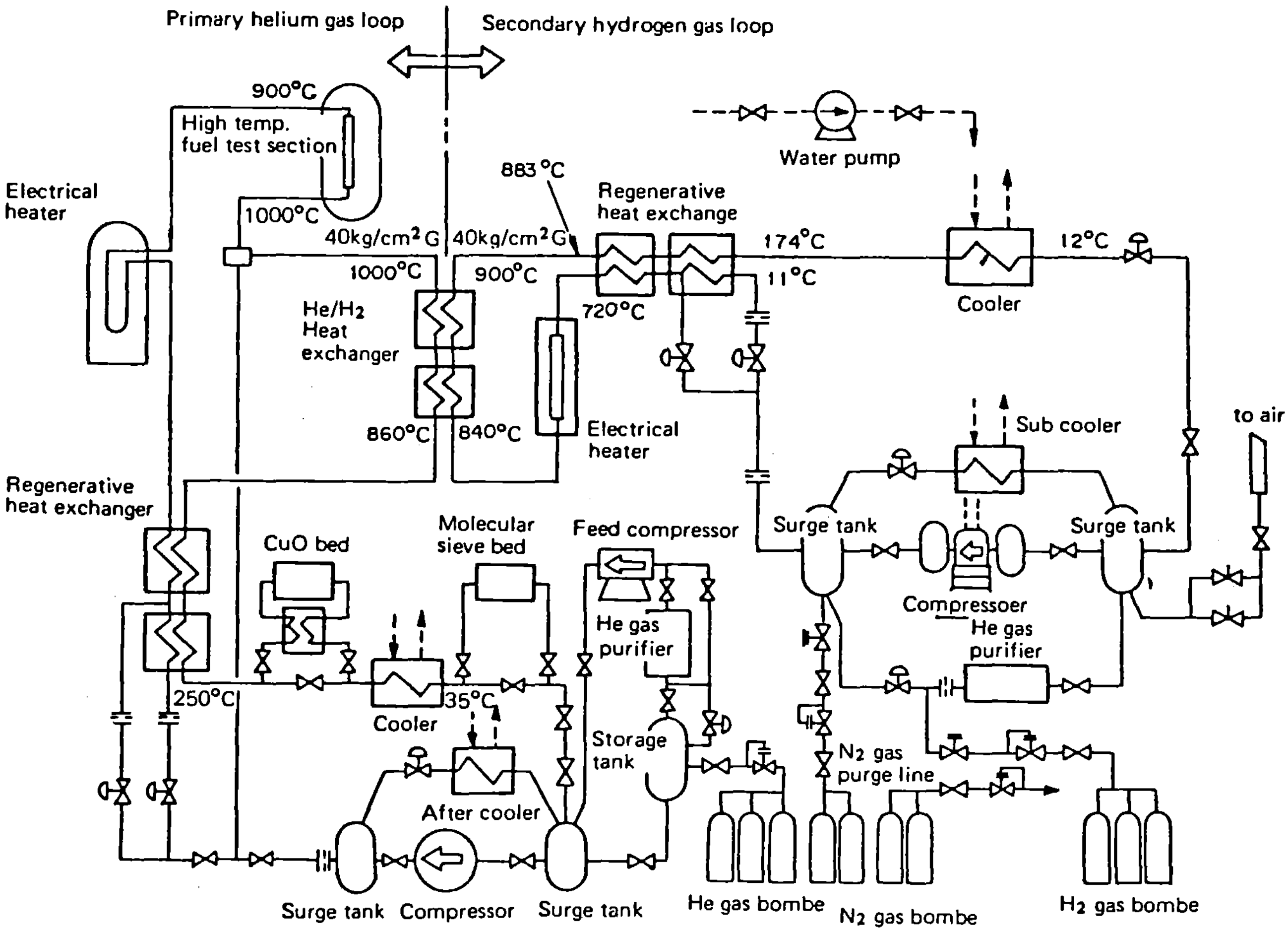

300 MW(e) GCFR
NUCLEAR STEAM SUPPLY SYSTEM
CONCRETE CLOSURE
CONTROL ROD PENETRATIONS
STEAM GENERATOR
PRESTRESS PANELS
MAIN CIRCULATOR
REACTOR CORE
PRESTRESSING TENDON
AUXILIARY HEAT EXCHANGER
AUXILIARY CIRCULATOR
PCRV

Fig.9:



Design parameters of hydrogen gas loop

Maximum operating pressure	; 42 kg/cm^{2}G	Head of the compressor ;	5 kg/cm^2
Maximum operating temperature	; 900°C	Heat input ;	80 kW (30 kW at He/H_2 heat exchanger 50 kW at electrical heater)
Flow rate	; ~30 g/sec	Diameter of main pipe ;	6 or 8 inches (pressure pipe) 1 inch (heat resisting pipe)

Fig.10: Flow Diagram of Secondary Hydrogen Gas Loop and Primary Helium Gas Loop

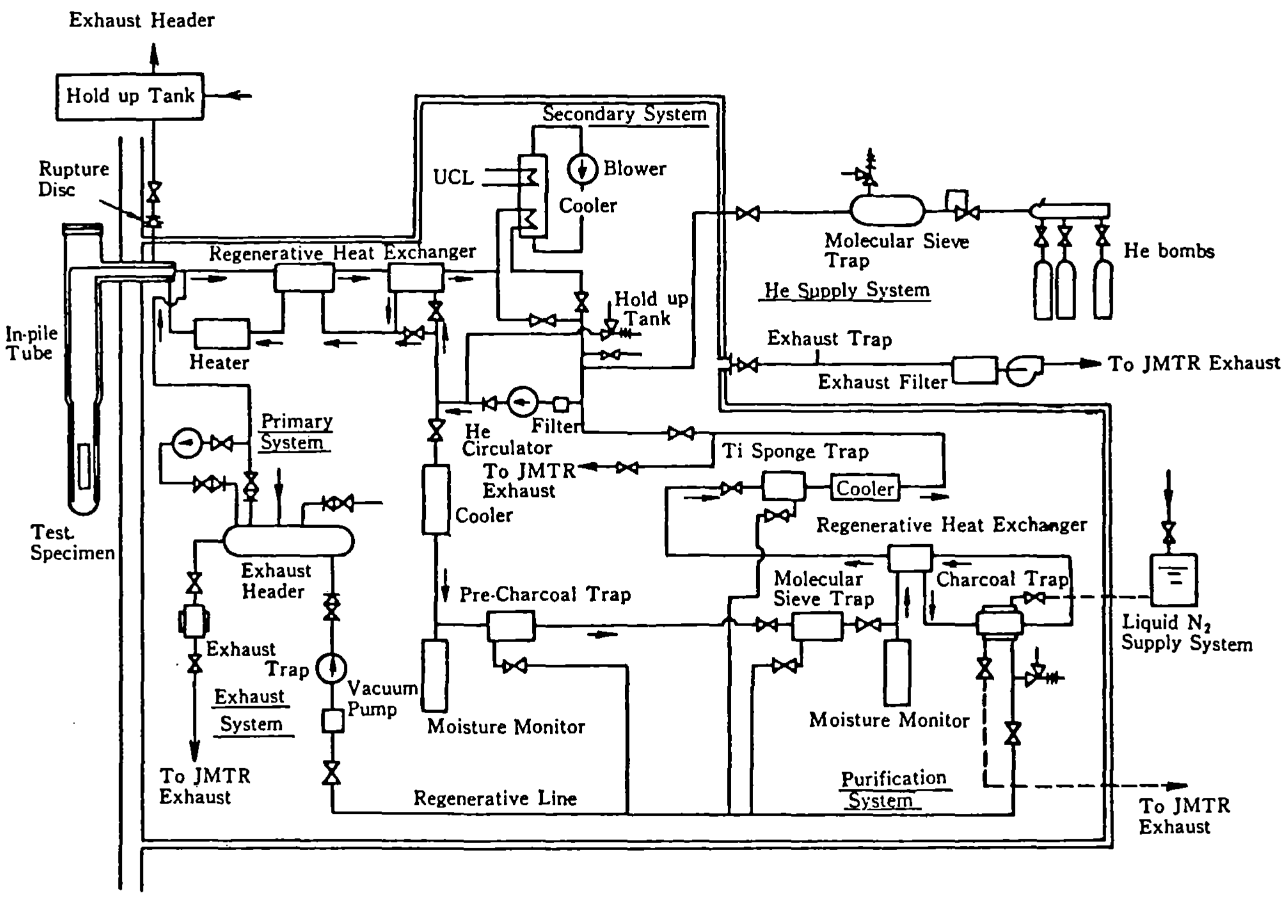


Fig.11: Flow Diagram of the OGL-1 Loop

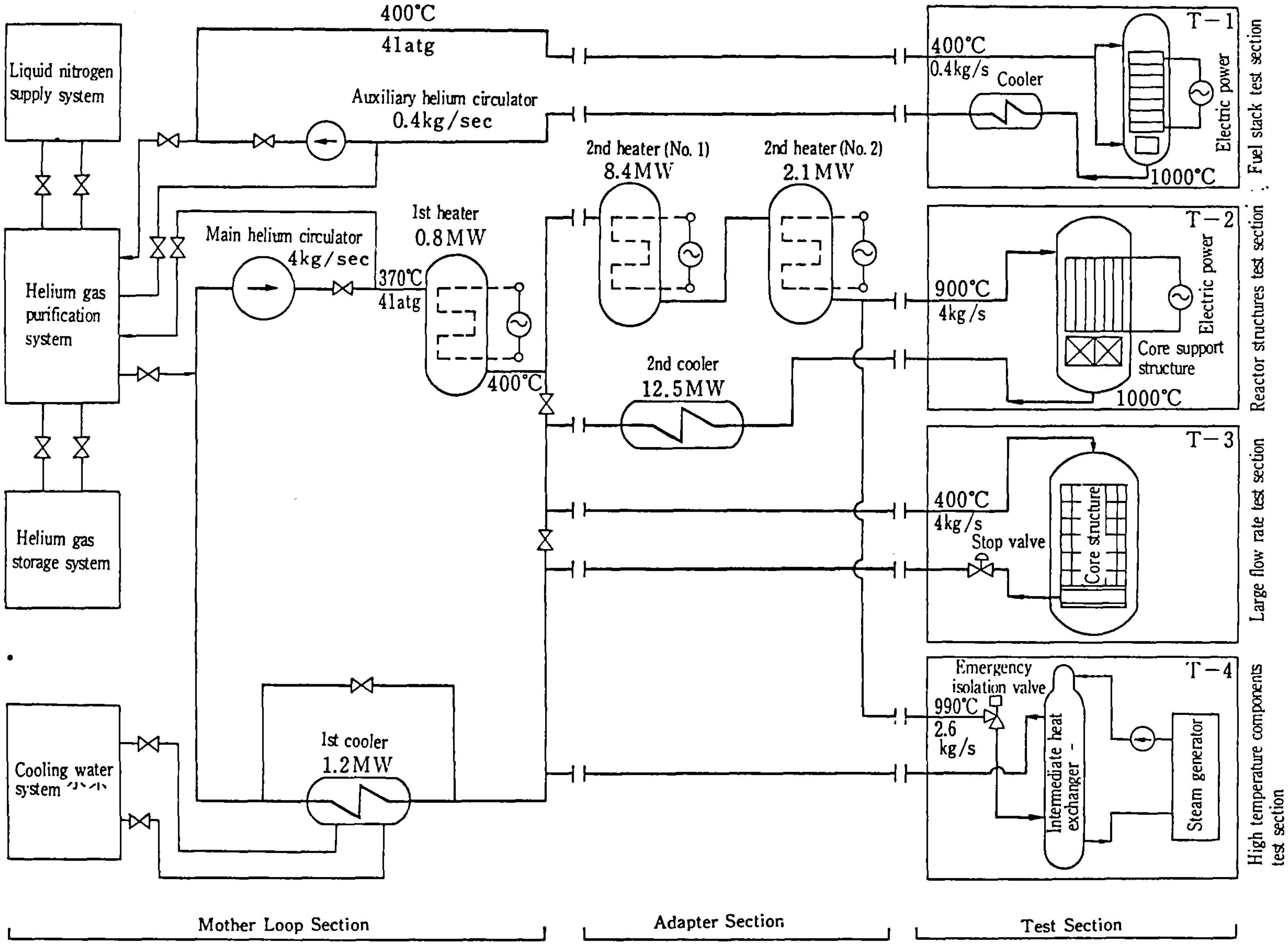
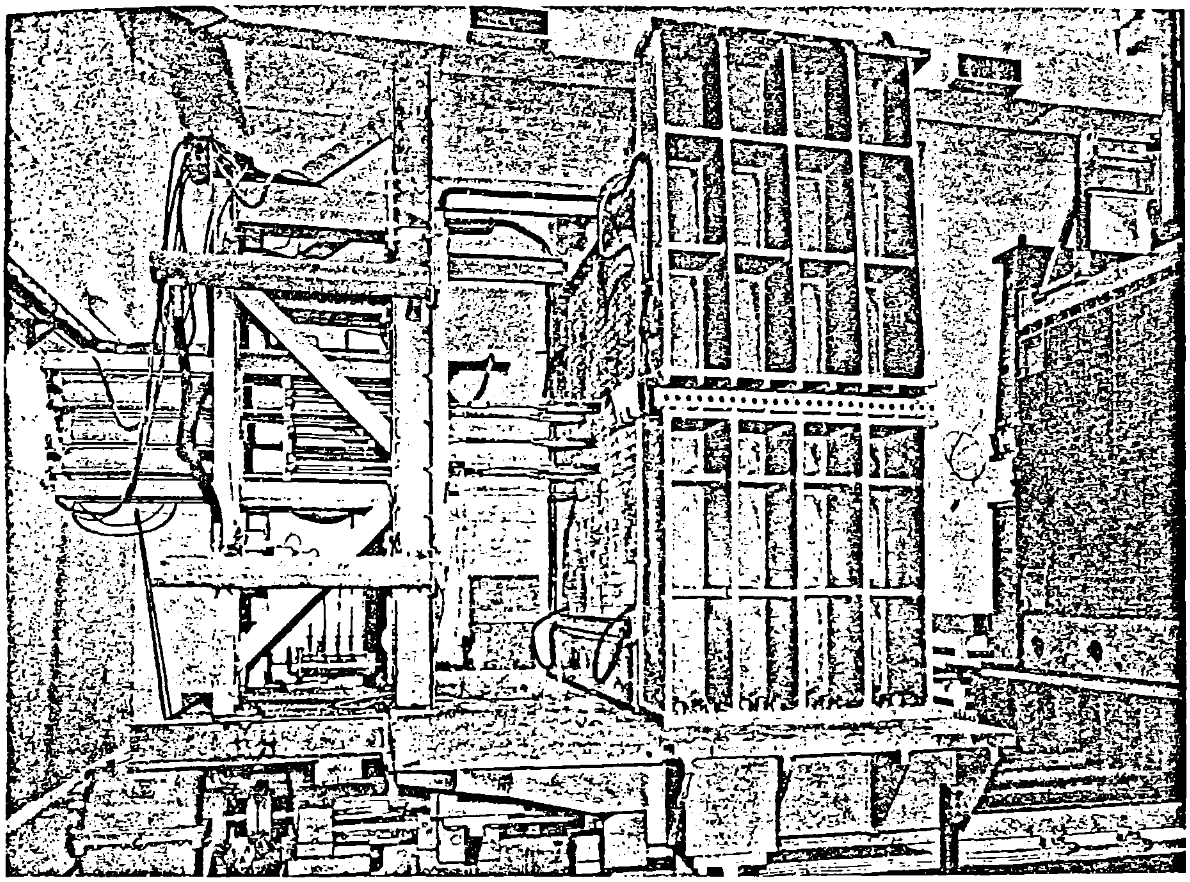
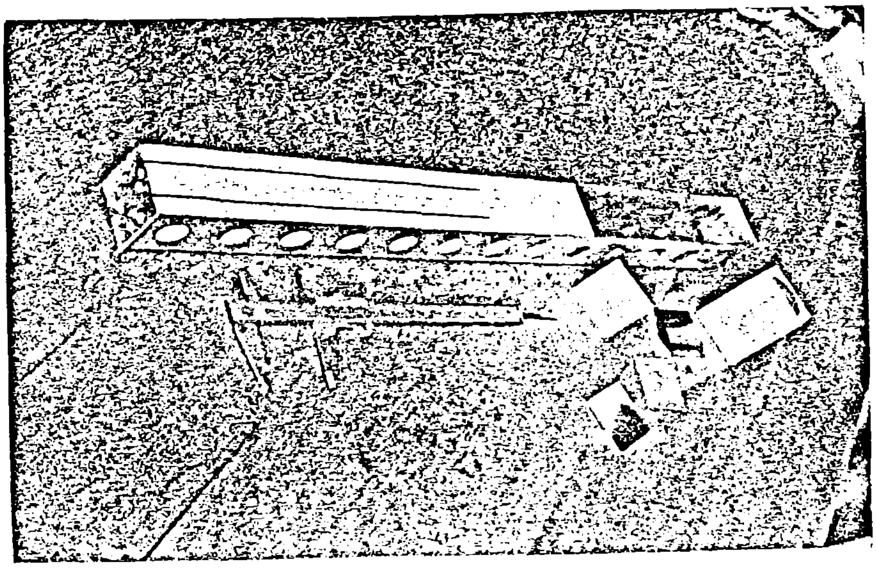


Fig.12: Flow Diagram of Helium Engineering Demonstration Loop



Fast Critical Assembly



Drawer

Fig.13: Views of fast critical assembly and its drawer

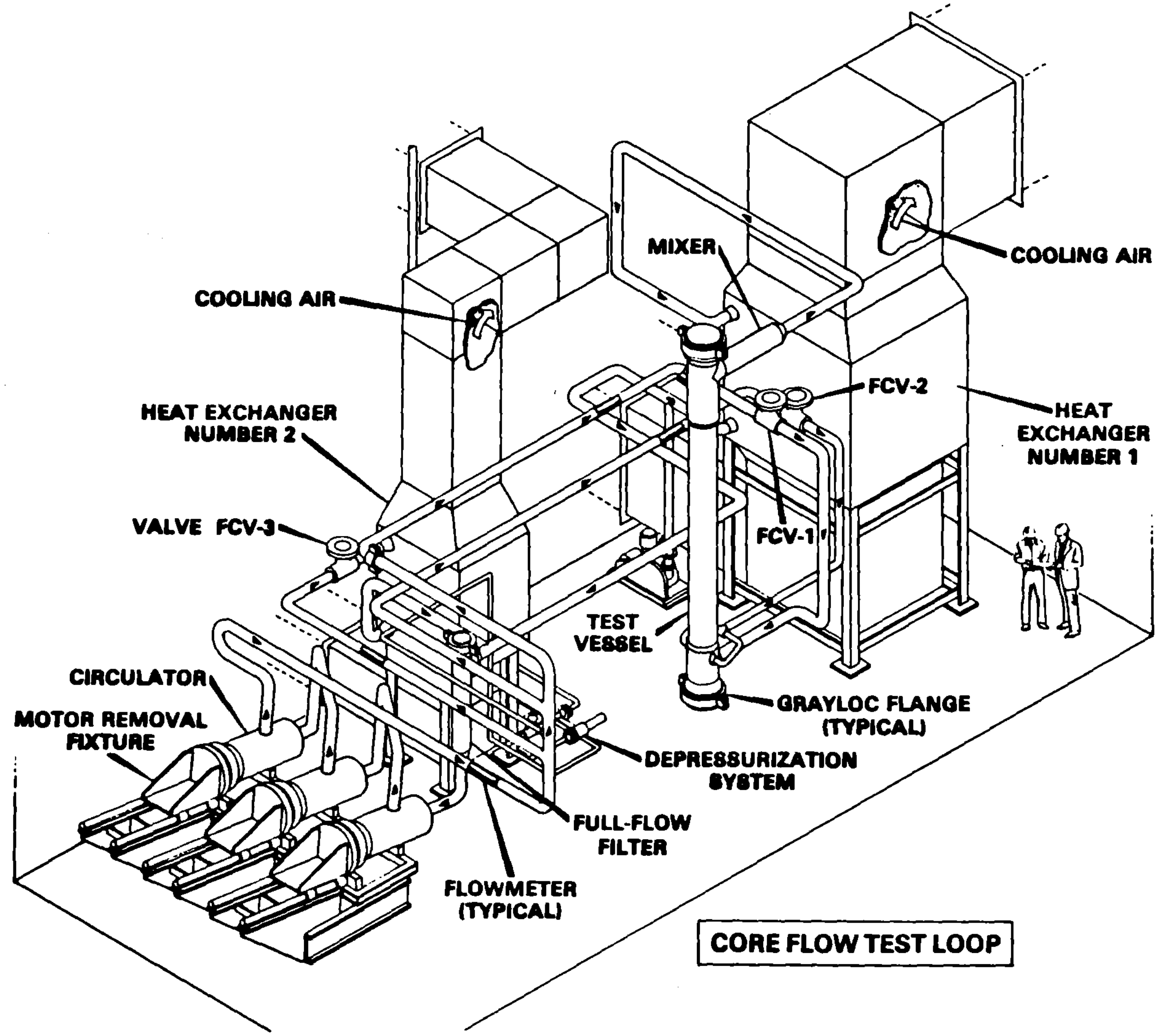

COOLING AIR
MIXER
COOLING AIR
FCV-2
HEAT EXCHANGER NUMBER 1
HEAT EXCHANGER NUMBER 2
VALVE FCV-3
FCV-1
TEST VESSEL
CIRCULATOR
MOTOR REMOVAL FIXTURE
GRAYLOC FLANGE (TYPICAL)
DEPRESSURIZATION SYSTEM
FULL-FLOW FILTER
FLOWMETER (TYPICAL)
CORE FLOW TEST LOOP

Fig. 14:

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