GCFR-FUEL-ELEMENT-IRRADIATIONS IN THE GSB-He-LOOP-MOL

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SUMMARY

The Gas Cooled Fast Breeder with vented fuel elements can be considered as a promising alternative solution to Sodium Cooled Breeders.

This variant of the GCFR needs the smallest amount of development-potential, considering that on one hand the He-plant is similar to that of HTRs, and on the other hand the fuel element development of the SNR can be applied.

As part of the development of the vented fuel element, irradiation tests on fuel element bundles will be performed in the BR2 reactor in the so called He-Loop Mol.

These are the only in-pile tests which respect all the requirements of a GCFR-power plant, except the original fast neutron fluence. This will be realized by the following items :

- *installation of a cadmium screen, assuring an epithermal to fast neutron flux ;*
- *irradiation of a vented 12 rod-bundle with all essential characteristics and components of a GCFR fuel element ;*
- *a fuel-element-venting-system together with a fission-gas-separator-system connected to the fuel element.*

The general aim of this test is a burn up to 60,000 - 100,000 MWd.t^{-1} with a maximum rod power of 500 W.cm^{-1} and a maximum cladding temperature of 700° C. Mainly the tests involve the irradiation behaviour of the vented fuel element bundles and the fuel-element-venting-system and its components.

These irradiation tests are a joint project of KFA and KWU with participation of S.C.K./C.E.N. and GfK.

I. INTRODUCTION, EXPERIMENTAL FACILITY

According to the Gas Breeder Memorandum, the Gas Cooled Fast Breeder (GCFR) with vented fuel elements, oxide fuel and stainless steel claddings can be considered as a promising alternative solution to the Sodium Cooled Fast Breeder. This variant of the GCFR needs the smallest amount of development potential, because design and construction experiences of HTR-plants, as well as the results and knowledge of the fuel element development for the SNR, can be applied. Nevertheless the fuel element and safety represent key-problems for the GCFR. Therefore, a safety study for the main loop of a 1000 MWe - plant will be elaborated by GfK and KWU and irradiation experiments for the development and testing of vented GCFR fuel elements will be performed in the BR2 as a common task of KFA and KWU, in collaboration with GGA, C.E.N./S.C.K. and GfK.

Two collaboration contracts exist, the first between KFA and KWU and the second, a tripartite contract, between KFA-KWU-GGA. A third contract between C.E.N./S.C.K. and KFA has been discussed and is in preparation.
Corresponding to these contracts and other agreements KFA is responsible for the design and construction of the test facility and for performing the tests, while KWU is responsible for the development and manufacturing of the test fuel elements. C.E.N./S.C.K. is responsible for the neutron-calculations and -experiments, for the Fuel-Transfer-Device, for realizing the reactor conditions which are necessary to reach the specified test-conditions, and to give assistance for constructing and loop-operation.

These planned experiments are the first dynamic in-pile tests, which respect by all means the requirements of a GCFR-power-plant, except the original fast neutron-flux and fluence. The design of these experiments allows us to investigate all fuel element problems connected with this concept, which are concentrated in the following two main objectives :

- Irradiation behaviour of a vented fuel rod bundle up to high burn ups ;
- Operation behaviour of the venting system under all possible operation phases over long operation times.

This will be realized by the following items :

1. Installation of a cadmium screen,assuring an epithermal to fast neutron flux ;
2. Irradiation of a 12 rod-bundle with all essential characteristics and components of a GCFR-fuel-element ;
3. A fuel-element-venting-system including a fission gas separator system connected to the fuel element ;
4. A maximum rod power of 500 W.cm^{-1} and a maximum cladding temperature of 700°C ;
5. A burn up objective of at least 60,000 MWd.t^{-1}.

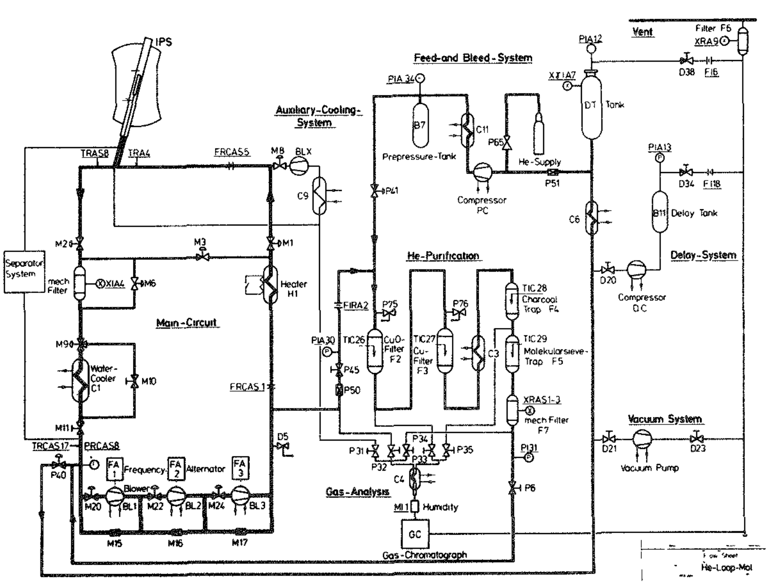
The specified rod power of 500 W.cm^{-1} could be easily reached in the central channel of BR2. But since this channel is occupied and we have to use the H4 channel, more investigations have to be done to find the correct core-configuration and the reactor power has to be increased to 100 MW or more.

These irradiations will be performed in the GSB-He-Loop-Mol, which is based on the old Siemens CO_2-Loop, which means that parts of this CO_2-Loop will be reused.

1. EXPERIMENTAL EQUIPMENT

The experimental equipment is composed by the following main components (Fig. 1) : In-pile-Section, Main Circuit, Secondary Circuits, Separator-System, Electrical-Instrumentation and Fuel-Transfer-Device.

FIG. 1



2. MAIN CIRCUIT

The main tasks of this circuit are to guarantee the continuous cooling of the test fuel element and to regulate the fuel element temperatures

The main components are three operating simultaneously gas bearing blowers in series A heater (normally not used), a mechanical filter, a cooler and a number of valves

A pressure control system guarantees a constant operation pressure of 60 bars It operates as a feed bleed system The mass-flow is kept constant by the rotational speed of the blowers

A three-way-regulating valve divides the mass-flow through the cooler and its by-pass so that the outlet gas-temperature is kept constant

In case of failure of all three main blowers, an auxiliary blower comes into operation to carry off the decay heat

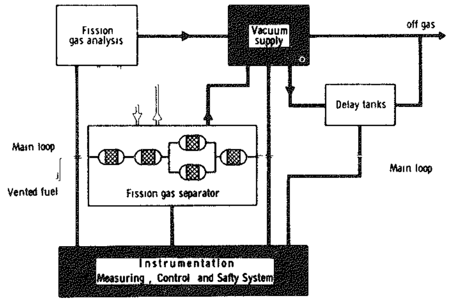
3. SECONDARY CIRCUITS

These consist mainly of the He-purification, the pressure-control, the discharge-system and the cooling water circuits The He-purification has to guarantee a specified impurity level in the cooling gas By the use of copper-, copper oxide- and molecular sieve beds, air water and ^{2}H levels should be regulated The impurity level can be measured by the gas analysis circuit
In the case of contamination of the cooling gas, this can be pumped into tanks by a compressor and from there it can be discharged into the reactors ventilation system
The vacuum pump is able to evacuate all parts of the loop

4. SEPARATOR SYSTEM

The separator system is part of the loop ventilation system A by-pass to the main loop between fuel element outlet and blower entrance sweeps all fission gases from the test fuel element into the separator system, where they are adsorbed

FIG. 2



Schema Spa tgassammelsystem SPAGS

The separator system (Fig 2) consists mainly of three liquid-N_2-cooled charcoal traps There are two traps in parallel, one operating and one spare The third is arranged in series as a safety trap
In addition there is a copper oxide- and a molecular sieve bed for removing humidity and tritium, and a measuring system with a Ge(Li)-detector which allows the analysis of the fission-gases
After each reactor cycle the traps have to be regenerated , the desorbed fission gases are pumped into delay tanks and, after a suitable delay time, flushed into the ventilation system

5. IN-PILE SECTION

The in-pile section (Fig 3) containing the test fuel element will be installed in the H4-channel of the BR2
In the core zone it is surrounded by a driver fuel element with a cadmium screen
The in-pile section, about 13 m long, is designed according to the reentrance and double barrier principle Therefore the gas-inlet- and -outlet-pipes are connected at the bottom, and the loading of test and driver fuel elements will be performed from the top
During normal operation the outer water-cooled pressure tube maintains the full pressure difference, whereas the inner hot pressure tube is unstressed Only in the case of failure of the outer pressure tube has the inner one to withstand the full pressure difference for a short time at full temperature The different thermal expansion of outer and

inner tubes is compensated by bellows.
The IPS is designed for maximum operation conditions of 600°C and 67 bars.

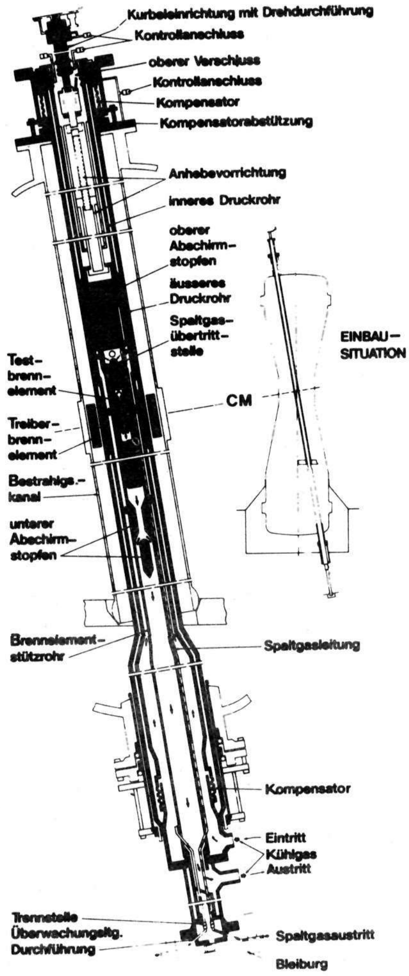


FIG. 3

6. FUEL TRANSFER DEVICE

The decay heat load of the test fuel element demands continuous cooling after reactor shut down for a definite time. Therefore, before unloading, the test fuel element will be lifted out of the core zone by a special lifting mechanism and stored in the upper part of the in-pile section for one more BR2-cycle. Then the fuel element can be unloaded and stored with the fuel transfer device without any further forced convection cooling.

7. THE ELECTRICAL INSTRUMENTATION

Contains the whole electrical power supply, the safety system, and all instruments for operation and for data collection.

8. STATUS AND TIME SCHEDULE

The design work for the He-Loop has been completed, and we are now in the phase of construction. The main components have been ordered, the manufacturing of several sub-assemblies has started, the old CO_2-Loop is dismantled as far as necessary and the mounting of the He-Loop will be started soon. The first approval-phase at the CEE has been accepted and the second phase is under preparation. Two tests, a short-term and a long-term test, are planned, and according to the latest time schedule irradiation will start in the middle of 1975.

II. TEST-ELEMENT, OPERATING CONDITIONS

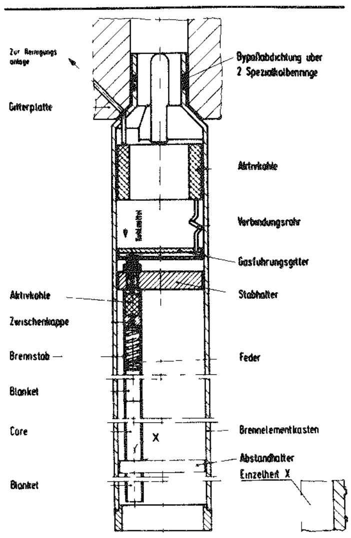
As part of the GCFR development work, KWU is investigating a power plant design based on fission gas-vented rod-bundle fuel elements. Performance testing of this fuel element type is to be done in the BR2 helium loop, i.e. test elements of typical GCFR design are to be irradiated under characteristic GCFR operating conditions.

A brief description of the GCFR fuel element is given, the test element derived from that is described, then the operating conditions of the He-Loop are compared, to those of a GCFR power plant, and, finally, the main questions to be answered by the irradiation experiments are listed.

1. GCFR FUEL ELEMENT

The GCFR fuel element designed by KWU is shown schematically in Figure 1.

FIG. 1

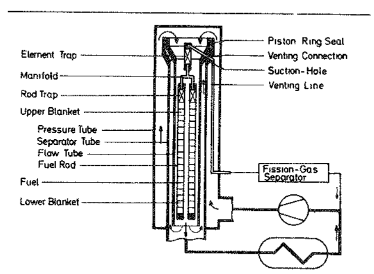


GCFR Fuel-element (schematic)

A hexagonal bundle of 270 (= 271-1) rods is fixed to a rod-holder at its upper end. Spacers are fixed to a flow tube surrounding the bundle, i.e. the rods may move axially through the spacer meshes as a result of differential thermal expansions. The fuel rod is composed of the fuel-stack (UO_2/PuO_2 pellets), axial blankets above and below (depleted UO_2 pellets), and a fission-product trap (charcoal bed) at its upper end. In the 1.5 m long fuel zone, the cladding tube (1.4981 stainless steel) is "roughened" by circumferential ribs to improve the clad-to-coolant heat transfer.

All fuel rods are manifolded by axial channels in their upper end-plugs and cross-channels in the rod-holder (Fig. 1)* to an additional charcoal trap, the exit of which is connected to the venting system. Above the venting connection, the head of the fuel element is sealed to the reactor grid plate by means of piston-rings.

FIG. 2



He-Loop Venting System (schematic)

Thus, as shown schematically in Figure 2** the venting system establishes a coolant by-pass starting at the fuel element outlet, flowing up outside the flow tube, entering the venting connection through a suction hole and picking up there the effluent from the charcoal traps, passing the fission gas separator where impurities are removed from the helium carrier-gas, and re-entering the coolant at the circulator inlet.

*The cross-channels may also be located in a separate manifold-grid above the rod-holder, as shown in Figs 1 and 3.

**In fact Figure 2 shows the He-loop venting system. This, however, is equivalent to the GCFR one, as confirmed below.

The benefits of this vented fuel design appear to be

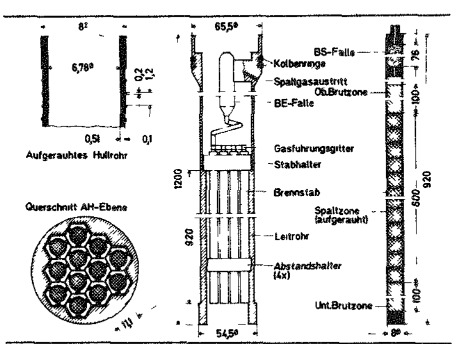
- pressure equalization inside and outside the fuel rods ; the cladding tubes are not subject to the coolant system pressure but only to a small pressure differential governed by the core coolant pressure drop ;
- no accumulation of fission-products released from the fuel matrix, due to continuous venting and helium in-leakage through cladding defects (due to the slightly lower inside gas-pressure and, therefore, substantially reduced fission-product release to the coolant in case of a cladding defect) ;
- feasibility of defect detection by activity monitoring in the venting line ; in case of a cladding defect, more short-lived fission-products will reach the monitor station due to convective instead of diffusive transport to the venting connection ;
- reduction in overall fuel element length and core coolant pressure drop, since no fission gas plenum is necessary ;
- improved safety in case of a loss-of-coolant accident.

2. He-CIRCUIT TEST ELEMENT

For irradiation testing, the GCFR fuel element as described above has to be adapted to the marginal conditions of the BR2 He-loop, i.e. to the space available in the in-pile test-section and the heat removal capacity of the loop, the active core height of the reactor, and the neutron flux and spectrum inside the driver element and Cd-screen in a reflector channel.

The test element resulting from these limitations is shown schematically in Figure 3*. The number of rods was reduced to 12 and the fuel length to 60 cm. Rod diameter (8 mm), triangular pitch of the bundle (11.1 mm), flow tube cross-section and inner profile, and the interrelations of these dimensions were optimized to achieve as flat a cladding temperature distribution as possible, across the bundle. Apart from these modifications, the keyfeatures of the GCFR fuel element design as described above have been perfectly realized in the He-loop test element.

FIG. 3



He-Loop Test-element (schematic)

To obtain the design heat-rating and target burn up without power degradation with time, a high enrichment of the test fuel uranium fraction is needed. The chemical fuel composition, however, will be representative for a GCFR.

Owing to cladding pressure equalization, the lower system pressure of the He-loop (60 bar with respect to 90-120 bar in GCFR lay-out), is not relevant for the test element. The venting system is designed to operate in the same way as in a GCFR, the reactor grid-plate being replaced by the test section separator tube.

Development work presently done in the KWU Erlangen laboratories is mainly directed towards examination techniques for tube roughening, charcoal trap design are performance, rod-manifold sealing, venting connection design, and rod-spacer interactions. By spring 1975, two test elements will be fabricated for irradiation in the He-loop.

3. OPERATING CONDITIONS

Design and operating conditions of the He-loop test-element are compared (Table 1) to those of a GCFR fuel-element. Linear heat-ratings and surface temperatures of the fuel rods must be regarded as the most relevant operational data.

It is expected that a linear heat-rating of 500 $W.cm^{-1}$ (axial maximum, averaged over the bundle) will be obtained. This would correspond to the hot-channel value in present GCFR lay-out. Final information will be gained by a BRO2 experiment. To some extent, the test element power will be optional adjusting the driver-element fuel content.

*With respect to the sizes indicated in Fig. 3, the upper blanket length and consequently, the total rod and element lengths and the number of spacers were increased. The intention was to make the upper blanket length (which might be relevant for the venting system performance) closer to that in a GCFR.

TABLE 1

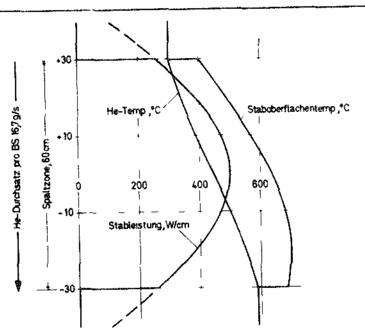
Design and Operating Data for GCFR Fuel-element and He-loop Test-element

Element for		1000 MWe GCFR	BR2 He-loop
Number of fuel-rods		270	12
Triangular pitch	mm	11	11.1
Rod diameter	mm	8.2	8
Rod length, total	mm	3000	1218.5
fuel	mm	1500	600
upper blanket	mm	600	420
lower blanket	mm	600	95
Rod surface roughening		Circumferential ribs	
Fuel material		UO_2/PuO_2-pellets	
Cladding material		Stainless steel 1.4981	
Maximum linear heat-rating	W/cm	500	
Maximum target burn up	GWd/t	100	
Helium system pressure	bar	90-120	60
Maximum helium outlet temperature	°C	580	
Maximum rod surface temperature	°C	685	600-750 (optional)

Rod surface temperatures (axial maxima) will be optional between 600°C and 750°C, approximately, by adjustment of coolant flow-rates and temperatures, thus covering the whole range of present and perhaps future GCFR lay-out. The methods of rod surface temperature determination will be verified in a calibration experiment using an electrically heated rod bundle.

Axial distributions of linear heat-rating, coolant temperature, and rod surface temperature are shown in Figure 4 as an example. The fuel zone will be located symmetrically with respect to the BR2 axial flux maximum, the linear heat-ratings at the ends of the fuel zone thus amounting to about half the maximum value. The maximum rod surface temperature will be reached in the lower part of the fuel zone.

FIG. 4



Axial Power and Temperature Distribution (example)

The fast neutron flux (> 100 keV) in the test-section will not be representative for a GCFR, i.e. typical fast fluence effects like stainless steel swelling are not expected to occur during irradiation.

The coolant pressure drop between test element outlet and circulator inlet which is most relevant for venting system performance will be controlled by means of a throttling valve, thus covering the whole range expected in a GCFR.

A target burn up of at least 60 $GWd.t^{-1}$ has been scheduled for the main irradiation experiment. It is planned, however, to extend this to 100 $GWd.t^{-1}$, provided that the experiment will run according to schedule. A short-time irradiation preceding the main experiment and using pure UO_2 fuel will serve as a final qualification test for the loop and the test-element.

4. EXPECTED RESULTS

The main objectives of the GCFR fuel-element performance testing in the BR2 He-loop, resulting from the main tasks for GCFR fuel development, are considered to be

- venting system performance with respect to charcoal traps, element-to-grid-plate connection, and fission gas-separator under both constant and transient conditions ;

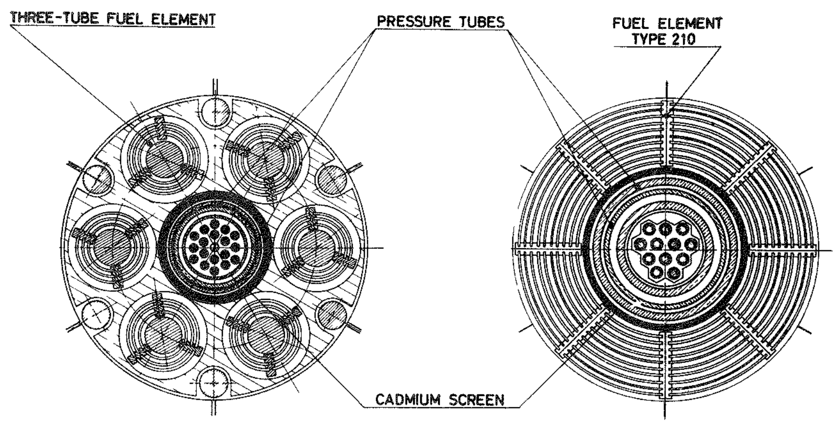
- mechanical and dimensional stability of the fuel element as a whole ;
- mechanical and thermohydraulic performance of roughened rod surfaces ; erosion resistance of the cladding ribs, and their interactions with the spacers ;
- chemical interactions between fuel and cladding which will establish the real upper limit for cladding temperatures, as the less corrosive coolant - with respect to sodium - would allow for higher values.

III. REACTOR PHYSICS STUDIES

1. The preparation work related to the irradiation of 12 fuel pin bundles in the cadmium screened helium-loop at BR2 is determined by the following main features :
   a) the outer diameter of the in-pile section (96 mm) makes necessary the utilization of one of the five 200 mm diameter H channels ;
   b) as the H1 central channel is generally occupied for the irradiation programme of mixed oxide fuel in sodium loops, the helium loop has to be introduced into a peripheral H channel ;
   c) a maximum linear power as close as possible to the value of 500 $W.cm^{-1}$ should be achieved in each fuel pin.

2. The loading of a 96 mm diameter loop experiment into a 200 mm channel requires the development of a new type of fuel element (element type 200) which completely surrounds the in-pile section of the loop and gives high fast neutron production in this region. The arrangement used up to now consisted of a 200 mm aluminium plug containing six three-tube fuel elements around a cadmium screened central channel ; the 500 kW MOL-7B sodium loop has been irradiated in these conditions since July 1972 (Fig. 1). The available diameter inside the cadmium is limited to 69 mm.

FIG. 1



MOL 7B SODIUM LOOP　　　　GSB HELIUM LOOP

The driver fuel element type 200* is made up of eight subassemblies around a cadmium screen, using the same fuel plate compositions as in the standard fuel elements of the reactor. The number of fuel plates in each subassembly and the diameter of the cadmium screen can be selected according to the dimensions of the irradiation device and the neutron flux conditions required. The arrangement adopted for the Helium loop is shown on the right hand side of Figure 1, the driver fuel element contains 10 plates per fuel assembly (element type 210).

3. Irradiation under fast neutrons in a peripheral H channel has never been performed in the past. Core configuration studies were therefore started in the nuclear mock-up BRO2 in order to design an appropriate BR2 core loading and to evaluate the flux level which can be obtained in a loop under cadmium in a peripheral H channel.
   In these studies, different requirements for the reactor operation conditions and the irradiation programme as a whole must be taken into account .
   - the reactivity investment must be sufficient to compensate a reactivity loss of about 6-7 % due to the irradiation devices and to achieve an operation time of three weeks without fuel reloading ;
   - the maximum heat flux, at present about 450 $W.cm^{-2}$, should never exceed 600 $W.cm^{-2}$, the total reactor power has to be as low as possible, with an upper limit at 110 MW, whereas a high value of the average to maximum flux ratio over the reactor core must be obtained.

4. In the early stages of the feasibility study of the helium loop experiment, it appeared that all means should be used to increase the flux in the selected channel H4 in order to realize an average power rating of 500 $W.cm^{-1}$ in the fuel pins at the hottest plane, i.e. high fuel concentration around the H4 channel and high power density (or maximum heat flux) in the core ; moreover, the ^{235}U enrichment of the pins, with a Pu content fixed at 15 % when mixed oxide is used, should reach 90 %, at least for the inner pins of the bundle.

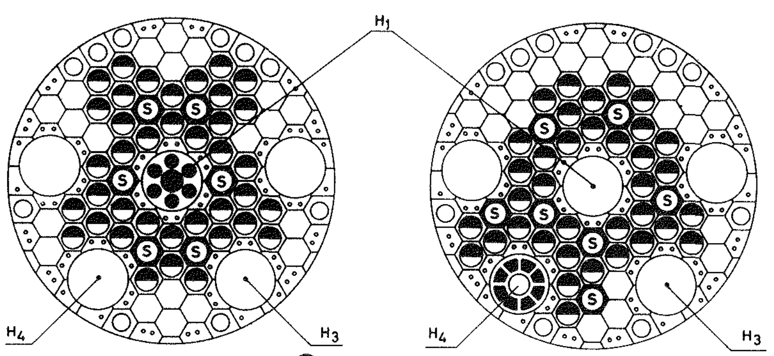
*General denomination of the elements type 210...213 which contain respectively 10...13 fuel plates per subassembly.

Neutron physics calculations of the loop were performed by means of a one dimensional transport code, using forty energy group cross-sections The calculations led to the definition of the characteristics of the cadmium screen and of the number of pins An estimate of the power rating in the pins was obtained by normalizing the calculations to the thermal data from the MFBS - 6 sodium loop, at that time under irradiation in the BR2 central channel H1 , an approximate value of the flux reduction from H1 to H4 was then used and a corresponding factor was applied to the results

5 Improving the accuracy in the predictions of the performance of the loop, in relation with the reactor operation conditions (core configuration, power density, total power), comprises the following steps

a) The acquisition of a fuel element type 200 for BRO2 was decided after the feasibility study of the loop , from March 1973 onwards core configurations were built with this element in a peripheral H channel and numerous measurements were performed In these tests, also performed in preparation to the irradiation of the MFBS - 7 sodium loop (start of the irradiation in March 1974), the element was made up of 13 plates per subassembly (element type 213) and the nuclear mock-up at the centre simulates a 250 kW sodium loop (outside diameter 53 mm) Although this arrangement differs from the device which will be used for the helium loop irradiation, these core configuration studies, complemented by calculations, provide information about the reactor operation characteristics which would satisfy the helium loop specifications
Figure 2 shows, on the left hand side, the present BR2 core configuration (configuration 7B) and, on the right hand side, the core configuration selected at present as being the most suited for the irradiation of a loop in H4 with respect to the different requirements, according to the measurements carried out in BRO2

FIG. 2



FUEL ELEMENT
SAFETY ROD

BR2 CORE CONFIGURATION CENTERED ON H_1

BR2 CORE CONFIGURATION WITH FUEL ELEMENT TYPE 200 IN H_4

b) The next step will consist of core configuration studies in BRO2 with a nuclear mock up of the helium loop inside a fuel element type 210 Fuel pellets will be made available to constitute a bundle of 12 pins in which detailed power density measurements will be performed The simulation of the device in H4 will then be as correct as possible
The manufacturing of the cadmium screen and of the structural parts for the BRO2 element type 210 has started , the fuel plates are already available The manufacturing of the BRO2 fuel pellets will be included in that of the pellets for the first UO_2 pin bundle to be irradiated in BR2 A decision must be taken in December 1973 about the two uranium enrichments of the fuel pins some pins (probably five) will indeed have an ^{235}U enrichment of 75 or 83 % and the remaining pins will be 93 % enriched It is expected that in this way a satisfactory power flattening will be achieved on the bundle by a proper choice of the enrichment in the different fuel pins, in spite of the macroscopic flux gradient in the H4 channel and of the self-shielding in the bundle

A survey of the activities is given in Table I

TABLE I SURVEY OF THE REACTOR PHYSICS STUDIES IN SUPPORT OF GCFR FUEL ELEMENT IRRADIATIONS IN THE HELIUM LOOP

Type of work	Objectives	
Neutron physics calculations		
Calculation of neutron spectrum and power rating in the loop	Decisions about	
Reactor core calculations	number of pins absorbing screen	1972
	driver fuel element pin enrichments	1973
Measurements in BRO2		
Core configuration studies with a driver fuel element type 213 in a peripheral H channel	Estimate of the flux level in H_4	1973
Measurements in a nuclear mock up of the loop with fuel pins inside a driver fuel element type 210	Power density distribution in the fuel bundle Gamma heating, fast neutron flux	1974
Core configuration studies with the mock up of the loop in the channel H_4	Reactor operation characteristics maximum heat flux, total power	1974 1975

Technical data and present estimate of the nuclear characteristics of the loop

a) Fuel UO_2 PuO_2
Outer pin diameter 8 mm
Pitch of twelve fuel pin bundle 11 1 mm
Pu/(Pu + U) = 15 % Pu_{fiss}/Pu_{tot} 85 %
$^{235}U/U$ = 75 % or 83 % for the highest rated pins
93 % for the other pins, among which the three inner pins of the bundle

b) Maximum linear power (average over the 12 pins) 500 W cm^{-1} (target)
The deviation of the linear power in each pin from the average value is expected not to exceed ≈ 5 % to ≈ 10 % when using two ^{235}U enrichments

Gamma heating contribution to the total (fission + gamma) power density in the fuel pins 7 %

Axial shape factor maximum/average power density over the fuel length (600 mm) 1 2

Cumulative energy distribution of the fissions in the inner pins according to the one-dimensional neutron transport calculations with 40 energy groups

28 % above 100 keV
47 % above 1 keV
82 % above 10 eV

Total neutron flux in the inner pins 6 73 10^{14} $cm^{-2}s^{-1}$

68 % above 100 keV
88 % above 1 keV

Damage production rate in the steel cladding atom displacements in steel WN 1 4981 (inner pins) 3 2 10^{15} $g^{-1}s^{-1}$
The corresponding figure for a typical 1000 MWe gas Cooled Fast Reactor (Siemens reference design) is 1 6 10^{16} $g^{-1}s^{-1}$, a factor 5 higher than in the helium loop