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GAS-COOLED FAST BREEDER REACTOR

QUARTERLY PROGRESS REPORT

FOR THE PERIOD FEBRUARY 1, 1971 THROUGH APRIL 30, 1971

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GA-8107  August 1, 1966 to July 31, 1967
GA-8787  August 1, 1967 to July 31, 1968
GA-8895  August 1, 1968 through October 31, 1968
GA-9229  November 1, 1968 through January 31, 1969
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ABSTRACT

The tasks of the AEC-supported Gas-Cooled Fast Breeder Reactor (GCFR) program are program planning, core engineering design and analysis, fuels and materials surveillance, irradiation tests, and nuclear analysis and reactor physics. The development program plan document for the design and construction of the GCFR demonstration plant is discussed. A review of results of world-wide fast flux irradiation tests of fuel rods is given. The current irradiation test results and analyses are reported for capsule GB-9, which is operating very satisfactorily in ORR and has reached 35,000 MWD/Te. Plans for the irradiation of a similar but improved capsule (GB-10) are discussed and an analysis of the gaseous fission-product release fractions expected from GB-10 is presented. The status of the fast-flux experiment Fl (X094) in EBR-II is also reported. Out-of-pile measurements of interdiffusion of krypton in helium are reported. Nuclear analysis and reactor physics included a review of the LMFBR critical assembly program.
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1. INTRODUCTION

Gulf General Atomic (GGA) is in its eighth year of U.S. Atomic Energy Commission sponsored work on the Gas-Cooled Fast Breeder Reactor (GCFR). The current program effort under USAEC sponsorship consists of Task 1000—Program Planning, Task 4100—Core Engineering Design and Analysis, Task 4200—Fuel and Materials, Task 4400—Irradiation Tests, and Task 4700—Nuclear Analysis and Reactor Physics. The broad objectives of these five tasks and the current efforts on each task during the period covered by this report are summarized in this section. More detailed accounts of the work are given for each task in the subsequent Sections 2 through 6. The publications issued during this reporting period are listed in the Appendix.

Gulf General Atomic is also engaged in the privately sponsored GCFR Utility Program that is centered on a 300-MW(e) demonstration plant as the focus for a GCFR development program. This and the USAEC sponsored work are complementary to each other.

1.1. TASK 1000—PROGRAM PLANNING

Work on this task is directed toward detailing plans for the technical development of the GCFR concept. A development program planning document is being prepared. It covers the developmental items necessary for the design and construction of a 300-MW(e) GCFR demonstration plant and a developmental test program plan. This document, which will be issued in three volumes, is to be submitted to the AEC for review and comments.

1.2. TASK 4100—CORE ENGINEERING DESIGN AND ANALYSIS

The objective of this task is the engineering development of the reactor core and associated components, including the core support structure and the fuel pressure equalization system.
During this reporting period, work on this task has involved converting the CYGRO-3 and the LIFE-1 codes to the 1108 computer for use in studying the GCFR fuel rod design.

1.3. TASK 4200—FUELS AND MATERIALS

GCFR fuel is based on LMFBR fuel development and technology and, therefore, the surveillance of LMFBR fuel and materials irradiation programs is important to provide the base technology for the GCFR. Of particular interest is the LMFBR work on fast flux irradiation-induced metal swelling, and surveillance of this work will continue under this task.

The current work on this program has been a review of the results of fast-flux fuel rod irradiation tests.

1.4. TASK 4400—IRRADIATION TESTS

Irradiation testing in the ORR of the pressure-equalized fuel rod in capsule GB-9 in cooperation with ORNL is continuing and the design of a second vented capsule (GB-10) has been initiated. The current test results on the irradiation testing of capsule GB-9 are reported in Section 5.1. The fuel rod has operated satisfactorily at 16 kW/ft and a maximum cladding temperature of 700°C to a burnup of 35,000 MWD/Te. The burnup goal was extended from 50,000 MWD/Te to 75,000 MWD/Te to obtain fission-gas release measurements at higher fuel burnup. The preliminary design of capsule GB-10 is being reviewed and analyzed.

In the fast-flux irradiation of the F-1 (X094) subassembly in EBR-II, the main objective is to study the behavior of fuel rods over a range of cladding temperatures higher than heretofore carried out in the LMFBR fast-flux tests. Irradiation of this fast-flux seven-rod subassembly was started on March 2, 1971, and received an exposure of ~600 MWD. Further irradiation of the subassembly has been delayed by trouble with other EBR-II experiments, but possibly it will be reinstalled for irradiation during Run 48E.

Fabrication of the hardware for the five replacement rods has been initiated. In addition, to aid in the interpretation of the data obtained from the temperature monitors in the F1 capsules, laboratory tests are being
initiated on $^{85}$Kr release from stainless steel cladding and on irradiated SiC.

To complement the irradiation work, out-of-pile experiments to measure interdiffusion of krypton in helium are being carried out.

1.5. TASK 4700—NUCLEAR ANALYSIS AND REACTOR PHYSICS

This task involves the evaluation of current fast reactor physics methods and data for GCFR application and the planning of a critical experiment program.

The status of the LMFBR critical assembly program was reviewed and areas of particular interest to the GCFR were defined. ZPPR assembly 2 and ZPR6 assembly 7 were selected for analysis.
2. TASK 1000 — PROGRAM PLANNING

The planning of a comprehensive research and development program that would support the design and construction of a GCFR demonstration plant was continued during this quarterly period. This planning effort will result in three volumes in which the development problems will be defined, the development efforts for resolving them will be described, and costs estimated.

The first volume is based on a systematic review of the current design of the 300-MW(e) GCFR demonstration plant to identify and delineate all of the problem items that might require developmental work before final design of a plant can be completed. The scope of the document is limited to the developmental needs of the first GCFR plant; thus it encompasses less than the previous GCFR development plan documents, which were intended to cover everything up to full commercial realization of the GCFR. The present document, on the other hand, is also more detailed and specific because it is focused sharply on the developmental needs of a particular design that has already been described and analyzed in some detail. The scheme for identifying development tasks follows that of the previous development plan document as closely as is consistent with the present needs.

In the first volume, under each development item identified there is an appropriate background discussion to point up the possible problems, a definition of the problem or problems if such exist, and an evaluation of the pertinent technology. Finally, there is a discussion of the approach or approaches to resolution of the problems. The first volume contains approximately 100 development items and a draft is currently being given project review. Preparation of the second volume has commenced. It will contain a description of a program of recommended experiments or analytical efforts necessary to provide answers to any problems or to supply the data needed to support the development. The organization, scope, and format of the second volume has been prepared and approximately 10% of the first draft of test descriptions has been completed.
3. TASK 4100 — CORE ENGINEERING DESIGN AND ANALYSIS

During this quarter the engineering of the core and related components has received considerable attention in connection with identification of specific development problems as necessary for preparing the program planning document of Task 1000. Similarly, the engineering of the pressure equalization system has received attention in conjunction with the analysis of results coming in from the irradiation of capsule GB-9 as well as with the design of capsule GB-10 (see Task 4400).

Work charged specifically to this task was on fuel mechanical performance codes. The CYGRO-3 code was converted for the 1108 computer and a double precision version is now available. This version gives a closer check of a test problem furnished by Westinghouse. Use of the program for comparison with BRITL calculations, however, is hampered by the long running time of CYGRO, with machine time of 5 minutes for only 20 days of real irradiation time. The LIFE-1 code, an Argonne National Laboratory code received from the Argonne Code Center, is now being converted for the 1108. This code, which has been adjusted to reflect results of EBR-II irradiations, may prove more useful than CYGRO for fast flux irradiations. There is hope that a steady power calculation will give results reasonably close to those obtained by a detailed cyclic calculation.
To guide GCFR fuel-rod development, results of several thousand fast-flux fuel-rod irradiation tests have been reviewed. These irradiations have been both single rods and assemblies in the U.S., U.K., and French LMFBR programs, and much of the resulting data are applicable to GCFR. The combination of the goals of fuel burnup (~100,000 MWd/Te) and fluence (~3 x 10^{23} n/cm^2) (see Fig. 4.1) and cladding temperatures (~700°C) for either GCFR or LMFBR have not been achieved simultaneously in any of the tests. Single-rod tests for General Electric in EBR-II have reached ~110,000 MWd/Te and fluences of ~1 x 10^{23} at linear powers in the range 11.5 to 16 kW/ft, but at maximum cladding temperatures of only about 530°C. In fact, nearly all of the LMFBR irradiations have been operated in a narrow range of 500° to 600°C. The results of LMFBR fast-flux fuel-rod irradiations are summarized in Table 4.1.

Table 4.1
STATUS OF FAST-FLUX ROD IRRADIATION TESTING

<table>
<thead>
<tr>
<th>Country</th>
<th>Reactor</th>
<th>Number of Rods Irradiated</th>
<th>Number of Failed Rods Reported</th>
</tr>
</thead>
<tbody>
<tr>
<td>USSR</td>
<td>BR5, BOR60</td>
<td>4,600 (mostly carbide fuel, some oxide)</td>
<td>~150</td>
</tr>
<tr>
<td>UK</td>
<td>DFR</td>
<td>800 (oxide fuel)</td>
<td>50</td>
</tr>
<tr>
<td>France</td>
<td>Rapsodie and DFR</td>
<td>8,000 (oxide fuel)</td>
<td>1</td>
</tr>
<tr>
<td>USA</td>
<td>EBR-II and SEFOR</td>
<td>1,500 (oxide fuel)</td>
<td>10</td>
</tr>
<tr>
<td>Other</td>
<td>Rapsodie and DFR</td>
<td>200 (oxide fuel)</td>
<td>3</td>
</tr>
<tr>
<td></td>
<td></td>
<td>15,100</td>
<td>214 (1.4% of total)</td>
</tr>
</tbody>
</table>

Data were selected for further analysis from ~230 U.S.-U.K. single-rod fast-flux irradiation tests and 12 single-rod thermal-flux tests. The typical
Fig. 4.1 Comparison of burnup vs fluence for fast-flux fuel rod tests
test conditions for the fast-flux tests were a cladding OD/ID of 1.15 and a cladding outer surface temperature of 500° to 600°C (with a maximum of 680°C). Heat ratings ranged to extremes from 9 to 24 kW/ft, but most were in the range from 10 to 16 kW/ft. Figure 4.2 illustrates the relationship between linear heat-generation rating and burnup for successful and failed fuel rod tests in fast flux. Under these conditions in fast flux, fuel-rod swelling rates increase rapidly with fuel burnup >3 at-% and cladding fluences >2 x 10^{22} n/cm^2 (E > 1 MeV). Figure 4.3 illustrates the difference in fuel-rod swelling rates in fast and thermal fluxes and indicates that at high burnups (>10 at-%), ~80% of the fuel-rod swelling in fast-flux tests is attributable to cladding swelling rather than strain due to fuel swelling. Figure 4.4 illustrates cladding strain as a function of neutron exposure in fast-flux fuel rod tests.

In the eight 77-rod bundles (616 rods total) tested in the Dounreay Fast Reactor (DFR), a statistical degree of confidence for survival can be assigned to operation under the combined conditions of burnup <6.2 at-% (58,000 MWD/Te) at linear heat-generation ratings of 9 to 13 kW/ft and maximum cladding OD temperatures ranging from 545° to 665°C. The results of these tests indicate a significantly lower rod swelling rate (approximately one-half) for rods clad with cold-worked 316 stainless steel (SS) as compared to rods clad with annealed 316 SS. It should be noted that one 77-rod bundle was irradiated to 5.1 at-% burnup at maximum cladding OD temperatures up to 690°C at a linear heat-generation rating of 12.5 kW/ft. Unfortunately, as a result of a leak in the coolant outlet pipe from the DFR and entrainment of cover gas in the coolant, the test assembly failed. However, two monitor rods being irradiated individually as controls achieved 8.0% burnup without failure. Several rods irradiated in DFR have currently achieved fuel burn-ups up to ~10 at-% and a cladding exposure of ~1 x 10^{23} n/cm^2, but the details of these results are not as yet available.

Although the details of fuel-rod irradiation behavior in the French Rapsodie reactor have not been made available, sixty 37-rod fuel bundles (2220 rods) have been irradiated to ~6 at-% burnup with no reported failures and only "minor" swelling. The maximum conditions for the Rapsodie fuel rods—316 SS clad, OD/ID of 1.16, (Pu-U)O_2, and smear density 88% of
Fig. 4.2 Linear power vs burnup for successful and failed FBR fast-flux tests
Fig. 4.3 Comparison of fast- and thermal-flux tests for cladding diameter strain vs burnup
Fig. 4.4 Cladding diameter strain vs fluence for successful and failed FBR fast-flux tests
theoretical—are reported to include a peak cladding temperature of 635°C and a maximum linear power rating of 11.5 kW/ft. It has been reported that more than a hundred rods have been irradiated in Rapsodie beyond 6.5 at-% burnup without difficulty (a single failure has been reported), and some of the rods have reached burnups of 9.6 at-%. The Rapsodie results confirm the DFR data in that no failures have been experienced in fuel-rod bundles irradiated in fast flux to ~6 at-% burnup.

In the USSR, several hundred fuel rods irradiated with type 12R72HV austenitic stainless steel cladding have reached over 5 at-% burnup without failure. A high rate of failure between 3.6 to 5.0 at-% burnup (38 of 53 assemblies) was reported for rods clad in 321 stainless steel, however. Most of the USSR fuel-rod irradiations employed carbide rather than oxide fuel so they are not all directly applicable in a discussion of oxide-fueled rods, but the results do emphasize a significant difference in the performance of the two types of cladding materials. The USSR has reported examination of 19 failed oxide fuel rod-bundle assemblies exposed to 6 at-% burnup in which only one or two failed rods were found per assembly. (The number of rods per assembly was not stated in the reference.)
5. TASK 4400 — IRRADIATION TESTS

The GCFR fuel-development program includes thermal-flux and fast-flux irradiation testing of fuel rods. The irradiation test program is a joint effort of Gulf General Atomic and Oak Ridge National Laboratory. Thermal-flux irradiations are being performed in the Oak Ridge Research Reactor (ORR) and fast-flux irradiations are being performed in the Experimental Breeder Reactor (EBR-II). Although this program is focused on fuel-rod development for the GCFR, it also contributes to the Liquid-metal Fast Breeder Reactor (LMFBR) fuel-rod development program because of the commonality of the fuel-rod designs for these two fast breeder reactor concepts. Conditions specific to the GCFR that are being tested include features related to fuel-rod venting, such as the use of activated-charcoal fission-product traps, and surface-roughening of the rod cladding to improve heat transfer to the helium coolant.

5.1. THERMAL-FLUX TESTS

5.1.1. Irradiation Capsule GB-9

Capsule GB-9, the design and objectives of which have been previously reported, (1)* is an instrumented sweep-gas thermal-flux experiment that contains a vented 316 stainless-steel-clad (Pu, U)O₂-fueled rod. The fuel rod has operated very satisfactorily since March 1970 at 16 kW/ft and a maximum cladding temperature of ~700°C to 35,000 MWD/Te toward a burnup goal of 75,000 MWD/Te. One of the main objectives of the test is to evaluate the performance of the charcoal fission-product trap under steady-state and cyclic conditions of power, temperature, and pressure. Fission products released from the fuel are monitored at the inlet and the exit to the trap.

*References are listed at the end of each section.
Fission-gas release values from gas samples obtained under steady-state operation of the fuel rod to 30,000 MWD/Te are plotted in Fig. 5.1. The release values increase initially with time and are approaching steady-state values, which are close to analytical predictions. The release of short-lived fission gases is reduced by decay during diffusion through the fuel and blanket regions of the fuel rod. Their release is thus lower than for the long-lived species. A comparison of the measured fission-gas release values with predicted SLIDER code values at 50,000 MWD/Te (see Fig. 5.1) shows these values to be in reasonably good agreement. The overall effectiveness of the trap is indicated by the fraction of a specific fission-gas nuclide leaving the trap compared to the fraction entering the trap. These ratios, which are expected to vary with the half-life of the nuclide, range from unity for long half-lives (trap ineffective) to $10^{-4}$ for short half-lives (e.g., $^{89}$Kr, $T_{1/2} = 3.18$ min). Results to date are generally in accordance with theoretical predictions.\(^{(2)}\)

In addition to the steady-state fission-gas release values obtained during normal capsule operation, gas samples were taken at various times (after ~25,000 MWD/tonne burnup) following a reactor refueling shutdown and were analyzed for $^{133}$Xe and $^{135}$Xe. These data were then used to determine the amount of iodine present in the charcoal trap and in the purge lines downstream of the trap. Xenon-133 activity, plotted versus time in Fig. 5.2, shows a half-life in agreement with that of $^{133}$I, verifying the presence of iodine in the trap. Analysis indicates that approximately 1.5% of the steady-state iodine generated during the irradiation had diffused up to the trap. The data also indicate that there is a negligible amount of iodine activity in the purge lines downstream of the trap. Calculations have shown that iodine breakthrough in the trap should not occur under normal operating conditions until a burnup of at least 50,000 MWD/Te is reached.

The experimental measurements of the release of $^{135}$Xe from capsule GB-9 have been much smaller than predicted by the SLIDER calculations given in Ref. 1. The SLIDER calculations did not take into account the removal of $^{135}$Xe by neutron capture. Loss of $^{135}$Xe by neutron capture in a highly
Fig. 5.1 Steady-state fission-gas fractional release at the top of the trap in capsule GB-9
Fig. 5.2 Half-life value of $^{133}$Xe determined after reactor shutdown
thermalized neutron flux such as that which occurs in the reflector of the ORR is significant in comparison with loss by decay. The effect of burnup of the Xe$^{135}$ is commonly treated by using an effective half-life. An estimate of the effective half-life has been made taking into account the suppression of the flux in the fuel rod. The half-life of Xe$^{135}$ is 9.13 hr and the effective half-life of Xe$^{135}$ in capsule GB-9 has been estimated to be 5.76 hr. This effect has now been taken into account and the computed Xe$^{135}$ release fraction from the charcoal trap has thus been reduced from 10.5% to 7.2%. The measured released fraction in capsule GB-9 is ~1.8% after 30,000 MWD/Te, which still does not agree with prediction.

One of the major uncertainties in the design of the GCFR demonstration plant is the possibility of fission-product deposition in the pressure equalization system lines and the traps of the fuel elements. The fuel rod in capsule GB-9 provides a simulation of some aspects of the GCFR system which might provide information relative to this uncertainty. Thus, consideration was given to a flow test of the GB-9 rod trap and sample line to measure the constriction of the passages in these components that might result from plateout or deposition of fission products as a function of increasing burn-up.

Calculations show that a flow test is quite insensitive to constriction in the line and in the trap until very substantial amounts of deposition have occurred. It is expected that such an occurrence would be apparent from the normal sampling and depressurization procedures and hence flow testing of the trap and sample line in GB-9 is not warranted.

**Mass Spectrometer Measurements for Capsule GB-9**

Measurements of the fission gases evolved from capsule GB-9 to date have included only those that are radioactive. Mass spectrometry to also measure stable gases during future operation of the capsule is planned.

A stable fission-product gas from the fuel oxide matrix will be released from the fuel rod to the sweep gas and analytical system at the same rate as it is released from the fuel after a time sufficient to accumulate equilibrium concentrations along the release path the gas has passed.
This time in capsule GB-9 is ~3 days. Stated otherwise, the release fraction for stable isotopes is independent of the transport time, provided that the transport time is short relative to the life of the experiment or, as a practical matter, the time between ORR shutdowns. Thus, the release fractions measured can be regarded as the fractions of the stable isotopes released from the oxide fuel matrix. From the stable-gas release fractions, it may be possible to estimate an average migration time for each isotope through the fuel matrix for the prevailing fuel-rod conditions.

By plotting the ratio of the release fractions of the radioactive fission products to those of the stable fission products versus the decay constant, the delay time in the blanket and trap can be independently determined. This technique has been successfully used by others in the analysis of a fuel-rod irradiation experiment in which venting of gases was permitted through a "diving bell."(3) Thus far, information on the transport time has only been inferred by how well the measured release fractions agree with those predicted, assuming only gaseous diffusion through the blanket region to be important.

Thus the planned mass spectrometer measurements can yield important further information not available from the radioanalysis of GB-9 gases alone.

**Decay Heating in Charcoal Trap of Capsule GB-9**

Measurement of temperatures in and around the charcoal trap in capsule GB-9 were taken following a shutdown of the ORR. The capsule was withdrawn to the "out" position in the reflector region of the pool. The electrical heaters used to maintain the trap at 300°C during normal operation were shut off so that only decay heat was available to maintain the temperature in the trap above the pool water temperature. The temperature difference between the center of the trap bed at its bottom end and the pool water was ~13°C at reactor shutdown. Two and three-quarters hours later the temperature difference had decreased to 5.5°C. Because of the exponential decay relationship it was expected that fission products at the bottom of the trap (the inlet for fission products coming from the fuel below) would
generate most of the decay heat. This was confirmed by the temperature distribution along the length of the trap where, within the first inch from the bottom of the trap, the temperature fell essentially to the pool temperature.

In order to estimate the heat generation in the trap it was necessary to make simplifying assumptions. It was thus assumed that all of the temperature difference, \( \Delta T \), measured between the charcoal-bed center and the pool water occurs in the bed material because the thermal conductivity of charcoal is 2 orders of magnitude less than the conductivities of the materials (stainless steel, NaK, and Zircaloy) between the bed and the pool water and because the area of the concentric annuli through which heat flows radially is much greater than the area in the charcoal bed itself. It was also assumed that axial heat conduction in the charcoal was negligible and thus only radial heat flow was important. It was further assumed that the heat generation was homogeneously distributed over the first one-half inch of the trap. Using these assumptions, the decay-heat generation rate was determined as follows:

\[
q = \frac{4kV(\Delta T)}{r^2}
\]

where
- \( q \) = decay heating (W),
- \( k \) = thermal conductivity of charcoal (W/cm·°C),
- \( V \) = volume of 1/2 in. length of rod (cm\(^3\)),
- \( r \) = radius of charcoal bed (cm),
- \( \Delta T \) = temperature difference from center to surface of charcoal bed (°C).

Thus, the decay-heat generation is

\[
q = 0.35 \text{ W at shutdown},
\]

\[
q = 0.15 \text{ W 2-3/4 hr after shutdown.}
\]

From separate tests using the electrical heaters but no fission heat, a trap temperature of 200°C was attained when the pool temperature was 49°C and 109 watts of electrical power was being generated in the heaters. Since
the heaters are adjacent to the cladding in the trap region, the ΔT results from heat flowing outward from the heaters. Thus, the resistance of the heat path from the heater to the pool water was determined from the above data. The ΔT from decay heat generated in the charcoal trap following shutdown as computed above was then found to result in a ΔT of only 0.48°C or 3.7% of the 13°C ΔT observed at shutdown, which verifies the assumption that nearly all of the ΔT occurs across the charcoal trap bed material.

The decay heat generated in the trap of capsule GB-9 was also estimated from the measurement of iodine reaching the trap in the following way. From measurements of Xe$^{133}$ and Xe$^{135}$ as described above, it has been deduced that ~1.5% of the iodine generated in the fuel rod during normal operation reaches the trap. Furthermore, it is true in general that total decay heat from all the volatile fission products is equal to less than 1% of the fission heat and iodine decay accounts for one-fourth of this. Thus, for capsule GB-9,

- Total fission heat generated = 13,200 W
- Decay heat generated from volatile fission products (1% of fission heat) = 132 W
- Volatile-fission-product trap heating if volatiles all behave like iodine (1.5% of 132 W) = 2 W
- If only iodine reaches the trap (iodine produces 1/4 of the decay heat generated by volatile fission products) = 0.5 W

The last two numbers agree satisfactorily with the initial decay-heat generation of 0.35 W estimated above from direct measurement of trap temperature rise.

5.1.2. Irradiation Capsule GB-10

With the experience gained in the design and operation of the pressure-equalized fuel rod in capsule GB-9, plans for the irradiation of capsule GB-10 in the ORR were started. The primary objective with capsule GB-10 is to obtain important information on the release, transport, and trapping of the volatile fission products. The specific objectives of the GB-10 test are to

1. Observe the overall behavior of a vented fuel rod with roughened cladding up to high burnup (75,000 MWD/Te) with imposed power
and temperature cycling and compare its behavior with that of the vented rod irradiated in capsule GB-9 and with the sealed-can fuel rods irradiated in capsule P-8. Measure the cladding strain of the vented fuel rod with roughened cladding in GB-10 and compare it with the strain of the vented fuel rod with smooth cladding irradiated in capsule GB-9 to determine if the mechanical strength of the cladding is affected by the roughening. The GB-10 experiment will be conducted to high burnup with a neutron exposure equivalent to that for capsule GB-9; the operating conditions (i.e., cladding temperature, kW/ft, and pressure) will be the same as for GB-9.

2. Measure the adsorption characteristics (in the presence of thermal aging and irradiation exposure) of the fission-product trap for the GCFR vented fuel rod by analysis of fission-product content and distribution as a function of temperature and fluence and thus provide additional trap design data. Analyze the release fractions of krypton and xenon by measuring the fission gases at the fuel exit and the trap exit under steady-state and transient temperature and pressure conditions and compare the results with calculated release values to determine the efficiency of the trap for those fission products. Measure iodine and bromine at shutdown. During postirradiation examination, determine the distribution of fission products, such as cesium and tellurium, deposited within the trap and the connecting passages that are typical of the GCFR. Capsule GB-10 will have lines to the bottom of the fuel rod, to the bottom of the top blanket region, to the bottom of the trap, and to the top of the trap. These lines will have valving to provide a number of operating modes. More detailed and interpretable data on fission-gas release can be obtained during startup and shutdown (i.e., whether these conditions lead to bursts in fission gas release). Also, much needed information on volatile-fission-product release will be obtainable using purge gas that does not pass through the rod trap. Purging through the bottom of the rod will simulate operation with failed cladding and will provide information on trap performance.
3. Study fission-product condensation effects in the cooler (blanket) regions of the rod during postirradiation examination.

4. Investigate the possibility of fission products creeping against the sweep gas flow by postirradiation examination.

5. Measure the effect of trap temperature on fission-product sorption over a larger range of temperatures than is possible in capsule GB-9.

6. Employ solid fuel pellets instead of annular fuel pellets to determine whether fission-product release and fuel-rod swelling are different for solid pellets.

7. Perform mass spectrometer analyses to determine the release fraction of stable and long-lived isotopes of krypton and xenon.

8. By analysis of the results of postirradiation examination, compare the performance of 316 stainless steel cladding and (U,Pu)O₂ fuel in a vented rod with the Hastelloy-X cladding and 316 SS cladding used in earlier sealed rods, particularly with respect to the extent and type of fuel-cladding chemical interactions.

9. Determine the change in ductility (embrittlement) of 316 stainless steel at an irradiation temperature of 700°C by performing creep burst tests at 300°C to 750°C and also look for other effects or comparative changes in strength, dimensions, volume, metallurgical structure (aging), and corrosion as a result of fuel-fission-product-cladding reactions. The changes will be compared with control samples heated out of pile at 700°C in NaK for an equivalent length of time.

10. Observe the amount of fuel swelling accommodated by porosity or by redistribution of other voidage using an initial fuel smear density of ~85%.

11. Interpret the fuel microstructure for indications of the temperature profile. Determine the distribution of Pu and solid fission products in the fuel and the distribution of fission products in the blanket.

12. By direct observation of fuel rod GA-21 in capsule GB-10 and by comparison with observations of specimens irradiated previously,
determine the effect of venting and any burnup effects, such as microstructural changes, stoichiometry changes, fuel-cladding compatibility, and fission-product migration.

13. Compare the performance of sol-gel-derived mixed-oxide fuel or fuel blended as a sol with the performance of mechanically blended mixed-oxide fuel, which was used in capsules P-6, P-7, and P-8. Some gross anomalies in the behavior of the mechanically blended fuel have been observed but are not expected to occur in the homogeneous sol-gel-derived fuel.

To determine the effectiveness of the charcoal trap in adsorbing or chemisorbing volatile fission products, it is necessary to obtain information on the quantity and type of fission products released from the fuel or from the blanket region that actually reached the charcoal trap. Capsule GB-10 will be designed to permit sampling the effluent from each of the regions of the fuel rod. A proposed design of the capsule GB-10 vented fuel rod is shown in Fig. 5.3.

Figure 5.4 shows schematically the termini of the sampling lines at the various axial locations in the capsule. The permissible flow direction in the sample line is prescribed to accomplish the purpose of the line and to enhance the probability of obtaining useful information from operation of the capsule or to ensure that reliable data will be obtained during post-irradiation examination. Starting from the bottom of the capsule,

1. Sampling line 1 terminates at the lower fuel-rod cladding end cap and is valved to permit inflow only. This sample line is used to sweep gas over the fuel itself in order that the release fraction for gases and volatile fission products from the fuel may be studied. Additionally, during postirradiation examination, this line will be sectioned axially and analyzed for plateout of fission products that may have been released from the fuel and back-diffused into the inlet line.

2. The second sampling line enters the capsule at the top and passes down through the annular blanket pellets and terminates at the top of the fuel column. The purpose of this line is to sample directly
Fig. 5.3 Preliminary design of capsule GB-10
Fig. 5.4 He sweep- and monitoring-line arrangements for capsule GB-10
the release from the top of the fuel column. This line can also be used as an inlet line to sweep deposited fission products from the blanket into an exit sampling line. Since this line has a dual purpose, it is valved for either inflow or outflow from the upper fuel-blanket interface.

3. The third sampling line terminates at the interface between the uppermost blanket pellet and the charcoal trap. The purpose of this line is to permit the study of effluents from the fuel and blanket regions or to serve as an inlet line to permit sweeping out fission products deposited in the charcoal trap. Since this is the influent line for the charcoal trap, it is also valved for two-way flow.

4. Sampling line 4 terminates at the top of the charcoal trap and collects volatile and gaseous fission-product effluents from the charcoal trap. Under normal steady-state operating conditions, this is the exit line from the fuel capsule column itself. This line simulates the fission-product escape from a GCFR fuel rod under diffusion conditions. As this line is the normal exit for the fission products and the line along which steady-state effluent diffusion profiles from the trap may develop, this line is utilized only for outflow and is valved accordingly.

Calculated Release Fractions from Irradiation Capsule GB-10

To ensure that activity levels in the sampling system will not be excessive, release fractions for gaseous fission products expected to effuse from the vent line of capsule GB-10 during steady normal operation have been computed to verify that the activity release from capsule GB-10 will not exceed that released from capsule GB-9. The release fractions were computed using the SLIDER computer code and the results for both capsules GB-10 and GB-9 are tabulated in Table 5.1. The release fractions tabulated for capsule GB-9 are slightly different from those given in Ref. 1 because of slight changes in the diffusion coefficients, depending on the temperatures assumed in the various regions of the rod. The results indicate that the release fractions from the vent line of GB-10 are less than those from
Table 5.1
CAPSULE GB-10 RELEASE FRACTIONS
(With comparative GB-9 release fractions)

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Fuel</th>
<th>Blanket</th>
<th>Void</th>
<th>Trap</th>
<th>Connector</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Half-life</td>
<td>GB-10</td>
<td>GB-9</td>
<td>GB-10</td>
<td>GB-9</td>
</tr>
<tr>
<td>Kr$^{3+}$</td>
<td>1.87h</td>
<td>9.73</td>
<td>9.60</td>
<td>1.69</td>
<td>2.26</td>
</tr>
<tr>
<td>Kr$^{5+}$</td>
<td>4.4h</td>
<td>14.84</td>
<td>14.98</td>
<td>5.76</td>
<td>7.06</td>
</tr>
<tr>
<td>Kr$^{5}$</td>
<td>10.3y</td>
<td>92.11</td>
<td>93.04</td>
<td>93.70</td>
<td>94.62</td>
</tr>
<tr>
<td>Kr$^{7}$</td>
<td>1.3h</td>
<td>8.22</td>
<td>8.04</td>
<td>0.89</td>
<td>1.27</td>
</tr>
<tr>
<td>Kr$^{8}$</td>
<td>2.8h</td>
<td>11.81</td>
<td>11.74</td>
<td>3.14</td>
<td>4.01</td>
</tr>
<tr>
<td>Kr$^{10}$</td>
<td>3.2m</td>
<td>2.30</td>
<td>2.23</td>
<td>10$^{-5}$</td>
<td>10$^{-4}$</td>
</tr>
<tr>
<td>Kr$^{10}$</td>
<td>33.0s</td>
<td>1.67</td>
<td>1.62</td>
<td>10$^{-10}$</td>
<td>10$^{-9}$</td>
</tr>
<tr>
<td>Kr$^{91}$</td>
<td>10.0s</td>
<td>1.53</td>
<td>1.48</td>
<td>10$^{-14}$</td>
<td>10$^{-12}$</td>
</tr>
<tr>
<td>Xe$^{131m}$</td>
<td>12.0d</td>
<td>82.28</td>
<td>83.67</td>
<td>81.75</td>
<td>83.59</td>
</tr>
<tr>
<td>Xe$^{133m}$</td>
<td>2.3d</td>
<td>52.00</td>
<td>54.38</td>
<td>46.94</td>
<td>50.48</td>
</tr>
<tr>
<td>Xe$^{133}$</td>
<td>5.27d</td>
<td>68.16</td>
<td>70.75</td>
<td>65.72</td>
<td>69.13</td>
</tr>
<tr>
<td>Xe$^{135}$</td>
<td>15.3m</td>
<td>3.76</td>
<td>3.65</td>
<td>10$^{-2}$</td>
<td>10$^{-2}$</td>
</tr>
<tr>
<td>Xe$^{135}$</td>
<td>15.6h$^{b}$</td>
<td>15.72</td>
<td>15.92</td>
<td>6.58</td>
<td>7.99</td>
</tr>
<tr>
<td>Xe$^{137}$</td>
<td>3.9m</td>
<td>2.32</td>
<td>2.25</td>
<td>10$^{-5}$</td>
<td>10$^{-4}$</td>
</tr>
<tr>
<td>Xe$^{138}$</td>
<td>17.0m</td>
<td>3.92</td>
<td>3.81</td>
<td>10$^{-2}$</td>
<td>10$^{-2}$</td>
</tr>
<tr>
<td>Xe$^{139}$</td>
<td>41.0s</td>
<td>1.67</td>
<td>1.62</td>
<td>10$^{-10}$</td>
<td>10$^{-9}$</td>
</tr>
<tr>
<td>Xe$^{140}$</td>
<td>16.0s</td>
<td>1.56</td>
<td>1.51</td>
<td>10$^{-13}$</td>
<td>10$^{-12}$</td>
</tr>
</tbody>
</table>

$^{a}$ Length of tube from Y to top of trap present in GB-10 but not in GB-9.

$^{b}$ An effective half-life of 5.76 hr was used for the calculations of GB-10.
the trap exit of GB-9 except for the isotopes with very short half-lives. However, the release fractions for the very short half-lived isotopes are so small that their contribution to the total activity released is insignificant.

However, from preliminary calculations of the release fractions of gaseous fission products from the fueled region of GB-10, it was noted that these exceeded the release fractions from the fueled region of GB-9. In order to understand this difference, a study was made in which the length of the fuel rod was kept constant and the boundary between the fuel and blanket regions was varied along the length of the rod from short fuel and long blanket to long fuel and short blanket. The results are shown in Fig. 5.5. The figure shows that the overall release fraction decreases with increasing blanket length and decreasing fuel length. It also shows that the release from the fuel region passes through a minimum while the trapping effectiveness of the blanket region continually increases with increasing blanket length. The GB-9 and GB-10 design points shown on the curve indicate that the release from capsule GB-10 should be less than that from GB-9, as indicated by the more detailed calculations. However, the figure also indicates that the release fraction for GB-10 compared to that of GB-9 from the fueled region should also be less, contrary to that calculated in the more detailed calculations. The explanation for the difference arises from the fact that the parametric study was conducted with a constant trap configuration while for irradiation capsule GB-9 the trap in the analytical model is six times the size of the trap in capsule GB-10. The study shows that the blanket and trap are important in establishing the release to be expected from the fueled region of the rod. That is, the resistance of the blanket and trap regions limits the current through the rod and thereby the release fraction, which is the time integral of this current divided by the source integral over the same time. This explains why when the diffusion coefficient in the fueled region was changed by 3 or more orders of magnitude, the release fraction from the fuel region changed by ~2 percentage points. These results cast some doubt upon the assumption made in the original calculations that a large diffusion coefficient in the fuel region was equivalent to transport by convection in that
Fig. 5.5 Effect of complementary fuel and blanket lengths on gaseous fission-product release fractions
region. On the other hand comparison of calculated values with the experi-
mental results (Fig. 5.1) suggests that diffusion rather than convection
may indeed be the proper model as originally assumed.

The effect on the release fractions of the sweep and sample lines to be
included in capsule GB-10 is being evaluated (see Figs. 5.3 and 5.4). The
effect of these lines may be much more important in capsule GB-10 than they
were in capsule GB-9 because of the larger number of connection points and the
regions which have been penetrated. In GB-9 the sweep and sample lines
were all connected outside of the blanket region, which is the major deteren
to the release of fission products from the capsule. In GB-10, one line
will penetrate through the blanket region and another will be connected to
the bottom of the fuel region. Thus, these lines, which are ~80 ft long,
might appear as infinite volumes if their resistances are not sufficiently
high to diffusive flow to prevent their utilization over an effective time
period.

5.2. FAST-FLUX TESTS

The Fl (X094) fast-flux experiment for studying fuel-rod behavior under
a broader and higher range of cladding temperatures than heretofore carried
out in the LMFBR fast-flux test program is described in Refs. 4 and 5. The
design of the fuel rods, which are (Pu,U)O₂ fuel clad with 20% cold-worked
316 SS made to the specifications for the Fast Test Reactor cladding, has
also been previously reported in Refs. 4 and 5.

The seven encapsulated fuel rods in the Fl (X094) experiment have
received an exposure of ~600 MWd since EBR-II startup on March 2, 1971.
Further irradiation has been delayed because the F-1 subassembly and several
other subassemblies were removed from the reactor when gaseous fission-
product release indicated leakage in one of the subassemblies. Although fis-
sion-product leakage still was detected in the EBR-II after the various
subassemblies had been removed, the Fl (X094) subassembly was not reinserted
for continued irradiation during Run 48B. The reasons for this have been
stated to be that the experiments selected for reinsertion provide the
greatest recovery of reactivity and have higher priority. It is possible,
but not certain, that the Fl subassembly will be reinstalled for Run 48E.
Fabrication of the hardware for the five replacement fuel-rod capsules (which are to be inserted after the Fl experiment has achieved 50,000 MWD/Te exposure) has been initiated.

Laboratory tests are being initiated for the purpose of calibrating and gaining experience with the passive temperature monitors that are included in Fl (X094). Instrumented experiments in the TRIGA King Furnace are in preparation to measure the $^{85}$Kr release from stainless steel cladding impregnated with $^{85}$Kr by fission recoil (under measured temperature conditions) to determine the effectiveness of this method of monitoring temperatures of fuel-rod cladding in the Fl (X094) capsules. Also, samples of silicon carbide irradiated in a thermodispersed ETR experiment have been received and are being prepared for examination. Dimensional analysis and temperature calibration will be made, the results of which will aid in the interpretation of the data to be obtained from the SiC temperature monitors in the Fl (X094) experiment capsules.

Samples of surface-roughened cladding of the type to be used for two of the Fl (X094) replacement fuel rods will undergo creep burst tests. The strength of surface-roughened versus smooth cladding will be compared as soon as the instrumentation for these tests is received.

5.3. OUT-OF-REACTOR TESTS

Eleven interdiffusion coefficients measurements have been completed at pressures ranging from 15 psi to 1000 psi and at krypton concentrations of approximately $10^{19}$ and $10^{20}$ mol/cm$^3$. The experimental data were analyzed using the one-dimensional gaseous diffusion code SLIDER. The initial purpose of the experiments was to determine if the inverse pressure dependency of the diffusion coefficient is valid. Previous experiments had indicated an anomaly from this dependence (see Appendix E of Ref. 1). In the later experiments it was necessary to go to much higher krypton source concentrations in helium because of the experimental apparatus and techniques used. To eliminate leakage of helium that had previously occurred, an all-welded apparatus was used. Also, NaI scintillation counting equipment was required for detection of the gamma rays from the $^{85}$Kr decay rather than
the beta decay detectors previously used. Since gamma emission occurs only in about 1% of the disintegrations, higher concentrations were necessary to attain good counting statistics.

The results from six of the later experiments are listed in Table 5.2. Within experimental error, these results show the inverse pressure dependence of the diffusion coefficient to be valid, as shown in Fig. 5.6. Also, the experimental deviation of the measured values about their mean was significantly reduced from earlier measurements, which, for comparison, are given in Table 5.3.

To conduct experiments at a pressure of 1 atm it was necessary to reduce the concentration of krypton by a factor of about 10. It should be noted that this concentration is still about $10^5$ times that used in the earlier experiments, but it satisfied the requirements for good counting. The 1-atm tests, however, resulted in measurement of a much higher diffusion coefficient than that reported in the literature and higher than that obtained at the higher pressure measurements corrected to 1 atm by assuming the inverse pressure relationship. These results raised the question of concentration dependence of the diffusion coefficient. The measurements at the $10^{19}$ mol/cm$^3$ concentration were thus extended to the corresponding higher pressure levels. The results at the $10^{19}$ mol/cm$^3$ concentration level are shown in Table 5.4. Again, it is noted that the deviation of the individual measurements about the mean of the measurements is small compared with those taken in the earlier apparatus (Table 5.3) and agrees well with those taken with the present apparatus at the higher concentration levels (Table 5.2).

Several conclusions may be drawn from the results shown in Tables 5.2 through 5.4. The next to the last column is the diffusion coefficient that one obtains when correcting the measured coefficient to 1 atm in accordance with the inverse pressure relationship. The last column shows the deviation of the reduced diffusion coefficient from the mean of the measured values in each individual table. It is noted that the average value of the measurements made at the high ($10^{20}$) and low ($10^{14}$) krypton concentrations agree, but that the average value at the intermediate concentration ($10^{19}$) is greater by 11%. In addition, the average value for the intermediate
### Table 5.2

**KRYPTON-HELIUM INTERDIFFUSION COEFFICIENTS WITH SOURCE CONCENTRATIONS OF \( \sim 10^{20} \)**

<table>
<thead>
<tr>
<th>Run No.</th>
<th>Pressure (atm(psi))</th>
<th>Source Concentration (Kr mol/cm³)</th>
<th>Theoretically Calculated Diffusion Coefficient (cm²/sec)</th>
<th>Measured Diffusion Coefficient (cm²/sec)</th>
<th>1.0 Atmosphere Reduced Diffusion Coefficient (cm²/sec)</th>
<th>Deviation (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>53</td>
<td>10.88(160)</td>
<td>( 1.865 \times 10^{20} )</td>
<td>0.0592</td>
<td>0.0634</td>
<td>0.690</td>
<td>-1.97</td>
</tr>
<tr>
<td>52</td>
<td>11.22(165)</td>
<td>( 1.865 \times 10^{20} )</td>
<td>0.0574</td>
<td>0.0621</td>
<td>0.697</td>
<td>-0.94</td>
</tr>
<tr>
<td>49</td>
<td>34.69(510)</td>
<td>( 1.865 \times 10^{20} )</td>
<td>0.01855</td>
<td>0.0203</td>
<td>0.703</td>
<td>-0.05</td>
</tr>
<tr>
<td>50</td>
<td>34.69(510)</td>
<td>( 1.865 \times 10^{20} )</td>
<td>0.01855</td>
<td>0.0203</td>
<td>0.703</td>
<td>-0.05</td>
</tr>
<tr>
<td>51</td>
<td>67.35(990)</td>
<td>( 1.865 \times 10^{20} )</td>
<td>0.00956</td>
<td>0.0109</td>
<td>0.731</td>
<td>+3.75</td>
</tr>
<tr>
<td>47</td>
<td>67.69(995)</td>
<td>( 1.865 \times 10^{20} )</td>
<td>0.00951</td>
<td>0.0103</td>
<td>0.697</td>
<td>-0.94</td>
</tr>
</tbody>
</table>

### Table 5.3

**KRYPTON-HELIUM INTERDIFFUSION COEFFICIENTS WITH SOURCE CONCENTRATIONS OF \( \sim 10^{14} \)**

<table>
<thead>
<tr>
<th>Run No.</th>
<th>Pressure (atm(psi))</th>
<th>Source Concentration (Kr mol/cm³)</th>
<th>Measured Diffusion Coefficient (cm²/sec)</th>
<th>1.0 Atmosphere Reduced Diffusion Coefficient (cm²/sec)</th>
<th>Deviation (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>1.0(14.7)</td>
<td>( 5.22 \times 10^{14} )</td>
<td>0.665</td>
<td>0.665</td>
<td>-5.81</td>
</tr>
<tr>
<td>13</td>
<td>1.0(14.7)</td>
<td>( 5.22 \times 10^{14} )</td>
<td>0.665</td>
<td>0.665</td>
<td>-5.81</td>
</tr>
<tr>
<td>3</td>
<td>10.5(154)</td>
<td>( 5.22 \times 10^{14} )</td>
<td>0.0549</td>
<td>0.577</td>
<td>-18.35</td>
</tr>
<tr>
<td>17</td>
<td>11.0(162)</td>
<td>( 5.22 \times 10^{14} )</td>
<td>0.0652</td>
<td>0.717</td>
<td>+1.58</td>
</tr>
<tr>
<td>16</td>
<td>34.7(510)</td>
<td>( 5.22 \times 10^{14} )</td>
<td>0.0216</td>
<td>0.750</td>
<td>+6.15</td>
</tr>
<tr>
<td>14</td>
<td>66.0(970)</td>
<td>( 5.22 \times 10^{14} )</td>
<td>0.0116</td>
<td>0.766</td>
<td>+8.48</td>
</tr>
<tr>
<td>15</td>
<td>68.1(1001)</td>
<td>( 5.22 \times 10^{14} )</td>
<td>0.0118</td>
<td>0.804</td>
<td>+13.82</td>
</tr>
</tbody>
</table>

average max. range

average max. range

max. range

33
Fig. 5.6 Effect of pressure on diffusion coefficient
$$\begin{array}{|c|c|c|c|c|}
<table>
<thead>
<tr>
<th>Run No.</th>
<th>Pressure (atm/psi)</th>
<th>Source Concentration (Kr mol/cm^3)</th>
<th>Theoretically Calculated Diffusion Coefficient (cm^2/sec)</th>
<th>Measured Diffusion Coefficient (cm^2/sec)</th>
<th>1.0 Atmosphere Reduced Diffusion Coefficient (cm^2/sec)</th>
<th>Deviation (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>55</td>
<td>1.0(14.7)</td>
<td>$2.5 \times 10^{19}$</td>
<td>0.6437</td>
<td>0.800</td>
<td>0.800</td>
<td>+2.46</td>
</tr>
<tr>
<td>56</td>
<td>1.0(14.7)</td>
<td>$2.5 \times 10^{19}$</td>
<td>0.6437</td>
<td>0.800</td>
<td>0.800</td>
<td>+2.46</td>
</tr>
<tr>
<td>58</td>
<td>11.22(165)</td>
<td>$2.5 \times 10^{19}$</td>
<td>0.0573</td>
<td>0.0699</td>
<td>0.784</td>
<td>-0.08</td>
</tr>
<tr>
<td>59</td>
<td>34.69(510)</td>
<td>$2.5 \times 10^{19}$</td>
<td>0.0186</td>
<td>0.0220</td>
<td>0.764</td>
<td>-2.15</td>
</tr>
<tr>
<td>57</td>
<td>68.02(1000)</td>
<td>$2.5 \times 10^{19}$</td>
<td>0.00946</td>
<td>0.0111</td>
<td>0.756</td>
<td>-3.18</td>
</tr>
</tbody>
</table>

The concentration is ~20% greater than the theoretical value at 1 atm. Since the deviations from the mean values in each table is about ±3%, the 11% difference between the numbers indicates a real difference in the measured values. Theoretical considerations indicate that the concentration dependence of the diffusion coefficient cannot account for more than 3% variation between the measurements. Thus, the results seem to indicate that there are still experimental difficulties present and that further improvement will be required before diffusion coefficients can be measured with sufficient accuracy. The presently suspected principal source of error is that thermal convection loops occur as a result of the diurnal temperature cycle in the laboratory. Modifications are being made which include mounting the diffusion tube in a copper block in the vertical position and imposing a small temperature gradient along the tube from the top to the bottom.
REFERENCES


6. TASK 4700 — NUCLEAR ANALYSIS AND REACTOR PHYSICS

Work during this period was concentrated on reviewing the status of the LMFBR critical assembly program and defining areas of special interest to GCFR physics. ZPPR assembly 2 and ZPR6 assembly 7 have been selected for particular analysis. Aside from the fact that both assemblies contain large proportions of sodium, their compositions are quite typical of those for GCFR designs. Additionally, experiments are planned for the near future in each of these assemblies in which a large portion of the central region will be voided of sodium. Calculations have been performed which indicate that the central spectrum will then be close to that anticipated for GCFR. Hence, measurements made in these regions, such as rod worths, material worths, and reaction-rate ratios will be of direct use to the GCFR program in that it will then be possible to determine the current status of ability to predict these quantities in a GCFR situation. It is planned to precalculate these experiments.
Appendix

PUBLICATIONS
