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AEC RESEARCH AND
DEVELOPMENT REPORT

GA-10517

GAS-COOLED FAST BREEDER REACTOR

PROGRESS REPORT

FOR THE PERIOD FEBRUARY 1, 1970 THROUGH JANUARY 31, 1971

by

Project Staff

Prepared for the
U.S. Atomic Energy Commission
San Francisco Operations Office
under
Contract AT(04-3)-167
Project Agreement No. 23

Submitted to AEC - February 26, 1971

GULF GENERAL ATOMIC COMPANY
P.O. BOX 608, SAN DIEGO, CALIFORNIA 92112

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ABSTRACT

The tasks of the AEC-supported Gas-Cooled Fast Breeder Reactor program are program planning, core engineering design and analysis, fuels and materials surveillance, irradiation tests, and nuclear analysis and reactor physics. A review of in-reactor creep data in austenitic stainless steels is given. Progress to date is reported on the irradiation test results and analyses of the instrumented, (U-Pu) O_2 -fueled, pressure-equalized GCFR fuel rod being irradiated in capsule GB-9 in the ORR, which is a cooperative program with ORNL. In the fast-flux test program, a seven-rod subassembly (F-1) has been inserted in the EBR-II for start of irradiation in the near future. Materials surveillance work reported is a review of data on in-reactor creep in austenitic stainless steels. A review of recent developments in cross-section evaluations for Pu^{239} , Pu^{240} , and U^{238} is summarized.

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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 below. 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 core engineering test program, the safety program, and the experimental physics program and will involve updating the core development and reactor system development plans previously prepared.

The general development tasks defined in the GCFR Development Plan are being reviewed, updated, and supplemented by specific items that define the development requirements that may necessitate experimental or analytical work.

1.2. TASK 4100—CORE ENGINEERING DESIGN AND ANALYSIS

Work on this task is directed toward the technical development of the reactor core, with current emphasis on the interaction between the fuel pressure equalization system design and the core support structure.

During this reporting period, work on this task has involved development of the computer program BRITL, which analyzes strains in the fuel cladding during a constant power level irradiation. Other work has involved converting the CYGRO-2 and CYGRO-3 programs to the 1108 computer.

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 metal swelling, and surveillance of this work will continue under this task.

The current work on this program has been a review of the data available on in-reactor creep in austenitic stainless steels. This review is presented in Section 4.

1.4. TASK 4400—IRRADIATION TESTS

Irradiation testing in the ORR of the pressure-equalized fuel rod in capsule GB-9 (formerly designated 04-P9) in cooperation with ORNL will continue and the design of a second vented capsule (GB-10) will be initiated. 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 wide range of temperatures and at a higher range of cladding temperatures than heretofore carried out in the LMFBR fast-flux tests.

The current test results on the irradiation testing of capsule GB-9 are reported in Section 5.1. A neutron radiograph taken of the capsule after a burnup of ~20,000 MWd/Te indicated that the fuel appeared to be as it was in its preirradiated condition and there was no evidence of fission-product deposition. The burnup goal was extended from 50,000 MWd/Te to 100,000 MWd/Te because of the need for fission-gas release measurements at higher fuel burnup; a factor that outweighed the desire for early postirradiation examination results.

The fast-flux seven-rod subassembly F-1 was installed in EBR-II for run 47B in November, 1970, but start of irradiation was delayed because of maintenance operations in the EBR-II. A brief description of the test rods and

test objectives are presented in Section 5.2. The test conditions and objectives for the five replacement rods are also discussed.

The out-of-reactor experimental work has been on interdiffusion of gases. The results of this work are discussed in Section 5.3.

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.

A review of recent developments in cross-section evaluations for Pu^{239} , Pu^{240} , and U^{238} has been completed. Conclusions from this review have been included in a reactor physics report that is to be published. A summary of this work is given in Section 6.

2. TASK 1000-PROGRAM PLANNING

The general development tasks defined in the GCFR Development Plan (GA-8009) are being reviewed, revised, and supplemented by specific items that define the development requirements that may necessitate experimental or analytical work to demonstrate feasibility or to permit confident completion of a final design. A preliminary list of about 85 development items for the GCFR program has been prepared, and drafts on approximately 50% of these have been completed. Each item will be discussed in four parts:

1. Problems (i.e., identification of the development problems),
2. Technology status (i.e., what is known, not known, and what will be expected from related programs),
3. Planned approach (i.e., development philosophy, alternate approaches, etc.),
4. Reference program (a specific list of experiments, tests, and analyses, assuming a likely course of events, to be used for estimating the cost of the program).

3. TASK 4100—CORE ENGINEERING DESIGN AND ANALYSIS

3.1. FUEL-ROD BEHAVIOR MODEL

Work on interpretation of fuel-rod irradiation experiments has resulted in development of the BRITL program.^{(1)*} This relatively simple code analyzes a constant power level irradiation. Recently the CYGRO-2⁽²⁾ code was obtained from the Argonne Code Center and converted to the 1108 computer at GGA. This is a much more elaborate program for cyclic operation, which was developed for the light-water breeder project by Westinghouse. However, preliminary results from this code show large tensile stresses in the outer parts of the fuel, which would probably result in crack formation. We have now obtained the updated CYGRO-3 code⁽³⁾ and are in the process of converting it to the GGA computer. This version has provisions for fuel cracking, as well as a number of other improvements. It is planned to use this code as a standard for comparison with faster running codes, such as BRITL, in order to develop a reasonably economic method for analyzing the results of GCFR fuel-rod irradiations.

*References are listed at the end of each section.

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4. TASK 4200—FUELS AND MATERIALS

4.1. IN-REACTOR CREEP RATES IN AUSTENITIC STAINLESS STEELS

A review of the current understanding of in-reactor creep in austenitic stainless steels is given here and appropriate creep equations for use in design calculations are suggested.

Very little experimental data on creep in stainless steels in a fast-flux environment are documented, and the theoretical treatment of irradiation creep in cubic materials is still controversial. However, there is general agreement on the following points:

1. At temperatures above $\sim 500^{\circ}\text{C}$, irradiation in a thermal flux has little or no effect on the steady-state creep rate of austenitic stainless steels.
2. Irradiation in a fast or thermal flux below $\sim 500^{\circ}\text{C}$ causes a significant increase in creep rate compared with an out-of-reactor test.
3. The irradiation-induced steady-state creep rate increases with decreasing temperature.
4. There is a recoverable irradiation-induced transient creep strain that is higher for cold-worked material than for annealed material.
5. Stress-rupture lifetimes and elongations are much reduced by fast-neutron irradiation.
6. The creep strain in a stressed sample in a neutron field is equal to the thermal creep strain (i.e., the creep strain that irradiated material would experience in an out-of-reactor test) plus the creep strain resulting from irradiation-induced deformation mechanisms. Under most conditions, the thermal creep rate of irradiated samples is similar to that of unirradiated materials.

These observations are consistent with the hypothesis that neutron irradiation adds a component to the creep strain through the enhancement of diffusion-controlled deformation mechanisms such as the climb of edge dislocations past barriers. Such an enhancement can occur if the steady-state flux of interstitials to dislocation sinks exceeds that of vacancies, although whether this is a sufficient condition for radiation enhancement of creep is a matter of controversy.⁽¹⁻³⁾ This condition is met under fast-flux irradiation and is also believed to be the driving force for void formation.

4.1.1. Irradiation Creep

4.1.1.1. Data on Stainless Steels. Results on the creep of helical springs of 316 stainless steel, both annealed and 20% cold-worked, in the Dounreay Materials Testing Reactor have been reported by Lewthwaite, et al.⁽⁴⁾ The test temperature was 100°C and the stresses were between 5500 and 9000 psi. The maximum fluence was 3×10^{20} n/cm² ($E > 1$ MeV). A transient creep strain, proportional to the applied stress and higher for the cold-worked material than the annealed material, was observed but no steady-state creep. Similar experiments on annealed 316 stainless steel in the Dounreay Fast Reactor⁽⁵⁾ conducted at ~250°C under 1900 to 6800 psi at fluences up to 1.5×10^{21} n/cm² ($E > 0.82$ MeV) showed a steady-state creep strain increasing linearly with fluence at a rate proportional to the applied stress. Out-of-reactor control experiments showed no significant creep under equivalent conditions.

Data on the creep of stainless steel in the EBR-II reactor are being obtained from the analysis of the profiles of annealed 304L stainless steel tubes internally pressurized to give hoop stresses from 10,000 to 35,000 psi.⁽⁶⁾ The operating temperature is 380°C and the maximum fast fluence reported so far is 9.9×10^{21} n/cm² ($E > 0.1$ MeV). The measurements show that in the part of the tube in the high flux zone a small diametral expansion occurs which is higher than can be explained by irradiation-induced swelling alone. The strain increases approximately linearly with fluence and increases with the applied stress. The stress-dependence data have much scatter, but a least-squares fit indicates that the strain may be proportional to the square of the applied stress.

Several studies have shown that steady-state creep rates of stainless steel irradiated in thermal reactors at temperatures between 500° and 750°C are essentially the same as in out-of-reactor tests at the same temperature. (7-9)

Gilbert, et al., have reported some results of creep experiments on annealed 304 stainless steel samples irradiated in the Hanford-K reactor at temperatures between 175° and 370°C. (10-12) The samples showed considerable transient creep and a small but significant steady-state creep. These results, together with the other data available from the literature are summarized in Fig. 4.1, which is taken from Ref. 12. The data were normalized for stress, assuming a linear dependence of creep strain on stress. A tentative plot of the steady-state creep constant versus temperature is shown in Fig. 4.2, also taken from Ref. 12.

Irradiation creep rates appear to be affected by the stress state, with hydrostatic or tensile stress causing more creep than the equivalent shear stress, i.e., some increase in volume occurs during creep under a tensile stress. Gilbert and Straalsund⁽¹³⁾ analyzed creep-relaxation data from annealed 304 stainless steel springs and tensile samples irradiated under identical conditions and concluded that creep relationships could be expressed in terms of effective stress, $\bar{\sigma}$, and effective strain, $\bar{\epsilon}$, where the effective stress and strain for various stress states are given by:

$$\left. \begin{array}{ll} \text{Uniaxial tension:} & \frac{\bar{\epsilon}}{\bar{\sigma}} = \frac{\epsilon_t}{6\sigma_t} \\ \text{Shear:} & \frac{\bar{\epsilon}}{\bar{\sigma}} = \frac{\epsilon_s}{3\sigma_s} \\ \text{Pressurized tube} & \frac{\bar{\epsilon}}{\bar{\sigma}} = \frac{4\epsilon_h}{33\sigma_h} \\ \text{(hoop stress):} & \end{array} \right\} \quad (1)$$

4.1.1.2. Equation for Irradiation Creep. The relationship between effective stress, $\bar{\sigma}$, and effective irradiation-induced creep strain, $\bar{\epsilon}$, is well represented by

Fig. 4.1 Irradiation-induced creep of austenitic stainless steels (taken from Ref. 12; reference numbers apply to Ref. 12)

Fig. 4.2 Temperature dependence of the irradiation-induced creep rate coefficient, C (taken from Ref. 12; reference numbers apply to Ref. 12)

$$\bar{\epsilon} = A\bar{\sigma} \left[1 - \exp \left(- \frac{\phi t}{B} \right) \right] + C\bar{\sigma}\phi t \quad , \quad (2)$$

where A is the transient creep constant, B is the time constant for transient creep, C is the steady-state creep constant, ϕ is the neutron flux, and t is the time. Equation (2) represents creep under a constant stress. The transient component of creep strain is recoverable and will adjust to the new stress level if the stress is changed.

Empirical best-fit values for the parameters in Eq. (2), based on Gilbert's summary of the existing data,⁽¹²⁾ have been obtained for annealed and cold-worked 316 stainless steel.⁽¹⁴⁾ The equations incorporate the observation in Ref. 4 that the transient creep strain and the transient creep time constant are higher for cold-worked material than for annealed material. They also incorporate the temperature dependence for the steady-state creep constant shown in Fig. 4.2. It is assumed that the transient creep constant has the same temperature dependence as the steady-state creep constant. The appropriate values are as follows:

Parameter	Annealed 316	20% Cold-worked 316
A	$0.18 \times 10^{-8} e^{(1.405-0.0027T)} (\text{psi}^{-1})$	$0.78 \times 10^{-8} e^{(1.405-0.0027T)} (\text{psi}^{-1})$
B	$1.47 \times 10^{21} \text{ n/cm}^2$	$6.37 \times 10^{20} \text{ n/cm}^2$
C	$1.87 \times 10^{-30} e^{(1.405-0.0027T)} (\text{psi}^{-1})$	$1.86 \times 10^{-30} e^{(1.405-0.0027T)} (\text{psi}^{-1})$

where T = temperature in °K; all fluences are in n/cm^2 ($E > 0.1 \text{ MeV}$).

There are not enough data to establish the effect of the neutron energy spectrum on the irradiation creep rates. Gilbert⁽¹²⁾ suggests using the mean neutron energy ($\bar{E}$) as a weighting factor on the fast fluence in Eq. (2). The constants listed above refer to the spectrum calculated for the Fast Test Reactor (FTR) driver fuel:

$$\left(\bar{E} = \int_0^{\infty} E \phi(E) dE / \int_0^{\infty} \phi(E) dE = 0.43 \text{ MeV} \right)$$

For a softer spectrum, the fluence should be weighted by a factor less than one; for a harder spectrum, the weighting factor would be greater than one.

Another set of constants that are in a summary of LMFBR core design requirements⁽¹⁵⁾ gives somewhat different creep constants and temperature dependence from those listed above. However, the values listed above correspond better with the reported data and it is recommended that they be used as best representing the scanty available data.

4.1.2. Thermal Creep

4.1.2.1. Data on Austenitic Stainless Steel. In contrast to irradiation creep, thermal creep experiments are relatively easy to perform and there is no shortage of experimental data on steady-state (secondary) creep rates above ~500°C, where creep rates are high enough to measure with reasonable accuracy ($> 10^{-6} \text{ hr}^{-1}$). Several reviews are available that summarize the experimental data on the steady-state creep rates in austenitic stainless steels.⁽¹⁴⁻¹⁷⁾ Data on the transient (primary) creep are rarer, but some measurements at 650° and 700°C were reported by Palentine⁽¹⁸⁾ for annealed and 20% cold-worked M 316 stainless steel.

4.1.2.2. Equations for Thermal Creep. To extrapolate creep data to temperatures and stresses outside the ranges used in obtaining experimental data, it is desirable to use equations that have a physical basis related to the mechanisms of creep.

It is now generally agreed⁽¹⁹⁾ that the dependence of the steady-state creep rate, $\dot{\epsilon}_s$, on the applied stress, σ , is best expressed by

$$\dot{\epsilon}_s = D(\sinh E\sigma)^n \quad (3)$$

where D, E, and n are constants for a given temperature. This type of expression avoids the complication inherent in power-law-type stress dependences that the stress exponent is not constant over the whole stress range. This type of stress-dependence is expected on theoretical grounds for several diffusion-controlled mechanisms of creep. The temperature dependence of the steady-state creep rate may be expressed by an Arrhenius

type of relationship, with the activation energy similar to that for self-diffusion. In practice, the activation energies for creep are found to be somewhat higher than the activation energies for self-diffusion in austenitic stainless steels (~100 kcal/mole compared with ~70 kcal/mole for self-diffusion). Figure 4.3 shows a summary of 1200°F data on the steady-state creep rate of annealed 316 stainless steel fitted to Eq. (3).⁽¹⁷⁾

Primary creep strain, ϵ_p , may be expressed either as an exponential approach to a limiting transient creep strain or by an Andrade type of relationship:

$$\epsilon_p = F\sigma^m t^{1/3}, \quad (4)$$

where F and m are constants, σ is the stress, and t is the time. The Andrade relationship was used by Palentine⁽¹⁸⁾ for analyzing primary creep at 650° and 700°C. These data are insufficient to determine the temperature dependence of F, but it is reasonable to assume that F will have an Arrhenius type of temperature dependence with the same activation energy as that for steady-state creep.

A reasonable expression for the total thermal creep strain in tension, ϵ_t , is therefore

$$\epsilon_t = \left[G\sigma^m t^{1/3} + Ht(\sinh E\sigma)^n \right] \exp\left(-\frac{Q}{RT}\right), \quad (5)$$

where R is the gas constant and T is the absolute temperature. Values for m are available from Ref. 18, and values for H, E, n, and Q are from several analyses of available data on the LMFBR program.⁽¹⁵⁾ Values for G were calculated to give the best match between Eq. (4) and Palentine's data on primary creep.⁽¹⁸⁾ These values are given in the following tabulation.

Since these thermal creep equations are based on data obtained between ~500° and ~800°C, they should be applied with caution below 500°C. The equations will tend to underestimate the thermal creep strain at low temperatures because the activation energy for creep normally decreases with decreasing temperature.

Fig. 4.3 Stress vs secondary creep rate of annealed Type 316 stainless steel at 1200°F (taken from Ref. 17; reference numbers apply to Ref. 17)

Parameter	Annealed 316	20% Cold-worked 316
G	$8.7 \times 10^8 \text{ (hr}^{-1/3} \text{ psi}^{-m})$	$1.3 \times 10^7 \text{ (hr}^{-1/3} \text{ psi}^{-m})$
m	3.2	2.14
H	$2.951 \times 10^{24} \text{ (hr}^{-1})$	$6.55 \times 10^{12} \text{ (hr}^{-1})$
E	$\frac{0.02}{T} \text{ (psi}^{-1})$	$\frac{0.30}{T} \text{ (psi}^{-1})$
n	6.371	1.0
Q	$110,935 \text{ (cal mole}^{-1} \text{ }^\circ\text{K}^{-1})$	$88,843 \text{ (cal mole}^{-1} \text{ }^\circ\text{K}^{-1})$

4.1.3. Conclusions

Irradiation-induced creep strain in austenitic stainless steels may be regarded as supplying an additional component to be added to the thermal creep strain. The irradiation-induced creep component increases with decreasing temperature and generally dominates the thermal creep component at temperatures below 500° to 600°C. Analysis of the very limited data indicates that suitable equations for estimating the total creep strain, ϵ , in 316 stainless steel components under a tensile stress are as given below. Each line of the equation represents a different creep component; these are written in the following order: irradiation-induced transient creep; irradiation-induced steady-state creep; thermal transient creep; and thermal steady-state creep.

Annealed 316 Stainless Steel:

$$\begin{aligned} \epsilon = & 1.08 \times 10^{-8} \sigma \exp(1.405 - 0.0027T)[1 - \exp(-2.45 \times 10^{-18} \phi t)] \\ & + 4.15 \times 10^{-26} \sigma \phi t \exp(1.405 - 0.0027T) \\ & + 8.7 \times 10^8 \sigma^{3.2} t^{0.333} \left(\exp - \frac{55800}{T} \right) \\ & + 2.951 \times 10^{24} t \left[\sinh \left(\frac{0.02\sigma}{T} \right) \right]^{6.371} \left(\exp - \frac{55800}{T} \right) . \end{aligned}$$

20% Cold-worked Stainless Steel:

$$\begin{aligned} \epsilon = & 4.68 \times 10^{-8} \sigma \exp(1.405 - 0.0027T)[1 - \exp(-5.65 \times 10^{-18} \phi t)] \\ & + 4.15 \times 10^{-26} \sigma \phi t \exp(1.405 - 0.0027T) \end{aligned}$$

$$+ 1.3 \times 10^7 \sigma^{2.14} t^{0.333} \left(\exp - \frac{44700}{T} \right) \\ + 6.55 \times 10^{12} t \left[\sinh \left(\frac{0.30\sigma}{T} \right) \right] \exp \left(- \frac{44700}{T} \right)$$

where ϵ = tensile strain (in./in.),

σ = tensile stress (psi),

T = temperature ($^{\circ}\text{K}$),

ϕ = fast flux ($\text{n/cm}^2\text{-sec}$ ($E > 0.1 \text{ MeV}$)),

t = time (hr).

Some typical calculated strains are shown in Fig. 4.4 for a fast flux of $10^{15} \text{ n/cm}^2\text{-sec}$.

When applying the above equations, the following points should be borne in mind:

1. The values of the constants in many cases are based on extrapolations of scanty data.
2. The equations probably underestimate the thermal creep components below $\sim 500^{\circ}\text{C}$.
3. The irradiation-creep components are based on the neutron spectrum calculated for the FTR driver fuel ($\bar{E} = 0.43 \text{ MeV}$). For harder or softer spectra the fluence, ϕ , should be weighted by a factor $\bar{E}/0.43$.
4. For stress systems other than pure tension, the irradiation-induced creep strain components should be adjusted according to Gilbert's "effective stress and strain" [Eq. (1)] and the thermal creep components calculated according to the usual Von Mises effective stress formula.
5. The transient strain components are reversable if the applied stress is reduced during a test.
6. The steady-state creep rate of cold-worked 316 stainless steel will increase as recovery of the cold-worked microstructure takes place after long times at temperatures above $\sim 650^{\circ}\text{C}$. Stress-rupture data indicate that recovery in 20% cold-worked 316 stainless steel becomes important for times above $\sim 3000 \text{ hr}$ at 650°C and $\sim 1500 \text{ hr}$ at 700°C .⁽²⁰⁾

Fig. 4.4 Fractional creep strain vs temperature for 20% cold-worked 316 stainless steel at 20,000 psi for 30,000 hr

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5. TASK 4400—IRRADIATION TESTS

An irradiation test program employing both thermal and fast flux tests is being conducted to develop fuel elements for the GCFR. Although this program is focused on the needs of the GCFR, it also contributes to the general background for the Liquid-Metal Fast Breeder Reactor (LMFBR) fuel-development program because the fuel-rod designs for these two fast breeder reactor concepts have much in common. The thermal flux irradiation program is a joint effort of Gulf General Atomic and Oak Ridge National Laboratory; the irradiations are being performed in the Oak Ridge Research Reactor (ORR) poolside facility. The fast-flux irradiations are to be performed in the Experimental Breeder Reactor (EBR-II).

5.1. THERMAL FLUX TESTS

5.1.1. Irradiation Capsule GB-9

The irradiation of an instrumented, (U,Pu)O₂-fueled, vented fuel rod containing a charcoal trap in Capsule GB-9 (formerly designated 04-P9) in the ORR was started in March 1970 and has reached a burnup exposure of 27,000 MWd/Te toward a burnup goal greater than 50,000 MWd/Te at a linear heat-generation rate of 16 kW/ft. After ~20,000 MWd/Te burnup, this goal was adjusted to ~100,000 MWd/Te.

Fission gases at the inlet and exit of the trap are being monitored under steady-state operation and cyclic conditions of power, temperature, and pressure. The cladding outer surface temperature is 700°C and the system pressure is 1000 psi. The design of the fuel rod and capsule are given in Ref. 1. The fuel rod is a 10-in.-long fuel column, with a 3-in. stack of UO₂ blanket pellets above the fuel column, and a charcoal trap above the blanket section. The interior of the rod is connected to a radio-analytical sampling system through the charcoal trap, as shown schematically

in Fig. 5.1. Fission-product decay heat in the charcoal trap is supplemented by electrical heat to maintain trap temperatures in the range of 200° to 400°C (normally 300°C). Thermocouples along the length of the fuel rod and in the charcoal trap region monitor the temperatures.

One aspect of the GCFR fuel rod currently being studied experimentally in Capsule GB-9 is the pressure equalization system. Although the test arrangement is not a complete simulation of the GCFR fuel pressure equalization system, it does possess many aspects of the system that make test information useful in evaluating the system design and expected performance. At steady operating conditions, gaseous diffusion of xenon and krypton isotopes out of the rod and through the trap is similar to that expected in the GCFR, except for the lengths of fuel and blanket regions and the trap and the fission rate. The sweep of helium across the top of the rod trap into the sampling line is analogous to helium being sucked into the fuel element to grid plate connection, which is past the outlet from the annular trap of the fuel element, and on into the monitor lines. The radiation detector on the sample line is representative of a monitor station in the GCFR pressure equalization system. The monitor in the sample line (which corresponds approximately to a monitor in the GCFR pressure-equalization system) is sensitive enough to discern breathing resulting from pressure changes of 1 psi at a system pressure of 1000 psi. The alternate flow path of sweep gas through the rod trap is analogous to a leak of primary coolant into a rod between the upper blanket and the trap. With this equipment, such aspects as breathing, depressurization, monitor response, etc., are being studied and evaluated.

During normal steady operation, radioactivity in the sample line indicated by the detector has been essentially steady, with a small rate of increase, which corresponds to the increasing release with burnup.

Sample-line detector response during shutdown at the end of an ORR cycle is shown in Fig. 5.2. The radioactivity level falls very rapidly with practically zero lag to about 20% of its steady value and then decays more slowly to zero; no bursts of fission-gas release are apparent. This is an example of release during inhalation.

Fig. 5.1 Schematic diagram of capsule GB-9

Fig. 5.2 Radioactivity in the sample line of capsule GB-9 during ORR scram

In the process of returning to power, the capsule was inserted into the flux to increase the power generation in steps, as indicated by the thermocouples monitoring the cladding temperature (50°C/step). The activity in the sample-line flow responded by step increases without indications of bursts in the release of gases within the fuel rod. This is an example of exhalation with rising power level.

The activity in the sample line indicated by the radiation detector during operation with three pressure cycles from 1000 to 750 psia at 6%/hr depressurization and rapid repressurization is plotted in Fig. 5.3 at the times indicated by the arrows. The initial peak during the first cycle is gas-borne activity being flushed out of the trap. The isotopic analyses are given in Table 5.1. The sample line activity during the second and third cycles rises rapidly and then more slowly as if to approach a steady level as expected. However, an upward inflection to the curves occurs near the end of each of these cycles, which was not anticipated and which has not yet been satisfactorily explained. Further analysis and experiments are planned to elucidate this behavior. The sharp cutoff noted at the end of each cycle results from repressurization, which is analogous to inhalation.

In the process of taking samples of the fission gases effusing from the top of the blanket region, sweep-gas flow was diverted so as to enter the rod at that point and flow up through the trap. This is analogous to what would occur with a large leak just above the blanket region in the GCFR fuel rod. The activity in the sample line as measured by the radiation detector is shown in Fig. 5.4. The large peak immediately following initiation of sweep-gas flow through the trap results from sweeping gas-borne activity, which was proceeding through the trap by gaseous diffusion, out of the trap into the sample line and from activity from desorption of the trap. The higher equilibrium level finally attained shows the total effectiveness of the trap.

Measurements of the purge gas sweeping past the trap outlet and sweeping through the trap under steady-state conditions after operation for ~170 effective full power days (EFPD) and a burnup exposure of 20,000 MWd/Te

Fig. 5.3 Sample-line activity and internal fuel-rod pressure during three successive breathing cycles

Table 5.1
RELRELEASE RATE TO BIRTH RATE (R/B) VALUES FOR FISSION-GAS RELEASE FROM CAPSULE GB-9
DURING PRESSURE-CYCLING EXPERIMENTS

Cycle No.	Sample No.	Pressure (psi)	R/B					
			Kr ^{85m}	Kr ⁸⁸	Kr ⁸⁷	Xe ¹³³	Xe ¹³⁵	Xe ¹³⁸
1	1	963	4.61×10^{-2}	1.66×10^{-3}	5.89×10^{-3}	4.09×10^{-1}	3.10×10^{-2}	4.48×10^{-5}
	2	775	3.47×10^{-2}	1.22×10^{-3}	5.15×10^{-3}	1.76×10^{-1}	1.97×10^{-2}	5.20×10^{-5}
2	3	960	1.47×10^{-2}	3.06×10^{-4}	7.03×10^{-3}	1.87×10^{-2}	2.72×10^{-3}	4.53×10^{-5}
	4	770	3.50×10^{-2}	1.42×10^{-3}	5.35×10^{-3}	3.60×10^{-1}	2.40×10^{-2}	5.11×10^{-5}
3	5	965	1.41×10^{-2}	6.25×10^{-4}	2.87×10^{-3}	1.42×10^{-2}	1.79×10^{-3}	5.57×10^{-5}
	6	770	3.55×10^{-2}	1.23×10^{-3}	4.56×10^{-3}	2.02×10^{-1}	2.56×10^{-2}	1.29×10^{-4}

Fig. 5.4 Sample-line monitor response to purge flow through the rod trap

were analyzed for krypton and xenon radioactivity. The absolute values of the fractional release from the fueled region measured in the bottom of the trap are approximately proportional to the square root of half life, within the precision of the data available. These results indicate that the mechanism controlling the release is a diffusion process. The release values for krypton and xenon shown in Fig. 5.5 slowly approach a steady-state value at a constant cladding temperature of 686°C and a charcoal-trap temperature of 300°C. The fractional release at the trap outlet relative to the trap inlet in Table 5.2 are compared with calculated values, which are in good agreement.

When the purge gas was switched from sweeping across the outlet of the trap to sweeping through the trap, the ionization chamber monitoring the sample line responded by going from a steady reading of 15 mr/hr, through a peak of 4000 mr/hr as the activity in the trap was swept out, and back to a steady level of 58 mr/hr. When the purge stream was switched back to the top of the trap, the original level (15 mr/hr) was reestablished in a few minutes.

Additional data were obtained for the release of gaseous fission products from the charcoal trap at a bed temperature of 200°C. Standard gas samples were obtained first at 200°C and finally at the normal operating temperature of 300°C. Inspection of the data indicates a factor of 3 increase in release when going from 200°C and 300°C, which is in agreement with expectation.

Neutron radiographs of Capsule GB-9 were taken after a burnup of ~20,000 MWd/Te. The fuel appears to be as it was in its preirradiated condition. Analysis of densitometer scans taken of the neutron radiograph revealed no detectable evidence of fission-product deposition, not even in the charcoal trap. The main value of the neutron radiographs and scans is for comparison of the capsule and traps as burnup proceeds.

Some concern has been expressed relative to the loss of postirradiation evaluation information if cladding failure should occur prior to reaching the established burnup goals. The advent of such a failure would be immediately apparent when the gas pressure within the rod is maintained at

Fig. 5.5 Steady-state fractional release of fission gases at the top of the trap vs burnup exposure for capsule GB-9

Table 5.2
RATIO OF THE FRACTIONAL RELEASE AT THE TRAP OUTLET
RELATIVE TO THAT AT THE TRAP INLET

Sample Series No.	Release _{outlet} /Release _{inlet}						
	Kr-85m	Kr-88	Kr-87	Kr-89	Xe-133	Xe-135	Xe-138
Calculated ratio	(0.71)	(0.59)	(0.37)	(10 ⁻⁴)	(0.98)	(0.81)	(0.10)
1	0.51	0.89	0.46	----	0.79	0.37	0.18
2	0.74	0.72	0.43	----	0.72	0.41	----
3	0.64	0.70	0.55	----	0.84	0.33	----
4	0.78	0.64	0.60	----	0.99	0.35	----
5	0.69	0.58	0.54	----	~1	0.39	0.21
6	0.61	0.51	0.45	----	0.79	0.30	0.06
7	0.71	0.83	0.56	----	~1	0.38	----
8	0.75	0.78	0.47	----	~1	0.40	----
9 (a)	0.71	0.69	0.53	----	0.99	0.38	----
13	0.72	0.71	0.50	----	~1	0.48	0.06
14	0.85	0.85	0.60	----	~1	0.48	0.10
15	0.68	0.70	0.44	----	~1	0.36	0.06
16	0.59	0.89	0.59	----	0.88	0.44	0.10
17	0.51	0.73	0.45	----	----	0.40	0.06

^aSeries 10, 11, and 12 not included because data questionable due to insufficient delay time between sample periods to allow the trap activity to reach steady-state equilibrium.

approximately 25 psi greater pressure than the NaK external to the rod. Failure of the cladding would immediately leak high activity levels in the cover gas of the NaK system, which is monitored. Ingress of NaK into the fuel rod can be prevented as long as the pressure relationship is maintained. Methods of handling the capsule following cladding failure until the rod can be removed from the NaK in the hot cell, which will maintain the internal overpressure relationship or solidify the NaK, are being considered.

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, irradiation of Capsule GB-10 in the ORR is being planned. The major objective with Capsule GB-10 is to obtain important information on the release, transport, and trapping of the volatile fission products. Because of the effectiveness of the charcoal trap in absorbing 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 before they pass through the trap. A preliminary proposed design of the Capsule GB-10 vented fuel rod is shown in Fig. 5.6.

5.2. FAST-FLUX TESTS

The general objective of the F-1 experiment in the EBR-II fast-flux environment is to study fuel-rod specimens over a wider range of temperatures and to a higher range of cladding temperatures than heretofore carried out in the LMFBR program. Variables that have been kept constant are the cladding material (316 stainless steel, 20% cold-worked), the fuel (mixed $(U,Pu)O_2$), the fuel fabrication technique, and the fuel-rod design. These fixed test conditions are summarized in Table 5.3. The range of temperatures cited, ranging from 612° to 783°C, are for the cladding outside surface. The midcladding temperatures for the Capsule G-1 through G-8 rods will be ~20°C higher than the listed value for the surface temperatures. The temperature range will overlap that already being studied under the LMFBR program to provide a point of commonality for comparison of results. The reference condition selected is the same as that being used in the

Fig. 5.6 Preliminary design of capsule GB-10

Table 5.3

FIXED TEST CONDITIONS FOR THE INITIAL F-1 SUBASSEMBLY

Cladding	316 SS
Cladding OD	0.3 in.
OD/ID ratio	~1.15
Linear power generation	~15 kW/ft
Fuel	85 UO_2 - 15 PuO_2
Fuel smear density	$\leq$ 85% T.D.
External pressure	~70 psig
Bond-gap fluid	Helium
Initial internal pressure, hot	40 psig
Maximum end-of-life internal pressure, hot	< 310 psig

thermal irradiation of Capsule GB-9 so that a direct comparison can be made of a fast versus a thermal neutron test environment. The rods will operate at linear heat ratings from 14 to 16 kW/ft to burnup goals ranging from 25,000 to 100,000 MWd/Te.

The standard EBR-II B-7 core subassembly was chosen as a design basis because it provided space within each capsule for a thermal barrier to raise the cladding temperatures to the desired levels. The design of the capsules is shown in Fig. 5.7. Detailed information on this experiment is given in Ref. 2.

The active charcoal fission-product traps in three of the rods will provide information on the ability of hot charcoal to remove condensable fission products and to delay short-lived gaseous fission products in the fast flux environment. In addition to the active carbon traps, two sealed containers of the same charcoal are provided at the ends of each capsule to assess the effects of the fast neutron environment on the sorption properties of charcoal, apart from effects due to fission-product loading, by comparison of preirradiation and postirradiation measurements. Fluence monitors employed are aluminum, cobalt, and vanadium-iron in the form of wires and carbon-coated (Th, enriched U)C₂, ThC₂, and depleted UC₂ particles in order to cover a rather wide range of threshold reactions and improve the accuracy of the measurement. Temperature monitoring will employ two techniques. In one technique, SiC rods are used, which are sensitive to the end-of-irradiation temperature. During postirradiation annealing, the SiC experiences a dimensional change at the temperature at which it was irradiated. In the other technique, Kr⁸⁵ is released from the cladding; this release is dependent on the thermal history. Kr⁸⁵ is introduced into the cladding during irradiation by fission recoil; during slow postirradiation heating of the cladding, the Kr⁸⁵ is released at or above the temperature at which the cladding was irradiated.

The GCFR fast-flux seven-rod subassembly test, designated F1 (X094), was installed in the EBR-II for run 47B in November 1970. However, start of irradiation of the subassembly was delayed because of maintenance work on the EBR-II.

Fig. 5.7 Typical F-1 capsule with active charcoal traps

The fuel-rod cladding material (20% cold-worked 316 stainless steel), the fuel $[(\text{Pu},\text{U})\text{O}_2]$, the fuel fabrication technique, and the design are the same for all of the rods, except that three rods—G-4, G-6, and G-7—contain active fission-product traps (Fig. 5.7); the other rods—G-1, G-2, G-3, G-5, and G-8—do not. The test conditions and objectives for these rods and for five planned replacement rods are also summarized in Table 5.4. Two of the planned replacement rods, G-9 and G-10, will be similar in design to the above rods except that the cladding outer surface will be roughened, as shown in Fig. 5.8, instead of smooth. One rod will contain fuel of low oxygen-to-metal ratio ($\text{O}/\text{M} \approx 1.94$ instead of 1.98). Two rods will contain solid pellet fuel instead of annular pellet fuel. One of these two rods will operate at the GCFR reference maximum cladding temperature of 700°C and the other at 750°C . All the other replacement rods will operate at maximum cladding temperatures of 700°C .

5.3. OUT-OF-REACTOR TESTS

5.3.1. Krypton-Helium Interdiffusion

The investigation of krypton and helium interdiffusion coefficients has continued utilizing the all-welded diffusion tube apparatus described in Ref. 3. The apparatus has been further modified to incorporate a Na(I) scintillation detector, thus allowing direct gamma counting of the Kr^{85} 520-keV photopeak. This modification necessitated increasing the Kr^{85} source activity by a factor of 5 to compensate for the loss in sensitivity when changing from an internal ionization detector to an external scintillation detector.

Previous experiments utilizing the electrometer were conducted using a charcoal bed in the simulated GCFR fuel-rod diffusion tube. The current experiments are being conducted using an empty diffusion tube to simplify analysis of the data using the SLIDER computer program. A total of 6 runs (duplicate runs at 1000, 500, and 150 psia helium pressure) has been completed and is currently being analyzed. Additional experiments will be conducted under varying diffusion tube conditions of pressure, temperature, and geometry.

Table 5.4

IRRADIATION CAPSULE LOADINGS FOR F-1 SUBASSEMBLY

Capsule	Test Condition		Burnup (MWd/Te x 1000)				Purpose (Status)
	Cladding Temperature (°C)	Heat-generation Rating (kW/ft)	25	50	75	100	
G-1	783	16.1					Temperature effect (being irradiated)
G-2	753	15.5					Temperature effect (being irradiated)
G-3	702	15.5					Temperature calibration (being irradiated)
G-4	708	14.9					Burnup effect (being irradiated)
G-5	669	14.3					Temperature effect (being irradiated)
G-6	721	14.4					Reference/fast flux-thermal flux comparison (being irradiated)
G-7	612	13.8					Overlap with LMFBR/temperature effect (being irradiated)
G-8 ^a	700	15.5					Burnup effect (replacement rod at EBR-II)
G-9	700	14.5					Surface roughening effect
G-10	700	14.5					Surface roughening effect (reproducibility)
G-11	700	15.0					Low fuel oxygen to metal effect
G-12	700	15.0					Solid pellet effect at reference cladding temperature
G-13	750	15.0					Solid pellet effect at cladding temperature of 750°C

^aG-8, which will replace G-3, is already at EBR-II.

Unetched

50 X

Etched

50 X

Roughening dimensions: Width - 0.003 in.; height - 0.002 in.
Pitch - 0.020 in.; root radius - 0.0015 in.

Fig. 5.8 Longitudinal section of electrochemically etched surface-roughened 316 stainless steel tube

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6. TASK 4700-NUCLEAR ANALYSIS AND REACTOR PHYSICS

During the period covered by this report, the major effort on nuclear analysis and reactor physics was the review, assessment, and documentation of methods and data used for reactor calculations.⁽¹⁾ Closely related to this review of internal methods is the surveillance of the LMFBR critical-assembly program, which serves to summarize the methods and data used in other fast-breeder reactor investigations. Finally, the program for the remainder of this fiscal year as well as the next fiscal year has received attention and the major problems and goals have been delineated.

6.1. ANALYSIS OF METHODS AND DATA

A review of the reactor physics input to the GCFR demonstration plant design was initiated with the intent of double-checking routine calculations (volume fractions, U/Pu ratios, atom densities, etc.) and then ascertaining the accuracy of the methods and the quality of the basic cross-section data utilized. The latter part of this task included an assessment of not only the data and methods used in the reference design up to that time, but also an assessment of new data and methods and their effect on design work completed up to that time.

The results of this review have been documented and are in the process of being published.⁽¹⁾ The most important new material in the review concerns the effects of new data on GCFR reactor development in general and the demonstration reactor in particular. In general, it is accepted that the quality of computer codes available for fast-reactor calculations exceeds the quality of presently available nuclear data.

6.1.1. Cross-section Data

Pu²³⁹

Until recently, the uncertainty in Pu²³⁹ alpha (α) values below about

20 keV has been the most important fast-reactor cross-section uncertainty, due to the extreme sensitivity of the breeding ratio to this quantity. Measurements, however, performed by Gwin⁽²⁾ and coworkers at Oak Ridge National Laboratory in the energy range from 0.3 to 30 keV have apparently settled that question. These measurements are supported by experimental work of others (Lottin,⁽³⁾ Carr,⁽⁴⁾ Hitchcock,⁽⁵⁾ Shunk,⁽⁶⁾ etc.). The new unresolved resonance parameters developed by Hitchcock and Hutchins⁽⁵⁾ for ENDF/B, Version II, have been used to prepare those parameters for use in the GANDY option of the GGC-5 spectrum code. By thus modifying the KFK-750 data,⁽⁷⁾ which have been the basis for the Pu²³⁹ cross sections heretofore, a set of (GGC-5 produced) cross sections has been obtained which agrees very well with the new measurements. The new data increase the Pu²³⁹ α significantly in the 0.3 keV to 10 keV range. The effect of this increase on GCFR performance will be briefly assessed later in this section.

Pu²⁴⁰

For this nuclide, a significant amount of uncertainty continues to exist. In fast reactors, these uncertainties imply an uncertainty up to about 1% in multiplication factor. Current appraisals of various Pu²⁴⁰ capture-cross-section evaluations indicate that ENDF/B, Version II, agrees best with experimental evidence. These data lead to a capture cross section significantly lower than that used for GCFR calculations⁽⁸⁾ up to this time for energies from 3 keV to 4 MeV. The effects on the GCFR of these new data are being assessed.

U²³⁸

Of all the cross section uncertainties presently under study, those in the capture cross section of U²³⁸ are the most significant to fast-reactor performance.

In the energy range from 1 keV to 1 MeV, the uncertainty is up to 20%. Between 1 and 20 keV, experimental data are quite scattered, but the agreement between all experimental data and values derived from predicted statistical resonance parameters is reasonable.

Above 20 keV, however, the story is different. The data of Barry⁽⁹⁾ has heretofore been generally accepted in the range above 20 keV. Recent data, however, conflicts with these measurements in the range from 20 keV to 1 MeV. To further complicate things, various sources of these recent data conflict with each other. Past U^{238} capture data in use at GGA (the so-called "old" GGA-NASA set⁽¹⁰⁾) were based essentially on Ref. 9 for 20 keV and up.

In the range from 20 keV to 100 keV, the new measurements fall into one of two patterns, about 15% apart in capture-cross-section value. In the higher set fall the "new" GGA-NASA measurements (1970),^(11,12) preliminary Oak Ridge National Laboratory results,⁽¹³⁾ and the Karlsruhe measurements of Menlove and Poenitz.⁽¹⁴⁾ All of these are within a 5% band. In the second of the two patterns, the Moxon measurements⁽¹⁵⁾ stand alone, about 15% lower. It is over the choice of which of these two patterns should be followed that much of the current controversy rages. Two cross-section evaluations have made opposite choices: Schmidt⁽¹⁶⁾ has opted for the Moxon data, whereas Davey⁽¹⁷⁾ prefers the Karlsruhe data. Since the latter are further supported by the "new" GGA-NASA data and the Oak Ridge data, the choice of Davey seems to be the better.

In the range from 100 keV to 1 MeV, the controversy involves basically the Karlsruhe and "new" GGA-NASA data versus the Barry data, with the Moxon data playing a supporting role. Because of the agreement between two independent experimenters (Karlsruhe and GGA), because of the better connection with the data selected below 100 keV (Karlsruhe again), and finally because Moxon's data give further evidence of a lower capture cross section than the Barry data in the range 100 keV to 1 MeV, the Karlsruhe and GGA-NASA data are selected for use in the energy range from 100 keV and 1 MeV. The Barry data continue to be accepted for energies above 1 MeV.

In summary, for U^{238} captures, the cross section data as generated by the "new" GGA-NASA measurements seem to be the best choice from 20 keV to 1 MeV and the Barry data are the best choice above 1 MeV, whereas there is no disagreement below 20 keV. As far as implementing these conclusions in

analytical data, the ENDF/B, Version II, data are incorporated into the GCFR data sets, except that the resonance parameters based on the "new" GGA-NASA data are used in preference to the ENDF/B, Version II, from 20 keV to 1 MeV. The analysis of the effects of these changes is continuing. The effects are expected to be significant and important because of the sizable reduction in captures in the range 100 keV to 1 MeV as the breeding ratio, Doppler coefficient, steam-entry worth, and reactivity excess can all be influenced.

6.1.2. Effects of New Cross-section Data

As has been mentioned previously in reference to each individual nuclide for which new data have become available and new uncertainties have arisen, the effects are under continuing analysis. However, some important preliminary interpretation of the effects can be made. The broad-group cross sections used to make these determinations were derived using the new GGC-5 spectrum code with the GAROL-GANDY resonance options. In the comparison to be made, the "old" GGA-NASA data set (for U^{238}) represents the GCFR calculations performed up to the time of the recent cross-section adjustments. Additional calculations have incorporated the new findings in all three nuclides, Pu^{239} , Pu^{240} , and U^{238} , with U^{238} probably being the major contributor to the new results. These effects need and will receive additional attention and appraisal.

Tables 6.1 and 6.2 show the effects of the different cross-section evaluations on the 300-MW(e) GCFR demonstration plant and on the 1000-MW(e) designs. The 1000-MW(e) GCFR plant used for the analysis has a thermal output of 2564 MW(t), a core volume of 5272 liters, a specific power of 1.294 MW(t)/kg fissile, and an average core enrichment of 15.8%.

The new GGA-NASA data have the greatest impact on reactor performance. For the demonstration plant, the reactivity exceeds by 0.0149 the value obtained with the ENDF/B data. The breeding ratio drops by 0.07 with a resulting detrimental effect on doubling time and fuel costs.

The results for the 1000-MW(e) plant show the same trend as for the demonstration plant. The breeding ratio drops by 0.08 and the doubling time increases by about 20% when the new GGA-NASA set replaces ENDF/B, Version II.

Table 6.1
EFFECT OF VARIOUS U^{238} CAPTURE CROSS-SECTION DATA
FOR THE 300-MW(e) GCFR DEMONSTRATION PLANT

Data Set	k_{eff}	$\% \Delta k/k^a$	$\% \Delta M/M^b$	$\bar{a}^c$	Breeding Ratio ^d	Doubling Time ^e (yr)
ENDF/B-II ^f	1.0390	---	---	0.209	1.34	15.9
New GGA-NASA ^g	1.0554	+1.49	-2.3	0.215	1.27	20.0
Moxon ^g	1.0498	+0.99	-1.5	0.214	1.29	18.6
Old GGA-NASA ^h	1.0491	+0.93	-1.4	0.174	1.33	16.9

^aWith respect to ENDF/B-II.

^bChange in critical mass with respect to ENDF/B-II, based on the empirical relationship: 1% fissile mass $\rightarrow$ 0.66% $\Delta k/k$.

^cAverage fissile capture-to-fission ratio.

^dBreeding ratio based on $(U^{238} + Pu^{240}) / (Pu^{239} + Pu^{241})$.

^eLinear doubling time = $2.74/S(1 + \alpha)(BR - 1)(1 - F)L$, where S = specific power (MWt/kg fissile), L = load factor (0.8), F = fertile fraction of total fissions (= 0.161 for the demonstration plant).

^fWith high Pu^{239} alpha and low Pu^{240} capture.

^gWith ENDF/B-II U^{238} data, high Pu^{239} alpha and low Pu^{240} capture.

^hWith low Pu^{239} alpha and high Pu^{240} capture.

The comparison in reactor performance shown in Table 6.2 refers to the GCFR. As far as performance of a sodium-cooled fast breeder reactor is concerned, the low U^{238} capture cross sections decrease the breeding ratio by about the same amount, but the percentage drop of the breeding gain (BR-1) is larger because of the smaller breeding ratio (~ 1.25). Since the doubling time is inversely proportional to the breeding gain, fuel cycle economics are more affected by the low capture than is the case for the GCFR.

In summary, the use of the recent low-capture U^{238} data has only a small adverse effect on the economics of the GCFR. On the other hand, calculations indicate a 30% larger Doppler coefficient and a more negative steam-entry reactivity effect.

Table 6.2
EFFECT OF VARIOUS U^{238} CAPTURE CROSS-SECTION DATA
FOR THE 1000-MW(e) GCFR PLANT

Data Set	k_{eff}	$\% \Delta k/k^a$	$\% \Delta M/M^b$	$\bar{\alpha}^c$	Breeding Ratio ^d	Doubling Time ^e (yr)
ENDF/B-II ^f	0.9850	---	---	0.210	1.50	5.4
New GGA-NASA ^g	1.0024	+1.76	-2.7	0.217	1.42	6.5
Moxon ^g	0.9966	+1.18	-1.8	0.216	1.45	6.0
Old GGA-NASA ^h	1.0010	+1.63	-2.5	0.175	1.50	5.6

^aWith respect to ENDF/B-II.

^bChange in critical mass with respect to ENDF/B-II, based on the empirical relationship: 1% fissile mass $\rightarrow$ 0.66% $\Delta k/k$.

^cAverage fissile capture-to-fission ratio.

^dBreeding ratio based on $(U^{238} + Pu^{240})/(Pu^{239} + Pu^{241})$.

^eLinear doubling time = $2.74/S(1 + \bar{\alpha})(BR - 1)(1 - F)L$, where S = specific power (MWt/kg fissile), L = load factor (0.8), and F = fertile fraction of total fissions (= 0.182).

^fWith high Pu^{239} alpha and low Pu^{240} capture.

^gWith ENDF/B-II U^{238} data, high Pu^{239} alpha and low Pu^{240} capture.

^hWith low Pu^{239} alpha and high Pu^{240} capture.

6.2. SURVEILLANCE OF LMFBR DEMONSTRATION-PLANT CRITICALITY PROGRAM

During this reporting period, monitoring of the progress and activities of the LMFBR critical-experiment program has continued. Although the major effort has been on methods and data review, as described in Section 6.1, the information received on the criticality program has been under continuing surveillance. Preparations are under way for performing calculations on some of these assemblies to see how well GGA's methods and data are able to calculate the experiments being conducted. Also, many of the methods and data utilized in GCFR reactor physics calculations are utilized by other reactor projects at GGA that have built or are building critical facilities and full-scale operating reactors. Hence, the methods and data

in use have to a great extent been tested against reality. But the opportunity to calculate the LMFBR criticalities permits making the benchmark current and in a fast spectrum environment.

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Appendix
PUBLICATIONS

"High-temperature Fast-flux Irradiation Experiment for Mixed-oxide Fuel Rods," by P. W. Flynn, et al., GA-10264, October 15, 1970.

"Gas-cooled Fast Reactor Fuel-element Development—Fuel-rod Irradiation Tests," by R. H. Simon, et al., paper presented at the International Meeting on Fast Reactor Fuel and Fuel Elements, Karlsruhe, Germany, September 28-30, 1970 (GA-10262, August 7, 1970).

"GCFR Fuel-rod Design Considerations," by W. I. Thompson, paper presented at the Winter Meeting of the American Nuclear Society, Washington, D.C., November 15-19, 1970 (GA-10361, October 30, 1970).

"Pressure Equalization System for GCFR Fuel Elements," by R. J. Campana, paper presented at the Winter Meeting of the American Nuclear Society, Washington, D.C., November 15-19, 1970 (GA-10343, November 2, 1970).

"Gulf General Atomic Contribution to the Thirty-first Meeting of the High-temperature Fuels Committee, Argonne National Laboratory, December 8-10, 1970," by J. R. Lindgren, et al., GAMD-2361, Part 31, November 9, 1970 (Official Use Only).

"Instantaneous and Time-dependent Collapse of Externally Pressurized Circular Tubes with Initial Out-of-Roundness," by D. R. Buttemer, GAMD-8219 Rev., November 6, 1970.