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AEC RESEARCH AND
DEVELOPMENT REPORT

GA-10099
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40-MW(E) PROTOTYPE HIGH-TEMPERATURE
GAS-COOLED REACTOR
POSTCONSTRUCTION RESEARCH AND DEVELOPMENT PROGRAM

MASTER

QUARTERLY PROGRESS REPORT
FOR THE PERIOD ENDING
APRIL 30, 1970

Prepared under
Contract AT(04-3)-314
for the
San Francisco Operations Office
U.S. Atomic Energy Commission

May 28, 1970

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Gulf General Atomic

Incorporated

P. O. Box 608, San Diego, California 92112

AEC RESEARCH AND
DEVELOPMENT REPORT

GA-10099
UC-80 Reactor
Technology

40-MW(E) PROTOTYPE HIGH-TEMPERATURE
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POSTCONSTRUCTION RESEARCH AND DEVELOPMENT PROGRAM

QUARTERLY REPORT SERIES

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GA-9360-February, 1969, through April, 1969
GA-9494-May, 1969 through July, 1969
GA-9797-August, 1969 through October, 1969
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INTRODUCTION AND SUMMARY

The Peach Bottom reactor has remained shut down for the entire quarter while the 450-EFPD (effective full-power day) core component examinations and reactor refueling was continued. The major operations at the reactor site involved the removal of the remainder of the 78 failed fuel elements (which were identified following the 450-EFPD shutdown) and initiation of Core 2 loading.

The fuel element that became lodged in the spent fuel chute January 26 was successfully removed. After the spent fuel elevator was repaired, the fuel element was placed in the spent fuel pit on February 11.

The fuel element in core position B16-10, which was inadvertently dropped into a vacant core location on December 18, 1969, was successfully removed during the quarter. The upper two thirds was removed in March with a special probing tool, and the bottom one-third was removed on April 6 using a specially designed jacking tool.

Failed element removal was completed on April 9. The Division of Reactor Licensing approval of the Core 2 loading document was received in March, and Core 2 loading began on April 10, 1970. By the end of April, 270 Core 2 elements had been loaded at a rate of about 15 elements per day.

New Po-Be neutron sources were placed in the reactor during loading and the reactor was brought critical with 192 Core 2 elements loaded.

Samples of particulate matter in the main loop dust collectors were transferred to a shipping cask and shipped to Gulf General Atomic on March 31,

1970. Thus far they have not been received because of the Teamster strike. When received, they will be analyzed according to test procedure JM-14, "Removal and Analysis of Main Loop Particulate Matter."

During April two reflector elements were cut at the Peach Bottom site and the bottom 15 in. of each (which contain Charpy steel vessel irradiation surveillance specimens) were shipped to Gulf General Atomic on April 28, 1970. These specimens, which were exposed for 450 EFPD, will be used to further study the NDTT shift. The results will be correlated with the specimens exposed 300 EFPD.

CORE 2 LOADING AND PHYSICS TESTING

Core 2 fuel loading started on April 10, 1970. The Type III elements were loaded by April 23, and a total of 270 Core 2 elements were loaded by April 30. The variation of the subcritical multiplication as determined from the startup channel counting rates was about as expected. Experimental values are shown in Fig. 1.

These results showed clearly the reduction in core reactivity resulting from the loading of the type III elements followed by an increase in the reactivity due to the filling up of vacant locations (holes) in the core. The number of holes was constant after 172 elements were loaded and the $1/M^*$ value remained essentially constant during the loading from 172 to 258 elements as had been predicted.

The initial value of $1/M$, normalized to $1-k_{\text{eff}} = 0.13$, was based on the best estimate of the actual core reactivity at the start of loading. The core had 101 holes at the start of loading which was calculated to reduce the fully rodded k_{eff} of the depleted Core 1 by $0.072 \Delta k$.

Two 35-Ci Po-Be sources were inserted into reflector locations on April 17, after the Type III elements were loaded. These sources increased the counting rates of the startup channels from an average of approximately 0.77 cps to about 2.20 cps.

The initial reduction in $1/M$ that occurred when the first six Type III elements were loaded is believed to be due to noise in the startup channel counting rates from the operation of the fuel handling equipment.

*M is the subcritical multiplication.

Fig. 1. Subcritical multiplication for Core 2 loading (36 control rods inserted)

The normalizing count rates ($1/M = 0.13$) were taken with no equipment operating while almost all subsequent counts were taken during loading operations. A noise level corresponding to only 3% or 4% of the counting rate would explain the initial decrease in $1/M$.

During the loading interval from 20 to 60 Type III elements, the number of holes was being reduced in the core and the positive reactivity of filling holes was nearly balanced by the net negative reactivity of exchanging fresh Type III elements with depleted Type III elements.

The loading of the last 22 Type III elements (60 to 82) reduced the core reactivity due to the net negative reactivity of exchanging fresh Type III elements for depleted Type III elements.

The loading of the next 90 elements involved filling vacant locations and the resulting increase in the core reactivity was clearly shown from the $1/M$ plot. Loading from 172 to 258 elements resulted in a very flat $1/M$ response as expected.

A critical rod position of 40 in. withdrawn on rod group 5 was made at 192 elements loaded. This measurement indicated a fully rodded shutdown margin of about $0.106 \Delta\rho$ (well below the technical specification limit of $0.04 \Delta\rho$), and in good agreement with the predicted shutdown margin of $0.112 \Delta\rho$.

Task 3

REACTOR MATERIALS BEHAVIOR

Work this quarter was primarily directed toward studies relating to Tests JM-2 through JM-6 (the 450 EFPD in-vessel examination), Test KM-3, "Performance of Control Rod Guide Tube and Thermally Released Shutdown Rod," and Test LM-1, "Reactor Vessel Steel Irradiation Surveillance." Some work has also been performed on Test JM-19, "Distribution of Fission Products in Fuel Elements" and Test KM-4, "Performance of Control Rod Absorber."

450 EFPD IN-VESSEL EXAMINATION

Between March 31 and April 2, 1970 the third Peach Bottom in-vessel examination was performed. This examination is intended to determine the soundness of fuel elements, reflectors, thermal barrier, and control rod and emergency shutdown rod guide tubes. The examination has been performed after each 150 EFPD of Core 1 operation and includes test procedures JM-2 through JM-6.

TEST JM-2, "IN-VESSEL FUEL INSPECTION"

Test procedure JM-2 covering fuel elements B13-07 and A16-15 was performed on March 31 and April 1, 1970. The results of this examination indicate that the fuel elements that were inspected do not appear to have suffered any visible damage after 450 full-power days of reactor operation.

Vertical markings were observed on the upper spacer ring, upper reflector, and sleeve. It is thought that these markings were caused by adjacent fuel element spacer rings that contacted the inspected element when the elements were removed and reinserted. Also, a circumferential ring was observed in the area of the upper reflector and sleeve joint; several rings were observed on the sleeves, most pronounced in the areas of the spacers; and in addition, a few rings were observed between spacers. It is generally concluded that

these rings were the result of the final machining operation on the sleeve. Both of these elements had been previously examined during the 300 EFPD shutdown for Test JM-2 (see earlier quarterly report GA-9237). No visually detectable change could be noted between the 450 and 300 EFPD examinations. No cracks were evident; such things as dust on the surface, surface erosion, and pitting were not within the resolution of the Peach Bottom viewing device in mode 1. Figures 3.1, 3.2, and 3.3 show typical top, middle, and bottom views of a fuel element taken during the 450 EFPD shutdown examination.

The fuel elements were straight within the limits of the reticle mounted in the viewing device. These limits are estimated to be $\pm 1/2$ in. in 12 ft, the fuel element length.

TEST JM-3, "IN-VESSEL CONTROL ROD GUIDE TUBE INSPECTION"

Test procedure JM-3 relating to the control rod guide tube locations was performed on April 1, 1970. All control rod guide sleeves were inspected to determine whether any were broken or had been displaced above the core. None could be visually observed to be displaced or broken. Figure 3.4 is a view of the F segment, which is typical of the rest of the core.

The top of the core and surrounding side reflector appeared generally to have an observable darker shade than at the 168 and 300 EFPD shutdowns. The dark shading appears to be a thin soot-like deposit that is most noticeable on the tops of the side reflector blocks. Previous analysis of this deposit has shown it to be primarily graphite and alpha iron.

TEST JM-4, "IN-VESSEL HEXAGONAL REFLECTOR INSPECTION"

Test JM-4 pertaining to reflector elements B18-12 and C17-03 was performed March 31, 1970. The reflector elements inspected appear to be sound after 450 days of equivalent full-power irradiation. The line contact marks were observed as in the previous 168 and 300 EFPD examinations; the in-vessel examination revealed nothing new since these reflector elements had been examined during the 300 EFPD shutdown (see earlier quarterly report GA-9237). No

Fig. 3.1. Upper portion of fuel element B13-07; note contact marks caused by adjacent fuel element spacer rings when element B13-07 was lifted

Fig. 3.2. Middle portion of fuel element B13-07; note circumferential rings

Fig. 3.3. Bottom portion of fuel element B13-07

Fig. 3.4. F Segment of the core after 450 EFPD including control rod and emergency shutdown rod guide sleeves; note the dark appearance of the side reflector blocks

cracks of any kind were observed. The reflector elements were thought to be straight within the limits of the reticle mounted in the viewing device (estimated to be $\pm 1/2$ in. in 12 ft, the reflector element length). Typical top, middle, and bottom views of a reflector element taken during the 450 EFPD shutdown are shown in Figs. 3.5, 3.6, and 3.7.

TEST JM-5, "THERMAL BARRIER IN-VESSEL INSPECTION"

Test procedure JM-5 pertaining to the visual inspection of the thermal barrier was performed on April 1 and 2, 1970. For the 450 EFPD inspection, the viewing device was again in the west viewing port in Mode 1 configuration as in the 168 and 300 EFPD examinations (see Ref. 1). During the 168 and 300 EFPD examinations (see Ref. 1), several areas of the thermal barrier were singled out for special attention during subsequent inspections. Because of fabrication discontinuities at these points, they were considered likely to exhibit the first evidence of degradation. One of the areas was the north helium inlet port. As Fig. 3.8 shows, the throat and adjacent wall panels appear to be in excellent condition (see Fig. 24 of Ref. 1 for comparison). Another area was the haunch panels on the west side. Figures 3.9 through 3.11 show that these have not changed significantly from their 300 EFPD appearance (see Figs. 27 and 28 of Ref. 1 for comparison).

The east side wall is shown in Fig. 3.12. It appears to be in good condition (see Fig. 20 of Ref. 1 for comparison). The north access port and adjacent haunch and dome areas are shown in Fig. 3.13. No indications of any changes in appearance from the previous inspection were apparent (see Fig. 25 of Ref. 1 for comparison).

The south fuel loading port and adjacent areas of the dome and haunch are shown in Fig. 3.14. Here again there was no evidence of disturbed or loose strips covering the panel joints; the barrier was considered to be in excellent condition.

In summary, from its appearance the thermal barrier is considered to be in excellent condition and no degradation of any areas was noted during the inspection.

Fig. 3.5. Upper portion of reflector element C17-03; note contact mark caused by adjacent fuel element spacer rings when reflector element C17-03 was lifted

Fig. 3.6. Middle portion of reflector element C17-03

Fig. 3.7. Lower portion of reflector element C17-03

Fig. 3.8. North helium inlet port at AZ 50-deg CCW, SH 030. → 035.

Fig. 3.9. West side haunch panels at AZ 50-deg CCW,
SH 930. → 918.

Fig. 3.10. West side haunch panels at AZ 50-deg CW, SH 940. → 920.

Fig. 3.11. West side haunch panels at AZ 10-deg CCW, SH 930. → 916.

Fig. 3.12. East side-wall above the core

Fig. 3.13. North access port and haunch area at AZ 40- to 50-deg CCW, SH 080. → 100.

Fig. 3.14. South fuel loading port and haunch at AZ 50-deg CW, SH 080. → 090.

TEST JM-6, "FUEL COMPACT DIMENSIONAL CHECK"

Gamma scans of fuel elements B13-07 and A16-15, which were inspected in test JM-2 above, were made according to test procedure JM-6 to determine the fuel compact axial growth during irradiation. The data are currently being reduced.

TEST JM-19, "DISTRIBUTION OF FISSION PRODUCTS IN FUEL ELEMENTS"

The purpose of Test JM-19 is to obtain more information on the behavior of fission products in fuel element graphite under reactor conditions. This information will aid in predicting the amounts of various fission products released from HTGR fuel elements. The information is obtained by determining fission-product concentration profiles in the graphite sleeves, spines, and reflectors of Peach Bottom fuel elements. Currently being studied are the distributions of fission products in fuel element D06-01 and in selected areas of control rod guide tube C08-01.

Fuel Element D06-01

A systematic evaluation of data from failed fuel element D06-01 is being made. The data consist of radial concentrations of fission-product and neutron-induced nuclides at a number of axial positions in the spine, sleeve, and upper reflector; bulk concentrations in the fuel; and axial concentrations in the internal fission-product trap. A number of preliminary observations are discussed below.

At almost all positions along the axis of the element, Cs^{134} profiles in the spine and sleeve parallel Cs^{137} profiles. This appears consistent with rare-gas precursor transport. Strontium-89 and Sr^{90} profiles are less well-matched, but differ in a regular fashion, perhaps as a result of the relatively short half life of Kr^{90} . Assuming a diffusion coefficient of $0.1 \text{ cm}^2 \text{ sec}^{-1}$ and a distance of 0.85 in. the transport time required for gaseous nuclides to traverse the spine is estimated to be 47 sec. This is

comparable to the 33-sec half life of Kr^{90} . Half lives of the rare-gas precursors of Sr^{89} , Cs^{134*} and Cs^{137} are much longer.

Cesium and strontium concentrations are much higher near the surface of the acoustic cavity that runs down the center of the spine than they are in the body of the spine. This may result from a large number of open adsorption sites on this surface and/or deposition following decay of gaseous precursors in the cavity. Hot-cell contamination is a possibility, but unlike the sleeve and spine surfaces discussed below, cavity surfaces were relatively well-protected during disassembly of the element.

Cesium and strontium concentrations at the inside sleeve and outside spine surfaces are high with respect to concentrations in the body of the sleeve. These surfaces were in contact with the fuel compacts and are presumably contaminated with fuel. Concentrations are also high on the outside sleeve surface relative to the body of the sleeve. There is strong evidence that this also is a result of fuel contamination. Concentrations of other fission products decrease rapidly a short distance from the sleeve-fuel interface, exhibit a rather flat profile through the body of the sleeve, and increase sharply at the outside surface. This behavior is common to most of the nuclides that were detected. It might be argued that such profiles represent steady-state transport from the fuel compacts to the outside of the sleeve where increased adsorption sites might exist. However, at the operating temperatures of the reactor, it is unlikely that such high concentrations of volatile cesium, for example, could exist at that surface. The presence of Pa^{233} at the surface is consistent with fuel contamination. Fuel contamination of the outside sleeve surface could have occurred during operation, or during subsequent disassembly and sectioning of the element in the hot cell.

A comparison of $\text{Sr}^{89}/\text{Sr}^{90}$ ratios in fuel element DO6-01 calculated from average concentrations at various places in the element is given below:

* Cs^{134} is not a fission product; it is produced by the reaction $\text{Cs}^{133}(n,\sigma)\text{Cs}^{134}$. The precursor is Xe^{133} .

Location	Ratio
Fuel	0.09
Fuel-spine interface	0.13
Fuel-sleeve interface	0.13
Internal trap	0.16
Outside sleeve surface	0.32

The trend of values is consistent with the transport of Sr^{89} and Sr^{90} being defined, at least partially, by their krypton precursors. The high value at the outside sleeve surface indicates a combination of fuel contamination and transport of the kryptons from the fuel compact through the sleeve with a lag period of the order of the half life of Kr^{90} .

Gamma counting and radiochemical analysis of graphite from the lower region of D06-01 were completed this month. No further work in obtaining fission-product concentration profiles in D06-01 graphite will be performed unless required for the interpretation of existing data discussed above.

Control Rod Guide Tube C08-01

Two graphite cylinders were cut from near the top and bottom, respectively, of control rod guide tube C08-01. The cylinders were sectioned radially and the resulting samples gamma counted. Fission-product concentration profiles in the graphite near the inlet and outlet of the core have been determined. Tentative results show slightly higher Cs^{134} concentrations in the graphite near the inlet rather than the outlet of the core. The Cs^{137} concentrations were too low to be determined by gamma counting except at surfaces exposed to the coolant. Radiochemical analysis of selected samples for $\text{Sr}^{89,90}$ and Cs^{137} has been completed. Except for interpretation of data, no other work on C08-01 is planned.

TEST KM-3, "PERFORMANCE OF CONTROL ROD GUIDE TUBE AND THERMALLY RELEASED SHUTDOWN ROD"

Control Rod Guide Tube

The examination of control rod guide tube C08-01 was continued during this report period in the Gulf General Atomic hot cell. The outside diameter of the upper and lower guide tube measured at 12-in. intervals was within the design dimensions and tolerance of 3.545 ± 0.002 in. The upper tube end was also within manufactured tolerance and dimension of 3.558 ± 0.002 in. The over-all length was $147\text{--}1/16$ in., which was within the manufactured tolerance of $147 \pm 7/64$ in. Six contact spots equally spaced around the outside diameter were visible on the surface of the guide tube at $57\text{--}5/8$ in., $95\text{--}7/16$ in., and $142\text{--}11/16$ in. above the bottom end of the tube where fuel elements made contact with the guide tube. Wear at the contact spots, which was undetectable by micrometer, was estimated to be less than 0.001 in. of surface recession. A photograph of a typical contact spot is shown in Fig. 3.15. The dark, soot-like deposit around each wear spot was very adherent and appeared to outline regions where the axial coolant flow was disrupted by the contact between fuel elements and guide tube.

The axial bow in the guide tube was measured and determined to be at a maximum (0.122 in.) about 6 ft from the bottom end. At 90° around the circumference, the maximum axial bow was 0.032 in., about 11 ft from the bottom of the tube. Since the maximum design tolerance for bow along the entire guide tube assembly was 0.070 in., it seems likely that some permanent bow of the guide tube took place in the reactor.

The guide tube was cut into six sections (see Fig. 3.16) for further examination. The bottom two sections are shown in Fig. 3.17. The sections containing the latching lugs showed no sign of wear or indication of malfunction. The other section shown in Fig. 3.16 was retained for fission-product analysis (see Test JM-19, which appears earlier in this task).

K303-4

Fig. 3.15. Upper tube cap of control rod guide tube C08-01 after 300 EFPD showing black deposit at point of contact with an adjacent fuel element. Six similar areas were equally spaced around the circumference of the rod at $57\frac{5}{8}$ in., $95\frac{7}{16}$ in., and $142\frac{11}{16}$ in. above the bottom end

Fig. 3.16. Control rod guide tube and adjacent fuel elements showing location of radial cuts made in the guide tube

K303-24

Fig. 3.17. Sections 1 and 2 (see Fig. 3.16) cut from the bottom portion of control rod guide tube C08-01. No damage or wear was detected in the outer portion of the guide tube

The central section (No. 3 in Fig. 3.16) of the guide tube was slit longitudinally on opposing sides and the inside surface examined. No wear or material deposits were observed on the inside surface. A typical example of the condition of the inner surface is shown in Fig. 3.18, at the termination of the grooves machined into the inner guide tube walls to control the position of the thermally released shutdown rod in the core after release.

The upper section of the guide tube contained the thermally released shutdown rod, which was disconnected by cutting through the stainless steel rod at the retaining nut in the tip of the guide tube. The thermally released shutdown rod was then removed by sliding it out the bottom of the tube. A photograph of the upper guide tube and the boronated-graphite thermally released shutdown rod is shown in Fig. 3.19.

A 4-in.-long section was cut from the guide tube (No. 5 in Fig. 3.16) and retained for fission-product analysis (see Test JM-19, this Task). The remaining portion of the rod was slit longitudinally and examined. The inner surface showed no signs of wear or material deposits. A photograph of the opened upper guide tube and the tip of the guide tube is shown in Fig. 3.20. The threaded joint between the upper and lower guide tube, visible in Fig. 3.20, showed no effects of irradiation. A high-magnification photograph of the cross section of the joint is shown in Fig. 3.21.

Thermally Released Shutdown Rod

The thermally released shutdown rod consisted of a boronated-graphite absorber containing 30 wt-% B in the form of B_4C and held in the reactor in such a manner that an over temperature condition would cause the absorber to be dropped into the core by gravity. The metal components of the thermally released shutdown rod assembly were covered with a thin, black adherent coating, but otherwise appeared unchanged from the manufactured condition. The boronated graphite absorber also appeared unchanged, but the graphite (HLM-85) absorber support had a single radial crack as shown in Fig. 3.22.

K303-33

Fig. 3.18. Inner surface of lower guide tube C08-01 showing bottom of grooves cut in the inner surface to guide thermally released shutdown rod and control its final position in core. No evidence of wear or damage was detected on the inner surface of the guide tube

M32669-4

Fig. 3.19. Top portion of guide tube (left) with thermally released shutdown rod removed (right). Metallic components of shutdown rod were covered with thin, black deposit. No damage or unusual features were observed in the boronated graphite absorber; however, graphite absorber support had radial crack (see Fig. 3.22)

M32669-1

Fig. 3.20. Inner surface of guide tube showing joint between upper and lower guide tube. Thermally released shutdown rod was held in the smooth walled section. No damage or wear was detected

K303-39

Fig. 3.21. Cross section of threaded joint between upper and lower guide tube. Locking pin and carbonaceous cement can be seen in the joint. No irradiation effects were detected

K303-29

Before Test

K303-46

After Test

Fig. 3.22. Lower portion of thermally released shutdown rod showing radial crack observed in HLM-85 graphite absorber support. The crack probably developed during reactor operation as a result of thermal expansion mismatch and inadequate gap between support and stainless steel weldment. No increase in damage took place during release test

The fusible link (an alloy of 85% Ag - 15% Mn) had a relatively thick, dark scale on the surface that could be easily scraped off. Otherwise, the link was unchanged from the manufactured condition. Photographs of the fusible link in place in the stainless steel weldment are shown in Fig. 3.23.

The release temperature for the thermally released shutdown rod was determined experimentally in the hot cell in a furnace apparatus shown in Fig. 3.24. The apparatus held the shutdown rod vertically in a helium-purged tube furnace while the fusible link and surrounding components were heated. Temperature was monitored by a millivolt potentiometer and a chromel/alumel thermocouple, which was touching the stainless steel weldment within about 1/4 in. of the fusible link. The shutdown rod assembly could be heated slowly to the release temperature while the temperature of the link was measured continuously. When released, the rod dropped to the bottom of the apparatus and made contacts that caused a bell to ring. There was an audible sound as the rod struck the bottom of the apparatus and the cooler portion of the rod fell into position around the thermocouple, which caused the indicated temperature to drop rapidly. The maximum temperature before these events was taken as the release temperature.

Prior to heating the shutdown rod from location C08-01, a complete unirradiated thermally released shutdown assembly weighing 5.9 lb was heated in the apparatus to 1743°F in 2-3/4 hr, when release took place. A new fusible link was installed, and the assembly was heated a second time. Release was achieved in 2-1/4 hr, when the indicated temperature reached 1740°F. The results of these two experiments were in good agreement with the reported FHSR (Final Hazards Summary Report) melting point of the fusible link which was $1745 \pm 3^\circ\text{F}$. From the results of the experiments and consideration of the experimental procedure, the accuracy of the temperature measurements was estimated at $\pm 5^\circ\text{F}$.

The irradiated thermally released shutdown rod from location C08-01 was loaded into the apparatus and heated until at the end of 2 hr the indicated temperature reached 1740°F and the rod was released. The release temperature was therefore set at $1740 \pm 5^\circ\text{F}$. The rod was undamaged by the experiment. A photograph of the rod after the experiment is shown in Fig. 3.25.

K303-23

(a)

K303-25

(b)

M32493-1

(c)

4X

Fig. 3.23. Fusible link in thermally released shutdown rod of guide tube C08-01: (a) over-all view of link and lower portion of rod; (b) close view of link showing scratches in the scale formed on the link; (c) an unused link as cast from braze alloy (85% Ag-15% Mn)

M32669-2

Fig. 3.24. Furnace apparatus used to determine release temperature of thermally released shutdown rod. Unirradiator shutdown rod is at right with holding bracket attached. The unirradiator absorber was made of two sections having a total length equal to that of the rod in C08-01

M32669-3

Fig. 3.25. Thermally released shutdown rod after release temperature experiment

After the thermal release experiments, the boronated absorber was removed from the stainless steel weldment and the dimensions of components were taken. The results of this work and the deviation from design values, if any, are shown in Table 3.1. Determination of the outside dimensions of the snubber, tube weldment, and absorber support was of primary importance in order to ensure that there was no interference and that an adequate gap existed between the shutdown assembly and the guide tube.

The absorber support diameter showed the greatest deviation from design, probably as a result of a crack through the wall mentioned above and shown in Fig. 3.22. The crack in the absorbersupport probably resulted from inadequate clearance and a thermal expansion mismatch between the graphite and the stainless steel weldment that passed through the support and the absorber. At the operating temperature of 900°F, the maximum inside diameter of the support would be about 0.002 in. less than the minimum outside diameter of the stainless steel weldment. The resulting minimum circumferential tensile strain in the support (about 0.2%) would be sufficient to cause failure. This explanation is consistent with the observation that after heating of the unirradiated assembly to 1743°F during preliminary work on determination of the rod release temperature, a previously unbroken absorber support was found to have broken in the same manner as the one shown in Fig. 3.22.

Support of the boronated graphite absorber resting on the ring was not affected by the crack, which was parallel to the compressive load, and the forces experienced during the release test did not cause further damage (see Fig. 3.22). During service in the reactor, the crack in the absorber support opened slightly and, at 90° to the crack, the outside diameter of the support was 0.005 in. greater than design maximum. The resulting decrease in the minimum clearance gap between the guide tube and the support (0.066 in. on opposed sides) was relatively small (about 3%). Results of a detailed analysis of the consequences of a cracked absorber support on the operation of the system indicate that it should cause no adverse effect on the safe operation of the reactor.

Table 3.1
DIMENSIONS OF THERMALLY RELEASED SHUTDOWN ROD COMPONENTS
FROM GUIDE TUBE C08-01 AFTER 300 EFPD

Component	Position in Component	Outside Diameter				Inside Diameter				Length	
		Minimum		Maximum		Minimum		Maximum		Length	
		Inches (± 0.001)	Deviation from Design Limits (%)	Inches (± 0.001)	Deviation from Design Limits (%)	Inches (± 0.001)	Deviation from Design Limits (%)	Inches (± 0.001)	Deviation from Design Limits (%)	Inches (± 0.001)	Deviation from Design Limits (%)
Absorber (boronated graphite, 30 wt-% B)	Top	2.234	-0.3	2.238	-0.1	1.198	---	1.200	---	19.151	0
	Middle	2.238	-0.1	2.238	-0.1	(a)	---	(a)	---		
	Bottom	2.237	-0.1	2.240	0	(a)	---	(a)	---		
Absorber support ^b (HLM 85 graphite)	Across lobes	2.688	0	2.693	+0.2	(a)	---	(a)	---	(a)	-
	Across flats	2.318	0 ^c	2.328	+0.3 ^c	(a)	---	(a)	---		
Tube weldment, at snubber (stainless steel)	Across lobes	2.684	0	2.687	0	(a)	---	(a)	---	(a)	-
	Across flats	2.315	0 ^c	2.316	0 ^c	(a)	---	(a)	---		
Snubber	Across lobes	2.683	0	2.684	0	(a)	---	(a)	---	(a)	-
	Across flats	2.324	+0.1 ^c	2.325	+0.1 ^c	(a)	---	(a)	---	(a)	

^aNot determined.

^bThe absorber support had a single radial crack through the wall. The maximum diameter was measured at 90° around the circumference from the crack.

^cAssumes ± 0.005 tolerance distance from center.

No irradiation-induced dimensional change was expected in the thermally released shutdown absorber, because the fast-neutron exposure was not significant. The fast-neutron fluence and the operating temperature of the thermally released shutdown rod is shown in Fig. 3.26. (The fast fluence and operating temperature of the control rod absorbers is also shown in Fig. 3.26.)

Samples of the tube weldment and the boronated graphite absorber are being prepared for metallographic examination. Upon completion of that work, hot-cell work concerning the examination of control rod guide tube C08-01 will be finished.

TEST LM-1, "REACTOR VESSEL STEEL IRRADIATION SURVEILLANCE"

The irradiation-damage surveillance program was initiated to provide a means of assessing the continuing serviceability of the Peach Bottom reactor vessel. In accordance with the Technical Specifications, Charpy impact specimens prepared from the vessel's plate and plate welds were placed in regions of high neutron flux to obtain integrated doses of radiation equal to or greater than that experienced by the vessel.

A total of three sets of irradiation-surveillance samples have been previously tested. The results of these tests, which are described in more detail in earlier quarterly reports (GA-9797 and GA-9898), are summarized in Table 3.2.

Charpy Specimens from Reflector B17-03

During this quarter, further irradiation surveillance impact specimens from reflector B17-03 were tested. These specimens, which represented parent material, were exposed in Peach Bottom for 300 EFPD at a temperature of 660° to 690°F. Analysis of accompanying dosimetry wires showed that these specimens had received a fast-neutron fluence of $7.63 \times 10^{17} \text{ n/cm}^2$ ($E > 1 \text{ MeV}$).

Fig. 3.26. Neutron absorber material operating temperature and neutron exposure profile after 300 EFPD in Peach Bottom HTGR core position C08-01

Table 3.2

RESULTS OF PREVIOUSLY REPORTED CHARPY IMPACT IRRADIATION SURVEILLANCE SPECIMENS
(300 EFPD Exposure)

Sample Location	Material Represented	Irrad. Temp. (°F)	Fluence (E > 1 Mev)	Orig. NDTT (°F)	Final NDTT (°F)	Apparent Shift (°F)
Nozzle access plug	Weld	500-633	1.3×10^{16}	-56	-46	+10
	Parent	500-633	1.3×10^{16}	-60	-40	+20
Reflector B16-01	Weld	660-690	9.7×10^{17}	-56	+6	+62

The results of the impact tests on these specimens are shown in Fig. 3.27, and photographs of the fracture faces of the specimens are shown in Fig. 3.28. In addition, as a calibration exercise, a set of unirradiated control specimens were broken at the same time as the above tests, using the same equipment and techniques as were used for the irradiated samples. The results of the latter calibration tests are shown in Fig. 3.29, together with a photograph of the specimen fracture faces (Fig. 3.30).

These results indicate that the NDTT of the irradiated parent metal samples has shifted by approximately 45°F. This shift is greater than would be expected for these exposure conditions, on the basis of the published literature. However, the results are generally consistent with those previously obtained on the weld metal specimens from reflector B16-01.

Plans are currently under way for testing of additional irradiation surveillance samples to further study this NDTT shift. In this regard, two reflectors exposed for 450 EFPD have been cut at the Peach Bottom site, and the bottom 15 in. (which contain Charpy specimens) are currently enroute to Gulf General Atomic. Samples of both weld and parent material are included.

References

1. Hildebrand, J. F., "Inspection of Thermal Behavior in the Peach Bottom Reactor Vessel at 168 and 300 EFPD," USAEC Report GAMD-8212, Gulf General Atomic Incorporated, March 31, 1969.

Fig. 3.27. Results of Charpy V-notch impact tests on irradiation surveillance specimens of parent metal (plate 2B) removed from Peach Bottom reflector B17-03 after 300 EFPD. Numbers on specimens refer to Fig. 3.28

Fig. 3.28. Fracture faces of parent metal Charpy V-notch impact test specimens from reflector B17-03 removed from Peach Bottom after 300 EFPD. Numbers under photographs refer to test points on Fig. 3.27

Fig. 3.29. Results of Charpy V-notch impact tests on unirradiated control specimens from parent metal (plate 2B). Numbers on specimens refer to Fig. 3.30

Fig. 3.30. Fracture faces of unirradiated Charpy specimens from plate 2B. Numbers under photographs refer to test points identified on Fig. 3.29

Task 5

DETERMINATION OF THE EFFECTIVENESS OF THE HELIUM PURIFICATION SYSTEM

Emphasis on this task during the quarter was placed on further analysis of samples from diffusion probe No. 3, further analysis of potential sources of hydrogen in the reactor, and initiation of the preparation of several topical reports summarizing Core 1 data for various Task 5 test procedures. Because of the continuing 450 EFPD shutdown, no measurements of fission-product release or low-level chemical impurities were made during the quarter.

TEST JM-1, "GASEOUS FISSION PRODUCT RELEASE MEASUREMENTS"

No gaseous-fission-product release measurements were made this quarter because of the continuing 450 EFPD shutdown which began on October 3, 1969. However, a topical report summarizing Core 1 gaseous-fission-product release data is being prepared that will include release data from Tests DM-1 (30% power), GM-1 (initial measurements at 100% power), and JM-1, and iodine release data from Test JM-8 (iodine decay studies).

TEST JM-14, "REMOVAL AND ANALYSIS OF MAIN LOOP PARTICULATE MATTER"

Particulate matter is periodically removed from the main-loop dust collectors at Peach Bottom and examined at Gulf General Atomic. The purpose is to characterize the particulate matter found in the primary coolant circuit and to determine its role in the transport of fission products.

The primary circuit contains two bypass cyclone dust collectors, one in each loop. About 1% of the primary coolant flow is continually passed through these collectors. The dust has consisted primarily of construction debris (pieces of metal, fibrous wool-like material, weld spatter, and fine black dust). Tables 5.1 and 5.2 compare the weights, carbon content, and specific activities of particulate matter samples collected to date.

Table 5.1
PARTICULATE MATTER REMOVED FROM MAIN LOOP DUST COLLECTORS

	July 1967	August 1967	January 1968		October 1968	
Loop	1	2	1	2	1	2
Weight, g	21	11	6	22	60	56
Carbon content, wt-% ^a	10	13	28	25	<10	(b)

^a-Balance construction debris.

^b-Not measured.

Table 5.2
ANALYSIS OF PARTICULATE MATTER LOOP NO. 1 DUST COLLECTOR

Nuclide	mCi/g		
	July 1967	January 1968	October 1968
Co ⁶⁰	9.5×10^{-4}	6.0×10^{-3}	2.5×10^{-2}
Sr ⁸⁹	1.4×10^{-4}	0.15	3.7
Sr ⁹⁰	6×10^{-6}	2.9×10^{-3}	0.16
Zr ⁹⁵	(a)	0.35	2.2
Cs ¹³⁷	1×10^{-6}	1.1×10^{-2}	9.5×10^{-2}
Ce ¹⁴⁴	$< 3 \times 10^{-5}$	0.16	4.2
Pa ²³³	(a)	≤ 0.43	23

^aNot detected.

The carbon content has ranged from 10% to 30%. Construction debris has not decreased as had been expected.

The total weights of particulate matter are small for a power reactor with complex gas circuitry. The concentration of dust in the primary coolant has been approximately 10 $\mu\text{g/lb He}$. This compares with 0.1 to 1 $\mu\text{g/lb He}$ in GAIL. GAIL, however, had a full-flow filter and was a much less complex system.

The specific activity of the dust has increased considerably with time, mainly because of failure of fuel elements. Nuclides such as Zr^{95} and Pa^{233} indicate the presence of fuel material released from the elements. The October 1968 dust contained a small number of recognizable fuel particle coating fragments, 0.16 g of thorium and 0.027 g of U^{235} and U^{233} . This is only 10^{-7} fraction of the fuel in the core. Activation products such as Cr^{51} , Mn^{54} , Fe^{59} , Co^{60} , and $\text{Ag}^{110\text{m}}$ have not increased markedly in specific activity.

Particulate matter collected during the 150 EFPD period of operation ending October 3, 1969, was removed from the main loop dust collectors in March and shipped to Gulf General Atomic for analysis. Receipt has been delayed by the Teamsters' strike. Results of earlier analyses are being summarized in a topical report now being prepared.

TEST JM-15, "DIFFUSION PROBE EXPERIMENT"

The purpose of Test JM-15 is to provide information on the behavior of condensable fission products in the Peach Bottom primary coolant system. Fission products and induced activities in the loop can be identified, plateout concentrations and partial pressures determined, and some information regarding chemical form obtained. A better understanding of plateout processes and activity distributions can be of value in safety, repair, routine maintenance, decontamination, and waste disposal considerations.

Calculations of partial pressures and release fractions for condensable fission products in the Peach Bottom primary circuit during the last 150 EFPD of Core I operation are nearly complete. Fractions of various fission products released from the core during that period (when 78 fuel elements failed) are in the range 10^{-7} to 10^{-3} . Failed fuel elements account for nearly all of the fission-product activity in the primary circuit. Diffusion probe data are being compiled for inclusion in a summary report.

TEST JM-16, "SURVEILLANCE OF LOW-LEVEL CHEMICAL IMPURITIES"

No measurements of chemical impurities have been made in the reactor since the shutdown in October 1969. Studies have continued at Gulf General Atomic on determining possible sources of impurities that have persisted in the main coolant at steady state. In particular, the source of hydrogen is under consideration.

Graphite outgassing and hydrocarbon or oil cracking have been proposed as possible sources of hydrogen. Analyses of core graphite for hydrogen before and after irradiation in Peach Bottom have been performed and yield some information on these two speculative sources. Results of analyses are given in Table 5.3.

The original hydrogen content of the sleeve graphite from fuel element D06-01 was 20 ppm, or a total of $1/2$ lb H_2 . (Reflector and fuel compact carbon contributed an additional estimated 6 lb H_2 .) The amount of hydrogen remaining in the sleeve graphite after 300 EFPD is about 0.3 lb (based on a 12-ppm average). It is estimated that at least 6 lb of hydrogen has been removed from the main coolant (based on a constant 10 vpm for 7600 hr). Although additional analyses will be performed on both sleeve and reflector material, these preliminary data suggest that graphite outgassing probably contributes very little to the total H_2 impurity content in the loop He.

An additional conclusion that may be drawn from these data is that little or no oil cracking has occurred on this element.

Table 5.3
ANALYSIS OF HYDROGEN IN SLEEVE GRAPHITE FROM FUEL
ELEMENT D06-01 AFTER 300 EFPD EXPOSURE

Sample	Reactor Conditions			H_2^a Content (ppm)
	Temp (°F)	Neutron Fluence (10^{20} nvt)		
		<2.38 eV	>0.18 MeV	
H-I	1250-1380	13.1	16.5	17.3
H-II	1250-1380	13.1	16.5	14.3
N-I	780-840	9.6	11.2	8.9

H_2^a content prior to irradiation: 20 ppm.

The amount of hydrogen observed, approximately 12 ppm, is in reasonable agreement with literature data on hydrogen sorption on graphite at these temperatures. Furthermore, the thermal decomposition of oil on graphite sleeve surfaces would lead to hydrogen-rich coke deposits that are reasonably stable at fuel element temperatures. It may be speculated, therefore, that oil decomposition on sleeve surfaces would lead to hydrogen concentrations somewhat higher than those observed.

Appendix A
PROJECT PERSONNEL

This postconstruction research and development program on the 40-Mw(e) prototype HTGR is being carried out by the following project personnel:

C. L. Allen	R. J. Lansley
E. E. Anderson	W. L. Lefler
W. E. Bell	P. T. Mattson
G. E. Besenbruch	J. W. McLean
J. R. Brown	R. J. Nirschl
R. D. Burnette	R. E. Norman
D. D. Busch	R. Riley
G. Buzzelli	D. I. Roberts
H. R. W. Cobb	W. J. Scheffel
B. P. Cross	A. S. Schwartz
R. C. Dahlberg	R. H. Smith
J. N. Graves	O. M. Stansfield
A. S. Hilbert	L. L. Stewart
L. J. Hull	K. R. Van Howe
J. F. Hildebrand	F. E. Vanslager
M. E. Kantor	W. H. Weitzel
R. K. Lane	G. L. Wessman
	P. Winchell