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GULF GENERAL ATOMIC

AEC RESEARCH AND
DEVELOPMENT REPORT

GA-10400

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

QUARTERLY PROGRESS REPORT
FOR THE PERIOD ENDING
OCTOBER 31, 1970

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

November 30, 1970

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

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Gulf General Atomic Project 166

November 30, 1970

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POSTCONSTRUCTION RESEARCH AND DEVELOPMENT PROGRAM
QUARTERLY REPORT SERIES

GA-5700-Summary Progress Report for the Period Ending July 31, 1964
GA-5924-August, 1964, through October, 1964
GA-6134-November, 1964, through January, 1965
GA-6406-February, 1965, through April, 1965
GA-6647-May, 1965, through July, 1965
GA-6829-August, 1965, through October, 1965
GA-6995-November, 1965, through January, 1966
GA-7232-February, 1966, through April, 1966
GA-7426-May, 1966, through July, 1966
GA-7595-August, 1966, through October, 1966
GA-7830-November, 1966, through January, 1967
GA-7988-February, 1967, through April, 1967
GA-8153-May, 1967, through July, 1967
GA-8370-August, 1967, through October, 1967
GA-8500-November, 1967, through January, 1968
GA-8679-February, 1968, through April, 1968
GA-8816-May, 1968, through July, 1968
GA-9080-August, 1968, through October, 1968
GA-9237-November, 1968, through January, 1969
GA-9360-February, 1969, through April, 1969
GA-9494-May, 1969 through July ,1969
GA-9797-August, 1969 through October, 1969
GA-9898-November, 1969 through January, 1970
GA-10099-February, 1970, through April, 1970
GA-10300-May, 1970 through July, 1970

INTRODUCTION AND SUMMARY

As reported in the previous quarterly report (GA-10300), the Peach Bottom Atomic Power Station reached full power on July 14, 1970 and remained at essentially this level until September 18, when it was shut down to repair a leaking dump valve bonnet on the No. 2 steam drum. This scheduled shut-down terminated the longest sustained power operation of the reactor at 66 days. The electrical and thermal power histories for this quarter are given in Figs. 1 and 2. During this exposure of Core 2, the system activity remained at approximately 0.3 Ci, showing excellent fuel performance. The main loop activity level for this quarter is shown in Fig. 3.

After the plant was restarted on September 23, a high steam generator bottom head temperature was noted, and power operation was limited to about 78%. It was concluded that the bottom head cooling unit damper on the No. 1 steam generator was malfunctioning. On October 1 the plant was shut down to effect repairs on this unit and to correct an increasing helium leak that was attributed to an incompletely closed valve in the helium purification system. This maintenance was readily accomplished, and the plant was returned to full power on October 4 where it remained until October 26.

During electrical testing on October 26, the 220-kV line breaker was inadvertently opened. The turbine-generator tripped on overspeed because of the rapid loss of load. The turbine-generator set went into the coast-down mode and house-load shedding and transfer occurred as programmed by the electrical protection system. The motor-driven boiler feed pump was shut down and both steam generators were supplied water from the turbine driven boiler feed pump. However, because of the changeover and the drum level control action in the transient, the No. 2 loop automatically isolated on low drum level.

Fig. 1. Thermal power history for period

Fig. 2. Electrical power history for period

Fig. 3. Main loop activity history for period

This incident provided an excellent example of the ability of the plant to withstand significant transients and recover quickly without damage. Within 3 hr of the trip, the plant was resynchronized to the line and full-power operation was recommenced.

On October 28 plant load was reduced while maintenance was done on the Freon refrigeration system supplying the low-temperature delay beds of the helium purification system. During this time the delay bed warmup raised the primary coolant activity to slightly over 1 Ci, but this was reduced to approximately 0.3 Ci on October 29, when normal system operation was achieved. The plant stayed at full power through the end of the quarter.

CORE 2 LOADING AND PHYSICS TESTING

The analysis of the Core 2 rise to power physics testing has been completed. The results indicate an overall reactor performance very close to that predicted. Figure 4 shows the major plant parameters recorded during the rise to power.

CORE TEMPERATURE CORRELATIONS

From the Core 1 physics testing and analysis, it was determined that the core average fuel element temperature for any operating condition could be simply expressed by the following equation:

$$\bar{T}_{\text{fuel}} = T_{\text{inlet}} + C_1 (\text{Power}) + C_2 \left(\frac{\text{Power}}{\text{Flow}} \right) .$$

This expression appeared to be valid for individual fuel elements in the core if the local power densities ($P/\bar{P}$) of the individual elements were factored into the two constants C_1 and C_2 . This equation has also been used to compare the predicted and measured Core 2 temperatures. In Core 2, the location of the radial traverse of the thermocouples is not a typical radius. Figure 5 shows the location of Core 2 thermocouples with a representative control rod pattern and a power distribution along the "E" radius. Figures 6 through 10 show measured temperatures along the "E" radius at several power levels during the rise to power. The presence of inserted control rods along the "E" radius results in the measured radial temperature profile deviating from the average profile. Temperatures measured along this radius must be corrected (by the above equation) for the ratio of the power density in the element containing the thermocouple to the core average power density at the same radius. The resultant experimental radial temperature traverse is then volume weighted to obtain the core average temperature.

Fig. 4. Rise to power

Fig. 5. Rod pattern and power distribution along E radius

Fig. 6. Radial temperature profile spine/fuel interface

Fig. 7. Radial temperature profile spine/fuel interface

Fig. 8. Radial temperature profile spine/fuel interface

Fig. 9. Radial temperature profile spine/fuel interface

Fig. 10. Radial temperature profile spine/fuel interface

The measured (corrected) average fuel temperatures and those predicted during the rise to power are shown in Fig. 11. The agreement is very good.

CONTROL ROD WORTHS

When reactor power rises to a point at which core temperatures are measurably increased, sustained period measurements can no longer be used for calculating control rod worths. During the rise to power, therefore, the Gulf General Atomic model R-20 reactivity computer was used for determining differential rod worths. The reactivity computer accepts input from the power range instrumentation and generates an essentially instantaneous output of core reactivity. A record of core reactivity and control rod position was obtained from strip recorders from which differential rod worths were determined. Two methods were used: in the single-bump method, rod positions and reactivity changes were recorded during the rod withdrawals necessary to increase reactor power; in the double-bump method, rod withdrawals were followed immediately by rod insertions to the original position. At higher reactor powers, the double bumps were performed with rod insertions followed by withdrawals. A double-bump rod calibration tends to eliminate the effects of core temperature changes during the rod motions, and hence they were weighted more heavily than single-bump data when fitting differential rod worth curves.

In the calibration of rod group 8, differential worths calculated with the reactivity computer overlapped data previously obtained by period measurements. As can be seen in Figs. 15 through 17, the agreement between the two methods is quite good.

The calculated differential rod worths for all rods measured during the startup testing are shown in Figs. 12 through 26. These curves were integrated to obtain the integral worth curves shown in Figs. 27 through 31. In general, the measured rod worths agreed well with the predicted worths, as shown in Table 1. However, group 5 was approximately 6% higher and group 1 was about 9% lower than expected. The total reactivity change for the five groups measured was within $0.003 \Delta\rho$ of that predicted.

Fig. 11. Average fuel temperature during Core 2 rise-to-power

Fig. 12. Rod 6A differential worth

Fig. 13. Rod 6B differential worth

Fig. 14. Rod 6C differential worth

Fig. 15. Rod 8A differential worth

Fig. 16. Rod 8B differential worth

Fig. 17. Rod 8C differential worth

Fig. 18. Rod 5A differential worth

Fig. 19. Rod 5B differential worth

Fig. 20. Rod 5C differential worth

Fig. 21. Rod 7A differential worth

Fig. 22. Rod 7B differential worth

Fig. 23. Rod 7C differential worth

Fig. 24. Rod 1A differential worth

Fig. 25. Rod 1B differential worth

Fig. 26. Rod 1C differential worth

Fig. 27. Rod group 6, integral worth versus inches withdrawn, total worth = 0.02746 $\Delta\rho$

Fig. 28. Rod group 8, integral worth versus inches withdrawn, total worth = 0.02160 $\Delta\rho$

Fig. 29. Rod group 5, integral worth versus inches withdrawn, total worth = 0.02976 $\Delta\rho$

Fig. 30. Rod group 7, integral worth versus inches withdrawn, total worth = 0.02272 $\Delta\rho$

Fig. 31. Rod group 1, integral worth versus inches withdrawn, total worth = 0.01454 Δp

TABLE 1
ROD GROUP INTEGRAL WORTHS

Rod Group	Expected ($\Delta\rho$)	Measured ($\Delta\rho$)	Ratio	Core 1 Worth
6 ^(a)	0.02899	0.02746	0.947	0.0253
8	0.02244	0.02160	0.963	0.0217
5	0.02817	0.02976	1.06	0.02738
7	0.02335	0.02272	0.973	0.0226
1 ^(a)	<u>0.01605</u>	<u>0.01454</u>	0.906	0.01617
Total	0.1190	0.1161	--	--

(a) Measured over part of the rod stroke.

TEMPERATURE COEFFICIENT MEASUREMENTS

The temperature defect during the rise to power was obtained from calibrated control rod position and average core temperature. Reactivity corrections for xenon buildup during the rise to power was obtained, and Fig. 32 shows the measured temperature defect and that predicted by a two-dimensional diffusion calculation. The total temperature defect measured agrees quite closely to that predicted for Core 2.

The derivative of the temperature defect curve shown in Fig. 33 was taken to obtain the temperature coefficient, shown in the figure. There was some scatter in calculated derivatives, and Fig. 33 represents a polynomial fit to the calculated derivatives of the measured data on the temperature defect. The measured temperature coefficient is in reasonable agreement with predictions from the dimensional diffusion calculations, as indicated on Fig. 33.

XENON BUILDUP MEASUREMENTS

Xenon reactivity was followed by calibrated rod motion after attaining full power operation. Figure 34 shows the xenon reactivity as a function of time, at full power. A prediction of the full power xenon reactivity from basic data, with no adjustment for Core 1 xenon measurements, yields 0.0215 $\Delta\rho$. This predicted xenon reactivity is about 8% lower than the calculated (measured) xenon reactivity of 0.0233 $\Delta\rho$, obtained in the startup testing.

Fig. 32. Temperature defect versus fuel temperature, no xenon, beginning of life Core 2

Fig. 33. Temperature coefficient versus temperature

Fig. 34. Xenon buildup, Core 2 beginning of life

TASK 3
REACTOR MATERIALS BEHAVIOR

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.

The transport of Sr-89 and Sr-90 in Peach Bottom fuel element D06-01 graphite has been shown to be consistent with a fission-gas (Kr) deposition model described by the following equation:

$$\ln\left(\frac{C}{C_o}\right) = -\left(\frac{\lambda}{D}\right)^{1/2} x, \quad (1)$$

where x is the penetration distance, C is the concentration of the Sr isotope at distance x , C_o is the surface concentration, λ is the decay constant of the Kr precursor, and D is the diffusion coefficient. That is, a plot of the log of the measured strontium concentration versus distance results in a straight line. Furthermore, the measured concentration ratios

$$\frac{\ln(C/C_o)_{\text{Sr-89}}}{\ln(C/C_o)_{\text{Sr-90}}}$$

are equal to the expected value for the square root of the ratio ($\lambda_{\text{Kr-89}}/\lambda_{\text{Kr-90}}$). A diffusion coefficient of krypton in the graphite matrix can be calculated from Eq. (1) and experimental data similar to that presented in an earlier quarterly report, GA-9797, p. 33. The calculated D is about

$10^{-3} \text{ cm}^2/\text{sec}$, 1/200 of the interdiffusion coefficient D_{12} expected for Kr in He at 23.8 atm and spine temperature. The fact that the coefficient is small can be attributed to the graphite pore structure, i.e., tortuosity and orificing effects.

The Cs data do not fit this model for precursor release. As discussed in earlier quarterly report GA-10300, these data indicate that the movement of Cs itself controls the Cs profiles. It should be noted that the diffusion of both Cs and Sr is very slow, however.

A limited analysis of fuel element D13-05 is planned for early next year in order to amplify and confirm the conclusions drawn from the examination of element D06-01. This element was exposed for 300 equivalent full-power days of reactor operation and D13-05 for 452 equivalent-full-power days. Detailing planning of disassembly and sampling operations for the hot-cell examination of the element has begun.

TASK 5

DETERMINATION OF EFFECTIVENESS OF HELIUM PURIFICATION SYSTEM

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

Monitoring of gaseous-fission-product release was resumed with the Core 2 startup. Core 2 average Kr-85m release (R/B) values decreased slightly initially and then levelled out at about 2.3×10^{-4} ; the data are reported in Table 2. Release values for the various krypton and xenon isotopes show an approximate square-root-of-half-life dependence.

Samples are taken by Philadelphia Electric Co. and counted, and the gamma spectrometry data are recorded on punched paper tapes. The tapes are forwarded along with other necessary data to Gulf General Atomic, where spectral analysis and R/B calculations are performed with the aid of the Sigma 2 computer and PB R/B code.

A comparison of Kr-85m R/B data for Core 1 and Core 2 shows that initial release values began at approximately the same level, but as expected, Core 2 values have not shown the increase evident at a comparable point in Core 1 operation. The data are compared in Fig. 35.

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

Samples of particulate matter removed from the main loop dust separators following the 452 EFPD shutdown of Core 1 were received at Gulf General Atomic last quarter. These samples will be analyzed early next year. The purpose is to characterize particulate matter found in the primary circuit and to determine its role in the transport of fission products.

TABLE 2
PEACH BOTTOM CORE 2
R/Bs DETERMINED FROM MAIN PURGE (A-301) SAMPLES

Date	Time	Total Reactor Thermal Energy MW-hr	(%) Full Power	R/B						
				Kr-85m 4.4h	Kr-88 2.8h	Kr-87 76m	Kr-89 3.2m	Xe-133 5.27d	Xe-135 9.2h	Xe-138 17m
7-15-70 ^a	1103	2,974	92	3.0×10^{-5}	3.1×10^{-6}	1.8×10^{-6}	1.3×10^{-6}	b	b	2.0×10^{-6}
7-16-70 ^a	1030	5,510	92	2.7×10^{-5}	4.9×10^{-6}	3.2×10^{-6}	2.1×10^{-6}	b	5.9×10^{-5}	1.7×10^{-5}
7-17-70	1535	8,028	92	4.7×10^{-4}	2.3×10^{-4}	1.4×10^{-4}	2.0×10^{-5}	b	4.0×10^{-4}	4.4×10^{-5}
7-18-70	1135	10,543	92	4.4×10^{-4}	2.0×10^{-4}	1.1×10^{-4}	1.4×10^{-5}	b	3.2×10^{-4}	3.7×10^{-5}
7-20-70	1143	15,620	92	2.7×10^{-4}	1.3×10^{-4}	1.2×10^{-4}	1.9×10^{-5}	1.6×10^{-3b}	2.8×10^{-4}	3.4×10^{-5}
7-20-70	1448	15,620	92	2.5×10^{-4}	1.3×10^{-4}	1.2×10^{-4}	1.6×10^{-5}	1.8×10^{-3b}	2.8×10^{-4}	3.0×10^{-5}
7-21-70	1405	18,161	92	3.2×10^{-4}	1.6×10^{-4}	1.2×10^{-4}	1.7×10^{-5}	1.8×10^{-3b}	2.7×10^{-4}	3.3×10^{-5}
7-22-70	1415	20,688	92	3.1×10^{-4}	1.6×10^{-4}	1.2×10^{-4}	2.2×10^{-5}	1.7×10^{-3b}	2.6×10^{-4}	3.3×10^{-5}
7-24-70	1112	25,733	94	3.1×10^{-4}	1.5×10^{-4}	1.1×10^{-4}	1.5×10^{-5}	1.7×10^{-3}	2.5×10^{-4}	2.8×10^{-5}
7-25-70	0800	28,263	94	2.8×10^{-4}	1.3×10^{-4}	1.0×10^{-4}	8.6×10^{-6}	1.8×10^{-3}	2.6×10^{-4}	2.3×10^{-5}
7-27-70	1442	32,684	93	2.4×10^{-4}	1.2×10^{-4}	1.1×10^{-4}	1.8×10^{-5}	1.5×10^{-3}	2.3×10^{-4}	3.0×10^{-5}
8-03-70	1050	51,126	95.7	2.5×10^{-4}	1.2×10^{-4}	9.3×10^{-5}	1.3×10^{-5}	1.2×10^{-3}	2.0×10^{-4}	2.4×10^{-5}
8-10-70	1015	69,545	94.7	2.5×10^{-4}	1.2×10^{-4}	9.3×10^{-5}	1.4×10^{-5}	1.2×10^{-3}	2.1×10^{-4}	2.8×10^{-5}
8-18-70	0900	90,566	94.5	2.1×10^{-4}	1.1×10^{-4}	9.6×10^{-5}	1.0×10^{-5}	1.2×10^{-3}	2.0×10^{-4}	1.9×10^{-5}
8-31-70	1335	122,600	95.0	2.2×10^{-4}	1.0×10^{-4}	8.6×10^{-5}	1.2×10^{-5}	9.8×10^{-4}	1.8×10^{-4}	2.4×10^{-5}
9-11-70	0935	152,450	95.0	2.3×10^{-4}	1.1×10^{-4}	8.8×10^{-5}	1.1×10^{-5}	1.2×10^{-3}	2.0×10^{-4}	2.3×10^{-5}
9-15-70	1439	162,930	95.0	2.4×10^{-4}	1.2×10^{-4}	9.3×10^{-5}	1.2×10^{-5}	1.2×10^{-3}	1.9×10^{-4}	2.4×10^{-5}
Average	8/3 - 9/15		95	2.3×10^{-4}	1.1×10^{-4}	9.2×10^{-5}	1.2×10^{-5}	1.2×10^{-3}	2.0×10^{-4}	2.4×10^{-5}
9-25-70	1410	178,400	82.1 ^c	1.6×10^{-4}	7.7×10^{-5}	6.0×10^{-5}	9.2×10^{-6}	7.9×10^{-4b}	1.3×10^{-4}	1.7×10^{-5}
9-30-70	1341	188,000	84.4 ^c	1.8×10^{-4}	8.7×10^{-5}	6.8×10^{-5}	8.9×10^{-6}	8.1×10^{-4b}	1.4×10^{-4}	1.9×10^{-5}

^a Questionable results.

^b Not at equilibrium.

^c Note lower power.

Fig. 35. Kr-85m R/B data for Core 1 and Core 2

Radiation monitors opposite the two main loop dust collectors show that during Core 2 operation there has been relatively little activity associated with particulate matter in the primary coolant gas.

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

Core 2 primary coolant chemical impurity levels are well below the technical specification limits. For example, after about a month of operation, the measured impurity concentrations for CO, CH₄, and N₂ were 1.2, 0.8, and 1.4 ppm by volume, respectively.

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