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GENERAL ATOMIC

DIVISION OF **GENERAL DYNAMICS**

AEC RESEARCH AND
DEVELOPMENT REPORT

GA-4512

MASTER

40-MW(E) PROTOTYPE HIGH-TEMPERATURE
GAS-COOLED REACTOR

RESEARCH AND DEVELOPMENT PROGRAM

QUARTERLY PROGRESS REPORT
FOR THE PERIOD ENDING MARCH 31, 1963

Contract AT(04-3)-314
U.S. Atomic Energy Commission

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GENERAL ATOMIC
DIVISION OF
GENERAL DYNAMICS

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HTGR QUARTERLY REPORT SERIES

- GA-1235—October, November, December, 1959
(including a summary of work between
January 1, 1959, and September 30, 1959)
- GA-1378—January, February, March, 1960
- GA-1640—April, May, June, 1960
- GA-1774—July, August, September, 1960
- GA-1982—October, November, December, 1960
- GA-2204—January, February, March, 1961
- GA-2493—April, May, June, 1961
- GA-2747—July, August, September, 1961
- GA-2861—October, November, December, 1961
- GA-3132—January, February, March, 1962
- GA-3396—April, May, June, 1962
- GA-3676—July, August, September, 1962
- GA-4247—October, November, December, 1962

INTRODUCTION

This report covers the work performed by General Atomic Division of General Dynamics Corporation under U.S. Atomic Energy Commission Contract AT(04-3)-314 during the period from January 1, 1963 through March 31, 1963. This contract calls for research and development specifically directed toward the construction of a 40-Mw(e) prototype power plant employing a high-temperature, gas-cooled, graphite-moderated reactor known as the HTGR.

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TASK II. DEVELOPMENT OF REACTOR CORE AND REFLECTOR

2.1. NUCLEAR DESIGN AND THEORETICAL PHYSICS

Reactivity-lifetime Studies

As pointed out in the previous quarterly progress report, some alteration in the specification for the initial core loading was required in order to achieve the desired core life. The major changes from the previous specification are

1. A reduction in the total thorium loading from 1,652.2 kg to 1,450.0 kg.
2. A reduction in the initial rhodium loading from 8.0 kg to 5.0 kg.
3. A reduction in the number of fuel elements containing compacts with high rhodium concentration from 198 to 54.
4. A reduction in the total graphite inventory in the active core from 21,331 kg to 20,506 kg.
5. Inclusion of U^{234} and U^{236} impurities in the fully enriched uranium.

With these changes, the specification for the initial core loading is

Th ²³²	1,450.0 kg
U ²³⁴	3.54 kg
U ²³⁵	220.0 kg
U ²³⁶	1.18 kg
U ²³⁸	11.46 kg
Rh ¹⁰³	5.0 kg
Boron	1.10 kg
Carbon	20,506.0 kg

The over-all C:Th:U²³⁵ ratios are 1727:5.44:1.0 for the central region, and 3087:22.7:1.0 for the outer row of fuel elements.

These total fuel loadings are zoned as shown in Figs. 2.1 and 2.2. Fig. 2.1 is an elevation view of the core. In the heavy rhodium region the initial rhodium concentration is three times as great as it is in the light rhodium region. There is no initial rhodium in the remaining regions. In the region from 36.1 cm to 130.47 cm, 60 of the fuel elements contain 18.3 g of burnable poison (boron) in each of their spines. The fuel elements in the outer row (heavy thorium, light uranium region) have 53% of the uranium concentration of the other fuel elements and an increased thorium concentration. Fig. 2.2 is a plane view of the core giving the location of each type of fuel element.

Fig. 2.1--Elevation view of core showing zoning

Fig. 2.2--Plan view of core showing location of each type of fuel element

A graph of the hot unrodded multiplication throughout the core life with this specified initial loading is given in Fig. 2. 3.

Control-rod-worth Studies

The dependence of the control-rod worth on the boron content in the rods has been investigated using the GAMBLE code in XY geometry. In addition, the decrease in the rod worth with life due to the burnout of the boron in the rod has been obtained using one-dimensional FEVER depletion calculations. The maximum worths of a rod in the central control ring at the beginning of life with various amounts of boron and at the end of life with both various amounts of initial boron and various exposures are given in Table 2. 1.

Table 2. 1
DEPENDENCE OF ROD WORTH ($\Delta\rho$) ON BORON
CONTENT AND EXPOSURE

Initial Boron (wt-%)	Beginning-of- life Worth	End-of-life Worth		
		0-Day Exposure	500-Day Exposure	900-Day Exposure
5	0. 00721	0. 00800	0. 00735	0. 00655
10	0. 00807	0. 00896	0. 00874	0. 00854
30	0. 00965	0. 0107	0. 0107	0. 0107

The initial boron content in the six control rods in the central ring has been reduced from 30 wt-% to 10 wt-%. This reduces their worth by 16% of their previous worth with no serious depletion due to exposure, gives an improved power distribution throughout life, and reduces the magnitude of the rod withdrawal accident at the end of life.

2. 2. REACTOR ENGINEERING ANALYSIS

Thermocouple Specification

The metallurgical examination of the thermocouple cladding materials which had been subjected to a 1000-hr high-temperature test has been completed. Although all of the materials showed some degree of carburization, the niobium coating over a molybdenum rod appeared to offer the best possibility for satisfactory thermocouple lifetime. Based on these results, the specifications for the fuel element thermocouples were written. These specifications call for the thermocouples in the active fuel region to have tungsten-5% rhenium versus tungsten-26% rhenium wires, magnesium oxide

Fig. 2.3--Hot unrodded multiplication throughout core life

insulation, a molybdenum sheath, and an outer cladding of niobium. The thermocouples in the lower reflector and internal trap are to be normal Chromel-P-Alumel type.

Fission-product Release Analysis

In order to anticipate the behavior of the fission-product sampling system, an estimate has been made of the activity in this system as a function of time. These activities, which are listed in Table 2.2, were calculated using the reference design release curve and the reference design sleeve release fraction of 2×10^{-4} . The preliminary data from the GAIL III-B experiment have indicated that the shape of the fission-product release curve can be somewhat reasonably approximated by this reference design release curve with the height of the curve increasing throughout life. With this assumption, a rough estimate of the activity values throughout life is as follows: At 1 month after reactor startup, the activities will be 0.1% of the values listed below; at 4 months, the fraction will be up to 1%; at 8 months up to 10%; and at end of life, it will possibly reach only 50% of these values. The reactor is, of course, designed for approximately 20 times these coolant activities and 3 times these purge activities.

Table 2.2
ANTICIPATED ACTIVITY IN SYSTEM

Isotope	Coolant Activity ($\mu\text{c}/\text{cm}^3$)*	Purge Activity ($\mu\text{c}/\text{cm}^3$)*
Kr ^{85m}	0.0072	54
Kr ⁸⁵	0.12	0.13
Kr ⁸⁷	0.022	210
Kr ⁸⁸	0.021	160
Kr ⁸⁹	0.024	1700
Xe ¹³⁵	0.017	120
Xe ¹³⁸	0.044	810

* At 700°F and 350 psia.

Fast-neutron Damage and Element Bowing

Deformation of fuel and reflector elements due to thermal and fast-neutron flux gradients was re-evaluated this period on the basis of more recent graphite damage data and more sophisticated fast-neutron flux distribution analysis. The following element locations were analysed:

1. Fuel element adjacent to an inserted control rod.

2. Fuel element adjacent to the core-reflector interface.
3. All three rows of hexagonal side reflector elements.

At each of the locations, a BEAF code analysis of deformation and stress was made for a new element subjected only to thermal gradients, then for the element at the end of core-life exposure to fast neutrons with the same thermal gradient, and, finally, for the element at the end of life with residual neutron damage only.

Core restraint reactions and other fast-neutron contraction effects were analyzed also.

2.3. EXPERIMENTAL ENGINEERING

Design modifications in the acoustic thermometer fuel element to improve the acoustic signal were completed this period, and the test fuel element assembly was modified to reflect these changes.

The test element with a revised wave guide was tested this period with the modified standoff assembly. Testing was conducted at design helium pressure, at temperatures up to approximately 1800°F, and with purge flow rates up to 2 lb/hr. The signal was satisfactory in all cases. All associated instrumentation duplicated the Peach Bottom readout system.

TASK IV. DEVELOPMENT OF REACTOR EQUIPMENT

4. 1. CONTROL-ROD DRIVES

During this quarter the test program on the prototype emergency-shutdown rod and drive indicated that the dry-film lubricant used on critical components of the device considerably lowered friction. A test was completed in which the emergency-shutdown-rod drive was subjected to 1000 strokes in rapid succession under battery power. On completion of the test there was no deterioration of the lubricant, and the frictional drag, which was low at the start of the test, was even lower.

The Acme screw friction test rig was reassembled and purged to reproduce the high-purity helium environment of the reactor. The wear test was then resumed to test the friction and wear compatibility of the two nitrided "Nitr alloy 135 Modified" materials. Test results showed considerable improvement over the previously tested Meehanite ductile cast iron against Nitr alloy. Neither of these combinations, however, approached the performance obtained with Stellite 6B against Nitr alloy.

Early results of the O-ring seal testing, which is underway at simulated reactor temperatures and pressures, indicated that Thiokol rubber is still the best among the elastomers tested. Of the plastics, Nylon 6 appears most promising. Testing is being continued to determine the long term effects of temperature and pressure on the sealing ability of the most promising elastomer and plastic materials.

4. 2. FUEL-HANDLING EQUIPMENT

Routine and demonstration testing of the fuel transfer machine was nearly completed during the quarter. The machine was tested with full-size dummy fuel elements assembled on a grid plate representing a one-sixth segment of the Peach Bottom core. In one test to demonstrate that the machine can clear the fuel elements from a large volume of the core, 23 elements were successfully removed at one time from the core segment and were later replaced satisfactorily.

TASK VI. DEVELOPMENT OF PLANT AUXILIARY SYSTEMS

6. 1. FISSION-PRODUCT CONTROL AND PRIMARY LOOP SYSTEMS

GAIL III-B Krypton and Xenon Release Data

Cycle 39 Data. During this period one additional set of krypton and xenon fission-product release data has been worked up from the period of cycle 39 (sample taken late in November, 1962). These data were delayed due to the fact that insufficient sampling data had been reported with the activity data. It was necessary to return to recorder charts for that period and search out the required data.

These data show the general trend of an increasing release rate versus time, but with a decreasing rate of increase with time. Duplicate samples were taken and are shown in Table 6. 1 for an accumulated power of 196. 4 Mw-hr of loop operation.

Table 6. 1
FISSION-PRODUCT RELEASE AT 196. 4 MW-HR

Isotope	Release Fraction	
	Sample 1	Sample 2
Kr ^{85m}	2.33×10^{-3}	3.40×10^{-3}
Kr ⁸⁷	1.22×10^{-3}	1.59×10^{-3}
Kr ⁸⁸	7.16×10^{-4}	7.80×10^{-4}
Kr ⁸⁹	2.53×10^{-4}	7.05×10^{-4}
Kr ¹³³	2.97×10^{-3}	3.68×10^{-3}
Xe ¹³⁵	7.71×10^{-4}	9.22×10^{-4}
Xe ¹³⁸	1.36×10^{-4}	1.51×10^{-4}

Kr⁸⁵ Release. Late in October, 1962, the liquid-nitrogen-cooled charcoal trap (No. 3) was heated up and the trapped materials were flushed into the hot helium storage tank. A sample was taken from this gas at that time and allowed to decay. This sample was recently analyzed for Kr⁸⁵ content and the release fraction calculated. The activity was collected on the trap for a period of two reactor cycles, 35 and 37 (52. 01 Mw-hr), before being flushed into the hot-helium storage tank.

The release fraction value obtained from this analysis was 1.51×10^{-2} which appears about a factor of ten low when data obtained for shorter-lived krypton isotopes are extrapolated to give an expected Kr^{85} release. The sampling method is rather crude, however, and some error can be expected in the results.

Distribution of Fission Products in the GAIL III-A Element

Fission-product Concentration Profiles. Analyses of Cs^{137} in the GAIL III-A sleeve and spine samples have been completed and the longitudinal profiles of the sleeve and spine have been plotted in Figs. 6.1 and 6.2. The dark data points shown in Fig. 6.1 are from analyses of corings taken as whole samples, whereas the light data points are the averages of radial slices taken from the corings. These curves are quite similar to those obtained in the laboratory diffusion tests. The data indicate little Cs^{137} in the spine and sleeve (less than 1/2%) and a negligible quantity in the internal trap.

Radial concentration profile curves obtained from the analysis of Cs^{137} in corings taken from the GAIL III-A spine showed a Cs^{137} concentration of between 10^{-5} and 10^{-3} mg of isotope per gram of graphite with a relatively flat profile through the spine.

It was decided not to do any further analyses for Ba^{140} , because the decay corrections necessary are enormous and may not give valid results. It was further decided to dispense with Sr analysis at the present time in order to place all effort in the Cs^{137} analyses. Spot analyses have been made on the spine for Ce^{144} and Zr^{95} , but these data have not yet been put in their final form.

Analysis for I^{129} . I^{129} has been determined in two of the fuel particles from the GAIL III-A compact and in samples from the internal trap. The I^{129} content of the two fuel particles was found to be 0.01 and 0.02 μg . A sample from the internal trap (presumed to be from the top section) was found to contain 0.003 μg of I^{129} per gram of trap material. Samples of the trap material taken from the bottom and mid-point of the trap were found to have virtually no I^{129} .

The method used for this determination incorporates leaching and extraction of the I^{129} with iodide carrier followed by neutron activation and counting of the extracted iodide. I^{129} is a long-lived fission product ($t_{1/2} = 2 \times 10^7$ years) with an appreciable thermal-neutron cross section (20 barns) and is therefore readily determined by neutron-activation analysis. Iodide carrier is added to the sample. The iodine is removed from the sample by double leaching in an alkaline cyanide solution. Silver iodide is soluble by decantation. Silver carrier is added to the solution

Fig. 6.1--Ba¹⁴⁰, Sr⁸⁹, Cs¹³⁷ concentrations in sleeve as a function of position (GAIL III-A)

Fig. 6.2-- Cs^{137} concentration in spine as a function of position (GAIL III-A)

and acidified causing the precipitation of silver iodide. The precipitate is washed and dissolved with alkaline-cyanide solution, and silver iodide is again precipitated and washed. The silver is reduced to the metal upon addition of zinc dust and H_2SO_4 , thus:

This mixture is transferred to a separatory funnel through a filter to remove the metals. Iodide is oxidized to iodine with peroxide and extracted into xylene. The iodine is then reduced and stripped from the organic phase.

The separated iodine was irradiated in the TRIGA reactor along with an I^{129} standard (obtained from ORNL) at 250 kw for 2-1/2 hr. The solutions were counted the next morning for I^{130} ($t_{1/2} = 12.6$ hr). Natural iodine is monoisotopic, having a mass number of 127. Neutron activation of the iodide carrier leads to formation of I^{128} ($t_{1/2} = 25$ min), which is ultimately used for the yield determination in this method. The I^{130} was counted after the I^{128} had decayed. The amount of I^{129} found in the fuel particles (0.01 and 0.02 μg) is approximately the quantity expected based on an average U^{235} content and burnup for other particles. The detectable limit by this technique is 0.5 ngm (where an ngm = 10^{-9} grams) based on the above irradiation parameters.

An attempt to repeat the analysis on three other particles led to questionable results. The I^{129} determination will be repeated simultaneously along with Cs^{137} , U^{235} , and Zr^{95} in two other particles.

Distribution of Cs^{137} in GA-309 Capsules

Analyses on the remaining graphite parts (homogenized by grinding and mixing) of capsules GA-309-4 and GA-309-8 were completed during this period for Cs^{137} . For capsule GA-309-4 the total release from the fuel body was 22.2%; for capsule GA-309-8 it was 28.0%. These values for total release from the fuel bodies are somewhat higher (about two orders of magnitude) than the estimated values for the GAIL III-A fuel element. Analyses are now being performed on samples of the GA-309-4 and GA-309-8 fuel bodies to determine the amount of Cs^{137} retained in order to check the validity of the values presented here. (GAIL III-A fuel-particle analyses indicate that essentially all of the Cs^{137} is retained by the fuel particle.)

TASK VIII. OVER-ALL PLANT ANALYSIS

8.1. HAZARDS ANALYSIS

The RAT code computation of core temperatures following loss of forced circulation indicated that seven of the thermally released absorbers would drop almost instantly after loss of flow and six more would drop in a half minute. The RAT code emergency-cooling accident calculation averaged temperatures in large lumps and did not take into account the initial local reduction in temperature due to coolant flow within the control-rod guide sleeves. During this quarter a detailed calculation of temperatures within the control-rod guides has been completed. The results indicate that the fusible alloy in the hottest location will not melt for six to eight minutes following loss of coolant flow, with scram either instantaneous or delayed 25 seconds. This appears to be very satisfactory and removes concern about the need for raising the alloy melting point in the central seven or 13 rods to prevent inadvertent dropping of fusible absorbers.

Appendix I

GENERAL ATOMIC REPORTS ISSUED ON THE HTGR PROJECT
FROM JANUARY 1, 1963, THROUGH MARCH 31, 1963

FORMAL REPORTS

- GA-2318 A PROCESS FOR MAKING GRAPHITE LESS PERMEABLE TO GASES AT HIGH TEMPERATURES, G. R. Tully, Jr. and B. F. Disselhorst, February 1, 1963.
- GA-3069 ANALYSIS OF ACTIVATION MEASUREMENTS OF Th^{232} RESONANCE CAPTURES IN THE PEACH BOTTOM (40-MW(E) PROTOTYPE HTGR) CRITICAL ASSEMBLY, J. B. Sampson, September 14, 1962.
- GA-3125 EXPERIMENTAL TECHNIQUES FOR ACTIVATION MEASUREMENT OF RESONANCE INTEGRALS, M. K. Drake and J. R. Brown, January 21, 1963.
- GA-3197 RADIATION STUDIES OF GRAPHITE-MATRIX FUEL BODIES CONTAINING $(\text{Th}, \text{U})\text{C}_2$ FUEL PARTICLES, Walter V. Goedel, November 16, 1962.
- GA-3422 MEASURED DOPPLER COEFFICIENT OF THORIUM DISPERSED IN GRAPHITE, J. R. Brown and J. B. Sampson, February 6, 1963.
- GA-3599 FISSION-PRODUCT RELEASE FROM $(\text{Th}, \text{U})\text{C}_2$ GRAPHITE FUELS, L. R. Zumwalt, E. E. Anderson, and P. E. Gethard, December 4, 1962.
- GA-3799 CORRELATION OF CALCULATIONS AND EXPERIMENTAL MEASUREMENTS MADE USING A PARTIAL CORE MOCKUP (CRITICAL ASSEMBLY) OF THE PEACH BOTTOM HTGR, J. R. Brown, M. K. Drake, and K. R. Van Howe, March 6, 1963.

INFORMAL REPORTS

- GAMD-2221 (Rev.)   PROCEDURES FOR THE PREPARATION OF GRAPHITE MATRIX FUEL COMPACTS CONTAINING CARBON COATED (Th, U)C₂ FUEL PARTICLES, W. V. Goeddel, August 15, 1962.
- GAMD-2605 (Rev. 2)   PRIMARY COOLANT SYSTEM BLOWDOWN ACCIDENT FOR PEACH BOTTOM HTGR, J. A. Landoni and M. L. Skinner, December 15, 1962.
- GAMD-2638   HELIUM INLEAKAGE THROUGH POROUS-WALLED FUEL ELEMENTS, R. F. Turner, November 21, 1961.
- GAMD-3378   HTGR IN-PILE LOOP EXPERIMENT ACTIVITY ANALYSIS GAIL III - A FUEL ELEMENT, D. R. Lofing and L. R. Zumwalt, September 28, 1962.
- GAMD-3779   VIEW FACTORS FOR HEAT RADIATION IN HTGR CORE CONFIGURATION, J. A. Landoni, December 10, 1962.
- GAMD-3977   PEACH BOTTOM HTGR EMERGENCY COOLING ACCIDENT TEMPERATURES AND THERMAL RELEASE OF NUCLEAR POISONS AND RADIOACTIVE NUCLIDES, J. A. Landoni and T. R. Moffette, January 30, 1963.

Appendix II

PROJECT PERSONNEL

This research and development program on the prototype HTGR is being carried out by the following:

<u>Assistant Director of the Laboratory</u>	Duffield, R. B.
<u>Project Manager</u>	Born, M. R.
Administrative Assistant	Ledbetter, R. J.
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<u>Reactor Section</u>	
Fuel-handling Equipment	Nicoll, D.
Control Rod and Control-rod Drive	Ledin, E.
<u>Fuel Element Design</u>	Turner, R. F. Vanslager, F. E.
<u>Engineering Analysis Section</u>	Rockenhauser, W. Hunt, P. S.
<u>Chemistry Section</u>	Zumwalt, L. R. Anderson, E. Busch, D. D. Riedinger, A. B.
<u>Hazards Analysis</u>	Moffette, T. R. Landoni, J. A.
<u>Physics Section</u>	Stewart, H. B. Merrill, M. H. Van Howe, K. R. Nordheim, L. W. (part time)
<u>Metallurgy</u>	Wallace, W. P.
<u>Experimental Operations and Operations Analysis</u>	
Operating Procedures	Paget, J. A.

In-pile-loop Experiments

Capsule Irradiation

Drafting and Graphic Arts

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Kemper, J. S.
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Turner, R. F.
Baumgartel, R.

Engle, G.

Dorion, G. O.

Knell, S. W.

Ringsmuth, R. J.