175
7-21-77

GENERAL ATOMIC

GA-A14380
UC-77

PUBLIC SERVICE COMPANY
OF COLORADO 330-MW(e)
HIGH-TEMPERATURE GAS-COOLED
REACTOR RESEARCH AND
DEVELOPMENT PROGRAM

MASTER

QUARTERLY PROGRESS REPORT
FOR THE PERIOD ENDING
MARCH 31, 1977

Prepared under
Contract E(04-3)-633
for the San Francisco Operations Office
U.S. Energy Research and Development Administration

DATE PUBLISHED: APRIL 1977

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Energy Research and Development Administration, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights.

Printed in the United States of America
Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road
Springfield, Virginia 22151
Price: Printed Copy \$3.50; Microfiche \$3.00

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

GENERAL ATOMIC

GA-A14380
UC-77

PUBLIC SERVICE COMPANY
OF COLORADO 330-MW(e)
HIGH-TEMPERATURE GAS-COOLED
REACTOR RESEARCH AND
DEVELOPMENT PROGRAM

QUARTERLY PROGRESS REPORT
FOR THE PERIOD ENDING
MARCH 31, 1977

Prepared under
Contract E(04-3)-633
for the San Francisco Operations Office
U.S. Energy Research and Development Administration

NOTICE
This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Energy Research and Development Administration, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

GENERAL ATOMIC PROJECT 1900

DATE PUBLISHED: APRIL 1977

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

QUARTERLY REPORT SERIES

GA-6830-July, 1965, through September, 1965
GA-6950-October, 1965, through December, 1965
GA-7086-January, 1966, through March, 1966
GA-7314-April, 1966, through June, 1966
GA-7453-July, 1966, through September, 1966
GA-7634-October, 1966, through December, 1966
GA-7939-January, 1967, through March, 1967
GA-8038-April, 1967, through June, 1967
GA-8270-July, 1967, through September, 1967
GA-8420-October, 1967, through December, 1967
GA-8600-January, 1968, through March, 1968
GA-8725-April, 1968, through June, 1968
GA-8879-July, 1968, through September, 1968
GA-9130-October, 1968, through December, 1968
GA-9261-January, 1969, through March, 1969
GA-9440-April, 1969, through June, 1969
GA-9720-July, 1969, through September, 1969
GA-9875-October, 1969, through December, 1969
GA-10010-January, 1970, through March, 1970
GA-10202-April, 1970, through June, 1970
GA-10313-July, 1970, through September, 1970
GA-10444-October, 1970, through December, 1970
GA-10560-January, 1971, through March, 1971
Gulf-GA-A10754-April, 1971, through June, 1971
Gulf-GA-A10850-July, 1971, through September, 1971
Gulf-GA-A10980-October, 1971, through December, 1971
Gulf-GA-A12030-January, 1972, through March, 1972
Gulf-GA-A12200-April, 1972, through June, 1972
Gulf-GA-A12325-July, 1972, through September, 1972
Gulf-GA-A12477-October, 1972, through December, 1972
Gulf-GA-A12559-January, 1973, through March, 1973
Gulf-GA-A12667-April, 1973, through June, 1973
Gulf-GA-A12733-July, 1973, through September, 1973
Gulf-GA-A12822-October, 1973, through December, 1973
GA-A12951-January, 1974, through March, 1974
GA-A13058-April, 1974, through June, 1974
GA-A13167-July, 1974, through September, 1974
GA-A13285-October, 1974, through December, 1974
GA-A13422-January, 1975, through March, 1975
GA-A13528-April, 1975, through June, 1975
GA-A13997-July, 1976 through September, 1976
GA-A14270-October, 1976 through December, 1976

ABSTRACT

This quarterly report is a continuation of the series describing the research and development program for the Public Service Company of Colorado 330-MW(e) High-Temperature Gas-Cooled Reactor. The work reported here includes preliminary results of the rise-to-power program.

INTRODUCTION

The objective of the research and development program work reported here is to develop and verify the information required to design, construct, operate, and maintain the Public Service Company of Colorado power plant as provided in U.S. Energy Research and Development Administration Contract E(04-3)-633.

Part I of this series of reports has included the work described in Appendix B of the contract and Part II covered the work described in Appendix K of the contract on the steam generator. Only Task 2 of Part I has had reportable work during this quarter and is included herein.

By agreement with SAN and RRD, the R&D monthly report under this contract has been discontinued and any matters previously covered in the monthly report will now be covered in the R&D Quarterly Progress Report for Contract E(04-3)-633.

CONTENTS

ABSTRACT	iii
INTRODUCTION	v
TASK 2: NUCLEAR ANALYSIS	1
Operation	1
Fuel Accountability	1
Fission Product Release from Buffer Elements During Initial Cycles	1
References	6

TABLES

1. Kr-85m R/B for 16 buffer elements as computed by SURVEY . . .	4
2. Column-average metallic release for 16 buffer elements as computed by TRAFIC	5

TASK 2 NUCLEAR ANALYSIS

OPERATION

The reactor operated at about 225 MW(t) (27%) and 70 MW(e) (21%) for the month of January. The plant was shutdown during February and most of March for various maintenance operations including tightening the circulator seals and repair of purification train valves. Operation recommenced on March 19 and a power of 210 MW(t) and 50 MW(e) had been attained by the end of the quarter.

FUEL ACCOUNTABILITY

At the end of the first quarter of 1977, the reactor has accumulated 401,013 MW(t) hr of operation, equivalent to 19.8 full power days. A total of 21.2 kg (46.7 lb) of government-owned leased U-235 has been consumed, and 13.4 kg (29.5 lb) of U-233 have been bred from the thorium (assuming full decay of Pa-233).

FISSION PRODUCT RELEASE FROM BUFFER ELEMENTS DURING INITIAL CYCLES

The FSV initial core design employs a special fuel blend, high in thorium content, used in the four rows of fuel rods nearest the side reflector. This thorium-rich fuel blend makes possible a power density in these peripheral fuel rods which does not exceed the power density in other fuel elements in peripheral regions. However, the depletion code GAUGE (Ref. 1) does not explicitly model this "buffer" fuel blend. The GAUGE model uses the average fuel loading in peripheral elements. The buffer blends are assumed to be mixed with fuel in the remainder of the peripheral blocks. Therefore, the fission cross section used in the

buffer fuel is too high. This modeling yields local power density, temperature, and, on occasion, fission product release estimates which are unrealistically high. In the more detailed BUGTRI model of the FSV core, the buffer fuel is represented explicitly (Ref. 1). Therefore, more realistic performance estimates are obtained when the BUGTRI power histories are used than when GAUGE is the source of the power distributions.

Comparable burnup calculations for the first three cycles of FSV core operation covering 2 years using both GAUGE and BUGTRI are available and were described in the previous quarterly report (Ref. 2). During this quarter comparisons have been made of calculated fission product release from the FSV core using, as input to TRAFIC (Ref. 3) and SURVEY (Ref. 4), power distributions from these GAUGE and BUGTRI 2-year depletion analyses. These analyses are well suited for such comparisons because the GAUGE calculations were used to determine critical rod positions for the subsequent BUGTRI calculations. Since power distributions calculated by these two methods are in good agreement except in the buffer fuel, the thermal and fission product calculations were limited to buffer elements.

Analyses were performed for the following 16 columns:

<u>Region</u>	<u>Columns</u>	<u>Region</u>	<u>Columns</u>
21	3,4	28	5
22	3	29	2
23	6	30	6,7
24	5	31	7
26	7	34	7
27	5,6	35	4
		37	2

All calculations were done at the bottom of layer 3 only where the highest fuel temperatures are encountered. The following fission product release comparisons were made for GAUGE versus BUGTRI input:

1. Kr-85m R/B versus time - all 16 elements.
2. Sr-90 column-average release - all 16 elements.
3. Sr-90 release from region 31, column 7 - all local points.

A comparison of the BUGTRI and GAUGE Kr-85m R/B values is shown in Table 1. As can be seen, the ratio between BUGTRI and GAUGE results decreases with increasing time. At zero days the difference in the R/B values is due solely to temperature effects on the R/B of the as-manufactured fuel. As burnup proceeds, however, fuel particle coating failure increasingly contributes to the R/B. Since BUGTRI analyses show lower failure rates, the BUGTRI/GAUGE ratio decreases with time.

Table 2 shows the strontium column-average release at 2 years for the two input cases. The total reduction obtained is a factor of 9 for Sr-90 release. One would expect a large reduction factor for strontium because strontium release is known to be quite sensitive to fuel element temperature (Ref. 5). The factor of 9 decrease in Sr-90 release, while large, does not have a great impact on the total FSV strontium inventory because the amount of strontium directly released to the coolant will be small compared to that produced by Kr-90 gaseous precursor decay.

Calculations were also made of Sr-90 release from all local points of region 31, column 7, a location which is expected to experience high local fuel temperatures. The total reduction factor (GAUGE/BUGTRI) for Sr-90 release from the element was 1.2, while the release reduction factor calculated on a column-average basis for this element was 90. Thus, it can be seen that the reduction in release obtained from individual local points is not as great as that obtained on a column-average basis. The calculation based on individual local points is considered to be more accurate. This latest result indicates that FSAR predictions of a small direct release of Sr-90 were probably slightly too high.

Thus, the following conclusions can be made regarding calculated fission product release from the FSV core using BUGTRI versus GAUGE input:

1. Circulating (i.e., Kr-85m) activity is lower by about 15% after 2 years when BUGTRI input is employed. The amount of reduction increases as the equilibrium cycle is approached.

TABLE 1
Kr-85m R/B FOR 16 BUFFER ELEMENTS AS
COMPUTED BY SURVEY (REF. 4)

Time (days)	R/B		BUGTRI/GAUGE
	BUGTRI	GAUGE	
0	7.379×10^{-6}	7.549×10^{-6}	0.977
60	1.042×10^{-5}	1.089×10^{-5}	0.957
150	1.453×10^{-5}	1.581×10^{-5}	0.919
224	1.113×10^{-5}	1.305×10^{-5}	0.853
314	1.316×10^{-5}	1.564×10^{-5}	0.841
620	2.165×10^{-5}	2.610×10^{-5}	0.830

TABLE 2
COLUMN-AVERAGE METALLIC RELEASE FOR 16
BUFFER ELEMENTS AS COMPUTER BY TRAFIC (REF. 3)

Region	Column	Sr-90 Release at 2 yr (Ci)	
		GAUGE(a)	BUGTRI(a)
21	3	8.14×10^{-9}	1.98×10^{-10}
21	4	4.66×10^{-9}	4.25×10^{-11}
22	3	1.04×10^{-8}	3.10×10^{-10}
23	6	6.52×10^{-7}	7.77×10^{-8}
24	5	1.82×10^{-10}	1.29×10^{-11}
26	7	6.13×10^{-7}	4.52×10^{-8}
27	5	5.23×10^{-9}	1.40×10^{-10}
27	6	2.48×10^{-9}	1.85×10^{-11}
28	5	2.07×10^{-7}	4.12×10^{-9}
29	12	5.40×10^{-7}	9.86×10^{-8}
30	6	8.70×10^{-10}	1.18×10^{-11}
30	7	7.77×10^{-10}	1.16×10^{-11}
31	7	2.47×10^{-10}	2.68×10^{-12}
34	7	2.65×10^{-8}	5.52×10^{-10}
35	4	1.63×10^{-9}	2.37×10^{-10}
37	2	4.02×10^{-9}	3.03×10^{-10}
Total		2.08×10^{-6}	2.27×10^{-7}
		Total Sr $\frac{\text{BUGTRI}}{\text{GAUGE}} = 0.11$	

(a) Power history source.

2. Strontium release calculated on a column-average basis is about a factor of 9 lower when the more accurate BUGTRI power distributions are used. However, the overall reduction in strontium inventory is not substantial because the amount of strontium directly released to the coolant is small compared to that produced by Kr-90 precursor decay.
3. When metallic release is calculated on a local point basis and summed, the reduction factor obtained with BUGTRI is less than that obtained for column-average calculations.
4. The considerably cheaper analysis based on GAUGE gives conservative results.

REFERENCES

1. Merrill, M. H., "Nuclear Design Methods and Experimental Data in Use at Gulf General Atomic," Gulf General Atomic Report Gulf-GA-A12652 (GA-LTR-2), July 1973.
2. "Public Service Company of Colorado 330-MW(e) High-Temperature Gas-Cooled Reactor Research and Development Program Quarterly Progress Report for the Period Ending December 31, 1976," ERDA Report GA-A14270, General Atomic Company, January 1977.
3. Alberstein, D., P. D. Smith, and M. J. Haire, "Metallic Fission Product Release from the HTGR Core," General Atomic Report GA-A13258 (GA-LTR-20), May 15, 1975.
4. Shenoy, A. S., and D. McEachern, "HTGR Core Thermal Design Methods and Analyses," General Atomic Report GA-A12985 (GA-LTR-17), December 31, 1974, Section 7.
5. "HTGR Accident Initiation and Progression Analysis Status Report, Volume V, AIPA Fission Product Source Terms," ERDA Report GA-A13617, General Atomic Company, February 1976, Section 3.