



NUREG/CR-4402
Volume 2
ORNL/TM-9798/V2

OAK RIDGE
NATIONAL
LABORATORY

MARTIN MARIETTA

High-Temperature Gas-Cooled
Reactor Safety Studies for the
Division of Accident Evaluation
Quarterly Progress Report,
April 1-June 30, 1985

S. J. Ball
J. C. Cleveland
R. M. Harrington
J. H. Wilson

Prepared for the U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Under Interagency Agreements DOE 40-551-75 and 40-552-75

8604010169 860228
PDR NUREG
CR-4402 R PDR

OPERATED BY
MARTIN MARIETTA ENERGY SYSTEMS, INC.
FOR THE UNITED STATES
DEPARTMENT OF ENERGY

NOTICE

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, or any of their employees, makes any warranty, expressed or implied, or assumes any legal liability or responsibility for any third party's use, or the results of such use, of any information, apparatus product or process disclosed in this report, or represents that its use by such third party would not infringe privately owned rights.

Available from

Superintendent of Documents
U.S. Government Printing Office
Post Office Box 37082
Washington, D.C. 20013-7982

and

National Technical Information Service
Springfield, VA 22161

NUREG/CR-4402
Volume 2
ORNL/TM-9798/V2
Dist. Category R8

HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR
THE DIVISION OF ACCIDENT EVALUATION QUARTERLY
PROGRESS REPORT, APRIL 1-JUNE 30, 1985

S. J. Ball, Manager
J. C. Cleveland
R. M. Harrington
J. H. Wilson

Manuscript Completed — January 21, 1986
Date Published — February 1986

NOTICE: This document contains information of a preliminary nature. It is subject to revision or correction and therefore does not represent a final report.

Prepared for the
U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Under Interagency Agreements DOE 40-551-75 and 40-552-75

NRC FIN No. B0122

Prepared by the
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37831
operated by
MARTIN MARIETTA ENERGY SYSTEMS, INC.
for the
U.S. DEPARTMENT OF ENERGY
under Contract No. DE-AC05-84OR21400

CONTENTS

	<u>Page</u>
PRIOR HTGR SAFETY REPORTS	v
FOREWORD	vii
ABSTRACT	1
1. HTGR SYSTEMS AND SAFETY ANALYSIS	1
1.1 Investigations of Modular HTGR Dynamics	1
1.2 Fission-Product Release from HTGRs	4
1.3 HTGR Safety Handbook	4
1.4 Review of FSV Reactor Technical Specifications on Limiting Maximum Core Temperatures	5
1.5 Cooperative Programs with the Federal Republic of Germany	5
2. TRIPS MADE UNDER PROGRAM SPONSORSHIP	7
2.1 Mid-Year Program Review, NRC-Bethesda, Maryland, April 2-3, 1985	7
2.2 IAEA Specialists Meeting on Gas-Cooled Reactor Safety and Accident Analysis, Oak Ridge, Tennessee, May 13-15, 1985	7
2.3 DOE/NRC Meeting on Licensing Plans for the Advanced HTGR Plant Design, Bethesda, Maryland, May 31, 1985	7
REFERENCE	8

PRIOR HTGR SAFETY REPORTS

Quarterly Progress Reports

<u>Ending date</u>	<u>Designation</u>
September 30, 1974	ORNL/TM-4798
December 31, 1974	ORNL/TM-4805, Vol. IV
March 31, 1975	ORNL/TM-4914, Vol. IV
June 30, 1975	ORNL/TM-5021, Vol. IV
September 30, 1975	ORNL/TM-5128
December 31, 1975	ORNL/TM-5255
March 31, 1976	ORNL/NUREG/TM-13
June 30, 1976	ORNL/NUREG/TM-43
September 30, 1976	ORNL/NUREG/TM-66
December 31, 1976	ORNL/NUREG/TM-96
March 31, 1977	ORNL/NUREG/TM-115
June 30, 1977	ORNL/NUREG/TM-138
September 30, 1977	ORNL/NUREG/TM-164
December 31, 1977	ORNL/NUREG/TM-195
March 31, 1978	ORNL/NUREG/TM-221
June 30, 1978	ORNL/NUREG/TM-233
September 30, 1978	ORNL/NUREG/TM-293
December 31, 1978	ORNL/NUREG/TM-314
March 31, 1979	ORNL/NUREG/TM-336
June 30, 1979	ORNL/NUREG/TM-356
September 30, 1979	ORNL/NUREG/TM-366
December 31, 1979	ORNL/NUREG/TM-383
March 31, 1980	ORNL/NUREG/TM-397
June 30, 1980	ORNL/NUREG/TM-415
September 30, 1980	ORNL/NUREG/TM-429
December 31, 1980	ORNL/TM-7809
March 31, 1981	ORNL/TM-7889
June 30, 1981	ORNL/TM-8091
September 30, 1981	ORNL/TM-8128
December 31, 1981	ORNL/TM-8260
March 31, 1982	ORNL/TM-8443/V1
June 30, 1982	ORNL/TM-8443/V2
September 30, 1982	ORNL/TM-8443/V3
December 31, 1982	ORNL/TM-8443/V4
March 31, 1983	ORNL/TM-8921/V1
June 30, 1983	ORNL/TM-8921/V2
September 30, 1983	ORNL/TM-8921/V3
December 31, 1983	ORNL/TM-8921/V4
March 31, 1984	ORNL/TM-9267/V1
June 30, 1984	ORNL/TM-9267/V2
September 30, 1984	ORNL/TM-9267/V3
December 31, 1984	ORNL/TM-9267/V4
March 31, 1985	ORNL/TM-9798/V1

FOREWORD

High-temperature gas-cooled reactor safety studies at Oak Ridge National Laboratory are sponsored by the Division of Accident Evaluation (formerly the Division of Reactor Safety Research), which is part of the Office of Nuclear Regulatory Research of the Nuclear Regulatory Commission.

This report covers work performed from April 1 to June 30, 1985. Previous quarterly reports and topical reports published to date are listed on pages v and vi. Copies of the reports are available from the Technical Information Center, U.S. Department of Energy, Oak Ridge, TN 37831.

HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR
THE DIVISION OF ACCIDENT EVALUATION QUARTERLY
PROGRESS REPORT, APRIL 1-JUNE 30, 1985

S. J. Ball, Manager

J. C. Cleveland R. M. Harrington
J. H. Wilson

ABSTRACT

Modeling, code development, and analyses of the modular High-Temperature Gas-Cooled Reactor (HTGR) continued with work on the side-by-side design. Fission-product release and transport experiments were completed. Sections of an HTGR safety handbook were written.

1. HTGR SYSTEMS AND SAFETY ANALYSIS

Work for the Division of Accident Evaluation (formerly Reactor Safety Research) under the High-Temperature Gas-Cooled Reactor (HTGR) Systems and Safety Analysis Program began in July 1974, and progress is reported quarterly. Work during this quarter included continuation of model and code development for the modular HTGRs. Fission-product (FP) release and transport experiments and technical assistance work on a Fort St. Vrain (FSV) technical specification (tech spec) review were continued. Work was initiated on sections of an HTGR Safety Handbook.

1.1 Investigations of Modular HTGR Dynamics

R. M. Harrington S. J. Ball

A paper entitled "Simulation of Thermal Response of the 250 MWT Modular HTGR During Hypothetical Uncontrolled Heatup Accidents" was prepared and presented on May 14, 1985, at the International Atomic Energy Agency (IAEA) Specialists Meeting on Gas-Cooled Reactor Safety and Accident Analysis. The conference was held at Oak Ridge National Laboratory (ORNL) May 13-15, 1985. The abstract for this paper is reproduced below:

One of the central design features of the 250 MWT modular HTGR is the ability to withstand uncontrolled heatup accidents without severe consequences. This paper describes calculational studies, conducted at Oak Ridge National Laboratory under the auspices of the U. S. Nuclear Regulatory Commission's HTGR Research Program, to test this important design feature. A multi-node thermal-hydraulic model of the 250 MWT modular HTGR reactor core was developed and implemented in the IBM CSMP (Continucus System Modeling Program) simulation language.

The code is capable of predicting the peak fuel, reflector, and reactor vessel temperatures reached following permanent loss of forced primary coolant circulation accidents with or without concomitant loss of steam generator cooling water flow or accidental depressurization of the helium primary coolant.

Survey calculations show that the loss of forced circulation accident with loss of steam generator cooling water and with accidental depressurization is the most severe heatup accident. The peak hot-spot fuel temperature is in the neighborhood of 1600°C. Fuel failure and fission product releases for such accidents would be minor. Sensitivity studies show that code input assumptions for thermal properties such as the side reflector conductivity have a significant effect on the peak temperature.

A computer model of the reactor vessel cavity concrete wall and its surrounding earth was developed. This model was used to simulate the extremely unlikely and very slowly-developing heatup accident that would take place if the worst-case loss of forced primary coolant circulation accident were further compounded by the loss of cooling water to the reactor vessel cavity liner cooling system. The simulation results show that the ability of the earth surrounding the cavity to act as a satisfactory long-term heat sink is very sensitive to the assumed rate of decay heat generation and on the effective thermal conductivity of the earth. Soil properties at some sites may not be suitable.

As part of the continuing study of the dynamics of modular HTGR core heatup accidents, several runs were made to learn more about the significance of the heat loss from the exterior of the reactor vessel. The dynamic simulator code described in the recent quarterly reports was used to perform the necessary calculations; input applicable to the vertical in-line concept was retained.

The accident chosen for study is the depressurized loss-of-forced convection (LOFC) with the concomitant failure of the steam generator feedwater system. Heat flow from the exterior of the reactor vessel to the surrounding reactor cavity, and its liner cooling system, would play an important part in the eventual recovery from this accident. The calculations, described below, were performed to evaluate the effect of this heat loss on average and maximum fuel temperatures during the first day or two following accident initiation.

For the first calculation, a LOFC with loss of feedwater was assumed, and the heat removal from the exterior of the reactor vessel was reduced by assuming failure of the liner cooling system (LCS) and by assigning an artificially high value of thermal resistance between the cavity liner and its concrete wall. In this case the predicted maximum hot node fuel temperature of 1567°C, occurring after 24 h, is only 17°C higher than the peak for the analogous LOFC with the LCS running (and with no increase in liner-to-concrete thermal resistance).

To obtain a result with a more clear-cut difference, it was decided that the next calculation should be an LOFC with loss of feedwater in which the heat loss from the exterior of the reactor vessel is held at

zero. This cannot be done by any physical means but is easily accomplished by modifying the programming of the simulation model to set the heat loss from the reactor vessel to zero throughout the whole accident. In this case the predicted maximum hot node fuel temperature is only 30°C higher than for the analogous case with the LCS running, but the average fuel temperature peak is 60°C higher. As expected, there is a dramatic difference in reactor vessel temperatures for the two cases. The maximum vessel temperature is 532°C after 36 h for the no-heat-loss case, 247°C higher than the case with nominal vessel heat loss and LCS operation.

A more complete comparison between these two cases is presented in Fig. 1 for the first 36 h following accident initiation. From this figure we see that the difference between hot node temperatures is truly small but that the behavior of average fuel temperature is significantly different. The average temperature for the no-heat-loss case, although decreasing slowly at the 36-h point, would begin to increase again some time after 36 h and would continue to do so thereafter. The average fuel temperature for the nominal heat loss case is under control and decreasing rapidly after 36 h.

The conclusion from these cases is that, during the first several days after accident initiation, cavity cooling is not very important to the peak hot node fuel temperature, somewhat important to average fuel temperature, and very important to reactor vessel temperature. This

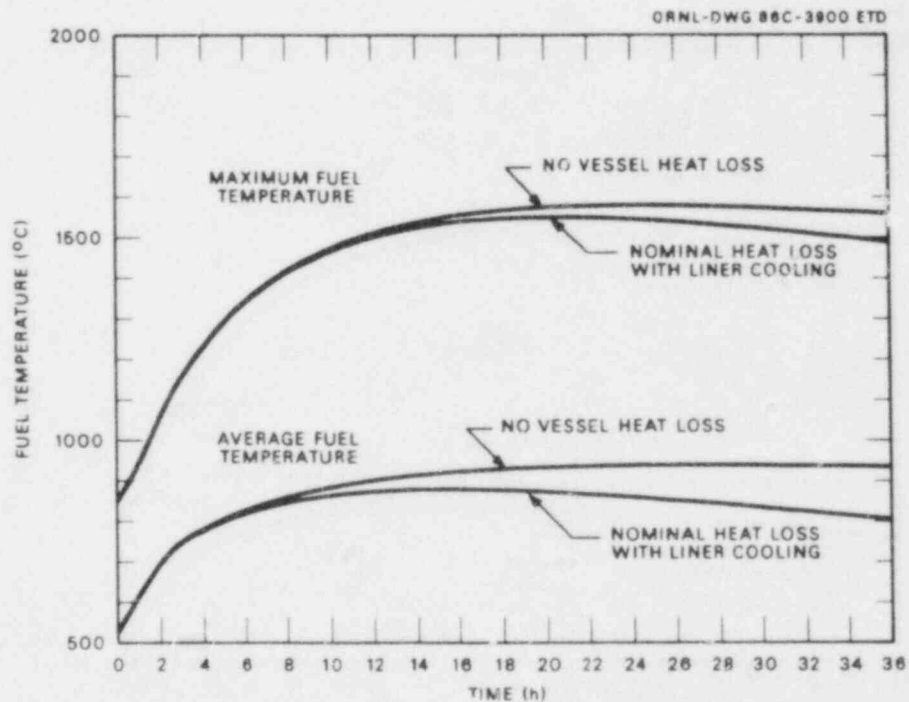


Fig. 1. Effect of reactor vessel heat loss on fuel temperatures for the loss of forced-convection accident.

result is reasonable because the peak fuel temperature occurs near the center of the core, which is separated from the outside of the reactor vessel by several meters thickness of fuel pebbles and graphite reflector blocks.

1.2 Fission-Product Release from HTGRs

J. H. Wilson C. M. Simmons

The objective of this task is to generate experimental data required for the analysis of FP release in HTGR severe accidents. Initial efforts involved the determination of FP vapor pressures and diffusion rates through graphite. These data are required for calculation of the rate of transport of FPs from failed fuel particles through graphite to the coolant. In the next phase, the interaction of FPs with prestressed concrete reactor vessel (PCRv) materials of construction will be studied. The experimental data will be used for the prediction of FP retention in the PCRv.

Several diffusion experiments using both the graphite tube and the open-end zirconia tube were performed after a reanalysis of experimental data showed the need for these additional tests. The results were incorporated into the report on the FP transport studies. As discussed in the last quarterly report, a draft report was written that covered the work for the past 1.5 years on FP vapor pressures and diffusion coefficients in graphite. The main conclusion of this report was that the transport of the FPs that were studied (Al, Ce, La, Pd, Pr, Rh, Ru, and Y) was described by a gas phase diffusion model. The draft report was prepared for submission and review.

Work on the study of FP interaction with materials of construction was initiated. The objective of the experimental design is to generate data that will be applicable to both HTGRs and PWRs. A preliminary design of an experimental apparatus was made. Plans are to use the high-temperature graphite resistance furnace to perform tests on the interaction of FPs with steel. This should provide information useful for the final equipment design.

1.3 HTGR Safety Handbook

S. J. Ball J. C. Mailen
 J. C. Cleveland D. L. Moses
 J. C. Conklin J. P. Sanders
 R. M. Harrington R. P. Wichner
 J. H. Wilson

ORNL was assigned to write major sections of the HTGR Safety Handbook (other contributors are Brookhaven National Laboratory and Los Alamos National Laboratory). This handbook is to be an up-to-date, complete, and reliable data base as well as a general reference source for HTGR safety technology. It is designed to be applicable to FSV as well as to the advanced (modular) HTGR designs. Drafts or finished versions of major parts of the following sections were written and

reviewed during the reporting period: Section 2 (NSSS Descriptions), Section 3 (Component and Subsystem Descriptions), Section 4 (Design Basis Accidents), Section 8 (Fission Product Release and Transport), Section 9 (Core Physics), Section 10 (Material and Equipment Aging), and Section 11 (Safety Issues). More work on sections 2, 8, and 9 is planned for the near term.

1.4 Review of FSV Reactor Technical Specifications on Limiting Maximum Core Temperatures

S. J. Ball

The ORNL review of the FSV tech spec limiting conditions for operation (LCO) 4.1.9 addresses limiting maximum core temperatures during low-power (0 to 15%) operation. It is sponsored primarily by a sub-contract with the Nuclear Regulatory Commission (NRC) Region IV and is part of a large tech spec upgrade program.

A series of low-power, low-flow "benchmark" scenarios, set up in a cooperative effort with GA Technologies (GAT), were run and analyzed using the modified version of the ORECA code,¹ which calculates intraregion flow redistributions. The main conclusions were that with equal refueling region orifice settings, the minimum flows needed to prevent stagnation, under the most conservative sets of high region peaking factors (RPF) conditions, are 4% at 1% power and 5% at 5% power. In the latter case, however, out-of-limits core exit and fuel temperatures occur at flows as high as 13%. Cases were also run for a worst-case minimum RPF (0.4) and maximum column power tilt factor (1.507). The ORECA code predictions, when the refueling region orifices are adjusted for nearly-equal region outlet temperatures, show that at least 4% flow is needed (with 1% power) to avoid intraregion flow stagnation. However, with an arbitrary minimum orifice setting of 10%, flow stagnation does not occur in the critical regions until the flow is reduced to ~2%. This indicates a possible desirable operating strategy. Benchmark comparisons showed that the differences between the ORNL (ORECA code) results and GAT's calculations were relatively small for the 1% power case. However, GAT's analysis of the 5% power case indicated that considerably more total core flow would be required to prevent flow stagnation in the critical region than ORNL's analysis had shown. Work on explaining this difference is continuing.

These and other aspects of the problem, such as heatup times and corrective measures, were discussed in a summary letter report to NRC.

1.5 Cooperative Programs with the Federal Republic of Germany

J. C. Cleveland

A meeting was held on May 17 at ORNL with W. Rehm and E. Teuchert of KFA-Jülich. ORNL personnel included S. J. Ball, R. M. Harrington, D. L. Moses, and J. C. Cleveland. The main focus of the meeting was to identify potential areas of cooperation between ORNL and KFA. Areas

identified included code development and validation, accident analysis, development of theoretical models for pebble bed heat transfer under accident conditions, and risk analysis. Technical discussions included KFA and ORNL dynamic analysis capabilities for HTRs and a presentation by KFA of results of their analysis of THTR safety.

2. TRIPS MADE UNDER PROGRAM SPONSORSHIP

2.1 Mid-Year Program Review,
NRC-Bethesda, Maryland,
April 2-3, 1985S. J. Ball R. M. Harrington
J. C. Cleveland J. H. Wilson

ORNL presentations at the meeting summarized accomplishments in the areas of FP diffusion experiments, code development and verification, and accident analyses. A major purpose of the meeting was to advise the Office of Nuclear Reactor Regulation Advanced Reactors Group of the tools and capabilities available to handle the upcoming licensing work on the Department of Energy's modular HTGR plant design. The meeting was held jointly with the BNL and LANL HTGR safety research programs reviews.

2.2 IAEA Specialists Meeting on Gas-Cooled
Reactor Safety and Accident Analysis,
Oak Ridge, Tennessee,
May 13-15, 1985S. J. Ball J. C. Cleveland
R. M. Harrington

Program personnel attended the sessions and presented two program-sponsored papers: "Simulation of Thermal Response of the 250 MWT Modular HTGR During Hypothetical Uncontrolled Heatup Accidents" (see Sect. 1.1) and "ORNL's NRC-Sponsored HTGR Safety and Licensing Analysis Activities for Fort St. Vrain and Advanced Reactors."

2.3 DOE/NRC Meeting on Licensing Plans for
the Advanced HTGR Plant Design,
Bethesda, Maryland,
May 31, 1985

R. M. Harrington

DOE and their contractors presented a proposed licensing plan and approach to developing an HTGR licensing bases for initial NRC review and comment.

REFERENCE

1. S. J. Ball, *ORECA-I: A Digital Computer Code for Simulating the Dynamics of HTGR Cores for Emergency Cooling Analyses*, ORNL/TM-5159, Union Carbide Corp. Nuclear Div., Oak Ridge Natl. Lab., April 1976.

NUREG/CR-4402
 Volume 2
 ORNL/TM-9798/V2
 Dist. Category R8

Internal Distribution

1-5.	S. J. Ball	26.	D. L. Moses
6.	N. E. Ciapp	27.	J. P. Sanders
7-11.	J. C. Cleveland	28.	R. S. Stone
12.	J. C. Conklin	29.	H. E. Trammell
13-14.	J. A. Getsi	30.	R. P. Wichner
15-19.	R. M. Harrington	31-35.	J. H. Wilson
20.	J. E. Jones, Jr.	36.	I&C Publications Office
21.	P. R. Kasten	37.	ORNL Patent Office
22.	A. D. Kelmers	38.	Central Research Library
23.	T. S. Kress	39.	Document Reference Section
24.	T. B. Lindemer	40-41.	Laboratory Records Department
25.	A. P. Malinauskas	42.	Laboratory Records, RC

External Distribution

43-46. Director, Office of Nuclear Regulatory Research, Nuclear Regulatory Commission, Washington, DC 20555

47. Chief, Advanced Safety Technology Branch, Division of Accident Evaluation, Office of Nuclear Regulatory Research, Nuclear Regulatory Commission, Washington, DC 20555

48. Office of Assistant Manager for Energy Research and Development, DOE, ORO, Oak Ridge, TN 37831

49-50. Technical Information Center, DOE, Oak Ridge, TN 37831

51-250. Given distribution as shown in category R8 (10-NTIS)

BIBLIOGRAPHIC DATA SHEET

NUREG/CR-4402
Volume 2
ORNL/TM-9798/V2

SEE INSTRUCTIONS ON THE REVERSE

2. TITLE AND SUBTITLE

High-Temperature Gas-Cooled Reactor Safety Studies for
the Division of Accident Evaluation Quarterly Progress
Report, April 1 - June 30, 1985

3. LEAVE BLANK

4. DATE REPORT COMPLETED

MONTH YEAR
January 1986

5. DATE REPORT ISSUED

MONTH YEAR
February 1986

5. AUTHOR(S)

S. J. Ball, J. C. Cleveland, R. M. Harrington,
J. H. Wilson

8. PROJECT/TASK WORK UNIT NUMBER

B0122

7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

Oak Ridge National Laboratory
P. O. Box X
Oak Ridge, Tennessee 37831

9. PIN OR GRANT NUMBER

10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

Division of Accident Evaluation
Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

1a. TYPE OF REPORT

Quarterly

b. PERIOD COVERED (Inclusive dates)

April 1 - June 30, 1985

12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

Modeling, code development, and analyses of the modular High-Temperature Gas-Cooled Reactor (HTGR) continued with work on the side-by-side design. Fission-product release and transport experiments were completed. Sections of an HTGR safety handbook were written.

14. DOCUMENT ANALYSIS - a. KEYWORDS/DESCRIPTORS

15. AVAILABILITY STATEMENT

Unlimited

16. SECURITY CLASSIFICATION

(This page)
Unclassified

(This report)
Unclassified

17. NUMBER OF PAGES

18. PRICE

b. IDENTIFIERS/OPEN-ENDED TERMS

120555078877 1 1AN1R8
US NRC
ADM-DIV OF TIDC
POLICY & PUB MGT BR-PDR NUREG
W-501
WASHINGTON DC 20555